Organisation for Economic Cooperation and Development

Nuclear Energy Agency

Committee on the Safety of Nuclear Installations

OECD–NEA–CSNI Specialist Meeting

HIGH TEMPERATURE GAS COOLED REACTOR SAFETY

Current Status and Perspective

Petten, The Netherlands, 13–15 May 1975

PROCEEDINGS

Volume B

— Programme
— Papers

REACTOR CENTRUM NEDERLAND
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Distribution: 80 participants
40 CSNI-members
20 NEA
20 RCN

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Petten, July 1976
OECD–NEA–CSNI  
HTGR Safety Meeting  
RCN–Petten, 13–15 May 1975

PROGRAMME OF THE MEETING

Tuesday, 13 May

Registration

Opening of meeting by J. Pelser, Reactor Centrum Nederland  
Welcoming address by K.B. Stadie, OECD – NEA – CSNI

Chairman: J. Pelser

| Invited       | F.R. Farmer (UKAEA–SRD):                                      |
| Paper         | Current safety considerations and their effect on HTGR safety |
|               | Page: 6                                                       |
| Invited       | O. Kellermann (FRG–IRS):                                     |
| Paper         | HTR – Safety philosophy in the Federal Republic of Germany    |
|               | Page: 15                                                      |
| Invited       | R.D. Schamberger (USNRC):                                    |
| Paper         | HTGR safety research objectives                               |
|               | Page: 36                                                      |
| Invited       | L. Shepherd (OECD Dragon Project):                            |
| Paper         | Design features of high temperature reactors                 |
|               | Page: 44                                                      |

Session I: Fission product behaviour

Chairman: R.D. Schamberger

| Paper                     | F.P.O. Ashworth and J. Kirk:                                  |
| SNI 4/1                   | Gas-cooled reactor safety technology                         |
|                           | Page: 58                                                     |
| Paper                     | R.L. Faircloth and A.N. Knowles:                             |
| SNI 4/2                   | Methods for the prediction and achievement of a low circuit activity |
|                           | Page: 116                                                    |
| Paper                     | F. Abbey:                                                    |
| SNI 4/3                   | Some comments on fission product behaviour in HTGR's and its impact on safety |
|                           | Page: 144                                                    |
| Paper                     | C.B. von der Decken, N. Iniotakis and K.H. Münchow:          |
| SNI 4/4                   | Behaviour of fission products in case of a depressurisation accident |
|                           | Page: 160                                                    |
Wednesday, 14 May

Session II : Specific accident considerations

Chairman: F.R. Farmer

<table>
<thead>
<tr>
<th>Paper</th>
<th>SNI 4/5</th>
<th>H. Eisele, H.W. Gabriel and J.A. Redondo: The environmental impacts of the HTGR 1160 and the risk assessment</th>
<th>page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Paper</td>
<td>SNI 4/7</td>
<td>K. Schifferstein: Requirements for the afterheat removal systems of high-temperature reactors in the Federal Republic of Germany</td>
<td>242</td>
</tr>
<tr>
<td>Paper</td>
<td>SNI 4/9</td>
<td>H.P. Drescher: Aspects of measuring, identifying and controlling steam generator leaks in HTGR's</td>
<td>269</td>
</tr>
<tr>
<td>Paper</td>
<td>SNI 4/10</td>
<td>A.W. Barsell, V. Joksimovic and M.B. Peroomian: Consequences of HTGR water ingress events into primary coolant system</td>
<td>275</td>
</tr>
</tbody>
</table>

Session III : Containment Aspects

Chairman: H. Sameith

<table>
<thead>
<tr>
<th>Paper</th>
<th>SNI 4/11</th>
<th>W. Kröger: Environmental and safety aspects of containment system</th>
<th>page</th>
</tr>
</thead>
<tbody>
<tr>
<td>Paper</td>
<td>SNI 4/12</td>
<td>R.K. Deremer and V. Joksimovic: Need for a conventional containment on HTGR's</td>
<td>313</td>
</tr>
</tbody>
</table>
Session IV: Causes and consequences of ultimate accidents

Paper
SNI 4/13
K.W. Otto, H. Bonka, J. Lukascevicz and R. Schulten:
Temperature behaviour of high-temperature reactors
in the case of hypothetical accidents

Paper
SNI 4/14
F.A. Silady and V. Joksimovic:
Consequences of unrestricted core heat-up event

Thursday, 15 May

Session V: Special problems arising from applications other than electricity production

Chairman: C.B. von der Decken

Paper
SNI 4/15
G.D. Bell and J. Shepherd:
Safety problems arising from process heat applications
of HTGR's

Paper
SNI 4/16
R. Schulten, K. Kugeler, C.B. von der Decken and
R. Hecker:
Studies for intermediate heat transfer loops in
process heat reactors

Paper
SNI 4/17
H. Barnert, K. Petersen, M. Schäfer, H.J. Scharf,
J. Engelhardt and G. Schroeder:
Safety aspects of the FR 500 for the combined production
of electricity and district heat

Session VI: Requirements in the field of HTGR safety

Paper
SNI 4/18
F.P.O. Ashworth and H.J. de Nordwall:
The status of information required for an independent
analysis of risks associated with an HTR

Paper
SNI 4/19
D.R.H. Fryer:
A review of certain gas-cooled reactor safety issues

Panel:
Future requirements and international collaboration
in the field of HTGR safety

E.V. Gilby (Chairman)
L.R. Shepherd
C.B. von der Decken
J. Fassbender
R.D. Schamberger
K.B. Stadie
Invited Paper

OECD/NEA/CSNI Specialist Meeting on High Temperature
Gas Cooled Reactor Safety - Current Status and Perspectives,
Petten, 13th-15th May, 1975

Current Safety Considerations and their Effect
on HTGR Safety

F. R. FARMER and F. ABBEY

Safety and Reliability Directorate, UKAEA,
Culcheth, Warrington, England
Current Safety Considerations and Their Effect on HTGR Safety

F. R. FARMER and F. ABBEY

Safety and Reliability Directorate, UKAEA,
Culcheth, Warrington, England

1. INTRODUCTION

For many years there have been attempts to show that a very serious reactor accident could not happen, or that should the accident occur the more serious consequences could be avoided. More recently many of the earlier proposals are being re-examined and the degree of incredibility of some events again being brought into question. In the USA this has been reflected in the preparation and issue in the last few months of Professor Rasmussen's assessment of accident risks in US commercial (light water) nuclear power plants(1).

The recognition that there is a price for risk taking carries with it the problem of developing acceptable standards of risk, i.e., of achieving some balance between risk and benefit which is judged fair by most people. In the field of normal operation of nuclear facilities a parallel situation is formulated in the ICRP's requirement that "all doses be kept as low as is reasonably achievable, economic and social considerations being taken into account". In the field of risks to people which could stem from accidents to nuclear plant we have no upper limit to the frequency of accidents and, as yet, there does not seem to be any clear consensus of opinion on "How safe is safe enough". Opinion will be varied in any one country and will differ from country to country depending on social conditions and previous and present history of hazards. It may well be that opinions will come closer together in the nuclear industry as a serious accident to any plant can threaten the framework of credibility established elsewhere.

My position on risk targets has been discussed in a number of presentations in recent years (2)(3). In the present paper I wish to re-state some of these views, to mention briefly the question of external accidents to reactors which is currently attracting attention, and to discuss the impact of these topics on the safety of the HTGR. Some key issues on which assurance will be needed for the HTGR are identified.

2. RISK STANDARDS

A release of a few hundreds of curies of I131 has a very low probability of causing harm to the general public and a release ten times as high - of the order of say 5,000 curies - from a typical reactor site is unlikely to cause one case of fatal illness among the general public within the ensuing ten or more years. It follows that the frequency with which the industry could tolerate accidents within the range zero to a few thousand curies will be determined in the long term by financial consideration, upset to plant or loss of production, rather than the very small risk of hurt to people. In the short term a more important factor may be the public concern arising from fear of radiation exposure even when carrying only a very small risk of harm.
suggested that we might use risk criteria of $10^{-5}$ per person per year applied to the few people in the first few kilometres around the reactor. It would be easy to show that a criterion of $10^{-6}$ for a major accident per reactor per year would imply a risk of harm to an individual at 1 kilometre from the site within the range $10^{-7}$ to $10^{-8}$ per year. To apply the individual risk limit of $10^{-5}$ per person per year implies the acceptance of a major accident in the range of $10^{-3}$ to $10^{-4}$ per reactor per year.

Apart from direct hurt to people, accidents to nuclear plant can result in disturbance of the normal patterns of living by contamination of surrounding areas. In the large accident referred to, contamination of pasture land by I131 could require control on the consumption of milk to a distance of 100 miles or over an area of 2,000 square miles. The dose level from contamination of ground, and particularly from Cs137, could affect the re-occupation of land evacuated at the time of the incident. In terms of a limit of 0.5 rads per year, the ICRP whole body gamma limit for individual members of the general public, the complete reoccupation of a sector out to a distance of 10 miles might well be delayed for a year or more. A simple costing exercise by my colleagues(4) has shown that at all levels of release it is the latter effect, the low level but persistent gamma dose caused by ground deposition, which makes the dominant contribution to costs and the data and methods used would have to be very much changed before this conclusion was affected. The financial penalty (claims from third parties) from a release of up to $10^3$ curies of Cs137 would be £$10^5$ or less which might not be thought excessive in relation to a plant which costs £$10^8$ in round terms to construct, and produces £$10^7$ in round terms in annual revenue. The possibility remains, of course, that the plant might be shut down and even written off as a result of the accident and in relation to this size of accident the need of the operator to protect his investment in the plant seems still to be of overriding importance. A release of $10^5$ curies of Cs137 could lead to a financial penalty in the range £$10^8$ - £$10^9$, probably much greater than the financial loss of the plant itself which would almost certainly have to be written off. The acceptable frequency of such a release on purely financial grounds might be $10^{-4}$ per reactor per year, corresponding to a loss rate of 1% of the revenue produced.

In summary, although individual risk and financial penalties from Cs137 deposition are important considerations, when the intangibles deriving from the special nature of nuclear power are taken into account it is the recognition of collective harm to society which in my subjective assessment will determine standards of risk for large accidents. On these grounds a frequency of no more than about $10^{-6}$ per reactor per year for major accidents appears desirable and may be at the limit of what can be attained in practice.

3. **EXTERNAL ACCIDENTS**

Any treatment of reactor safety must give consideration to hazards originating outside the reactor complex. The HTGR may have particular application to processes heat generation and adjacent to industrial areas there may be special hazards from, inter alia, vapour cloud explosions, missiles generated by failure of pressurised vessels, fire, or toxic gases or fumes(5). Elsewhere more general hazards may be more important, e.g., crashing aircraft. Studies in the UK, France and the USA have been carried out to determine the frequency of aircraft crashes on vital reactor areas. Provided that the reactor is not sited within about 5 miles of an airfield the risk is assessed at about $10^{-7}$ per year for large commercial aircraft. It may be somewhat
How sensitive are the economics of the system to the burn-up which can be achieved within this requirement and how adequate is the demonstrated experience.

An alternative approach would be to demonstrate that the activity deposited round the circuit is fixed and not removable in accidents. It would be necessary to guarantee this with a degree of certainty again commensurate with the magnitude of the amounts of activity potentially available for release and this approach would be of no value in facilitating access to the circuit. In the UK the Magnox system has produced negligible circuit activity from either fission or corrosion products and man access to regions shielded from direct radiation from the core has been possible. A custom of boiler inspection closely related to conventional practice has, therefore, been followed. However, remote inspection may be possible and it may be that circuit access should be regarded as an operational rather than a safety problem.

Coupled with the limit on failed fuel fraction there must be some method of demonstrating in service that the limit is being maintained. Observation of activity at the heat exchangers would not be very satisfactory since if only the silicon carbide layer and not the pyrocarbon were failed (so-called defective particles) Cs137 could accumulate in core graphite before the heat exchanger activity gave warning. For broken particles (all layers failed) the levels of several gaseous fission products would indicate the situation, but for defective particles more consideration is needed.

A further necessary feature of the fuel is that it should not appear to be intact in normal running, yet fail prematurely and release significant activity due to an accident transient. A similar problem has to be faced in most reactor systems and the main need is again for sufficient preliminary fuel proving.

4.2 The Cooling System

Shut-down cooling is more difficult in the HTGR than in earlier gas-cooled reactors. Because of the high gas temperatures flow may be downwards through the core so that loss of forced circulation may lead to flow reversal in some or all channels, with consequent overheating and possible hazard to circulators and boilers. The high gas temperatures may also indicate that the boilers are more vulnerable to loss of feed than the reactor proper would be to reduction of cooling. Specifically it may be difficult for once-through boilers to accept such an occurrence.

The high circuit pressure means that following depressurisation the disparity of cooling is greater and, more seriously, helium may be in laminar flow in the core and boilers.

A second key issue to be faced for the HTGR is, therefore, the reliability of the cooling system. The question of designing for and validating the high standards of reliability demanded by the risk targets discussed earlier has been considered elsewhere (8)(9) (both as regards cooling systems and pressure vessel closures) and an assessment of the reliability theoretically attainable by the cooling system provided for a gas cooled fast reactor has been reported(10). In the UK design of HTGR the reflectors above and below the core act as thermal sinks allowing extra time, and additional plant is also provided. The main boilers are not required to operate below about 20% full load, auxiliary boilers less susceptible to thermal shock and auxiliary
In the event that a containment system is required there will be a need to demonstrate its value. The choice might be for a vented or a tight building. In the former case the rate at which the helium escapes from the assumed circuit breach in a depressurisation accident is important. In either case a further key issue for a steam cycle system will be water ingress from damaged boilers in depressurisation and the possible formation of water gas because of the high core graphite temperatures in the HTGR. This problem is clearly analogous to that of hydrogen generation in the PWR case. It may be possible to show that water ingress will not occur concurrently with depressurisation (or if it does that the water cannot reach the core); that the quantity of water can be restricted by dumping boiling water; or that the containment can be inerted. In the large accident of low probability the advantages claimed for the HTGR may lead to easier containment. For instance, it may be that the high temperature capability of the core will reduce the possibility of containment melt through.

Following a depressurisation accident even slight contamination of the containment may make access difficult for emergency repairs and maintenance operations\(^\text{(7)}\). In the LWR's machinery to provide long term cooling is generally provided outside the containment. Consideration may need to be given as to whether arrangements allowing access can be made for the HTGR, e.g., a "sealed" floor at pile cap level with machinery in lower rooms that do not require access from above.

5. SUMMARY AND CONCLUSIONS

Some of the considerations important in setting risk standards for nuclear power plant have been reviewed with particular reference to high risk situations and the importance of collective harm to society as a whole highlighted. A target frequency of \(10^{-5}\) per reactor per year is suggested for major accidents.

Viewed against this background and with the experience of earlier gas cooled systems, some of the key issues in assessing the safety of the HTGR are identified as - fuel particle integrity and fission product behaviour in the circuit; the reliability of the cooling system, especially following depressurisation; and the possibility and consequences of water ingress from damaged boiler tubes following depressurisation. The treatment and relative importance of these and other topics will eventually depend on the safety strategy adopted for the reactor.

The increased attention being given to the analysis of major accidents of low probability for other reactor systems is noted and the need for similar studies for the HTGR proposed.
Invited Paper

HTR Licensing Procedure and
Safety Philosophy in the Federal Republic of Germany

O. Kellermann; W. Ullrich

Köln, May 1975
1. German licensing procedure including existing criteria, rules and guidelines

In the Federal Republic of Germany, a lot of experience is available with regard to evaluating and licensing practices for LWRs, i.e. PWRs and BWRs. Fig. 1 shows the German Atomic Licensing Procedure and the connections between the different partners. For HTR's (high-temperature reactors), however, there is rather little experience at hand. The gas-cooled (CO$_2$), D$_2$O-moderated reactor at Niederaichbach (KKN) has been shut down and will be dismantled. The AVR prototype reactor is in successful operation at the Jülich Nuclear Research Installation (KFA). For the THTR at Schmehausen, we are now in the licensing phase, in particular with regard to its construction. For the Gulf HTR, we have entered the phase of evaluating site and concept.

First works with regard to evaluating the Licensability of helium HTR's with integrated helium turbines have been initiated.

As a result of the experience at hand, criteria, rules, and guidelines are often LWR-oriented. The Federal Ministry of the Interior has issued a set of safety criteria which apply to nuclear power plants in general. They apply to both LWR's and HTR's. These criteria were issued in July 1974 after having been approved by the Länder Committee on Atomic Energy. Deviations will only be permitted when the safety goals are not impaired.

The Reactor Safety Commission (RSK), an advisory board to the Federal Ministry of the Interior, has so far only passed the guidelines for PWR's; those for BWR's will follow in the near future. The Nuclear Safety Standards Board (KTA) has displayed activities which are intended to establish rules which will also apply to HTR's.
A few measured values have been obtained from the operation of the AVR reactor at Jülich.

The following table quotes those values which have been applied for and which are expected. In some cases, a certain range is quoted for the values expected, since they have been estimated. In these estimates, manufacturers operators and safety evaluators employ different assumptions (cf. Tab. 3).

- Accident analysis considering internal influences

The cladding tube, i.e. the first barrier, must be designed so as to remain intact under normal operating conditions if anomalies of high or moderate frequency occur. An impairment of this barrier will only be accepted for accidents, in particular design basis accidents.

In any case, even worst accidents have to be controlled so that any radioactive material will be retained definitely by the last barrier which is, in general, the containment. Examples of such accidents are a loss of coolant, an uncontrolled withdrawal of control rods or a steam generator piping rupture.

The investigations of transients conducted by manufacturers and safety evaluators are, in general, not only effected with nominal data. Parameter investigations provide information on the relevant factors which will have a particular effect on the result. For these factors error estimates are made and used in a conservative way for further calculations. Analyses proceed from a broad basic, i.e. safety measures are inducted by different initiation criteria, and different failures are assumed with regard to the safety systems.
The aim of diversity in initiation criteria, e.g. a reactor scram as a result of neutron flux, reactor pressure or temperature initiation, is to assure a shutdown if entire initiation chains fail because of a common mode failure. However, all parties concerned know that complete diversity is very difficult, and frequently not at all, to achieve. These difficulties are, in particular, encountered at HTR's. The scram initiation in case of a rod with drawal accident may be quoted as an example. The first initiation is released by a high neutron flux. There is a second initiation available, the hot-gas temperature signal which, however, involves a considerable delay time. The first initiation will thus have to be designed at a higher standard.

Control is of great importance in functional processes. In most cases, control is not subject to the licensing and evaluating procedure. Frequently it is not even taken note of. This is why, in analyses, worst-case reactions of the control system are assumed. It is only a higher-standard design or a close investigation of the control system that will induce safety evaluators to assume sensible reactions of the control system during an accident.

With regard to LWR's, there is a growing discussion of ATWS problems in the Federal Republic of Germany, same as in the USA. Special investigations are being required which may have effects upon hardware. As far as the THTR is concerned, investigations of the functions of the first and second shutdown systems have been particularly strong. The same is done for the HTR. There is no doubt that in this case LWR's and HTR's are measured by the same standard. At present, comprehensive projects aiming at a detailed investigation of this complex are under way in the Federal Republic, financed by the Federal Ministries of the Interior and for Research and Technology. Results are expected in the course of this year.

In the Federal Republic, 2 independent and diverse shutdown systems are required; one of the two systems must be designed as a scram system.
Various possible developments of an accident are investigated, e.g. occurrence of a single failure and/or the fact that only the second initiation becomes effective instead of the first. Even less probable accidents with multiple failures are investigated. The result is a comprehensive survey of the hazard potential and the corresponding probability of an accident.

The allocation of failure and damage criteria to defined accident categories, as is provided by the ASME Code, has not yet been achieved in the Federal Republic. First measures to solve this problem are being initiated by different boards, e.g. within the scope of the activities of the KTA.

As far as the design of prestressed concrete pressure vessels is concerned, work is based on a collection of material prepared by a Civil and Nuclear Engineering Joint Committee within the German Standards Committee (DNA), and on ASME III/2 (Proposed Standard Code for Concrete Reactor Vessels and Containments). Furthermore, the Accident Prevention Rule "Pressure Vessels" (VGB 17) is used. A proof of the integrity of the liner of a concrete pressure vessel is not required if the pressurized crack is controlled and the containment is designed to resist aircraft crashes (leaktightness function). In the Federal Republic, the question of safety valves is judged from a somewhat different point of view as compared to the USA. A final opinion has not yet been reached.

- Accident analysis considering external influences

The term "external influences" covers a very complex range of different phenomena. According to their causes, they can be sub-divided into two main categories. One comprises effects caused by Acts of God, the other man-made effects in a very broad sense.
How to achieve such optimum protection in HTR's is not yet clear because there is still a certain lack of knowledge concerning the actual behaviour of large-scale nuclear power plants of this type.

In order to cover these uncertainties in a conservative way, the design criteria for external influences which are at present used for the HTR 1160 are the same as for LWR's. According to these criteria, a structural protection (building design) has to be achieved, and the machinery must be capable of compensating for those movements caused by external influences (e.g. vibrations, contour displacements). It is assumed in this context that all those components and systems which can definitely not accommodate these additional loads will fail.

The high safety requirements in the Federal Republic of Germany with regard to the control of internal accidents and external influences have favoured, as far as LWR's are concerned, the development of types where the above-mentioned protection is achieved, to a great extent, as a result of the overall concept of the design against external influences. This developmental advantage of LWR's has to be made good by HTR's, since the detailed design depends on the reactor concept and cannot simply be transferred e.g. from PWR's to the HTR 1160.

Contrary to the LWR's, the so-called reactor protection building (prestressed-concrete containment with liner) of the HTR 1160 has a double function: protection against external influences and vessel function. A further characteristic of the HTR which differs from LWR features is the following:

As a result of the large dimensions of the prestressed-concrete pressure vessel which holds the integrated steam generator systems, other auxiliary systems such as the fuel-element storage and the refuelling facilities are provided outside the reactor protection building.
Tab. 4 clearly indicates that more or less demeshed and redundant residual heat removal systems have been achieved, as is also the case for PWR's.

3. Experience obtained from the licensing procedure for THTR, Consequences for HTR's

The behaviour of a plant in operation under normal conditions and in accident conditions can be calculated with the aid of more or less detailed physical models.

For LWR's a great deal of such work has already been done. With an increasing number of such investigations and thus a growing amount of experience, a few parameters appear the modification of which will considerably influence the behaviour of a system. It will then be possible to make qualitative statements concerning the behaviour of a system, without any additional transients calculation if one or even two parameters are modified.

For HTR's, a comparable amount of experience is not yet available. Nevertheless, when evaluating the concept of the THTR-300 reactor, the decision was to proceed on the lines of the LWR's, i.e. there was only a first, less detailed examination of design and plant behaviour. People relied on their experience with LWR's which had shown that, in most cases, a more detailed examination prior to the individual partial construction phases had only resulted in minor changes of the plant.

Today, as the first partial construction phases have been completed, it has become obvious that, in some cases, tremendous difficulties have arisen which could have been avoided if more detailed investigations had been made in the concept phase.
<table>
<thead>
<tr>
<th>Barrier</th>
<th>LWR</th>
<th>THTR</th>
<th>HTR</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>cladding</td>
<td>multilayer Pyrocarbon</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>graphite matrix as a delay for escaping fission products</td>
</tr>
<tr>
<td>2.</td>
<td>primary system</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>(pressure vessel, piping, steam generator)</td>
<td>prestressed concrete pressure vessel, liner to assure leak tightness</td>
<td></td>
</tr>
<tr>
<td>3.</td>
<td>containment</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>building with a defined air circulation; additionally a second cover or a flow limiter for large vents (e.g. blower, steam generator) in the PCRV</td>
<td>containment</td>
</tr>
</tbody>
</table>

SAFETY BARRIERS
<table>
<thead>
<tr>
<th></th>
<th>into atmosphere</th>
<th></th>
<th>into water</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>noble gases</td>
<td>I-131</td>
<td>tritium</td>
</tr>
<tr>
<td>applied</td>
<td>expected values</td>
<td>applied</td>
<td>expected values</td>
</tr>
<tr>
<td>values</td>
<td></td>
<td>values</td>
<td></td>
</tr>
<tr>
<td>AVR</td>
<td>actually observed values: 9 Kr 85 21 Xe</td>
<td>–</td>
<td>–</td>
</tr>
<tr>
<td>(1972)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>THTR</td>
<td>$35 \times 10^3$</td>
<td>$3.5 \times 10^3$ up to $31 \times 10^3$</td>
<td>0</td>
</tr>
<tr>
<td>HTR</td>
<td>$19,1 \times 10^3$</td>
<td>$13 \times 10^3$</td>
<td>0</td>
</tr>
</tbody>
</table>

Release rates of radioactive nuclides into Water and atmosphere

Tab. 3
889
IMPACT LOAD-TIME FUNCTION OF AN AIR-PLANE CRASH UPON A RIGID WALL

Fig. 2
884
<table>
<thead>
<tr>
<th></th>
<th>PWR</th>
<th>AVR</th>
<th>THTR-300</th>
<th>HTR-1160</th>
</tr>
</thead>
<tbody>
<tr>
<td>number of ECCS</td>
<td>4x50%</td>
<td>-</td>
<td>2x100%</td>
<td>3x50%+</td>
</tr>
<tr>
<td>interconnection with primary or secondary systems</td>
<td>no</td>
<td>yes</td>
<td>no</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>4 separate loops</td>
<td>e.g. steam generator, helium blower and control devices of the primary system are required</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>3 separate loops</td>
</tr>
<tr>
<td>failure of all decay heat removal systems admissible?</td>
<td>no (confirmed by experiments)</td>
<td>yes</td>
<td>probably not</td>
<td>probably not</td>
</tr>
</tbody>
</table>

+ still in discussion

**Comparison between decay heat removal systems of LWRs and HTRs in the Federal Republic of Germany**
accident consequences. There is, however, much more information which should be obtained and incorporated in the development of quantitative guides and criteria to accurately reflect both the consequential public risk and the economic impact of achieving and maintaining that level of risk.

My objective in this presentation is to discuss briefly a number of those areas in which additional information will be required. In this attempt, I have not restricted consideration to items which are necessary for a conservative regulatory assessment but have included needs for a realistic risk and economic assessment.

Material Properties

Let me begin with brief comments on a number of items which may be quantitatively peculiar to the HTGR but are not qualitatively different from the requirements for other reactor systems. One such fundamental area is material properties. In the normal HTGR operating temperature range, as well as the temperature range involved in accident analyses, the information which is available for many of the materials in safety related components is inadequate. Legitimate questions can be raised about the confidence limits which can be placed on the material performance. The most difficult of these questions deal with the long-term dependence of the material properties upon environmental conditions, i.e., preconditioning, and, especially for graphites, the intrinsic variability of the material as manufactured. The lack of adequate data has an obvious impact on the inspection and surveillance requirements which a regulatory agency should demand. Such requirements may, of course, influence the design and the
If, as this capability matures and is utilized, our expectation is confirmed that the present HTGR design and licensing concept retains a significant safety advantage, a very challenging issue of regulatory policy may have to be faced. One may ask, for example, "Is there a valid rationale for maintaining at measurable economic cost a differential in expected public risk between competitive systems?" One of the many objectives of safety research efforts can be to develop sufficient understanding of the phenomena involved to make consideration of such a question more than just an academic exercise.

**Mechanical Response**

The response of safety related components to static and dynamic mechanical loadings is an area in which we encounter situations which are qualitatively unique to the HTGR. The seismic response of a prismatic core and its support structure is probably the most challenging of these. Since both a rigorous full core analysis and a full scale experiment are beyond our present capabilities, financially if not conceptually, the efforts to develop and validate approximate treatments must be pursued vigorously. The problem is, of course, complicated by the uncertainties in graphite properties, particularly late in life. This, too, is an area in which the regulatory research program is actively engaged. Also of special concern to HTGR safety is the integrity of the steam generator. The potential consequences of steam ingress into the primary system under both chronic low level and acute high level conditions place great demands on the ability to predict with confidence the behavior of the steam generator under both normal and accident conditions.
The determination of fission product release rates from failed fuel particles and fission product transport in the primary system under normal conditions also require validation.

The redistribution of fission products under postulated accident conditions is, on the other hand, an area in which we feel less confident. While, from a licensing standpoint, it may be quite acceptable to use very conservative estimates of fission product liftoff etc., the effect of such conservatism on the predicted magnitude of accident consequences in an otherwise realistic assessment of total public risk should be large enough to motivate a detailed investigation of fission product lift-off during a depressurization accident and the effect of a high moisture concentration in the primary coolant on the fission product mobility.

**Loss of Forced Cooling**

The last item in this incomplete list of safety concerns is the loss-of-forced-cooling accident which is now used as the bounding event for purpose-of-site evaluation. The regulatory staff has adopted an evaluation procedure which is asserted to be genuinely conservative. There has, to the best of my knowledge, never been an attempt to dispute this assertion. It may, of course, be more conservative than will ultimately be required. There are a number of aspects of such a hypothetical accident which would be investigated. These include an experimental verification of the fission product release rates from the core and investigation of the long term behavior of the core.

The foregoing can, I think, be summarized by noting that there are two types of safety questions which are being asked. The first are
and that it should be the objective of every component of the safety research activity whether developer, user or regulator to organize his activities to contribute effectively to that development.
DESIGN FEATURES of HIGH TEMPERATURE REACTORS

By L R SHEPHERD - Chief Executive, OECD High Temperature Reactor Project (DRAGON)

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1 INTRODUCTION

The common features of all proposed commercial HTRs, as presently understood, are -

i the use of helium, under high pressure, as a coolant;

ii the adoption of a core made up entirely of graphite fuel elements incorporating coated particle fuel; and

iii the use of prestressed concrete pressure vessels to contain the core and graphite reflector assembly.

Around these main themes an almost limitless set of variations may be composed. Perhaps the most fundamental, relating to the core structure, is the division into the categories of pebble bed and prismatic cores and the further subdivisions of the latter to cover a range of different fuel element types. The coated particle fuel itself can vary accordingly to the type of fuel cycle chosen (thorium or low enriched uranium); the sizes and types of fuel kernels; the nature of the coatings. In addition the manner in which the fuel bodies are made up from the coated particles and the carbonaceous matrix and the way in which these are incorporated into the graphite structure of the fuel elements provides further scope for variation.

The primary circuit layout and pressure vessel design, of course, depend upon the purpose of the reactor, whether it is intended for steam cycle or gas turbine cycle power conversion or, perhaps, for some process heat application. Naturally, this has resulted in a wide variety of possible designs which have tended in recent times to converge into a concept in which principal primary circuit components are contained within the main pressure vessel
\( \sim 1 \text{ vpm.} \)

Helium is a reasonably effective heat transfer medium, but it is, nevertheless, desirable to work with the highest possible primary circuit pressure to minimise the volume flow rate and thereby the sizes of circulators. Typically, the working pressures may be in the region of 50 atmospheres for commercial power systems and as in the case of earlier types of gas cooled reactors, it has been necessary to resort to prestressed concrete pressure vessels for the primary containment. This eliminates the risk of an explosive depressurisation of the primary circuit, though, of course, it does not eliminate the possibility of the failure of a closure leading to a less rapid loss of coolant. It is obviously desirable to make the circulators large enough to remove the heat from the shutdown system at ambient pressure in the event of such an accident. This, in turn, sets a useful upper limit to the primary circuit helium pressure in normal operating conditions and is one of the considerations leading to the choice of coolant pressure.

Returning to the question of the chemical activity of the primary coolant impurities, it should be noted that any inleakage of water vapour through faulty steam generator tubes, will lead to a corresponding amount of graphite being removed by corrosion of the exposed surface, the attack being obviously concentrated in the highest temperature regions. Considerations of the need to maintain the strength and integrity of the fuel element structure naturally set a limit to the amount of carbon gasification that can be permitted and, therefore, to the amount of water inleakage that can be permitted over a given period. In the event of a major failure of one or more steam generator tubes, of course, a large quantity of hydrogen and carbon monoxide will be generated. The significance of such an event will be considered elsewhere.

A favourable characteristic of helium, arising from its low
include oxygen getters which can reduce chemical interaction between kernel and coating if particles are run at very high temperatures (1400 - 1500°C).

The principal constituent of particle coatings is pyrocarbon which is deposited on the kernels by pyrolysis of an appropriate hydrocarbon in a fluidised bed furnace. There is a limitless range of pyrocarbons which may be formed in this way according to the temperature, the identity of the hydrocarbon and its concentration in the fluidising gas. The simplest type of coating, the so-called BISO variety, consists of an inner PyC buffer layer, with high porosity to accommodate fission product gases, surrounded by a strong dense shell of pyrocarbon which acts as pressure vessel. Originally methane was used to produce the latter layer at very high temperatures (up to 2000°C), but the ordered structure resulting is dimensionally unstable under fast neutron irradiation and a so-called high density isotropic coating deposited from higher hydrocarbons, notably propylene, is clearly preferred because of its stability over high accumulated fast neutron doses.

At temperatures in excess of about 1000°C all forms of HDI pyrocarbon, presently known, are virtually transparent to certain fission products, particularly caesium isotopes. Since caesium activity is particularly troublesome if it is deposited on primary circuit components requiring occasional maintenance, every attempt must be made to prevent its migration into the primary circuit. Matrix and structural graphite in the fuel elements themselves can be quite effective in doing this at the relatively low surface temperatures required in steam cycle HTRs; additives to improve kernel retention of caesium are also being investigated. However, the most effective way of preventing caesium migration is to include a silicon carbide layer in the coating. In the so-called TRISO particle the single HDI pyrocarbon layer of the BISO type is replaced by a PyC/SiC/PyC
the resulting compacts are inferior in strength, density and thermal conductivity. Possibly the oldest proposed method of forming high particle density fuel bodies, but still relatively untested, is that of bonding the coated particles and filling the interspaces by pyrolysis of a hydrocarbon gas at relatively low temperature (<1000°C). Some early Dragon fuel bodies were made by this process and more recently the technique is being pursued in France but so far without satisfactory outcome.

Forms of fuel element are almost as unlimited as types of coated particle. The simple spherical fuel element of the pebble bed reactor is exceptional in this respect in that all reactor designs from the original AVR to conceptual designs for commercial power reactors have retained spheres of a fixed size and it has only been the constitution and mode of fabrication of these spheres that has varied, in addition, of course, to the size of the bed (to match the reactor output).

Originally the spherical elements were made from machined spheres with the coated particles and graphite matrix incorporated in a hollow space within the spheres. Subsequently they have been made by a compacting process similar to the overcoating technique with an inner zone containing the coated particles and graphite powder and an outer zone consisting only of the matrix material; pressing being carried out in a semi-isostatic manner to achieve isotropy in the matrix. Thus these spherical fuel elements are essentially just fuel bodies with no additional structural graphite. As such they are the forerunners of directly cooled fuel pins and moulded integral block fuel elements for prismatic HTRs.

The first prismatic core reactors (Dragon Reactor Experiment and Peach Bottom 1) were based on assemblies made up of full core length rods stacked vertically and cooled by up-flowing helium over the outer surfaces. The Peach Bottom elements were single rods of circular outer section while the Dragon elements were
outer graphite tubes, with graphite end caps closing the fuelled annulus. An important variant is a pressed fuel pin (analogous to the pressed spheres now used in the pebble bed reactors) with an annular fuelled region and unfuelled matrix boundaries inside and outside separating the fuelled zone from the helium coolant. A number of moulded pins of this type have been made and tested in the DRE (Dragon Reactor Experiment) with so far completely satisfactory outcome.

The pin-in-block element has now been abandoned in favour of the integral block element which is the type employed in the Fort Saint Vrain Reactor. Similar block dimensions are involved in both varieties but in the integral block the graphite is drilled by a very large number of longitudinal channels about one third of these, uniformly spaced, are coolant channels while the remainder are filled with slender cylindrical fuel bodies (fuel sticks). In the FSV (Fort Saint Vrain) elements the coolant channels are 15.9 mm diameter and the fuel channels 11.4 mm diameter, the fuel sticks being made by the injection moulding process.

As in the case of the pins and spheres, a variant of the integral block has been developed in which the whole element is pressed from coated particles and matrix material with fuelled and fuel free regions as appropriate. Such 'moulded integral block' elements are currently undergoing irradiation testing and appear so far to behave in a satisfactory manner.

Some remarks are appropriate in relation to the conditions under which HTR cores may be operated. Commercial reactors are expected to operate with thermal power densities in the core ranging from 5 - 10 kilowatts per litre and because of this relatively low figure which is more than an order of magnitude less than in LWRs, the conditions in the HTR are very mild by comparison with the water systems. In fact, with a helium outlet temperature of 750°C peak fuel temperature in an HTR core need not exceed 1200°C. This
cases, the entry to the cavities containing the main stream generators is from below so that it has been necessary either to adopt down stream boiling in these or to build in a complicated coolant flow system which takes the hot helium to the top of the reactor and allows it subsequently to flow downward over the boilers.

The steel liners of the pressure vessel and its boiler cavities have to be thermally insulated from the hot helium and (water) cooled to keep the concrete of the vessel at a relatively low temperature. The problem is, of course, most severe in the hot helium ducts, the base of the main chamber and the lower regions of the steam generator cavities which are exposed to the hot outflowing coolant. Metal foil or non-metallic (kaowool) barriers are both used as thermal insulation in present designs. This is a development area, where one may expect new approaches to emerge in the near future.

It is an important criterion in the podded-boiler layout that the components in the cavities should be accessible for maintenance and should be removable. Of course, this implies that they are adequately shielded from the reactor assembly so that neutron activation is negligible, but in addition it requires that the helium coolant should not transport from the core significant quantities of radioactivity liable to be deposited on the exposed surfaces of heat exchangers, etc. Fortunately, gaseous coolants particularly a light inert gas like helium, are very favourable in this respect when compared to liquids. Experience in the DRE indicates that with appropriate fuel specifications and adequate quality control, it should be possible to remove a steam generator from a large HTR after prolonged operation, without shielding.

The foregoing remarks have been applied to steam cycle electricity generating HTRs. However, in the case of direct cycle gas turbine systems it has been shown, in conceptual design studies, that power turbines, turbocompressors units, recuperators and water
SLIDE LIST

1  Coated Particles
2  Liquid Route Kernels
3  BISO Coating
4  TRISO Coating
5  Annular Fuel Compact (X-Ray)
6  Fuel Bodies
7  Spherical Fuel Element Sections
8  Peach Bottom Dummy Core
9  Dragon Cut-Away of Core & Reflector
10 Part Assembled Dragon Core
11 Dragon Cut-Away Fuel Element
12 Dragon Sectioned Fuel Elements
13 Block Core
14 Pin-in-Block Element
15 Dragon D16 Fuel Element
16 Dragon D16 During Assembly
17 HOBEG Moulded Pin
18 HOBEG Moulded Pin Sectioned
19 Fort Saint Vrain Element
20 Dragon FSV-Type Element
21 Dragon/HOBEG Integral Block Elements
22 HOBEG Moulded Block Element
23 HOBEG Moulded Blocks for Dragon Test
24 Conditions in HTR & PWR
25 Dragon Primary Circuit Layout
26 Large HTR Steam Cycle Layout (DRAGON)
27 Large HTR Steam Cycle Layout (BNDC)
28 (Up and Over Picture) Direct Cycle Layout
29 Single Shaft Direct Cycle Layout
30 Process Heat HTR Layout
HIGH TEMPERATURE GAS COOLED REACTOR SAFETY TECHNOLOGY

1. INTRODUCTION

In common with all commercial reactor systems, the safety of the HTR largely depends upon the integrity of the fuel cladding. Whilst some limited release of actual fission products is usually permissible, in both normal and fault conditions, the vast majority are retained within the fuel and cladding. The demonstration that only acceptably small releases occur in the HTR requires an understanding of the transient responses of the core and associated plant to fault disturbances, and of the fission product retaining properties of the core and primary circuit.

This paper discusses these aspects of the HTR with particular reference to the integral graphite block system with low enriched uranium fuel clad in high integrity particle coatings, which is being developed in the UK [1].

The adequacy of any part of the technology is decided by the final stage in the process of acquiring basic data, predicting behaviour in specified conditions, proving these predictions and extrapolating the predictions to the power reactor. This extrapolation is often the most difficult and expensive step and the vital role of the Dragon Reactor Experiment at this stage will become evident in the brief survey of the current technology.

2. HTR SYSTEM CHARACTERISTICS

2.1 Transient Performance

The HTR utilises a ceramic fuel situated within graphite blocks which act as both moderator and fuel cladding. The absence of metal within the core means that short temperature transients in excess of 2000°C compared with a maximum operating fuel temperature of less than 1300°C could be tolerated before significant fuel damage would occur and this provides fundamental confidence that criteria related to accidental releases of fission products can be met. However, to prevent failure of the coatings of some of the highest temperature, highest burn-up fuel particles, the reactor control and safety systems are designed to prevent fuel temperature excursions from exceeding approximately 200°C.
2.2 Primary and Secondary Circuits

The high temperature of the core gas dictates that the core gas flows in a downward direction so that the control rod mechanisms and standpipes are in the lower temperature environment of the core inlet gas (approximately 280°C). The gas flow through the boilers may be upward [3] or downward [4] depending on the detailed circuit arrangement, but in neither arrangement can natural circulation of gas be guaranteed to flow in the normal direction should forced gas circulation be lost. Also, the once-through type of boilers employed, whether with upward or downward water/steam flow, suffer flow instabilities at steam flow levels of a few per cent of full power flow. Hence unless a high percentage of feed flow can be guaranteed available to these boilers following all types of fault conditions, particularly those involving interruptions in boiler feed flow, boiler flow instabilities cannot be prevented and their integrity as a post-trip heat sink cannot be guaranteed.

These two factors, the absence of guaranteed natural circulation of gas and low flow boiler instability lead to the need to consider alternatives to the normal full power systems for providing forced circulation of gas and heat removal from the gas in post-trip situations. A solution being pursued in the UK is to provide auxiliary boilers and gas circulators which are in operation with the reactor at power and which continue to operate following all types of reactor trip. This arrangement has been described in previous papers, [1, 3, 5 and 6] and is illustrated in schematic form in Fig. 2.6. The auxiliary boilers are arranged so that water/steam flows in the upward direction; they remove approximately 16% of the total reactor heat, the steam so produced being used to drive steam generators (which provide power for normal house load as well as essential duties), to drive steam driven feed pumps for the main and auxiliary boilers, and as reheat steam for the main turbo-generators. The primary objective of these auxiliary boilers is to enable the main circulators and boilers to be tripped out following all reactor trips and so avoid the complex problem of controlling the boiler conditions to give acceptable boiler and reactor gas inlet temperature transients following different initiating faults. The main boiler steam is blown down to low pressure following reactor trips from any cause (including detection of water ingress through boiler tube leaks), and the auxiliary boilers continue to operate and remove the post-trip heat in a controlled and stable manner. The response of this arrangement has
walls from top to bottom, and wire winding of the outside walls, ensure that the concrete is in compression at all times. The tendons and wire windings can be tested and tightened if necessary throughout the vessel life. Such vessels have been tested to destruction and it has been demonstrated that they can withstand pressures in the range 2½-3 times design pressure before failing [7, 8].

The closures to the vessel comprise the circulator/boiler closures, the man access closure, the fuelling and control rod standpipes and sundry small service pipes and instrumentation penetrations. Kinetic studies show that the temperature transients resulting from a depressurisation accident are almost independent of the assumed hole size [5]. However for large holes the forces imposed on some of the vessel internal structures become excessive and it is therefore necessary to ensure that large penetration failures do not occur. Consequently, the boiler/circulator and man access penetrations are doubly enclosed by an outer steel closure, domed inwards so that the forces are in compression, and by a separately anchored restrictor. The restrictors are situated inside the penetrations and, although they are not gas tight, the flow area which they would present to the primary gas should the outer penetration fail is only a few hundred square millimeters. This area is well below that which would give rise to unacceptable forces on the vessel internals. These types of closures are similar to those on some of the penetrations in the Advanced Gas Cooled Reactors.

The standpipes are anchored into the concrete in the penetration and, in addition, they have a secondary restraint system which holds the standpipes and its closure by anchoring into the concrete at the pile cap.

All closures are provided with double seals which enable any gas leaking from the inner seal to be collected and returned to the reactor via the gas clean-up system if necessary, thus giving essentially a zero release reactor in normal operation.

2.5 Vessel Internals

The core is made up of columns of free standing graphite blocks which are replaced during refuelling. The control rod guide tubes are withdrawn with the fuelling standpipe during the refuelling process, together with the gag system. Hence a large proportion of the equipment associated with the core is removed regularly and can be inspected and replaced as necessary.
in the following, an attempt has been made to present the status of parts of
the technology related to safety and to indicate, in addition, the routes along
which the current positions have been reached and also what are envisaged as
the on-going tasks and their outcomes. It is recognised that the objectives
set out in each area are now viewed with hind-sight but because the majority
of this work was closely linked with development projects, particularly the
UKAEA Mark III programme, the objectives were in fact focussed at an early stage.

4. FISSION PRODUCT TECHNOLOGY

The fuel goes into this reactor with a small, but finite, fraction of
imperfect particles and during irradiation there is a small, but finite,
probability of failure. The methods of controlling and estimating these small
fission product source terms are now well understood and are reported in detail
elsewhere [12-15].

Models have been developed to predict the fission product migration
processes from these sources with the objectives of estimating the release
rates of activities from the core taking into account the variations of burn-up,
temperature, rating and gas flow throughout the core. The first objective is
to provide convincing evidence that the prediction method for any particular
species is reliable and that the risk of hidden effects arising, for example,
from high burn-up, variations in fuel manufacturing methods or in the geometry
of the fuel assemblies is acceptably small.

Two complementary lines of investigation were adopted to provide this
validation. The first is the irradiation of a fuel assembly containing a
known proportion of exposed kernels with the objective of measuring as many
of the fission product species, both gaseous and metallic, released from this
single unambiguous source as possible.

This experiment - the Pegase Idyle 03 irradiation - was designed and
operated in collaboration with the Commissariat a l'Energie Atomique. The fuel
element was designed in hollow rod geometry to provide cylindrical symmetry
with a single heat and fission product transfer surface. The intact fuel
particles were manufactured to a specification which required an exposed uranium
fraction of less than 1 in $10^5$, an order of magnitude lower than the requirement
for power reactor fuel. The gaseous diffusion characteristics of the uncoated
kernels were measured separately by light irradiation and annealing and the
evaporation and diffusion properties of the matrix and graphite were measured by
current laboratory methods [16, 17]. Several of these properties over a range
of concentration and temperature were measured independently at Harwell and Grenoble.
(c) That burn-up and geometry have little effect on the release iodine and rare gas activities.

(d) That feasible methods for monitoring and locating the gaseous activities in the coolant of a power reactor have been demonstrated.

These conclusions are of particular relevance in two particular areas. The first is the assessment of the hazard resulting from a slow (instrument line) depressurisation. In this case the discharged iodine activity must be shown not to exceed four times the normal working limit. It is this criterion which sets the tolerable fraction of exposed fuel in normal operation. The second is the restriction of fuel operating conditions imposed by the access through-outlife criterion in order to limit the predicted circuit inventory of caesium at 30 years to a few hundred curies. This sets the tolerable fraction of defective silicon carbide. A proof of precision or of a margin in either case is important.

The bar-chart (Section 4; Fission Products) illustrates the sequence of the investigations described. Typical data derived from out-of-pile experiments on materials are shown in Figs. 4.2, 4.3 and Table 4.2. The emphasis now placed on in-pile measurements of properties is indicated in the work grouped under Current Tasks and two irradiation experiments now in hand are illustrated in Figs. 4.4 and 4.5.

The behaviour of strontium in the particle coatings, the matrix and the graphite is adequately understood up to fuel running at 1400°C in the matrix. On the other hand the migration of silver from particles exceeds the model predictions and the effects of gas flow and temperature gradient are currently being investigated.

5. FISSION PRODUCT RELEASES IN ABNORMAL CONDITIONS

The effects of persistent abnormally high temperatures and abnormally high concentrations of water have been investigated experimentally and the long time constants associated with coating damage have been demonstrated. The temperature effects are shown in Fig. 5.8 and the results of validation experiments on irradiated compacts are summarised in Table 5.9.

Methods for the detection and location of incipient failure of retention barriers in the fuel have been developed from the monitoring instrumentation used on the Dragon reactor and, in principle, both on-line activity detection and location of the source by control rod movement are feasible.
The working models used to predict the distributions of deposited activities are based on semi-empirical relationships, analogous to heat transfer, which describe the transport of species to a surface. Surface capacities and desorption rates can be represented and diffusion into the surface allowed for.

The mass transport model can be tested in a system in which this sticking probability is close to unity, as is the case for the isothermal (200°C) heat exchanger surfaces in Dragon, or the lower temperature regions of a plate-out probe. The results of applying such a prediction for mass transfer in a helical tube bundle are shown in Fig. 6.1 [19]; the diffusion-in-helium coefficients for silver, caesium, iodine and strontium have been deduced from Schmidt numbers and show agreement within a factor of x 2 with the predicted values [20].

The adsorption data at higher temperatures and for surfaces more typical of a power reactor have not yet been measured. Until this information is available, the time dependence of the plate-out factors and the profiles of deposition round the circuit cannot be estimated with confidence.

The effects of high water concentrations on deposited activities have been examined qualitatively in pilot experiments conducted in the Dragon plate-out probe [21]. These show that the deposits can be disturbed. High coolant velocities can also cause deposits to be disturbed.

This has led to the proposed High Flow Rig for incorporation into the Dragon Reactor. The basis of the proposal is the use of a 20 cm diameter branch in the primary circuit for sampling the reactor coolant in the upper plenum chamber. The objectives are to measure the adsorption of volatile fission products in controlled conditions and to simulate their mass transfer behaviour in heat exchangers. A further development of the experimental facility for the investigation of the behaviour of deposited activities in accident conditions has also been considered. Emphasis will be on deposition between 500°C and 900°C where solution of fission products in metals becomes important.

This proposal is shown in the context of the deposition technology in the Section 6 bar chart.

- 11 -
The samples of graphite used in these experiments were taken from adjacent positions in the original blocks and, in addition, the spread of reactivity of each type of material was determined. In the Loop A experiments, the burn-off was taken to the point where the reaction rate was sensibly constant.

The results indicate that over a wide range of the parameters which are assumed to affect corrosion rate there is agreement between the observed and predicted corrosion to within a factor of about three. However there is as yet no explanation of why any particular material should experience less or more weight loss than predicted. The measured density profiles within the graphite were not correctly predicted and the conclusion was drawn that some property other than the measured diffusivity was controlling the in-pore diffusion.

The next stage was to compare the measurements of corrosion in the in-pile chemistry experiments with the predictions based on the out-of-pile data. The chemistry experiments in Dragon have two principle objectives: (a) to determine the degree of protection provided by the structural graphite to the fuel compacts; and (b) to investigate the variation of reaction rates in typical candidate materials.

Chemistry experiments 1, 2 and 3 aimed at low corrosion (<10 mg/cm²) with flow varying from 0.25 g/s (Re ~200) in Chemistry 1 and 2 to 2.3 g/s (Re ~2,000) in Chemistry 3. The comparisons between the in-pile and out-of-pile results are shown in Table 7.2. These comparisons indicate that the rate of in-pile corrosion at 20 atm was significantly less than that predicted from out-of-pile measurements at 1 atm. In consequence, the weight loss of fuel compacts protected by the graphite was greater than predicted; the property affected was the compact strength and no particle detachment or increase in friability was observed. The fission gas release was measured by a regular purge monitoring during each experiment and no increase in release was observed.

The Chemistry 4 experiment was planned to examine the upper range of corrosion (100 mg/cm² peak) and the initial results indicate that the in-pile corrosion was again less than predicted.

The conclusions drawn from these results were that either the flow conditions in the experiments were affecting the pattern of corrosion, possibly by causing local depletion of the moisture in the helium in immediate contact with the graphite or that the assumed inverse square root pressure dependence did not hold.
In all cases there was no measurable density gradient in the interior of
the remaining material and the appearance was similar to that of mass transport
controlled corrosion observed out-of-pile at much higher temperatures.

It was clear, therefore, that the short term investigations should include
the re-measurement of samples of the in-pile corroded materials in the
out-of-pile loops, the inclusion of further high flow experiments in driver
channels and the examination of the highly corroded material for catalytic
effects, pore size distribution and pore wall corrosion.

In addition the future work lay in determining the properties of graphite
which control the migration of oxidising impurities and in examining the effects
of geometry on corrosion distribution in block geometry. An experiment designed
for the investigation will be irradiated during 1976 (Fig. 7.5).

Simultaneously, the integrated core reactivity of the reactor has been
measured at least twice during each irradiation by the slug injection of a
small quantity of water and the subsequent measurement of the reaction product
concentrations in the coolant. The most notable feature of these measurements
summarised in Table 7.4 - is their consistency. Clearly, the ideal validation
will be to predict these measured $R_c$ values from the component reaction rates
in each part of the core.

8. ACCIDENT CHEMISTRY

The graphite steam reaction is central to the consideration of burst heat
exchanger tube incidents. The detailed chemistry is complicated [22] and
relatively little work has been done on nuclear grade graphites with high
partial pressures of steam. At present the main reaction rate data is taken
from a thesis by Mienkina [23] and is compatible with work carried out at
ORNIL.

The following features of the graphite steam reaction should be noted:

(a) the reaction is endothermic by 20-30 kcal/g mole depending on
conditions so the reaction is self limiting

(b) it is strongly dependent for the in-pore diffusion controlled
reaction with an exponent in the order of 50 kcal/g mole

(c) the reaction of steam with bulk graphite is approximately zero order
with respect to steam pressure

(d) the products occupy a larger volume than the reactants and this retards
the diffusion and permeation of the reactant within graphite components

(e) hydrogen, one of the reaction products, markedly inhibits the reaction.
The calculation methods developed for the Dragon Reactor were found to be applicable to the study of incidents on power reactors almost without change. One important difference for accident analysis is that in a power reactor the secondary circuit pressure is above the primary circuit pressure which means that burst tube incidents have to be considered in which a tube burst occurs without a breach of the primary circuit. In such an incident there is complete containment (unless a pressure relief valve opens) and hence no safety hazard. Then the main problem is corrosion damage to the core.

The results of such a calculation in which 2,000 kg of water are assumed to enter a core of 1,500 MW(T) power are shown in Figs. 8.1, 8.2 and 8.3 [25]. In Fig. 8.2 the distributions of corrosion in the peak and average temperature channels are shown and in Fig. 8.3 the peak corrosion is plotted against ingress rate for incidents with pressure trips and water detector trips. It is clear that reliance on a pressure trip alone might lead to unacceptable corrosion at ingress rates below 5 kg/s, whereas a water detector giving a trip within about 60 seconds would prevent serious damage.

Studies have also been carried out of the coincident or consequential failure of a tube with the depressurisation of the primary circuit of a power reactor [25, 27]. The most sensitive parameter in such studies is the histogram of surface temperatures in the core at any time and therefore, although general conclusions can be derived, such as the highest water gas proportion arising in the case of low ingress rates (2 kg/s) and for delayed tube failure, the development of methods and the generation of data is more important than their application to notional cores.

This point is emphasised in the on-going work proposed in this field. Both from economic (core damage) and safety considerations there is a need for reaction rate data on the materials now preferred for core construction and the experience in the related field of low concentration chemistry emphasises the importance of simulating the conditions of pressure, flow and geometry as closely as possible.

With regard to method development, the OXIDE-3 code developed by GAI and the TUBER code stem from different basic equations and calculation procedures. An intercomparison of these codes would provide a valuable reference.

The refinement of TUBER, will lie in the direction of an improved thermal/hydraulic treatment.
correct core pressure drop. The centre pod also contains a central standpipe with a complete set of contract components. The linear insulation will be identical to the contract insulation.

The heat input to the rig is by the gas circulator power which will be capable of heating the facility up to approximately 310°C, and in its basic form heat will be removed by one or other of the boilers.

Contract boilers would be used to carry out the basic engineering tests, and when these are complete, they will be removed and erected in the reactor. The facility will then be re-established using the station spare circulators and small heaters and coolers to become a proving facility for the refuelling machinery, since it will allow a simulated refuelling operation at normal conditions.

We anticipate that this facility could be operational some 12-18 months before the site engineering tests could start.

11. CONCLUSIONS

Any accident analysis is related to a 'snap shot' situation with regard to the state of the fuel, the inventory of fission products, the corrosion distribution, the plated-out activity, and so on, which has been developed during the course of normal operation.

If the experimental data on abnormal conditions can be based, even in a limited way, on an operating reactor, the extrapolation to power reactor conditions can be made at less expense and with greater confidence than a synthesis from basic data.

The intention in this paper was to show, by presenting the status of several areas of safety related technology, how the in-pile experimental programme has been integrated with the overall research and development for High Temperature Reactors.

The current trend in the safety analysis of any nuclear power plant is towards a deeper understanding of phenomena and the examination of an increased range of abnormalities. Because we are becoming concerned with risk definition as well as accident analysis, this trend is recognised in the proposals outlined above for current and future work.

12. ACKNOWLEDGEMENTS

The authors wish to thank the Chief Executive of the Dragon Project and the management of BN/HC for permission to present this paper.


FIG. 2.2 REACTOR GAS FLOW REDUCTION TO \( \frac{1}{2}\% \) IN 15 SECONDS.
Fig. 2.4. Gas flow - stagnation and reversal in all channels - reactor tripped.

Temperature

Fuel temperature

Graphite temperature

Maximum temperatures in mean rated column.

Gas temperature in reactor inlet plenum.

1/2% reversed flow from 160 sec.

1% reversed flow from 160 sec.
FIG. 2.9 RUN THROUGH MAIN TURBO-GENERATOR TRIP. MAIN BOILERS TRIPPED AND 1 NILE WORTH OF CONTROL RODS INSERTED AT 5 SECONDS TO RAPIDLY REDUCE REACTOR POWER.
Fig. 4.4  In-PILE EXPERIMENT TO DETERMINE THE BEHAVIOUR OF VOLATILE METAL FISSION PRODUCTS IN GRAPHITE & MATRIX MATERIALS.
Fig. 5.8 Variation of $\text{XE}_{133}$ Release/Birth Ratio with Irradiation Time
FIG. 21 - TYPICAL REACTION RATE VARIABLES.

- REACTIVITY \( \mu g \cdot cm^{-2} \cdot h^{-1} \cdot \mu m^{-2} \) AT 1200°C

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<tr>
<th>NO. OF SAMPLES</th>
<th>NO. OF BLOCKS</th>
<th>CODE</th>
<th>GRAPHITE TYPE</th>
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<td>500</td>
<td>FINE GRAIN BATCH (1)</td>
</tr>
<tr>
<td>5 S S S S</td>
<td>1 1</td>
<td>510</td>
<td>FINE GRAIN BATCH (2)</td>
</tr>
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<td>5 S S S S</td>
<td>4 1 1</td>
<td>503</td>
<td>DRAGON FUEL TUBE</td>
</tr>
<tr>
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<td>4 1 4 1</td>
<td>505</td>
<td>DRAGON FUEL TUBE</td>
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<td>DRAGON FUEL TUBE BATCH (1)</td>
</tr>
<tr>
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<td>FINE GRAIN BATCH (3)</td>
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<tr>
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<td>4 1 1 1</td>
<td>516</td>
<td>FINE GRAIN BATCH (6)</td>
</tr>
<tr>
<td>5 S S S S</td>
<td>4 1 1 1</td>
<td>518</td>
<td>FINE GRAIN BATCH (7)</td>
</tr>
</tbody>
</table>

- FUEL BLOCK GRAPHITE
- FUEL BLOCK GRAPHITE
- FUEL BLOCK GRAPHITE
- FUEL BLOCK GRAPHITE
- FUEL BLOCK GRAPHITE
- FUEL BLOCK GRAPHITE
- FUEL BLOCK GRAPHITE

- GILSO EXTRUDED
- GILSO MOLDED
Oxidation rates of sample AGL 19 SOI at various pressures. Temperature 900°C.
Pressure and Temperature Changes in an Incident with a Steam

Fig. 8.1 Increase Rate of 20 kg/s

Thermal Power: 1500 MW
Gas Pressure: 54 bar
Helium Inventory: 5000 kg
Total Water Ingress: 2000 kg.
Water Detection Response Time: 60 s
Working Pressure: 54 bar
Trip Pressure: 56 bar

Graph showing the relationship between steam ingress rate (Kg/s) and peak corrosion (mg/cm²). The graph indicates two trip points: a pressure trip only and a water detection trip.
<table>
<thead>
<tr>
<th>Duration of wet run</th>
<th>Long Term</th>
<th>Short Term</th>
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</thead>
<tbody>
<tr>
<td>Moisture content of inlet gas</td>
<td>50 h</td>
<td>110 min</td>
</tr>
<tr>
<td>Compact temperature</td>
<td>2420 vpm</td>
<td>13-37 v/o</td>
</tr>
<tr>
<td>Inlet flow rate</td>
<td>1275°C</td>
<td>1230-1300°C</td>
</tr>
<tr>
<td>Weight loss of compact</td>
<td>1.1 l/min</td>
<td>0.9 l/min</td>
</tr>
<tr>
<td>8.5% (3.3g)</td>
<td>27% (10.6g)</td>
<td></td>
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Release measurements:

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<thead>
<tr>
<th></th>
<th>mCi</th>
<th>Fraction of inventory</th>
<th>mCi</th>
<th>Fraction of inventory</th>
</tr>
</thead>
<tbody>
<tr>
<td>Xe-133 - initial release rate of wet stage</td>
<td>0.33 h⁻¹</td>
<td>2.0 x 10⁻⁶ h⁻¹</td>
<td>0.33 h⁻¹</td>
<td>2.0 x 10⁻⁶ h⁻¹</td>
</tr>
<tr>
<td>- final release rate</td>
<td>40 h⁻¹</td>
<td>2.4 x 10⁻⁴ h⁻¹</td>
<td>1.25 h⁻¹</td>
<td>7.5 x 10⁻⁶ h⁻¹</td>
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<tr>
<td>total release</td>
<td>1,147</td>
<td>6.9 x 10⁻³</td>
<td>1.58</td>
<td>9.5 x 10⁻⁶</td>
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<tr>
<td>I-131 total release</td>
<td>363</td>
<td>4.9 x 10⁻³</td>
<td>0.53</td>
<td>7.2 x 10⁻⁵</td>
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<tr>
<td>Cs-137 total release</td>
<td>5.5</td>
<td>3.2 x 10⁻³</td>
<td>0.002-0.006</td>
<td>1.3 x 10⁻⁶</td>
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<tr>
<td>H-3 total release</td>
<td>0.6</td>
<td>4.2 x 10⁻²</td>
<td>0.14</td>
<td>1.0 x 10⁻²</td>
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<tr>
<td>Graphite</td>
<td>Loop A Oxidation</td>
<td>Observed/predicted weight loss at various burn-offs:</td>
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</tr>
<tr>
<td>----------------------------------------------</td>
<td>------------------</td>
<td>-----------------------------------------------------</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Temp °C</td>
<td>5</td>
<td>25</td>
<td>50</td>
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<tr>
<td>Fine grain Gilso-carbon moulded, low reactivity</td>
<td>1190</td>
<td>0.53</td>
<td>0.71</td>
<td>0.76</td>
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<td>As above, sample from lower reactivity block</td>
<td>1190</td>
<td>0.83</td>
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<tr>
<td>As above, sample from block of lower reactivity and lower diffusivity ratio</td>
<td>~ 1150</td>
<td>3.1</td>
<td>3.7</td>
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<tr>
<td>As above</td>
<td>1100</td>
<td>1.5</td>
<td>1.9</td>
<td>-</td>
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<td>Fine grain Gilso-carbon extruded</td>
<td>1200</td>
<td>0.79</td>
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<td>As above, higher reactivity</td>
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<td>0.34</td>
<td>0.35</td>
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<td>0.79</td>
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<tr>
<td>As above, with higher diffusivity ratio</td>
<td>~ 1170</td>
<td>-</td>
<td>1.5</td>
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<tr>
<td>Moulded Gilso-carbon moderator block, low density</td>
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<td>1.1</td>
<td>1.1</td>
<td>1.1</td>
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<td>Event</td>
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<td>Density g.cm^{-3}</td>
<td>$^{90}Y$ reactivity mg.g^{-1}.h^{-1}.atm^{-1}</td>
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<td>---------</td>
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<td>------</td>
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<td>&quot;          = &quot;</td>
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<td></td>
<td>IV, 2 4/6A</td>
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<td></td>
<td>= outer &quot;</td>
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<td>IV, 1 2/6</td>
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<td>100</td>
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<td>8.6</td>
<td>0.6</td>
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<td></td>
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<td></td>
<td></td>
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<tr>
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<td>=  &quot;</td>
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</tr>
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<td>IV, 11 1110</td>
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<td>TOTAL</td>
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<td>35.1</td>
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<td>IV, 8 1070</td>
</tr>
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<td>=  &quot;</td>
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<td></td>
<td>IV, 9 1090</td>
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<td>=  &quot;</td>
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<tr>
<td></td>
<td>=  &quot;</td>
<td></td>
<td></td>
<td>IV, 11 1110</td>
</tr>
</tbody>
</table>

TABLE 7.3
A comparison of the observed and predicted corrosion of Dragon tubular-interacting fuel tube graphite

- Observed corrosion obtained from weight change of graphite components weighed as a unit. Estimated precision ± 10%.
- Observed corrosion obtained by comparison of mg.cm^{-2} of exposed surface with value obtained from density and race specified dimensions. Precision ± 10 mg.cm^{-2}.
- Observed corrosion obtained as with FE923.
- Observed corrosion from measurement of weight change of inner tubes. Precision ± 0.005 mg.cm^{-2}.
- Observed corrosion from measurement of weight change of 17 cm long segments of inner tube. Precision ± 0.01 mg.cm^{-2}.
1. TO ASSESS THE IMPACT OF THE FIXED AND FLOWING \nGAS VELOCITY ON THE CAUSING OF VAPOR \nCONDUCTIVITY OF THE COAL.

2. TO ASSESS THE QUALITY OF WATER GAS PRODUCTION IF \nSUCH GAS CONDUCTION IS TO THE LOW-PRESSURE TIP \nOF THE COAL.

3. TO ESTIMATE THE POSSIBLE EXPANSION OF THE COAL \nOF WATER GAS PRODUCTION TO THE BLOWER \nAND OPTIMIZE THE ACCURACY WITH WHICH THE \nCOAL-CONDUCTIVITY IS CONSIDERED.

4. TO ESTIMATE THE ACCEPTABLE RADIUS OF THE \nCOAL-CONDUCTIVITY OF A REACTOR WITH AN ACCELERATING \nRATE OF STEAM FLOW.

5. TO DEVELOP A TECHNIQUE FOR THE ON-LINE WATER \nDETECTION.

6. TESTING OF VARIOUS \nINSTRUMENTS FOR ON-LINE WATER DETECTION.

7. DEVELOPMENT OF WATER GAS \nANALYZER FOR DRAGON REACTOR.

8. WATER GAS ACCIDENT \nANALYZER FOR DRAGON REACTOR.

9. RESPONSE TIMES AND TEMPERATURE LIMITS FOR GAS MISTERS.

10. REQUIREMENT FOR REACTION RATE \nDATA ON CURRENT MATERIALS.

11. REQUIREMENT FOR IMPROVED \nCALCULATIONAL MODEL.

12. REQUIREMENT FOR TOOLTILE \nDRAGON CONTROL FOR UPTIGHT \nBLOCK CORE AND FOR SALT- \nPERMANENT STRUCTURES.

13. FUTURE STUDIES NEEDED IN DESIGN AND FOR \nOPERATIONAL LIMITS.

14. RESPONSE TIMES AND TEMPERATURE LIMITS.
1. Introduction

1.1 Because of the very much larger number of basic fuel element units, the study of the release of fission products in the particle fuelled HTR requires rather different design approach as well as different analytical methods from those developed for use with the Magnox, LWR and AGR systems. The fuel units in HTR are not only four or five orders of magnitude more numerous, they are also physically very small, the particles being about one millimetre in diameter. This very small size requires automated inspection in production and since non-destructive inspection schemes can have only a limited effectiveness, the practical result is that a small fraction of imperfect particles is included in every fuel charge. These imperfections range from relatively gross defects in which the primary fission product barriers, the particle coatings, are either absent or broken, to minor imperfections which nevertheless can still cause premature failure of the coatings as burn-up proceeds and the damage from fast neutrons, etc., increases.

1.2 However although the particle coatings are the first barrier to those fission products which escape from the oxide or carbide fuel kernels, other barriers exist in a practical fuel element which significantly delay the passage of some fission products to the coolant. The study of the release of fission products from HTR fuel in normal operation therefore involves the study of release from the kernel, the nature and distribution of coating imperfections and the rate of diffusion of various fission products of importance through the perfect and imperfect coatings and also through the rest of the fuel element to the gas stream. The methods developed for assessing the actual gas borne burden of fission products used in the UK have evolved over a considerable period and are still being refined. Most people attending this conference will have some familiarity with them so the purpose of this paper is to review their current status. Since the UK has consistently aimed at fission product releases in normal operation of the very high standard set by the Magnox and AGR reactors, our particle manufacture has been geared to producing very high quality and consistent fuel particles. As will be mentioned again later, the work of the fuel fabrication technologists has been remarkably successful. Automatically manufactured and inspected fuel having a very narrow range of tolerances in material properties and dimensions (with a consequential low new fuel failed fraction) has been produced on a semi-production scale. It was originally feared that these high standards would result in very high fabrication costs which would have been a serious matter since the fabrication cost of HTR fuel is inherently high to the extent of absorbing most of the economic advantage of the system's very good neutron economy and thermal efficiency. In the event, the manufacturing development has broadly reached both quality and economic targets and since this high quality production equipment now exists, only a relatively small saving in fuel fabrication cost would be achieved by lowering the standards. This has an important bearing on the title of this paper "Methods for the Prediction and Achievement of Low Circuit Activity".

1.3 Another matter important to the prediction and achievement of a low circuit activity is an understanding of the behaviour of fuel particles. It is necessary to know how particles change from "perfect" to leaking as a result of burn-up, temperature and fast dose. This enables realistic models to be used in the computer programmes which integrate these effects spatially and temporally to provide estimates of the release which can be expected in normal operation over the life of the reactor. It also guides the fuel inspection and quality control during manufacture by highlighting the areas of major importance. This is another field in which progress is continuing in the UK and this paper brings up to date the accounts already published in many previous papers (for example, Ref11).
In order to predict the overall effect of these barriers it is necessary to obtain the relevant diffusion and evaporation parameters for the individual fission products. These data may then be incorporated into a computer model calculation which allows the release of the fission product to be predicted for a whole core over a wide range of operating conditions.

2.2 Release from kernel

During the fission process the bulk of the recoiling fission fragments are likely to come to rest within the UO₂ crystallite structure. The first major barrier to the release of fission products from the coated particle, and hence from the whole core, is therefore the diffusion process from the recoil sites.

a. Rare gases and iodine

In the past, when considering the release of the rare gases Kr and Xe, it has been assumed that the kernel consists of a large number of spherical grains of equal radius (a cm) and to this idealised structure is ascribed a reduced diffusion coefficient \( D' \) sec\(^{-1} \) defined by \( D' = D/a^2 \) where \( D(\text{cm}^2 \text{ sec}^{-1}) \) is the in-grain diffusion coefficient.

The theoretical treatment of this, (Ref 1) the so-called equivalent sphere model, assumes that the diffusing atoms, on reaching the surface of the spherical grain, are able to move rapidly along the grain boundaries thereby maintaining the grain surface concentration zero at all times. It is possible, under these conditions, to derive an analytical expression for the variation of the in-pile fractional release (defined as the ratio of the undecayed atoms outside the system to the number of undecayed atoms in the whole system) with time.

For isotopes whose half-life is short compared with the irradiation time, the fractional release expression simplifies to the steady state case

\[
F = 3 \sqrt{\frac{D'}{\lambda}} \left( \coth \sqrt{\frac{\lambda}{D'}} - \sqrt{\frac{D'}{\lambda}} \right) \quad \cdots (1)
\]

where \( F \) = fractional release
and \( \lambda = \text{decay constant (sec}^{-1}) \)

which, in the situation where the radioactive decay is much faster than the diffusion rate, further reduces to

\[
F = 3 \sqrt{\frac{D'/\lambda}{\lambda}} \quad \cdots (2)
\]

and therefore shows that the fractional release should vary as \( 1/\sqrt{\lambda} \).

These expressions have been confirmed by irradiations of bare kernels at Harwell and also from gas release measurements in the Dragon reactor and analysis of these experiments has provided data for \( D' \) as a function of temperature. No distinction has been found between Kr and Xe; I\(_2\) is also known to behave in a similar manner.

For long lived isotopes, the in-pile fractional release is given by
The results have been interpreted in the form of reduced diffusion coefficients and the results are generally very similar to those obtained for caesium (Ref 3). It seems likely that similar mechanisms of migration must operate in both cases, probably by diffusion of atoms through defects in the UO₂ grains, the entropies and enthalpies of the jump processes being similar for the two metals, neither of which has any significant chemical interaction with the host material.

The relative escaping tendencies for metal fission products from the coated particle kernel might be expected to be related to the thermodynamic activities and metal partial pressures existing in the kernel during irradiation. For example, metals forming a relatively stable oxide, such as strontium (or metals which are involatile such as molybdenum) should be retained by the kernel to a greater extent than caesium or silver. For carbide kernels it is found that the retention capability for the alkaline earths and rare earths is decreased, the strontium metal pressure being, for example, some two orders of magnitude greater than for the oxide kernel case (Ref 4).

It may therefore be concluded that oxide kernels are to be preferred to those made from carbide from the point of view of metal retention, and it is now increasingly recognised that the development potential of coated particle fuel in respect of increased temperatures and burn-up depends considerably on being able to increase the kernel retention even further by the addition of specific fission product getters which will produce compounds of greater stability than the simple oxide. Current work is largely concerned with the effect of the addition of Al₂O₃ and SiO₂ which can form stable compounds such as alumino-silicates.

2.3 Diffusion through pyrocarbon

a. Rare gases and iodine

It is now well established that the pyrocarbon layers of the coated particle serve as virtually complete barriers to the gaseous fission products. Although the pyrocarbon layers are frequently found to contain pores in the diameter range 0.01-1 μm it can be shown theoretically that such a tortuous system of small channels is very inefficient for the transmission of gaseous species.

b. Metal fission products

Metals which adsorb, if only to a small extent, on the defect surfaces of the pyrocarbon layer and therefore migrate by a combination of gas phase and surface diffusion, are known to permeate the layer relatively rapidly. In order to carry out reactor safety calculations in a rigorous fashion it is necessary to predict the escape rate of these metal fission products from which in certain circumstances can penetrate the silicon carbide layer of Triso particles and therefore for this purpose diffusion coefficient data for metals in pyrocarbon are required. Of the fission products which are known to be released from some 'Triso' (and also plain pyrocarbon '3iso') coated particles, namely caesium, silver and strontium, caesium has received the most attention and diffusion coefficients in pyrocarbon derived from both methane and propylene source gases have been obtained over the temperature range 1000-1500°C.

This information is being obtained in two ways. In the first unirradiated 'Triso' particles are heated in a caesium atmosphere at constant pressure,
surface migration on the overall diffusion process. Post-irradiation annealing experiments have shown that silver diffusion in pyrocarbon is also a relatively rapid process although laboratory work has indicated that the solubility is small. It seems likely that silver diffusion at temperatures in excess of 1000°C occurs mainly in the gas phase although, by analogy to the rare gases, this should result in the pyrocarbon acting as an almost perfect barrier. The reason for this apparent anomaly is not understood.

2.4 Prediction of the release of caesium, strontium and silver for pyrocarbon coated UO₂ particles

The diffusion data discussed above has been incorporated into a finite difference fission product diffusion code FIPDOL (Ref 6) in order to calculate the fractional release, as a function of time and temperature, for the three key metal fission products caesium, strontium and silver. These calculations were carried out on a particle with a 800 μm diameter UO₂ kernel coated with an inner 25 μm, thick porous buffer carbon layer and an outer pyrocarbon layer of 140 μm, thickness. In the case of caesium a partition factor of 10 was employed between the kernel and buffer layer, in favour of the latter, to allow for the known retention of caesium by the buffer layer. For the other two metals the partition factor was assumed to be unity. In the absence of any definite information for the silver diffusion coefficient in pyrocarbon a value of this parameter was chosen which was at least two orders of magnitude greater than the corresponding kernel diffusion coefficient thereby simulating the situation where the release from the particle was controlled entirely by the kernel diffusion rate. The same criteria was also applied to the buffer layer for all three metals. The release from the particle was calculated under infinite sink conditions by introducing a fast evaporation step at the particle surface thereby reducing the concentration to near zero at this point during the whole of the irradiation time.

The results of the calculations are summarised in Fig 2.

Because of the similarities between the kernel diffusion coefficient data for caesium and silver, the fractional release curves for the latter refer also to the release of caesium from bare UO₂ kernels.

Examination of the results shown in Fig 1 leads to the following general conclusions for caesium.

a. At 1300°C, which is a realistic peak fuel temperature in a steam-cycle HTR, the reduction in release, after a dwell time of 700 days (a likely maximum fuel irradiation time) produced by a propylene pyrocarbon coating, is small. In the case of the methane pyrocarbon material the fractional release is decreased by a factor of at least 30. The potential benefit to be gained by reducing the diffusion coefficient for the propylene derived pyrocarbon to a value close to that expected for the methane material is therefore obvious.

b. Retention by the kernel is almost complete at temperatures below 1100°C for irradiations of 700 days duration and the effect of loss of coating integrity on the total release from regions of the core with fuel temperatures below this value is likely to be small.

c. Conversely at temperatures above 1300°C the particle coating plays a more important role and the need for a silicon carbide layer is if a relatively clean circuit is to be achieved is self evident.
For caesium, however, the situation was more complex. The profiles obtained have in general indicated that at least two diffusion processes are occurring. The steep initial portion of the profile is probably caused by in-grain diffusion although the effect of open porosity at the surface increasing the available surface area may also be important. The remainder of the profile is ascribed to a relatively fast process occurring in major structural defects such as pores. The relative importance of these two processes during normal fuel element dwell times is difficult to assess from the results of laboratory experiments which usually last for only a few hours. The present recommendation is that data relative to the fast process should be used in release calculations but this may lead to a quite serious overestimate of the situation.

2.7 Metal vapour pressures over matrix and block graphite

The final barrier to fission product release is the evaporation process from the block surface into the helium coolant. The effect of this barrier is determined from a knowledge of the adsorption isotherm describing the equilibrium partition of the fission product between the graphite surface and the helium boundary layer, as a function of temperature. Evaporation coefficients are then calculated using the heat/mass transfer analogy in conjunction with a knowledge of the parameters describing the gas flow conditions such as Reynolds Number and hydraulic diameter.

Vapour pressure data for caesium and strontium over matrix graphite have been obtained (Ref 8) as a function of temperature and concentration, using the Knudsen effusion technique. The variation of effusion rate with time at constant temperature and also its dependence on orifice size has indicated that the rate of escape of the vapour species from the cell was, to some extent at least, controlled by the rate of diffusion of caesium to the surface of the solid. A model has been developed which allows the calculation of the surface concentration appertaining to each vapour pressure measurement and this has been used to construct the adsorption isotherm. Good agreement has been achieved over a wide variety of experimental conditions and the results indicate the applicability of a Langmuir type of adsorption isotherm for both caesium and strontium over matrix graphite at least up to a loading of 10 µg metal/g graphite.

Similarly satisfactory results have been obtained for strontium over block type graphite (Ref 8) with a possible transition to a Freundlich type of adsorption at loadings in excess of ~ 30 µg g Sr/g graphite.

The situation for caesium adsorption isotherm data is much less certain. Knudsen effusion experiments have shown a time/concentration dependence which is not simply interpretable by the diffusion model used for matrix material (Ref 8). In fact the effusion rate seems to be controlled by a complicated diffusion process which cannot be described by a single diffusion coefficient. This is in line with the conclusions for caesium diffusion in block graphite outlined earlier. The measured vapour pressure at the beginning of a run is somewhat greater than the corresponding matrix graphite value but a rapid decrease in pressure with time follows, the overall fall being as much as two orders of magnitude before a pseudo-steady state is reached. The associated bulk concentration decrease may be as little as 20-30%.

It is assumed that during the initial preparation of the specimens a considerable fraction of the caesium diffuses into relatively inaccessible regions of
It is, however, important to note that the interpretation of in-pile produced profiles is necessarily an unreliable process because of the uncertainties in the operating conditions and in a purged element such as FE307 the Cs-137 in the fuel tube could be affected very dramatically by transverse helical flow through the tube wall. This will be particularly true for the fast component of the caesium diffusion process where the migration may occur to a large extent in the gas phase. Its contribution to the profile in Figure 3 could therefore have been reduced to undetectable levels.

A further serious disadvantage arising in the interpretation of the results of in-pile experiments is that no data are available on the amount of fission product metal which has been released to the coolant, except in certain rather exceptional circumstances which will be referred to later. A check on the validity of the release prediction is therefore not possible and this highlights the need for well instrumented in-pile loop experiments used in conjunction with accurately defined fission product sources and trapping facilities.

The silver profile shown in Figure 3 demonstrates a further difficulty of interpretation on the basis of a simple diffusion model. There is no retention of the Ag-110m within the teledial box but a very marked hold-up occurs in the fuel tube producing a sharp peak in the profile. It seems probable that the silver enters the fuel tube in the form of a gas which then rapidly permeates the graphite structure. With the reduction in temperature experienced with increasing penetration of the tube wall an increased degree of adsorption on the graphite surface occurs. If the rate of surface diffusion is assumed to be very slow a peaked profile will result. Models are being developed (Ref 10) using this two phase diffusion mechanism which satisfactorily reproduces this type of behaviour for diffusion in a non-isothermal situation. It has been observed from this and other experiments that the position of the peak in the silver profile occurs at a graphite temperature of 1000-1050°C and may therefore be used to confirm the irradiation temperature of fuel elements. This type of diffusion behaviour has not been reproduced in laboratory experiments so far but, as in the case of caesium, evidence has been presented indicating that silver is generally more mobile in unirradiated graphite than in the as-prepared material.

The shape of the strontium profile between the fuel hole and the gap also cannot be represented by a simple diffusion model. Again it is necessary to take into account the effect of the temperature gradient on the mobility or thermodynamic activity of the diffusing species. Since strontium is believed to diffuse mainly by a surface mechanism a model is being developed using a simple diffusion phase but with chemical potential rather than concentration being the driving force. Profiles of the 'up hill' (or peaked) form shown in Figure 3 can be reproduced in this way.

It should be stressed that the main uncertainties in the basic caesium migration data occur in the block graphite. The data for the matrix material seen to be satisfactory and have in fact been tested in some detail in the form of release predictions on two elements in the Dragon Reactor fuelled with a high percentage of plain pyrocum coated particles which therefore dominated the caesium release into the coolant. These elements were both of the moulded form with fuel free zones and the calculated caesium releases as a function of time were in satisfactory agreement with the results of gas sampling probe analysis.
environment not included in the model. Examples of such sources of complexity are: pronounced lack of sphericity, local mechanical pressure between touching particles or between a particle and its container, and temperature gradients. Other effects which throw more stress on the SiC by damaging the PyC layers seem to be of importance. One is the fast dose failure of the PyC, which may either directly weaken the coating or allow the chemical attack of the SiC by uranium or by fission products either from the kernel within or externally from nearby failed particles.

Another important weakening effect is of course the so-called "amoeba attack".

3.4 Amoeba attack

Over the past 18 months some advances in the understanding of amoeba attack in the low enriched fuel particle have had a considerable effect on particle design and in the optimisation of the core as a whole. The most significant step forward in this time has been the realisation that the activation energy for the effect is quite low, much lower than had been previously assumed. This means that the effect is less sensitive to temperature but relatively more sensitive to other parameters such as the local temperature gradients, the burn-up and enrichment effects. Previously, to avoid the premature pressure failure which excessive amoeba attack produces, the reactor designers commonly adopted the technique of dropping fuel temperature appreciably towards the end of life with the objective of reducing the rate of attack in the highly stressed older particles to negligible proportions. It is now apparent that this solution is not likely to be very effective.

The studies on the nature of amoeba attack have confirmed the importance of free oxygen in the carbon transport observed. The use of oxygen getters within the kernel has been shown to be an effective way of limiting amoeba attack for the low enriched cycle HTR and since the use of these getters seems straightforward, it is the preferred solution to amoeba attack in the UK. However, a residual problem remains in that it is difficult to incorporate such getters in kernels made by the wet fabrication routes, these routes now being recognised as the most attractive to the manufacturers.

3.5 Escape from intact particles

Section 2.4 discusses the evidence which suggests that at high temperature strontium appears to diffuse through the composite coatings, including the SiC layer. The mechanism for this is under investigation but the temperatures at which this effect becomes important seem to be well above those occurring during normal operation of a UK type steam cycle station. A possibly more limiting situation may arise with silver, there being evidence that Ag 110m escapes from the fuel block into the coolant at temperatures close to those existing in some parts of the core in a steam cycle HTR. Ag 110m is of interest mainly as a boiler contaminant but the quantities created in the Low Enriched cycle mean that some care will be necessary in future designs to avoid boiler maintenance problems from this source. A more certain control of the release of Ag 110m can probably be effected by the provision of cooler graphite or by the use of a kernel getter if further evaluation of this effect indicates this to be necessary.

4. The Distribution of Temperature Burn-up and Fast Dose in an HTR Core

4.1 The previous sections of this paper discuss the factors governing the creation and transport of fission products from leaking fuel particles via the
5. The Calculation of the Distribution of Leaking Particles in the Core

5.1 The results shown in Figs 5 and 6 are illustrative only, each being tending to produce an individual distribution. However, for low enriched HTR's without an imposed axial variation in enrichment in the fuel columns, the shapes of the distributions are usually close to those shown, the shape for variation being mainly in the general level. Calculations linking these distributions with simplified particle failure models can provide estimates of the distribution of leaking fuel within a core at various stages in life. Computer programs exist for performing this calculation (Ref 13) but to date the results so obtained have not yet used particle "failure" models which reflect the most recent extensions of our knowledge of particle behaviour.

6. The Specification of the Allowable Number of Leaking Particles

6.1 It has already been mentioned that the calculation described above is part of a calculation cycle, feeding back into a revision of the conditions leading to leakage. Thus the analysis described provides both a basis for estimating the likely density of significant fission products in various parts of the reactor circuit at various stages in life and also plays a valuable role in determining the allowable failed fraction.

The HTR has an unusually large number of fuel units. While it is economically feasible to make 60,000 fuel pins (the approximate charge of a typical LWR) with the assurance that only one or two will prove to be defective, it is not attractive to attempt to produce an HTR charge with no defective particles. Nor is such an absolute proposition made necessary by a realistic appraisal of the health and safety considerations. Thus while the particle concept enables somewhat better predictions to be made of the response of the fuel to operational and fault conditions, it also requires the tolerance of a small but predictable fraction of leaking fuel.

6.2 Fuel develops leaks in service by the phenomena discussed earlier. Superimposed on the "in-service" failure pattern is of course the new fuel failed fraction. This, although small, must exist because there is no way of checking that the $10^5$ to $10^6$ particles in a fuel element assembly are all sound as effectively as a sealed metal canned fuel pin can be checked. Thus the allowable failed fraction in a core at a given time is made up of particles which were leaking from new, particles which developed leaks as a result of the various operational phenomena already described and of course particles whose failure occurs as a result of a combination of initial defects and operational effects. Thus a core typically contains leaking particles of all ages.

6.3 Both the reactor/fuel designer and the fuel fabricator have a measure of control over this situation. The designer can specify a particle's composition to give it a greater or lesser design life and he can also govern the temperature and dose history in the core, while the fabricator can control the variability of the product, leading to a lower scatter in endurance. However in both cases the exercise of this control in a direction to reduce failures usually results in a higher fuel cycle cost.

This increase in cost is not necessarily linear. For example, in the UK, HTR fuel is already made and inspected by largely automated processes and these would be extended in scope if full scale production were started. While such processes are "in control", ie the process is easily meeting the product specification, there is little to be gained by relaxing the specification. Tightening the specification, on the other hand, to a point where the reject rates...
mass transfer coefficient derived from heat transfer correlations using the analogy between heat and mass transfer and the second that the concentration in the gas at the surface is taken to be in equilibrium with the solid surface concentration at all times.

7.3 Comparison with in pile measurements

Information concerning the plate-out behaviour of fission products in PWR coolant circuits is potentially available from three sources. The AVR pebble bed reactor in Germany has a gas sampling probe (VAMPYR) as did also the Peach Bottom Reactor in the USA. No significant amount of data is currently available from these sources.

On the Dragon Reactor two separate types of measurement have been made. These have involved the analysis of heat exchanger profiles and also the analysis of profiles in gas sampling probes.

The theoretical ADDICT model has been tested (Ref 16) by comparing the predicted plate-out profiles for Cs and I₂ with those obtained experimentally by cutting up heat exchangers and carrying out radiochemical analysis of the various sections. The slope of the profile and the plate-out per pass are studied in detail since this information is important in determining the distribution of dose rates for inspection and maintenance operations and the amount of the fission product inventory which remains gas borne.

It was found that in the case of Cs the profile was largely a test of mass transfer effects since the sticking probability of Cs on the cool heat exchanger surfaces is very high, a plate-out per pass figure of around 50% being quoted. In the case of I₂ a suitable adsorption isotherm must be used in conjunction with the mass transfer coefficient. It was also observed that the plate-out per pass for I-131 is a rather sensitive function of this isotherm and cannot be calculated accurately until the isotherm is known with more certainty, a figure of 25% seems to be a likely approximation. In these calculations all the Cs and I₂ was assumed to enter the heat exchangers in atomic form. In the case of Sr the situation is more complicated. It seems likely that most of the Sr²⁺ activity measured outside the core was released as the inert gas precursor Kr-89 (T₁/₂ = 3.18 min) which decays to Rb 89 (T₁/₂ = 15 min). Since the circulation time of the helium is only a few seconds the Rb is likely to be formed uniformly round the circuit and is then deposited mainly in the heat exchangers where decay to Sr-89 occurs.

Gas-in-gas diffusion coefficients have also been derived from Schmidt numbers both from the heat exchanger analysis and probe profiles. These have generally given values which are in close agreement with those derived from the kinetic theory of gases which gives confidence in the mass-transport model.

It must be stressed that extrapolation from data obtained on the relatively cool surfaces of a Dragon heat exchanger (200°C) to the hotter surfaces of a power reactor steam generator (500°C) involves adsorption isotherm data which is not available at present. The time dependence of the plate-out factors and the profiles of deposition round the circuit cannot therefore be estimated with confidence.

Freck and Rodliffe (Ref 17) have attempted to calculate the likely Cs behaviour within the boilers of a 1000 MW(Th) unit cooled by He at 40 atmospheres with a mass flow of 540 kg/sec. Adsorption isotherm data for Cs on steel derived by Milstead and Zumwalt and by analysis of PLUTO loop experiments was used. The
For reactors containing only pyrocarbon coated fuel the peak fuel temperature should probably not exceed 1100°C although the uncertainties in the pyrocarbon behaviour make this suggested limitation somewhat tentative.

Considerable work still remains in the characterisation of both silicon carbide and pyrocarbon to obtain information on structural and chemical properties which may be related to the fission product diffusion rate. Information is also needed on the behaviour of the more volatile metal fission products silver and caesium in block type graphite and the development of diffusion models which are able to reproduce the observed fission product behaviour in irradiation experiments is also urgently required.

8.2 Failure mechanisms

On the recent developments in the understanding of the way fuel particles fail, it is worth noting two aspects. The better understanding of the "amoeba" attack in low enriched fuel particles has had a marked effect on core performance optimisation with consequential changes in the distribution of temperature, etc., within the core. The lower estimates of activation energy which have resulted from the analysis of recent experiments have given rise to renewed attention to the use of oxygen getters. Certain getters have been shown to be very effective in reducing the free oxygen in the particle and the use of these will certainly be beneficial.

The other aspect has only recently been shown up in the experimental programme. This may cause some revision in our ideas of the temperatures at which certain fission products (and Ag 110 seems to be the most important) escape through visually intact Triso fuel coatings. Work on this is still in progress and it is therefore premature to more than note the existence of a potential problem, and to remark that, if required, inexpensive counter measures seem to be available.

8.3 The distribution within a core of burn-up, temperature and dose

Section 4 of the paper gives examples of the kind of distribution of burn-up, temperature and dose for an HTR core which is not axially flattened. The long high burn-up or high temperature tails of the distributions are characteristic of such unflattened cores and some improvement would be possible by adopting a simple two level enrichment zoning. However this would further complicate the proposed fuel management routines (there are already radially disposed enrichment zones) and would not be adopted unless a problem with fission product release were definitely confirmed. For higher gas temperatures (direct gas turbine cycle and process heat applications) much more complex arrangements would, of course, be essential.

8.4 The determination of the allowable net failed fuel fraction

The total equilibrium release of fission products from the fuel in normal operation depends on a number of factors already discussed. In many respects the most important is the fraction of fuel particles which have damaged or imperfect coatings. The endurance of the coated particle under the operating conditions is one aspect of this, the new fuel failed fraction is another. A special feature of the design of an HTR core is the need for a continuous co-ordination between the magnitude of the permissible failed fuel fraction determined from consideration of the permissible release of fission products and that can be economically attained with a chosen set of production and inspection techniques.
REFERENCES


FIG. 2  SILVER RELEASE FROM BISO AND TRISO COATED PARTICLES
ANNEALED AT 1500°C.
TYPICAL SIC STRENGTH DISTRIBUTION (UNIRRADIATED)
Approximate Distribution of Burn-up

Low Enriched HTR
End of Year
3 Year, Annual Refuelling Cycle

Percentage of Particles in Core Having Less Than Stated Burn-up

Burn-up % FIMA

2% at 2% burn-up
99.5% at 14% burn-up
Some Comments on Fission Product Behaviour in HTGR's, and its Impact on Safety

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1. INTRODUCTION

In the years since the first assessments of the risks associated with nuclear reactors were made the size of both reactors and nuclear power programmes has grown and there has been an accompanying increase in public interest in safety. At the same time there have been advances in methods of risk assessment and attempts are being made to define acceptable standards of risk (1).

During the period in question both the techniques for predicting the likelihood of failures in the engineering systems and the volume of relevant data for nuclear plants have improved and the effect of this is being seen in a broadening of the approach in safety studies. Additional emphasis is being placed on the frequency of occurrence of accidents, on the reliability of protective systems and engineered safeguards, and on a quantitative evaluation of risk. As an inevitable part of this process increased attention is being directed to those larger accidents which in the past may have been considered so unlikely as not to need detailed analysis. These developments are exemplified for instance in the recent assessment of accident risks in US commercial (light water) power plants (2). A concomitant of the study of larger accidents is the possibility of significant releases of some of the less volatile fission products, notably caesium, and there is now improved understanding of the consequences of such releases (3).

Any HTGR will be judged against the safety standards current at the time that it is licensed. It may be that with the general growth in nuclear expertise and experience in the next few years there will be a tendency for safety standards to become more restrictive.

The present paper reviews the impact of fission product behaviour on the safety of the HTGR in accident conditions against this background. Various aspects of fission product behaviour are commented on in relation to the safety strategy adopted for the reactor, and some preliminary calculations are included for the loss of caesium from overheated fuel blocks, for deposition on heat exchanger surfaces, and for dose rates within a containment contaminated by caesium.

2. FISSION PRODUCT DISTRIBUTION IN NORMAL REACTOR OPERATION AND PLATE-OUT MECHANISMS

An understanding of the behaviour of the fission products in normal steady reactor operation is always a pre-requisite for the prediction of releases in accident conditions.

The unique feature of the HTGR is the fuel, consisting as it does of small particles of fissile or fertile materials coated with layers of pyrocarbon and possibly silicon carbide. As a consequence of the large number of these particles ($10^{10}$ in a large reactor) the HTGR has the advantage that fuel failures observed in normal operation will be progressive in nature, and there will be no sudden releases of large amounts of fission products. The other aspect of this, however, is that because failed particles are not easily separated and removed, and because there will always be some
and were those deposited in largest quantity. The results are illustrated in Figure 1-4, where the predicted profiles have been normalized to the same total area as the experimental results. The fission products were assumed to exist in atomic form in the helium coolant, mass transfer coefficients between the coolant and the heat exchanger surfaces were evaluated by means of a heat/mass transfer analogy, and interaction between the fission product and the surface was represented by an adsorption isotherm.

Heat exchanger tube surface temperatures are not measured in Dragon but estimates of these and of the gas-side heat transfer coefficients were made using the known thermal performance of the heat exchangers. The tube surface temperatures were quite low (200-300°C) and this leads to a basic distinction in behaviour between I\(^{131}\) and the other, less volatile fission products Cs\(^{137}\), Ag\(^{111}\) and Rb\(^{85}/Sr\(^{89}\). For I\(^{131}\) it appears that plate-out may be controlled by both mass transfer and adsorption mechanisms but for Cs\(^{137}\), Ag\(^{111}\) and Rb\(^{85}/Sr\(^{89}\) the adsorption isotherms imply that at the temperatures in Dragon the surfaces will act as a perfect sink. Atoms of these materials once deposited will not re-evaporate and plate-out will therefore be controlled entirely by the mass transfer mechanism. In order to derive profiles of the correct general shape for these less volatile materials, i.e. with a negative slope everywhere as shown in Figures 1-3, it was not sufficient to compute average heat and therefore mass transfer coefficients applicable to the whole of a heat exchanger, but these quantities had to be allowed to vary along the length of the heat exchanger as the gas and tube surface temperatures varied. Agreement between the slope of the calculated profile and the linear part of the experimental profile could then be optimised by the choice of the relevant diffusion coefficient for Cs\(^{137}\), Ag\(^{111}\) or Rb\(^{85}/Sr\(^{89}\) in the helium coolant. It was found that diffusion coefficients obtained in this way were up to 50\% larger than literature values for the atomic species based on simple empirical correlations of density and boiling point. Other diffusion measurements, e.g. in the Uranium probe, have since indicated that these higher values of diffusion coefficient are to be preferred (5). In the case of I\(^{131}\) both the diffusion coefficient and the adsorption isotherm can be varied to improve the agreement between experimental and theoretical behaviour and it is not clear which if either should be changed, though it may be noted that the information on which the isotherms is based is very limited.

The agreement between theory and experiment in Figures 1-4 is relatively good apart from the increased deposition which occurs experimentally, particularly for Cs\(^{137}\) and Rb\(^{85}/Sr\(^{89}\) on tubes near the entrance to the heat exchangers. The present work offers no explanation for this effect. Possibilities which have been suggested are flow disturbances associated with the entrance region and with changes in tube bank geometry which occur in that region, and the attachment of some fission products to particulate matter in the coolant which might deposit in a different manner to atomic species.

In summary the work described is valuable evidence that the fission products investigated probably exist largely in atomic form in the Dragon coolant, though some may be attached to particulate matter, and that mass transfer coefficients can be evaluated from known heat transfer coefficients. It throws no light, however, on the interaction between the fission products and the surface on which they deposit. Before confident predictions can be made for power reactors the nature of this interaction will need to be understood at the higher surface temperatures which will be characteristic of such reactors. It will need to be established, for instance, whether equilibrium surface adsorption can be assumed or whether the deposited fission products can migrate into the body of the material, and what the capacity of the surface is including possible effects of interference between competing species. Any effects of slow inleakage of water into the coolant from defective boiler tubes with any occur in a steam-cycle power reactor will also need to be understood.
code the actual fuel element was simulated by a single fuel pencil surrounded by a cylindrical annulus of fuel-free block graphite. An evaporation coefficient calculated from the appropriate reactor conditions applied at the outer edge of the fuel-free region. The initial distribution of Cs\textsuperscript{137} was appropriate to a fuel element approaching the end of life and was established by a FINFLO run corresponding to 730 days at full power with a fuel centre temperature of 1000\degree C, a graphite outer temperature of 800\degree C and a bulk coolant temperature of 650\degree C. The accident was then simulated by reducing the fission rate to zero and raising the temperature of the fuel pencil and graphite block to a uniform value of 1200\degree C. The evaporation coefficient at the outer edge of the block under accident conditions was that appropriate to a pure nitrogen coolant at 1 atmosphere pressure with a bulk temperature of 1000\degree C and a volume flow equal to 30\% of the full power value. All the relevant data and the initial and final distributions of the Cs\textsuperscript{137} after a period of about 5 hours at 1200\degree C are given in Figure 5. Figure 5 shows the fraction of Cs\textsuperscript{137} originally present outside fuel particles which is released as a function of time. It is clear that substantial releases can occur in times of interest, and notably in the first 5 hours.

In calculations of this kind for a complete reactor core account would have to be taken of temperature and other variations between different fuel elements. Nevertheless, the present results are indicative. Their value depends on the reliability of the data used. From the shape of the profiles in Figure 5 it is clear that using the currently recommended UK data it is the diffusion process in the fuel pencil matrix graphite, possibly partition between the matrix and block graphite, and evaporation from the outer edge of the block graphite which are the controlling steps in Cs\textsuperscript{137} release; diffusion in the block graphite is relatively rapid. At 1200\degree C evaporation is sufficiently rapid for the Cs\textsuperscript{137} content of the block graphite to be fairly quickly released. The Cs\textsuperscript{137} content of the matrix is then released much more slowly. There is currently considerable doubt about the models and data for caesium transport in block graphite. The caesium may be able to exist in at least two adsorbed states, each of which has its own associated diffusion and evaporation coefficients, and caesium transport may also be possible by diffusion in the gas phase within the graphite pores. The latter process may be affected by gas flow through the graphite. The data used in the present calculations are thought to be conservative in that they represent the highest measured values of diffusion and evaporation coefficient, though the effects of gas flow through the graphite are not explicitly allowed for. They are also measured out-of-pile on fresh graphite and there is some evidence that transport under in-pile conditions may be slower. Further work is necessary in this area to establish better data and it may be that when these are available the present results will be seen to be unduly pessimistic.

A further consideration is that any Cs\textsuperscript{137} release from overheated fuel blocks which does occur will take place largely after the initial depressurisation of the reactor is over. It may be that nitrogen will be injected into the reactor at this time and gas swept out of the breach in the pressure vessel to prevent air ingress. Nevertheless, there may be significant deposition on the circuit of any fission products released from the core. This will be influenced by factors of the kind discussed earlier and may limit the release from the pressure vessel.

Release of Cs\textsuperscript{137} from overheated fuel blocks in a depressurisation accident with impaired cooling is likely to remain of concern unless these various complexities can be resolved favourably.

4. \textbf{Permanent Loss of Cooling}

The high gas temperatures in the HTGR may constitute a potential hazard to the heat exchangers and circulators in certain accidents. Where the high gas temperatures have led to a choice of downward gas flow in the core and upward flow in the heat exchangers total loss of forced circulation may be followed by flow reversal and
5. CONTAINMENT DOSE RATES

In the case of a contained reactor a question which has to be considered is the possibility of access to the containment for repairs and maintenance of equipment required for long term cooling following a depressurisation. Then $^{131}I$ has decayed the dominant activity preventing access is likely to be $^{137}Cs$. Some simple calculations have been carried out based on a typical containment with an internal volume, not including the PCHV, of approximately $5 \times 10^4 \text{m}^3$. Two situations have been studied, with the activity distributed uniformly through the containment volume, and with the activity distributed uniformly over the superficial area of the containment and PCHV. In both cases the dose rate adjacent to the top surface of the PCHV was in the region of a few m Rem/hr per Ci of $^{137}Cs$ released. Clearly if the activity were localised in a particular region dose rates in the vicinity of that region would be substantially higher.

It follows that a release of 1000 Ci of $^{137}Cs$ to such a containment would give dose rates of several Rem/hr at least make access to the containment difficult. In a reactor in which advantage has been taken of the existence of a containment to substantially relax the fuel specification the $^{137}Cs$ inventories in the fuel matrix and block graphite in the circuit might be many thousands of curies.

6. IMPACT ON SAFETY

The eventual importance attached to the various aspects of fission product behaviour in the HTGR will clearly depend on the safety strategy adopted for the reactor and on the safety standards current at the time. On the basis of the material reviewed in the present paper various approaches can be envisaged:

(1) Limitation of the failed fuel fraction and hence the fission product release from the core in normal reactor operation to levels in line with those for other systems.

It is probable that the Advanced Gas Cooled reactor, which has on-load refuelling and has been licensed for urban sites in the UK, will be operated with a proportion of failed fuel pins of no more than about 1 in $10^4$. If a similar figure can be achieved for the effective broken and defect particle fractions in the HTGR the risks from an AGR and from an uncontained, steam cycle HTGR of typical UK design may be regarded as comparable, given comparable engineering reliabilities (8). The fission product release to atmosphere in depressurisation is then unlikely to exceed a few hundred curies of $^{131}I$ and $^{137}Cs$, even taking account of the matrix and fuel block inventories, (see Table 1). Limitation of the effective broken and defect particle fractions to these values is also likely to permit man access to the reactor circuit for inspection and maintenance purposes, though additional fission products, eg., Ag-110m, may have to be considered in this context. The fuel specification imposed by questions of coolant leakage in normal operation is less restrictive (4).

For a developed HTGR operating at higher average fuel temperatures, possibly for direct cycle gas turbine or process heat application, fission product release rates from a core with specified particle damage may be higher than for the current steam cycle system. If at the same time the need for access to the circuit is more acute a tighter fuel specification may be required (9).

The data on retention of $^{137}Cs$ by $\text{UO}_2$ particle kernels and pyrocarbon coatings are somewhat uncertain and the present assumptions may be conservative. If further investigation confirms this some relaxation on the fuel specification for defect particles may be possible. Further relaxation may also be possible from the development of getters specifically designed for retention of caesium within the kernel.
7. G. Preinreich, Dragon Project report.


9. G.D. Bell and J. Shepherd, Safety Problems Arising from Process Heat Applications of HTGR's, paper to this meeting.

FIG. 2. 

**REGION OF COOLING TUBES**

- **THEORY**
- **EXPERIMENT**

**NANOCURIES/cm²**

**SWEPT SURFACE AREA OF TUBE BUNDLES (cm² x 10^-4)**

**Ag-111 DEPOSITION PROFILE**
**FIG. 4.**

**I^{131} DEPOSITION PROFILE**

- **SWEPT SURFACE AREA OF TUBE BUNDLES (cm^2 x 10^-4)**
- **SURFACE CONCENTRATION (g/m cm^-2)**

**EXPERIMENTAL CURVE**

**THEORETICAL CURVE**

<table>
<thead>
<tr>
<th></th>
<th>EXPERIMENTAL</th>
<th>THEORETICAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>PLATE-OUT PER PASS</td>
<td>28%</td>
<td>24%</td>
</tr>
<tr>
<td>PLATE-OUT FACTOR</td>
<td>10^4</td>
<td>10^4</td>
</tr>
</tbody>
</table>
FIG. 6. FRACTIONAL RELEASE OF Cs-137 FROM MATRIX AND FUEL BLOCK AFTER DEPRESSURISATION AND OVERHEATING.

FRACTIONAL RELEASE OF Cs-137
could be achieved. The behaviour of the fission products nuclides found experimentally proved the model and the used basic physical assumptions to be right.

Using the program "PATRAS" it is now possible to calculate the distribution of the fission products in primary cooling cycles of HTR's. As an example this has been done for the AVR. The results are given in this paper.

Concerning the depressurisation accident no lift-off experiments under accident conditions have been done up to now. But as the plate-out model takes into account the adsorption and desorption as well as the non-reversible phenomena it is possible to calculate the desorption of fission products in case of a depressurisation accident. Two kinds of depressurisation accidents are discussed.

In one case the accident occurs out of normal operations of the reactor, in the second case it is assumed that at the same time or just before the depressurisation a large water inleakage into the primary cooling system takes place. Results from calculations for both cases are given.
or chemical reactions are considered:

A flux of particles, \( I_E \), hits the wall. A fraction of it, \( I_R \), is reflected. The complement fraction \( I_E - I_R = \alpha I_E \) stays on the wall, where \( \alpha \) is the accommodation coefficient. The flux \( \alpha I_E \) is divided into an adsorption flux \( I_A = \beta \alpha I_E \) and a diffusion flux \( I_D = (1 - \beta)\alpha I_E \), where \( (1 - \beta) \) is the penetration coefficient. \( M_0 \) is the number of the adsorbed particles. A fraction of it \( \mathcal{S}M_0 \) is desorbed later on. Thereby it is considered that an adsorbed particle can be desorbed later and is again available for adsorption or diffusion. \( \mathcal{S} \) is the desorption constant.

On the basis of the analytical solution and with boundary values considerations special evaluation methods have been developed. By this from the experimental data directly conclusions can be drawn about the important mechanisms of the different elementary processes of the interaction of fission products with the wall and the different plate-out parameters can be gained.

### 2. EXPERIMENTAL INVESTIGATIONS

The experimental facilities Vampyr in the AVR and Saphir in the reactor Pégase have been described already in detail /4, 5, 6/.

In Vampyr till today 14 experiments had been carried out. 11 had been utilized. In these experiments two main objectives were investigated:

1. The concentration of fission and activation products in the hot helium coolant of the AVR depending on gas outlet temperature of the AVR-core.
behaviour of isotopes of different chemical nature Cs 134 and I 131 and the results for the same isotope Cs 134 in different experimental arrangements (Vampyr, Saphir).

Fig. 2 shows the axial plate-out profile of Cs 134 on a 4541 tube of the experiment Vampyr 07. The temperature profile is also shown. The circles represent the experimental results, the curve the distribution calculated with the program "PATRAS". As one can see the plate-out concentration is decreasing with decreasing temperature. If one would assume pure adsorption the concentration should increase with decreasing temperature as the adsorption equilibrium is reached in this case.

Only with the assumption of additional non-reversible processes like diffusion or chemical reaction which are expressed by the penetration coefficient $1 - \beta$, the behaviour can be described. Over the full length there is an influence of the mass transfer coefficient. The evaluation of these experiments gave for $1 - \beta = 0.33 \%$ and for the desorption energy $Q = 45$ Kcal/Mol.

Fig. 3 presents the plate-out of Cs 134 again on 4541 but in the experiment Saphir 05. For the theoretical calculation the same values for the penetration coefficient $1 - \beta$ and the desorption energy $Q$ as in the Vampyr experiment have been used. One can see the influence of mass transfer coefficient at the change of geometry of the test tube and at the bends (K). Again there is a good agreement between experiment and theory.
part of this bridge the Vampyr experiment is installed. The helium is entering directly the heat exchanger and flows back through a circular gap to the blowers. On the pressure side of the blowers before the entrance to the core the gas for the gas purification is taken out. In the tube to the gas purification the dust filter is mounted, where in parallel to the Vampyr experiments the amount of dust in the helium and the concentration of solid fission products in the cold gas is measured.

The heat exchanger is divided into 8 parts. Three different materials, 10CrMo 910, 15Mo 3 and St 35.8, are used.

Tab. 1 gives a list of basic assumptions used for the calculation of the plate-out distribution /8, 9/.

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Surface area of the heat exchanger</td>
<td>$1763 \text{ m}^2$</td>
</tr>
<tr>
<td>Surface area of the circular gap</td>
<td>$460 \text{ m}^2$</td>
</tr>
<tr>
<td>Reynolds number heat exchanger entrance</td>
<td>$4.6 \cdot 10^3$</td>
</tr>
<tr>
<td>Reynolds number circular gap entrance</td>
<td>$7.7 \cdot 10^4$</td>
</tr>
<tr>
<td>Mean coolant core outlet temperature</td>
<td>$850 \text{ °C}$</td>
</tr>
<tr>
<td>Operation time of the reactor</td>
<td>10 years</td>
</tr>
</tbody>
</table>

Concentrations of fission products in the core outlet:

<table>
<thead>
<tr>
<th>Element</th>
<th>Concentration</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs 137</td>
<td>$0.25 \cdot 10^{-9} \text{ Ci/Nm}^3$</td>
</tr>
<tr>
<td>Cs 134</td>
<td>$0.13 \cdot 10^{-9} \text{ Ci/Nm}^3$</td>
</tr>
<tr>
<td>J 131</td>
<td>$9.6 \cdot 10^{-9} \text{ Ci/Nm}^3$</td>
</tr>
<tr>
<td>Ag 110 m</td>
<td>$3.5 \cdot 10^{-9} \text{ Ci/Nm}^3$</td>
</tr>
</tbody>
</table>
For Ag 110 m in the full temperature range the contribution of the adsorption is very small, e.g. the plate-out is only influenced by the penetration coefficient. The step in the heat exchanger results from change in the material. The other steps in the profile are due to different flow conditions and geometries.

One can define a feedback factor for the different isotopes as the ratio of particle flow released from the core to the particle flow which is transported back to the core by the helium. As shown in tab. 2 for the different isotopes this factor is calculated to

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Factor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs 137</td>
<td>0.17</td>
</tr>
<tr>
<td>Cs 134</td>
<td>0.16</td>
</tr>
<tr>
<td>J 131</td>
<td>0.11</td>
</tr>
<tr>
<td>Ag 110 m</td>
<td>0.21</td>
</tr>
</tbody>
</table>

5. CONCLUSIONS FOR DEPRESSURISATION ACCIDENTS

Based on the assumption that the model described above gives a sufficient understanding of the plate-out behaviour, one can draw some very interesting conclusions for the behaviour of the fission products in case of a depressurisation accident.

It is assumed that during such an accident within a period of seconds or a few minutes the depressurisation takes place. This can lead to high local gas velocities over a relative short time period. After this one has to assume relatively low gas velocities.

Two cases of depressurisation accidents are discussed. In one case it is assumed that the accident occurs out of the normal operation of the
Fig. 7 shows the desorption of Cs 134 and J 131 depending on time after the accident, assuming a sudden depressurisation from 10 to 1 bar.

Curve a represents in both cases the maximum possible amount of desorption at infinite time. These values correspond to the total adsorption amount in the circuit at the time of the accident, which is assumed to be 10 years of operation for this example. The values are for Cs 134 66 % and for J 131 99 % of the total deposited activity in the circuit.

Curve b is based on the very pessimistic assumption that the mass transport coefficient is infinite, e.g. the nuclide is leaving the circuit after desorption without any possibility of collision with the wall. 300 hours after the accident only 29 % of the Cs 134 is desorbed. For J 131 due to the radioactive decay after 50 hours a maximum of 50 % is reached. Then the curve decreases.

The results of calculation for a more realistic but still pessimistic case are given in the curve c. Here it is assumed that the desorbed nuclide can interact locally with the wall before it is leaving the circuit, still the possibility of adsorption on the wall of the circuit during transport through pipes and components is neglected. For this case the desorption is much more slowly. Only 1.4 % of Cs 134 is desorbed after 300 h and for J 131 a maximum of 11 % is reached after 100 hours.

For Ag 110 m the situation is different. Only a fraction of $7 \cdot 10^{-5}$ of the total deposited quantity is adsorbed reversible on the surface, e.g.
accident combined with a water inleakage.

The steps in the different curves are due to change in temperature and materials in the different parts of the circuit. In conclusion one can see that in the hot part of the circuit larger amounts of Cs 134 are fixed in the material and in the colder part nearly 80% of Cs 134 is desorbed.

The corresponding distribution for J 131 is shown in fig. 9. In contrary to Cs 134 only in the first part of the heat exchanger some J 131 is bounded non-reversible on the surfaces. The contribution of the diffusion of J into the material is very small but cannot be neglected. In case of a water inleakage a very high percentage of the J can be desorbed spontaneously. Only a very small percentage of the J in the first part of the heat exchanger is located in a 3 µ layer of the material and could be set free additionally assuming a corrosion by the water of 3 µ of the material.

For Ag 110 m the adsorbed fraction is very small, 7 \times 10^{-5} als already mentioned and can be neglected. As Ag is not soluble in water it can be assumed that even in case of 3 µ corrosion no Ag will be set free additionally.

First decontamination experiments at the EIR - Würenlingen and KFA Jülich using water as a decontamination agent indicate that the assumption of complete spontaneous desorption of the adsorbed amount of nuclides in case of water inleakage and the assumption of a 3 µ corrosion as used in the above estimates are very pessimistic. Further investigations are necessary to find more realistic values.
of adsorption of fission products on the dust. We hope we will get at least some informations from further experiments with the dust filter in the AVR.

7. SUMMARY CONCLUSIONS

Table 4 gives some estimated values of the contribution of the different isotopes in case of a depressurisation accident. In the second column the total amounts for the different isotopes are given which are released into the primary cooling circuit of the AVR from the core within a ten years operation time of the reactor. The values are based on the measurement of the experiment Vampyr. The third column indicates the total amount of gas from activity in the AVR - cooling system. This activity is assumed to be released immediately from the cooling circuit into the containment in case of a depressurisation accident.

Assuming in case of a depressurisation accident without presence of water or water vapour the depressurisation ends, e.g. pressure balance is reached, within a time period of 6 min column 4 gives the contribution of desorption from the surfaces at this time. These figures are indicating that the desorption is very slow as has been shown in the previous discussion. These are still upper pessimistic values.

Under the pessimistic assumption of spontaneous desorption in case of water inleakage one gets the data in the 5th column, the contribution of a 3 μ corrosion of the surfaces is negligible (see column 6).
e) For prediction of the consequences of depressurisation accidents of large HTGR systems the largest uncertainty today might be the prediction of fission product release from the core into the primary cooling system during long operation periods.
Iniotakis, Münchow
Theoretische Auswertung und Interpretation der Ablagerungs-
untersuchungen in den Reaktorexperimenten Vampyr/AVR und
Saphir/Pégase
Reaktortagung des DAtF, Nürnberg 1975

Vereinigte Kesselwerke A.G., Düsseldorf
Technische Beschreibung für den Dampferzeuger des BBC/Krupp-
Reaktors
15. 3. 1965

Private communication of AVR
Ratio between the gas borne activity to the total release in AVR after 10 years operation time

Cs 137 6.10^{-9}
Cs 134 7.10^{-8}
J 131 7.10^{-6}
Ag 110 m 2.10^{-7}

Table 3

Summary of results for depressurisation accident in AVR primary circuit after 10 years operation time

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Released Activity from Core</th>
<th>Gasborne Activity in Primary Circuit</th>
<th>Desorption Activity after 6 min</th>
<th>Released Activity after Water In-leakage</th>
<th>Released Activity 3μ corrosion</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs 137</td>
<td>10.7 Ci</td>
<td>69 nCi</td>
<td>76 μCi</td>
<td>5.8 Ci</td>
<td>6.0 Ci</td>
</tr>
<tr>
<td>Cs 134</td>
<td>1.9 Ci</td>
<td>130 nCi</td>
<td>13 μCi</td>
<td>0.97 Ci</td>
<td>1.0 Ci</td>
</tr>
<tr>
<td>J 131</td>
<td>0.7 Ci</td>
<td>4.6 μCi</td>
<td>1.6 mCi</td>
<td>0.6 Ci</td>
<td>0.6 Ci</td>
</tr>
<tr>
<td>Ag 110 m</td>
<td>29.2 Ci</td>
<td>6.8 μCi</td>
<td>0.3 mCi</td>
<td>1.7 mCi</td>
<td>1.7 mCi</td>
</tr>
</tbody>
</table>

Table 4
Deposition of Cs-134 on testtube of Saphir 05

Fig. 3

Deposition of J-131 on testtube of Saphir 06

Fig. 4
Desorption of Cs-134 and J-131 of AVR-primary circuit after depressurisation

Fig. 7

Release of Cs 134 from AVR-primary circuit after depressurisation accident

Fig. 8
THE ENVIRONMENTAL IMPACT OF THE HTGR-1160

AND THE RISK ASSESSMENT

by H. Eisele, H.W. Gabriel and J.A. Redondo

HOCHTEMPERATUR REAKTORBAU GMBH

Mannheim

CSNI Specialists Meeting on:

High Temperature Gas - Cooled Reactor Safety

Current Status and Perspective

Petten, 13th - 15th may 1975
1. Environmental influences during normal operation

1.1 Radiation exposure in the plant

All the major components of the primary system of an HTGR, including the steam generators, are housed in the prestressed concrete reactor vessel (PCRV). As a consequence of this, the radiation during normal operation is limited to $<2.5$ mrem/h. Thus, significant exposure for the personnel is possible only during refueling and during the repair procedures of integrated components. An approximate evaluation shows that an HTGR can be operated with a total exposure of 20 man-rem/year [Ref. 1].

1.2 Environmental radiation exposures

The environs of an HTGR receive a low radiological dosage. The paths which result in this dosage are the following:

**Path 1:**
deals with radioactive gas effluents whose sources are:
- Primary coolant from PCRV leaks
- Helium purification system regeneration

**Path 2:**
covers the discharge activity of the liquid waste system

**Path 3:**
deals with the radioactive discharge which could appear as a consequence of the use of distant process and domestic heating.
b) The total amount of Kr 85 occurring in the plant is re-fed into the PCRV and re-directed through the gas purification plant into the storage and gas recovery system.

c) The total amount of Kr 85 occurring in the plant is filled into containers and stored.

The way described under c) causes neither an accumulation of Kr 85 in the plant nor an additional environmental impact of the site.

Path 2

Radioactive liquid waste

Figure 2 shows the concept of the water control system.

Radioactive liquid waste comes mainly from the controlled area (reactor containment building and reactor service building).

Radioactive liquid waste coming from surveyed area (machine house) will be mentioned as well, only because the very severe licensing values for solids make it necessary to consider the side-effects.

The liquid waste reprocessing plant is so designed that the total amount of liquid waste occurring can be reprocessed down to \( \leq 10^{-7} \text{ Ci/m}^3 \), i.e. it can be discharged to the environment without dilution or isotope analysis.

The water of the secondary circuit is exposed to the following activities:

- Activation products from steam generators
- Tritium (due to diffusion from the primary to the secondary circuit).
For the purposes of considering environmental risk it will be both useful and instructive to consider certain "large area-effects", "smog-effect" and "urban-rural-effect".

"Smog-effect"

From Dec. 3rd to Dec. 7th of 1962 the concentration of SO₂ in the Ruhr-area increased up to 4 mg/m³. At the same time the number of deaths in that area increased by about 150 cases. During the smog period in December 1952 in London the number of deaths was even higher than above, reaching some 4000 cases more than normal (Ref. 4).

"Urban-rural effect"

Statistics show a growth in the frequency of cancer of about three-fold from the rural to the urban population. The concentration of chemical elements in the air (Ref. 5) shows the same trend.

A similar clear dependency between damage and natural radiation resulting from the natural radioactivity has not been found.

1.4 Environmental thermal influences

The loss of heat from nuclear power stations is in view of the thermal efficiency and the needs of cooling air, an ecological and economic factor of increasing importance. The use of waste heat for distant heating reduce the environmental thermal influences and the relative consumption of fuel, producing as well a desireable substitut for fossil energy. This concept with an efficiency of some 70 % presupposes consumer-near siting of the power station. This point brings us to the topic of risk analysis.
2.1 Relative Risk Reduction

With respect to the points 1 to 6 examples for way I, "relative risk reduction, are following:

to point 1:

The "bursting strength for the primary circuit" under discussion led to the result that for HTGR, as a precaution, failure of any metal component of the PCRV was assumed. Due to favourable conditions of in-service inspections at the closures (favourable accessibility), this event whose effects can be controlled, can be compared in its event probability, to the failure of the closure of a reactor steel pressure vessel. Such a depressurization can occur in an HTGR only over a cross section of 645 cm$^2$, since the larger PCRV penetrations are equipped with flow restrictors.

to point 2/3/4:

The activity released to the reactor containment is reliably limited to $< 10^{-4}$ of the activity inventory due to the high mechanical and thermal resistance of the fuel elements. Based on the usual calculations of propagation, a whole-body irradiation dose of $< 50$ mrem and a partial-body irradiation dose of $< 300$ mrem was obtained for the environment of the power plant.

to point 5:

The operational capability of the core auxiliary cooling systems represents one of the essential safety aspects, related to the total cooling function. We refer especially to the characteristics of the primary coolant and to the CACS. The heat transfer coefficient of He can not deteriorate under depressurized conditions.
Figure 5 shows, as a typical example of the risk-reducing behaviour of systems, the temperature curves of reactor components after total interruption of the removal of decay heat.

2.2 Risk Evaluation

A risk quantification and evaluation can be carried out at present only to a limited degree. There are, however, three reasons for promoting each step towards this aim:

- The large-scale introduction of nuclear energy will require a transparent risk strategy in future.

- The presently valid deterministic and system-specific guidelines can be evaluated with the help of probability considerations and can thus be analogously applied to different types of systems.

- Risk analyses permit optimization of individual requirements and will thus result in advantages regarding safety and economics. (Ref. 6)
Figure 1

reactor building

service building

AU - environment
ZL - clean air
AL - exhaust air

band filter
cloth filter
micro filter
ceramic filter AC-6 120

NL - auxiliary exhaust air
FL - waste air
grid
air conditioner
blower
shutting clack
stop valve

redundancy
SL - cleaning air

HTGR 1160 air cleaning system

HTR-1160
74.27 - 13e
### Figure 3

<table>
<thead>
<tr>
<th>PATH AND TYPE OF THE RADIOACTIVE DISCHARGE</th>
<th>ANNUAL RELEASE (Cl/a)</th>
<th>ANNUAL DOSAGE (mrem/a)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>E</td>
<td>D</td>
</tr>
<tr>
<td>1st RADIOACTIVE GASEOUS EFFLUENTS</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Xe Kr (CONT.)</td>
<td>~200</td>
<td>1000 - 2000</td>
</tr>
<tr>
<td>SOLIDS (CONT.)</td>
<td>&lt;2 \cdot 10^{4}</td>
<td>2 \cdot 10^{4}</td>
</tr>
<tr>
<td>Kr 85 (DISCONT.)</td>
<td>0</td>
<td>~5000</td>
</tr>
<tr>
<td>2nd RADIOACTIVE LIQUID EFFLUENTS</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SOLIDS</td>
<td>0.1</td>
<td>0.5</td>
</tr>
<tr>
<td>H 3</td>
<td>800</td>
<td>2000</td>
</tr>
<tr>
<td>3rd DISTANT HEATING (PROCESS AND DOMESTIC)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SOLIDS</td>
<td>0</td>
<td>~0</td>
</tr>
<tr>
<td>H 3</td>
<td>0</td>
<td>~0</td>
</tr>
</tbody>
</table>

E = EXPECTATION  
D = DESIGN

---

**Table:**  
**Radioactive Discharge and Dosage of the HTGR 1160 MWe During Normal Operation**  
**HTR 1160**  
**75.15 - 4**
Figure 5

Temperature of the reactor components after scram without core cooling

- Control Rods
- Max. Fuel
- Average Fuel
- Bottom Insulation
- Top Insulation

Safety Load Limit
Operational Load Limit
Contents

1. Introduction

2. The depressurization of the HTR - 1160 and the THTR - 300

3. Corrosion of graphite primary circuit components in HTR's by oxygen

4. Safety assessment of the consequences of the depressurization incident
2. The depressurization of the HTR-1160 and the THTR 300

Primary circuit and other essential reactor components of the high temperature reactors HTR-1160 and THTR-300 are contained within a prestressed concrete pressure vessel (fig. 1 and 2). Auxiliary systems, which are connected with the primary circuit, are set up outside the pressure vessel. Pressure vessel penetrations are closed either with a single or two independent steel covers. Since a failure of the concrete pressure vessel and of the double covers is not credible due to their method of construction, a loss of coolant is only conceivable at the penetrations with single steel covers or in the auxiliary systems which contain primary circuit gas. In order to reduce the gas egress rates if those single barriers fail, flow restrictors which reduce the cross sectional area are installed in all such pressure vessel penetrations of the HTR-1160 and the THTR-300. In this way the flow cross section is limited to 645 cm² for the HTR and to 7 cm² for the THTR. As opposed to the HTR-1160 the maximum possible egress rate for the THTR-300 is not determined by a failure of a penetration cover but by rupture of a tube which contains primary circuit gas outside the pressure vessel. The diameter of all pipes leading out of the THTR pressure vessel is limited to less than 65 mm. The largest possible leak can occur in the fuel element loading and unloading system whose connections to the reactor core lead through the bottom of the concrete pressure vessel.

The primary circuit gas that leaves the THTR in the course of a depressurization accident is released to the atmosphere either via the ventilating system or via the 150 m high stack. The coolant gas escaping from the HTR-1160 is collected in a containment which surrounds the pressure vessel and the associated plant.
The after-heat cooling is performed with two loops of each auxiliary cooling system. Conservatively it was assumed for the system transients shown in fig. 3 that only two cooling loops are available.

After pressure equilibration in the HTR-1160 a convective gas exchange begins between containment and primary circuit. In addition containment gas enters the primary circuit due to contraction of the remaining cooling gas. When the auxiliary cooling system is started the temperature of the core inlet gas and thus the average primary circuit temperature decrease instantaneously. Thereby ca. 720 m$^3$ (S.T.P.) are drawn into the primary circuit. The second period of contraction starts after the temperature of the primary coolant has passed through a maximum, 2 hours after beginning of the accident.

The gas exchange due to convection occurs only if a penetration cover in the top of the concrete pressure vessel fails, because in this case a gas layer of low density is positioned underneath a layer of higher density. The total oxygen inleakage into the primary circuit of the HTR-1160 due to gas contraction and convective inflow is shown in fig. 4 as function of time.

This figure also shows the oxygen ingress from the atmosphere into the THTR-300. Two effects were considered in the calculation of the inleakage transient: contraction of the gas in the primary circuit and fluctuations of the atmospheric pressure. Between four and six hours after pressure equilibration the core temperatures in the THTR pass through a maximum. Pessimistically it was assumed that during this period two rapid atmospheric pressure changes both of 25 mm Hg occur.
3. Corrosion of graphite primary circuit components in HTR's by oxygen

As previously mentioned oxygen can enter the primary circuit when the depressurisation process is finished. This will react with fuel element, reflector and bottom reflector graphite. In this chapter the method of calculation of graphite corrosion in the THTR - 300 and the HTR - 1160 is outlined and the data used in calculation is presented.

3.1 THTR - 300

The THTR is a pebble bed reactor and has spherical fuel elements with a diameter of 6 cm. The outer fuel-free layer is of A3-graphite and has a thickness of 0,5 cm. The coated particles are embedded in an A3-graphite matrix material. The corrosion of A3-graphite with oxygen was determined on unirradiated fuel free balls which were manufactured in the same way as the normal THTR fuel balls. They fitted the graphite specification for the THTR.

3.1.1 Determination of the rate of corrosion of A3-graphite by oxygen

The apparatus used is shown schematically in fig. 6. The sample is suspended on a platinum wire in a vertical tube furnace. The surface temperature of the sample is measured with a Pt-Pt/Rh - thermocouple.
In the oxygen partial pressure range investigated, the apparent reaction order lay between 0.67 and 0.7. Similarly the temperature dependence of reaction rate, determined on two other samples is shown in fig. 9. The apparent activation energy lay in the range of 23.6 to 25.2 kcal/mol. These results agree well with literature values for other types of graphite. Hawtin and Gibson (1,2) found apparent reaction orders of 0.54 - 0.6 and 0.78. Lewis et al. (3) give values of 0.6. Hoelscher and Effron (4) determined reaction orders of about 0.6. The activation energies determined by a number of workers lie between 37 and 58 kcal/mol (5-9). The values found by HRB appear to be somewhat lower than the literature values. However this can be explained by considering the reaction type. The literature values have been determined in the chemical or volume reaction regime. The HRB values have been measured in the in-pore-diffusion regime. It is well known that the activation energy for the in-pore-diffusion regime is only half the value of the activation energy in the chemical regime. Thus the agreement of our values and the literature values is reasonable.

3.1.2 The effect of radiation and catalysis by fission products on corrosion of graphite by oxygen

The influence of radiation and fission product catalysis on graphite corrosion by oxygen effects the rate of corrosion mainly in the low temperature regions of the core. This leads to a more equal attack on all the fuel elements.
Mc is the atomic weight of carbon (g), \( n_i \) is the number of fuel elements in each temperature region of the core, \( E \) [cal \cdot mol^{-1}] the activation energy, \( R \) [cal \cdot mol^{-1} K^{-1}] the gas constant and \( T_i (\Delta t_j) \) is the surface temperature of the fuel elements in the \( i \)th class during the time interval \( \Delta t_j \).

Using these equations the total graphite corrosion in the core was calculated. With a similar equation the burn off of hot fuel elements was calculated. The calculation was programmed in Fortran IV.

Similarly the extent of corrosion of the core bottom reflector was calculated. The area of the bottom reflector in most danger is the upper 25 cm layer which has a network of many 16 mm diameter holes and on which the fuel elements rest. The thickness of material between each hole is 20 mm. These blocks are manufactured from Gilsonite PXA2N-graphite, which has better corrosion resistance than A3 graphite. This is being predicted by the results of water corrosion tests. Conservatively we have used the A3 data.

3.1.4 Results

The calculations were made using the data in fig. 4 and 5. They show that almost all the oxygen, which enters the circuit, will react. Three corrosion rates were used (Sect. 3.1.2). However an increase in corrosion rate has little effect on the extent of attack, as may be seen in the table below.

<table>
<thead>
<tr>
<th>corrosion rate</th>
<th>% oxygen reacted</th>
</tr>
</thead>
<tbody>
<tr>
<td>( r_0 )</td>
<td>88</td>
</tr>
<tr>
<td>3 ( r_0 )</td>
<td>93</td>
</tr>
<tr>
<td>6 ( r_0 )</td>
<td>96</td>
</tr>
</tbody>
</table>
3.2.1 Data for the Calculation of Graphite Corrosion

The volume reaction rates for an American reference graphite were measured between 380° and 570° C. Approximately half of the corrosion experiments were carried out at a γ-flux of 10^6 rad/h. Irradiation had no effect on the corrosion rate. Corrosion of pre-irradiated specimens, which had received a neutron dose of 6.10^{21} cm^{-2} (E > 0.1 MeV) showed no change in corrosion properties. The findings of these investigations is shown in fig. 10. The results were fitted to the expression below using a least squares method:

\[ \tau_v = 0.54 \times 10^{-10} \exp(-\frac{4000000}{RT}) \cdot \left(\frac{p}{p_0}\right)^{n} \left[\frac{\%}{h}\right] \]

\[ p_0 = 0.21 \text{ atm} \]

It was shown by these experiments that the corrosion experiments were carried out in a volume reaction regime since change in the ratio of surface area to volume produced no detectable change in corrosion rate.

3.2.2 Relationship between the Volume and Surface Reaction Rates

Competitive material transport within pores and the chemical reaction rate is known to occur between volume reactions at low temperatures and surface reactions at higher temperatures. It is possible to relate \( \tau_v \), the volume reaction (3.2.1) and \( \tau \), the surface reaction rate (3.1.1) by the following equations:

\[ \tau_v = \hbar \cdot C^{n} ; \quad \tau = \sqrt{\hbar \cdot D_{\text{eff}}} \cdot C^{\frac{n+1}{2}} \]
4. Safety assessment of the consequences of the depressurization incident

The oxidation of graphite during a depressurization incident gives rise to three main consequences:

The mechanical strength of graphite core components is affected by corrosion.
Heat is produced. This and the residual heat must be removed.
Carbon monoxide is formed. Thus the possibility of formation of a combustible mixture must be examined. These points will be discussed in detail in the following sections.

4.1 Effect of corrosion on the strength of the graphite core components

Surface attack of oxygen on graphite reduces the strength of the material due to reduction of the cross sectional area. Attack in the interior of the solid is more serious. For this to happen the corrosive gas must penetrate deep into the inner of the component without reacting before reaction occurs. It was essential therefore to investigate the dependence of the loss of strength of graphite on the extent of corrosion. This work is described in this section.

The compressive strength of the THTR fuel elements must be sufficient to withstand both the static weight of the balls above them in the core, and the load exerted by the descent of the shutdown rods into the core. It was specified that the compressive resistance of the fuel elements determined between two parallel plates should be at least 1800 kp. In order to ascertain how the compressive strength was affected by corrosion 10 fuel elements were oxidised to between 0.4 and 1% weightloss using steam at 800 ºC.
The ungraphitised part is more rapidly corroded. This leads to a definite disintegration of the material. This data has been used to predict a < 5% compressive strength loss in the core bottom of the THTR-300 on air ingress (sec. 3.1.4). In view of the highly pessimistic safety factors used in these calculations, this small loss of strength during a depressurization incident can be considered as insignificant.

Initially this data is also being used for the HTTR-1160 calculations until the compressive strength of the graphite chosen for this reactor has been determined on both uncorroded and corroded samples.

Using this data, it follows that the loss of strength by the hot elements in a single channel can amount locally to about 20% and in the permanent bottom reflector to about 3%. In the core support structure it is negligible. Thus it can be safely said that no part of the primary circuit will suffer a serious strength loss in a depressurization accident.

4.2 Heat production by graphite corrosion

The generation of heat by the graphite - oxygen reaction may be expressed by use of the equation:

\[ \Delta H = \frac{\Delta m_{\text{cor}}}{M_c} \cdot \overline{\Delta H} \]

where \( \Delta m_{\text{cor}} \) is the mass of graphite that has reacted, \( \overline{\Delta H} \) the molar enthalpy for the reaction and \( M_c \) the gram atomic weight of carbon.

The maximum rate of evolution of heat may be written:

\[ \Delta H_{\text{max}} = \frac{\overline{\Delta H}}{M_c} \cdot \left( \frac{\partial m}{\partial t} \right)_{\text{max}} \]
In the case of the latter system, two curves are shown which were determined using upward and downward climbing flames. The former yields a more pessimistic, and hence preferable curve. In the helium system no determinations have been carried out using an upward burning flame. It was thus necessary to draw a corrected curve.

Fig. 14 shows that the limiting concentration of carbon monoxide in nitrogen, such that a mixture with air will not ignite, is 20%. Similarly the limiting value of carbon monoxide in helium is 28%. The ignition limits for the quaternary system containing both inert gases can be determined by interpolation.

4.3.2 Composition of the primary circuit atmosphere after the entry of air during a depressurization incident

It has been calculated that the maximum carbon monoxide concentration in the primary circuit of the THTR-300 is 3% by volume. It will be seen from fig. 14 that with such a gas mixture it is impossible to form an ignitable gas with air. Also only after the reactor has reached the maximum temperature and is cooling down, will air be drawn into the reactor. Under these conditions carbon monoxide will have to diffuse out of the reactor against a pressure difference. It is thus impossible that gas escaping from the leak may ignite outside the reactor.

In the case of the HTR-1160 it has been calculated that within 16 hours of the pressure equilibration, all the oxygen in the primary circuit will have reacted. At this time, the temperature of the core will be low. Thus the reaction rate will be slow. Hence although oxygen will continue to diffuse into the core, its contribution to the build-up of carbon monoxide will be insignificant.
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HTGR 1160 MW Nuclear Steam System

Pig.

-227-
Fig. 3
Fig.
1: SPHERE CORRODED AT 630°C IN OXYGEN
   TOTAL BURN OFF : 0.78 %
   BURN OFF IN H₂O BEFORE TEST : 0.25

2: SPHERE AS RECEIVED

Fig. 7
Fig. 9

TEMPERATURE DEPENDENCE FOR REACTION RATE
OF A3-GRAFHCIT WITH OXYGEN

HTR-1160
75.11 - 7
1: PARALLEL TO GRAIN
2: PERPENDICULAR TO GRAIN

COMPRESSIVE STRENGTH (kp)

BURN OFF \( \gamma \) (WEIGHT %)

Fig. 11
Fig. 13
Fig. 15

TOTAL OXYGEN MASS AND CO-MOLEFRACTION IN THE PRIMARY LOOP AFTER DEPRESSURISATION ACCIDENT

HTR-1160
75.11-13
With respect to the emergency core cooling system of a reactor (ECCS) the Safety Criteria require that it has to be designed to have no loss of its capability to sufficiently remove afterheat from the core in case of a loss of coolant accident including any additional single failure of an active or passive component, also when parts of the system are under repair or undergoing functional testing. This means that a nuclear power plant, which ought to be operating during repair or functional testing of one of the core auxiliary cooling loops, has to be equipped with an ECCS consisting of either 3 loops with a capacity of 100 % each or 4 loops with a capacity of 50 % each. Such a system can remove the afterheat sufficiently in the event of loss of main loop cooling even when two loops are not available. A 3 x 50 %-system does not meet this safety criterion.

In the FRG the requirements for the ECCS of light water reactors have been applied to the core auxiliary cooling systems of HTR's in spite of the fact that the main loop cooling has the capacity to sufficiently remove afterheat from the core even in case of the primary system depressurization accident.

During the construction of the THTR-300 the design criteria for nuclear power plants in the FRG changed essentially with respect to protection against man induced external impacts. The main events are airplane crashes, pressure waves from chemical explosions and sabotage. All parts of a plant which are necessary to ensure that the reactor can be safely shut down and held in the subcritical state, and that the afterheat can be removed, and that the eventual release of radioactive material can be kept below acceptable limits, shall be designed to perform their safety functions even with one of the above-mentioned impacts. The guide-lines of the German Reactor Safety Commission for pessurized water reactors /2/ define the external impacts to be regarded. These sections of the PWR-guide lines are also applied to other types of reactors.

With respect to airplane crashes a time-dependent load with a maximum of 110 GN acting on an area of 7 m² was defined (Fig. 1) /3/. The
and the reliability of the core auxiliary cooling system of a HTR, it is difficult to estimate the extent of damage if these systems fail during an accident. An extensive attempt was made for the LWR with the Rasmussen Study /4/. The cooling medium and the material of the core components of a HTR and their physical properties differ markedly from those of a LWR. The cooling medium of the HTR will not suffer a phase change and the heat capacity of the core and the insensibility against a temperature rise is higher compared with a LWR. The supplier of the HTR should analyse this in detail in order to derive benefit from these material characteristics in the licensing procedure. These considerations will be important during the licensing procedure for the HTR-1160.

4. Adaption of Plants under Construction to the Progress in Safety Standards

A special question is the applicability of new criteria or guide-lines to nuclear power plants under construction. In this regard we have to distinguish between a new requirement which reflects a new knowledge of an undue risk, and requirements due to general progress in safety philosophy. In the case of an undue risk any measures to overcome the risk have to be included also in plants under construction. In the case of general progress in safety philosophy, plants under construction should comply as far as possible with corresponding requirements. This means that the applicant should utilize additional safety equipment if the progress in the construction of the plant makes it possible without undue cost.

5. Comments on the THTR-300 MWe

The THTR-300 was faced with changing safety requirements during its construction which began in 1971. The protection of a nuclear power plant against pressure waves from chemical explosions was for the first time required in 1970 when, at Brunsbüttel, a nuclear power plant was erected in the neighbourhood of a chemical industry. As a general requirement protection against pressure waves from chemical explosions was introduced in 1972.
2) The redundant installations which are necessary for the afterheat removal, for the shut down of the plant, and for the supervision of the plant are locally separated or protected by constructional arrangements so that the probability becomes very low that two redundant installations would be destroyed by an airplane crash or a pressure wave.

Despite these improvements there are deviations from the safety criteria of the Federal Ministry of Interior for ECCS. There are some points in the secondary circuit (bypass flash tank, tubing) and in the component cooling water system (e.g. circulator motor cooling) which are common to the main loop cooling and to one of the two core auxiliary cooling systems. An initial failure of such a component makes the main loop cooling system and one of the core auxiliary cooling systems inoperable. Thus, a single failure of a passive component in the other core auxiliary cooling system or a single failure of an active component while the redundant one is being repaired or undergoing functional testing would cause a loss of function of the second core auxiliary cooling system. However, it should be mentioned that many failures can be overcome by emergency measures to supply single components or cooling circuits with water.

A final recommendation of the RSK and of the consulting engineers (TOV, IRS, ...) performing the safety analyses for the licensing authority has not yet been announced. The advice of the experts will depend on the reliability of the heat removal systems; this is now under investigation.

6. Comments on the HTR-1160

The concept of the General Atomic 1160 MWe nuclear power plant as it was presented by HRB for licensing in the FRG shows a clear

RSK-Leitlinien für Druckwasserreaktoren, Bundesanzeiger Nr. 144 vom 7. 8. 1974, Jahrgang 26

K. Drittler, P. Gruner und L. Sütterlin
"Zur Auslegung kerntechnischer Anlagen gegen Einwirkungen von außen, Teilaspekt Flugzeugabsturz - Zwischenbericht"
IRS-W-7 (Dezember 1973)

Fig. 2 Overpressure Versus Time Resulting from a Deflagrating Gas Cloud
Fig. 4 THTR 300MWe – Component Cooling Water System
1. INTRODUCTION

The experimental multi-purpose high-temperature gas-cooled reactor, now being planned in JAERI, should provide the functions permitting the development and demonstration test of process heat application. Over the past years, the design studies of the experimental reactor plant have been conducted and the preliminary conceptual design was completed in 1974.

The system of the reactor plant, obtained in the design work, is schematically shown in Fig. 1. The reactor is composed of a steel reactor vessel and pin-in-block type fuels, and is provided with two systems of primary cooling circuits, which are connected to the secondary cooling circuits, respectively, through intermediate heat exchangers. The steam reformer and steam generator are placed in one of the secondary circuits (reformer loop), and the reducing gas heater, gas turbine and cooler in the other circuit (test loop). By any combination of various types of component, it is possible to provide a several kinds of operation mode meeting with the predetermined experimental program.

The safety characteristics of the reactor plant are largely different from those of light water reactor. The behaviors of fission products and the operating responses of the reactor core and heat utilization components are considered to play an important role in characterizing the safety aspects of the reactor plant.

The objective of this paper is to examine the behaviors of fission products in the plant, especially on the release from fuel and the deposition on coolant circuit surface, and the overall dynamics of the reactor plant in transients and accidental conditions. Design of the engineered safety features is also presented.

2. BEHAVIORS OF FISSION PRODUCTS DURING NORMAL OPERATION

Characteristics which control the release of fission product from reactor plant may be classified into two groups, those inherent with fission product itself, such as release from fuel and deposition on coolant duct, and those inherent with plant system, such as leakage from cooling circuit and removal by purification and ventilation facilities.

In this section, the effects of the former on fission product behaviors are studied.

2.1 RELEASE FROM THE FUEL

The compact and sleeve of fuel pin are considered to be one of the barriers for the release of fission products, as well as the coatings of fuel particle. In practice, however, some metallic fission products diffuse and pass through the compact and the sleeve, and evaporate into the reactor coolant.
results for the non-volatile fission products (Zirconium and Ruthenium) except the region of the entrance effects. However, for the volatile fission products (Iodine) the agreement is less excellent. Based on the above results, it is concluded that the computer code can be used for predicting the distribution of non-volatile fission products deposited on duct surface from gas stream.

In Fig. 4, the calculational results of the deposition distributions of Sr\(^{90}\) on the inside surface of the primary coolant ducts in the reactor are shown. From this figure, it is seen that in the early stage of plant operation, the concentration distribution of Sr\(^{90}\) decreases with the distance from the entrance of the primary cooling circuit, while the distribution becomes relatively uniform with the operation of plant. It is interesting that the concentration of Sr\(^{90}\) in the low temperature region of the primary cooling system is higher than that in the high temperature region, and that the IHX may be a critical component in determining the permissible deposition activity in the primary system from the maintenance point of view.

2.3 ACTIVITIES IN THE PLANT

The amounts of activities which are retained in the plant and released to the environment were calculated. The schematic model of the calculation and the summary of the results for the experimental reactor plant are shown in Fig. 5 and Fig. 6 respectively. As seen from the latter, the activities nearly as much as that removed by the purification systems deposit on the duct surface of the cooling circuits. This large amount of the deposited activities will have important effects on the safety of the reactor plant.

3. DYNAMICS OF THE REACTOR PLANT

To avoid excessive fatigue of the components and structures and to minimize the possibility of the fission product leakage to the environment, the key variables of the plant, such as the temperatures and pressure of coolant, should be controlled precisely so that the temperatures of the components that compose pressure boundary of the primary cooling system are placed below the specified values, and so that the pressure of the primary cooling system is maintained lower than that of the secondary.

In this section, the operational characteristics of the plant obtained by digital computing simulations are presented, paying attention to temperature and pressure responses.
Taking into consideration the characteristics of the reactor plant in postulated accidents, the auxiliary and the reserve cooling systems are provided. The auxiliary cooling system is to cope with the type of accidents in which it is still possible to cool directly the core by means of forced circulation. The reserve cooling system is for coping with the type of accidents in which the direct cooling by forced circulation is no longer possible.

(3) Reactor Containment Facilities

This facilities suppress the radioactivities from scattering over the environment in the event that activities are released out of the primary cooling circuit due to its rupture or any other causes.

A complete double containment sytem is provided to minimize the hazard to environment during the normal operation and at the time of accident. In addition, leak tight inner compartments are prepared using the biological shield concrete structures, with a view to achieving several safety functions in the normal and accidental conditions.

5. REQUIRED STUDIES

Based on the fundamental safety characteristics of the experimental reactor obtained in this analysis, the following items should be examined in the future design work.

Establishment of the limitations on the amount of activities released from the fuel pin and deposited on the coolant circuit surface.

Design of the effective plant control systems which placed the temperatures of components below their accepted design levels, and the proper protection systems which suppress the transients in accidents so that the primary coolant pressure is maintained lower than the secondary.

Affirmation of the functions of the engineered safety features must also be conducted by simulation calculations and experiments.

ACKNOWLEDGEMENTS

The authors would like to thank members of the nuclear technology divisions of Fuji Electric Co., Ltd. and Kawasaki Heavy Industries, Ltd. who have conducted the analyses under the contract between JAERI and these companies.
Fig. 2  Retained fractions for Sr$^{90}$; thickness of compact and sleeve are 0.7 cm and 0.5 cm respectively.
Fig. 4  Distributions of $\text{Sr}^{90}$ in the primary cooling circuit. $K_a$ is the desorption coefficient.
Fig. 6 Activities retained in the plant and released to the environment.
Fig. 8 Plant responses to the loss of electric power accident.
ASPECTS OF MEASURING, IDENTIFYING AND CONTROLLING STEAM GENERATOR LEAKS IN HTGR's

by

H.P. Drescher

Leaks and ruptures in steam generator tubes of conventional power plants are reported rather often. The same is with smaller leaks of steam generators in Pressurized Water Reactors which happened often in the last time and which did not influence the normal operation of the plants very much. Contrary to this the heat exchangers of HTGR's have to be absolutely undamaged during operation. The $\text{H}_2\text{O}$-content of the primary coolant gas of the HTGR is assumed to be 0.1-1.0 Vpm. That means that the primary circuit containing approximately 40,000 cbm helium only contains 5-10 g $\text{H}_2\text{O}$. Even very small leaks (50-100 g/h) which cannot be detected in other power plants can lead to unspecified corrosion at the fuel elements and at the reflector graphite. The immediate rupture of a steam generator tube has to be considered as a strong damage after which the plant has to be shut down immediately.

Because of the high corrosion rate after a great ingress of water (500-1000 kg) it is advantageous not only to scram the reactor but also to cool it down rather quickly. The graphite surfaces of the fuel elements and the reflector should be cooled below 600 °C as fast as possible. This procedure has to be initiated and managed very carefully for reasons of not exceeding the allowed corrosion and the allowed temperature changes of the heat exchanger units and of the other metal components of the primary circuit. Care is necessary as these procedures differ from a "normal" scram and have to be initiated by additional input signals.
The temperature of the helium on the cold side of a steam generator can be changed according to the thermodynamic data of the penetrating water. This influence strongly depends on the construction data of the steam generator and on its operating conditions (full power, hot or cold water operation) and on the position of the rupture. In most cases one can expect the penetrating water cooling down the helium.

After a rupture of a tube the pressure of the primary circuit will be increased. That effect is important for the construction of the pressure vessel and the coolant gas circuits. The rupture will raise the pressure by approximately 0.1 bar/sec up to a maximum value of 2-3 bar. Very important for safety analysis is the positive reactivity effect of penetrating steam and the resulting increase of power level and temperature. This influence strongly depends on the nuclear design of the core. The greatest effects are at small cores with low moderation ratios and a small primary circuit volume. At big power plants it should be easily managed. After 30 seconds there is no further water joining and being mixed to the coolant gas. During this time the helium content of the primary circuit is circulated every 5-10 seconds. That means that there should be no "clouds" of water with a positive reactivity effect passing the core. The maximum positive reactivity will be less than 200 mWile. This may be less by a factor of 5 at power plants with other nuclear and technical design. If there is no compensation of the additional reactivity by the absorber rods the reactor will reach 110 % of its former power level after 5 seconds or more and thereby will be scrammed by exceeding its neutron flux limit. Probably the power level of the core will not exceed its former level (except of smaller oscillations) due to automatically setting the absorber rods. By this there will be no scram initiated. The same is with the pressure of the primary circuit which is increasing rather slowly and probably will not exceed its limit.
100 g/sec raises the H₂O-content of the coolant gas by 100 Vpm along the steam generator unit. A leakage rate of 100 g/sec thereby may be regarded as the smallest value which can be identified automatically by the safety systems. In some cases especially at leaks in the cold region of the steam generator one has to take into account that the coolant gas and the leakage is not mixed completely so that the measuring positions are not touched necessarily.

There may be problems if a steam tube does not rupture immediately but following a small leakage. Such a leakage will not be identified automatically but nevertheless will raise the average humidity of the primary circuit above 500-1000 Vpm within 5 minutes. Thereby the measuring range of all hydrometers is exceeded without identifying the leak. If after this delay the steam tube breaks there is no possibility for detecting the damaged unit by any instrumentation of the primary circuit. In this case the reactor will be scrammed and all heat exchanger units are cut off. The defect steam generator only can be identified now by the instrumentation of the secondary loops (reversed flow etc).

These facts show some problems of steam generator leaks in HTGR's. The scram of the power plant can be initiated very easily but the identification of the defect unit demands a lot of sensitive controlling procedures. The different effects which follow a great ingress of water have been mentioned. The positive reactivity effect will be compensated by setting the absorber rods. The pressure of the primary circuit is increased very slowly and will give a very late signal. The coolant gas temperature behind the steam generator units is also changed by the penetrating water rather rapidly but the amount of change is very difficult to predict and varies according to the accident conditions. It should be researched whether transients of coolant gas temperature could deliver a useful input signal for the identification of leaks. The change of the specific weight of the coolant flow by the additional water gives a remarkable effect on the power demand of helium blowers. Together with the H₂O-detectors and the instrumentation on the secondary side of a steam generator this effect can give useful information for identifying great leaks. Every of these effects strongly depends on the conditions especially on the power level under which the rupture occurs.
CONSEQUENCES OF HTGR WATER INGRESS EVENTS INTO PRIMARY COOLANT SYSTEM

by

A. W. Barsell, V. Joksimovic, M. B. Peroomian

This is a preprint of a paper to be presented at the Specialist Meeting on High-Temperature Gas-Cooled Reactor Safety, May 13-15, 1975, Petten, Netherlands, and to be printed in the Proceedings.

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1. ABSTRACT

An event of particular interest to the HTGR is inleakage of water/steam into the primary coolant system with attendant PCRV pressure rise and reactions of steam with core graphite and failed carbide fuel particles. A spectrum of leak rates ranging to a steam generator (SG) offset tube rupture was analyzed with expected response of the plant demonstrating no significant core damage, hence plant operation can be resumed after a clean-up. The bounding event within the plant design basis is an offset SG tube rupture with failure of the redundant moisture monitors to identify, isolate and dump the leaking SG. Safe shutdown of the plant and fission product activity release within allowable limits are predicted with high design safety margins. To illustrate inherent safety margins for so-called Class 9 events beyond the plant design basis, effects of multiple tube ruptures as well as concurrent PCRV depressurization plus steam leak are considered and shown to engender negligible risk to public health and safety.
rates up to 150 lb/sec. Also, analysis is presented for a tube rupture occurring coincidently with PCRV depressurization which indicates high safety factors for no potential flammability of steam-graphite reaction products accumulated in the containment.
2/3 power.

For a postulated depressurization event occurring simultaneously with a steam leak, the primary consequences are the containment pressure rise, which is not significantly increased over that for depressurization alone, and the activity release, which is about twice that for depressurization alone. The threshold of potential local plume flammability of gases egressing from the PCRV is approached at long times after initiation of the event. However, the volume of gas associated with such local burning is too small to impair containment integrity. Accumulation of non-combusted CO+H₂ is predicted to be 20 times less than that needed for a flammable concentration of mixed contaminant gas. It is concluded that safe shutdown of the plant and preservation of containment integrity is assured.
5. INITIATING EVENTS

For safety evaluations, the critical types of postulated steam/water leaks to the primary coolant system are those occurring at full power conditions when pressure and graphite temperatures are highest. The upper envelope for rate and quantity of steam inleakage is a postulated failure in the main bundle of the steam generator. The SG main bundle in the HTGR has thick-walled tubes designed to withstand the secondary-to-primary coolant pressure differential at maximum SG pressure. Analysis indicates that the tube design, which is constrained by long-term creep effects during normal operation, has substantial stress margin for short-term stresses under accident conditions. Only about 1/4 of the tube thickness is needed to withstand internal water pressure at atmospheric external pressure.

Steam generator tube leaks have been experienced in past gas-cooled reactor operating history, as summarized in Ref. 6. An offset tube rupture inside the steam generator cavity in the PCRV is a low probability event. However, for conservatism, the design philosophy governing HTGR plant protective systems is to guard against steam inleakage accidents of such severity as an offset rupture of a SG tube at the worst location under most adverse plant conditions.

At full power, the worst SG tube rupture occurs in a feedwater tube near the feedwater tubesheet. Calculated steam/water inleakage for this rupture shows that within several seconds a constant inleakage at a rate of 24 lb/sec is achieved. The corresponding leak rate for a superheater tube rupture is about 11 lb/sec.
under tentative technical specification limits. Higher leak rates up to 2.6x10^{-3} lb/sec correspond also to "normal" operating conditions but limit the plant operation within the context of tech spec limits. The condition of the primary coolant is monitored by the chemical analyzer instruments which record the oxidant levels. Alarm devices also assist the operator control.

Leak rates higher than 2.6x10^{-3} lb/sec correspond to equilibrium oxidant impurity levels above 1000 ppmv requiring operator or automatic action for plant shutdown and remedial cleanup. A spectrum of leak rates was analyzed ranging from slow inleakage through a small crack or hole in a tube (in the "upset" category) where the operator has time to perform an orderly shutdown to larger leak rates up to that for an offset tube rupture.

Operator action is assumed to occur at 30 min and PPS action at 1000 ppmv, whichever comes first. Results of maximum oxidation in fuel elements and core support posts are presented in Fig. 3. It was found that the limiting leak rate yielding 1000 ppmv moisture concentration in the primary coolant at 30 min is 0.067 lb/sec. At lower leak rates, the moisture concentration builds up to less than 1000 ppmv at 30 min; corresponding core graphite oxidation reduction is less than proportional to leak rate. Thus, a sharp local maximum burnoff is incurred at the limiting leak rate of 0.067 lb/sec. Faster PPS actions where the moisture detector set point of 1000 ppmv is reached (less than 30 min) cause lower oxidation at leak rates just higher than these critical upset leak rates. Beyond leak rates of 1 lb/sec, the response delay time of the moisture monitor system is assumed to be limiting (as opposed to the time to reach 1000 ppmv). Thus, the oxidation again increases beyond 1 lb/sec and the maximum burnoff for the worst offset tube rupture is nearly equivalent to that for the critical upset condition.
neither the fuel block nor the support post burnoff exceeds values for which safe shutdown of the plant would be jeopardized.

The primary coolant pressure is calculated to peak at 765 psia which is about 10 psi below the rupture disc set point. Therefore, no primary coolant or activity release to the containment is predicted for this event.

6.3 FAILURE TO ISOLATE AND DUMP STEAM GENERATOR

A hypothetical event involving steam inleakage into the primary system with concurrent failure of the automatic PPS action to isolate and dump the leaking SG is considered improbable because multiple failures of redundant safety system components are required. For example, the SG isolation and dump systems employ two out of three MMS instrument signals, redundant logic systems, and safety grade redundant closure or opening valves and actuators. These features make the systems single failure proof. Consequences of failure of the systems are analyzed to illustrate the extent to which these systems must respond.

The initiating event is assumed to be an offset tube rupture occurring at the worst time in the plant life at full power where the leaking loop is not automatically isolated and dumped by the PPS. Cause of the incorrect plant response is postulated to be failure of the moisture monitors in the leaking loop. In this event, the protective action results in isolation and dump of the first loop to indicate the high moisture level trip point. Isolation and dump of this wrong loop along with reactor trip are assumed to occur 98 sec after inleakage begins. The assumed sequence of events is depicted schematically in Fig. 1.

Since the steam leak is not terminated by PPS action, the increase in pressure due to the leak is sufficient to cause the opening of a PCRV pressure relief valve. Helium, steam, and any steam/graphite reaction
7. PLANT SAFETY MARGINS FOR CLASS 9 EVENTS

In the preceding sections, the steam leak events analyzed were those within the design basis of the HTGR. In order to demonstrate the ultimate safety of design beyond the design basis events, analyses have been performed for the extremely low probability (Class 9) events.

7.1 MULTIPLE TUBE RUPTURE

Analysis was performed for a hypothetical rupture of multiple steam generator tubes with expected plant protective action. Such an occurrence is beyond the design basis of HTGRs which includes the worst random failure of a SG tube.

An analysis for a typical 2000 MW(t) reactor was performed with selected steam inleakage rates of 30, 90, and 150 lb/sec. These leak rates can be associated with corresponding numbers of tubes failed. The exact flow rate from a number of tubes will depend on the plant system characteristics.

The sequence of events and input parameters are basically the same as in Section 6.1. Based on an expected moisture monitor response time, the steam generator begins to be isolated and its contents dumped at 22 seconds for leak rates of 30 and 90 lb/sec. For the 150 lb/sec leak rate, the reactor is tripped within 14 seconds on overpressure signal set at 752 psia, and the isolation and dump begins at 22 sec.

The results for all the cases are summarized in Fig. 2. Only a small amount of graphite is reacted with less than 1% of steam in each case. For the 30 lb/sec and 90 lb/sec leak rates the pressure rise in the PCRV is not large enough to lift the safety valve. However, for
stratified, calculated, graphically illustrated for steam egresses from this
examination agree, too, although it is low enough to be added to the
melting steam in the reaction. As expansion commences, the steam meets a
pressure of the reaction, and the remaining steam moves away from the
steam surface. The remaining steam is then expanded 100 percent during the
blowdown, without a shock wave. It is calculated that the heat release rate can be
dissipated to the containment heat sinks for pressure changes from the PWR and the
containment pressure increases are summarized in Table 4. Comparisons
are made with a depressurization event without a steam leak. The
ratio of H₂ to CO concentration needed for overall containment
depressurization is about 0.4 for the steam leak case in
PWR at 2 hours. Beyond 2 hours, the gas egress from the PWR
is larger since the molecular weights of the CO and containment
equalize. The final containment composition is calculated to have a
flammability of 0.4 with H₂ to CO₂ ratio of 20 in the
containment. The flammability of the reaction product gases
are illustrated in Fig. 4. After all steam has reacted
at 2 hours, the maximum concentration of CO and H₂ is attained in the
PCRV. The points in the lower right hand corner (0% air) in the
flammability diagram in Fig. 4 represent these maximum concentrations.
For the depressurization accident alone, there is a small concentration
of CO at 2 hours due to air ingress. A slow gas exchange between the
PCRV and the containment is predicted due to a molecular weight
difference. A conservative estimate of the initial (maximum) gas
exchange rate is 0.01 moles/sec (Refs. 3 and 5).

Expansion of the PWR exit gas into the containment is depicted
in Fig. 4. Assuming non-separation of the gases, the PWR egress gas
composition approaches the overall containment composition along the
SLOW STEAM LEAK
< 7 X 10⁻² LB/SEC FROM STEAM GENERATOR TUBE

STEAM ADDITION
≤ 1000 PPMV AT 30 MIN

ORDERLY PLANT SHUTDOWN BY OPERATOR AT 30 MIN

STEAM GENERATOR OFFSET TUBE RUPTURE AT FULL POWER

RAPID MASS AND PRESSURE ADDITION TO CERV

REACTOR TRIP AT 98 sec

EVENT SEQUENCES

RESULTS

<table>
<thead>
<tr>
<th>percent pressure rise to venting nominal rupture disc set point</th>
<th>2</th>
<th>54</th>
<th>84</th>
<th>100</th>
<th>100</th>
<th>100</th>
</tr>
</thead>
<tbody>
<tr>
<td>maximum core block burn-off (%)</td>
<td>5.6 x 10⁻³</td>
<td>5.5 x 10⁻³</td>
<td>0.53</td>
<td>0.028</td>
<td>0.28</td>
<td>1.2</td>
</tr>
<tr>
<td>maximum core support post burn-off (%)</td>
<td>2.9 x 10⁻³</td>
<td>5.5 x 10⁻³</td>
<td>0.31</td>
<td>0.025</td>
<td>0.25</td>
<td>0.79</td>
</tr>
<tr>
<td>containment pressure rise (PSI)</td>
<td>0</td>
<td>0</td>
<td>5.8</td>
<td>11.6</td>
<td>5.7</td>
<td>5.6</td>
</tr>
<tr>
<td>activity release to containment (Ci)</td>
<td>3.2 x 10⁶</td>
<td>2.7 x 10⁵</td>
<td>4.3 x 10⁹</td>
<td>2.8 x 10⁹</td>
<td></td>
<td></td>
</tr>
<tr>
<td>containment flammability safety factor δ</td>
<td>1020</td>
<td>210</td>
<td>630</td>
<td>1510</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

‡ Ratio of H₂+CO needed for overall containment flammability to actual H₂+CO accumulated.

***Venting low-tolerance relief pressure assumed.

Fig. 1. Summary of sequences and salient consequence parameters for design basis steam ingress events

8-1
FIG. 3. CALCULATED CORE
GRAPHITE BURNOFF DURING
STEAM INGRESS INCIDENTS
AS A FUNCTION OF LEAK RATE.

OFUEL ELEMENT
□ SUPPORT POSTS

MAXIMUM CORE BURN-OFF (10^{-3} %)

NORMAL CONDITIONS

UNLIMITED OPERATION

OPERATIONAL TIME
LIMITED BY TECH SPECS

ALARM

UPSET CONDITIONS

OFFSET TUBE RUPTURE

EMERGENCY CONDITIONS

STEAM INLEAKAGE RATE (LB/SEC)

EQUILIBRIUM OXIDANT LEVEL (PPMV)

ORDERLY PLANT SHUTDOWN AT 30 MIN

PPS TRIP AT 1000 PPMV

REACTOR TRIP BY PPS AT 98 SEC
<table>
<thead>
<tr>
<th>System</th>
<th>Design Features</th>
<th>Performance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Moisture Monitor System</td>
<td>Redundant rakes (two out three logic) and detection instruments for each loop.</td>
<td>response time = 22 sec (expected) or 98 sec (max. allowable).</td>
</tr>
<tr>
<td>SG Isolation and Dump System</td>
<td>Redundant logic, feedwater valves, check valves, and dump valves.</td>
<td>2 sec valve stroke times.</td>
</tr>
<tr>
<td>PCRV safety relief valve system</td>
<td>Two independent trains each with 27 in² flow area</td>
<td>45 moles/sec outflow for pure helium and 17 moles/sec for pure steam at nominal relief pressure</td>
</tr>
<tr>
<td>Core Auxiliary Cooling System</td>
<td>1% rated flow capacity per aux. loop</td>
<td>5 min delay on startup (max)</td>
</tr>
<tr>
<td>Case and Parameter Varied</td>
<td>Noble Gas Release (10^5 Ci)</td>
<td>Maximum Core Block and Support Post Burn-off (10^-3 %)</td>
</tr>
<tr>
<td>--------------------------</td>
<td>-----------------------------</td>
<td>-----------------------------------------------------</td>
</tr>
<tr>
<td>Total Steam Ingress (lb)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Steam Reacted (lb)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Core Graphite Oxidized</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Steam PCRV (psig)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Peak Core (psig)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- Reference case
- Superheater tube rupture
- PPS response at 22 sec.
- Initial cold leg temperature = 600°F
- 20%/10% afterheat increase
- 20%/10% afterheat decrease
- Factor of 10 increase in reaction rate
- Factor of 10 decrease in reaction rate
- 20% decrease in helium flow rate

Design basis steam leak, PPS response at 98 sec, initial cold leg temperature of 640°F, nominal afterheat and reaction rates. + 20% change from nominal afterheat up to 1000 sec; 10% change after 1000 sec.

(*) Value at 600 sec (some additional release occurs at later times).
10. REFERENCES


2. In the case of a failure of the primary circuit, the energy released into the containment is lower by 1 to 2 orders of magnitude than with similar procedures of LWR.

3. By using a prestressed concrete pressure-vessel (PCRV) of integrated or non-integrated design with additional safeguards, the burst of components subjected to compression can be excluded, and thereby also extreme mechanical impacts on the structure.

4. The thermal stability and high heat capacity of the core delay the course of events after hypothetical accidents and lead to a delayed release of a major portion of the activity inventory (Fig. 1).

![Fig. 1: Delayed Time-Dependent Activity Release](image)

5. The production of explosive gases within a containment filled with air cannot be ruled out on principle. They can come from gaseous products of a graphite corrosion in the primary circuit or from process-gas leakage in case of NPH. They can be controlled according to the present state of knowledge. For serious reactor accidents, however, the formation of explosive mixtures can also not be excluded for LWR.
Fig. 3 shows the influence of technical measures inside the containment system on the activity decrease of representative fission products after failure of the primary circuit. If the primary circuit (A) is not completely surrounded by a structural unit, a spontaneous activity release at ground-level under the worst propagation conditions has to be assumed. The atmospheric dilution can only be taken as a reducing factor. Using a not gastight containment (B) emission over its surface - average emission height 30 m - leads to a greater atmospheric dilution. Moreover, the plate-out of iodine, cesium and strontium on the containment wall and the good retention properties of the concrete - 90 % on the whole - effect another reduction. In case of a gas-tight containment (C), stack-release (100 m) and average propagation conditions can be considered. The efficiency of filters (\( \eta_J = 99 \% \), \( \eta_{\text{CH}_3J} = 90 \% \), \( \eta_{\text{Sr},\text{Cs}} = 99,9 \% \)) and depositions inside the structure (\( \alpha_{\text{Cs},\text{Sr},J} = 60\% \), \( \alpha_{\text{CH}_3,J} = 20 \% \)) are of great importance. They decrease significantly the emission values of the non-noble gaseous materials. The environmental exposure caused by leakage is decisive for cesium and strontium. Generally have to be pointed out the activity decrease by radioactive decay of short-lived fission products, as can already be recognized in the presentation for Xe-133,
If one considers the influence of reasons and consequences of accidents by technical measures as the strongest criterion, the free internal volume of the gas-tight containment is of safety relevance. During overflow after depressurization accidents, pressure and temperature are determined by the free volume. After this short-time process, pressure and temperature are reduced by heat transfer. Fig. 4 shows the curve of pressure and temperature under the assumption of isothermal expansion. The heat transfer properties of the coolant gas - a total loss of coolant can be excluded - are in case of forced convection proportional to the height of accidental pressure; thus a high pressure facilitates after-heat removal. With the hypothetical loss of emergency cooling, the heat-removal because of natural convection increases by a higher pressure /3/ after exceeding a threshold value which depends on the composition of the after-accident atmosphere. The mentioned safety advantage of a small containment is confronted by higher requirements for the gas-tightness of the system, in order to avoid higher leakage rates because of high pressure despite of a smaller surface. The safety importance of a smaller amount of oxygen within the containment has not yet been clarified. Lower oxidation rates can be expected in principle inside the core after an air-inflow.

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Fig. 4: Calculated Containment Atmosphere Pressure and Temperature Response Following a DBDA /1/
Fig. 5: Containment-Systems of Different Reactor Types

Safety reserves of the construction, one can make use of the results of PCEV-technology. A sudden rupture of the containment can be excluded. The use of steel concrete, which is possible in principle, is confronted by technical, economical, and safety disadvantages.

The efficiency of the entire containment system requires reliable functioning of the isolation system. Intensive research and developmental work has been done to achieve a high degree of reliability.

Additional studies have to be performed to quantify the residual risk of HTR and the influence of the containment, e.g. feedback of thermodynamic procedures inside the containment on the accident process and possible failures of the containment structure. The effects of most serious nuclear accidents involving a loss of all heat sinks are dependent upon the lay-out of the nuclear energy generation system with power density and power size as the most important factors. Accident process and accident consequences, however, are significantly influenced by safety properties of the containment. In quality have to be pointed out:

1. The enormous heat capacity of the concrete vessel \( (c\cdot M \approx 2 \cdot 10^{10} \text{ J/}^\circ\text{C}) \) and its installations;
2. The large surface \( (F \approx 15,000 \text{ m}^2) \) for retention of radioactive materials, and
These advantages of safety of an underground-sited HTR improve licenceability of sites near to consumers. Because of the topographical conditions in Germany, the "cut and fill concept" in soil is favoured, as is shown schematically in Fig. 6. The ceiling of the containment is protected by an impermeable layer of 2 m in thickness, and a coverage of 10 m in height. For relatively favourable underground conditions the slurry-trenches technique can be used. The slurry trenches do not only serve for protection of pit during construction time, but can also be integrated into the construction of the annular space. For unfavourable underground conditions the application of the freezing technique is recommended. The space between pit wall and the surface of the building can be filled with solid impermeable material of 2 m in thickness. The effects on costs and construction time are influenced by many factors. Preliminary estimations are not satisfactory and need to be stated more precisely. Preparation of the pit causes the strongest increase in cost and construction time - this potential disadvantage does not exist at all, if available underground spaces were used.

Fig. 7: Rhine Open-Cast Lignite Mining Area
NEED FOR A CONVENTIONAL CONTAINMENT ON HTGRs

by

R. K. Deremer and V. Joksimovic

This is a preprint of a paper to be presented at the Specialist Meeting on High-Temperature Gas-Cooled Reactor Safety, May 13-15, 1975, Petten, Netherlands, and to be printed in the Proceedings.

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May 6, 1975
1. SUMMARY

For high-temperature gas-cooled reactors (HTGRs) currently undergoing licensing review in the United States, the pre-stressed concrete reactor vessel (PCRV) is enclosed by a conventional, high pressure, low leakage containment building. The large HTGR possesses the same features as the Fort St. Vrain reactor, which was licensed without a conventional containment. The purpose of this paper is to illustrate that from the standpoint of acceptable off-site dose consequences following hypothetical design basis events, the conventional containment is not needed for most sites. For those sites located very near to highly populated urban areas, a single state-of-the-art conventional containment might be necessary.
Consistent with the philosophy which has been employed in Great Britain and Germany, GA has recently proposed\(^1\) to the Nuclear Regulatory Commission (NRC) that for the 3800 MW(t) plant the design basis depressurization accident (DBDA) for the integral PCRV arrangement be defined as the rupture of the largest pipe connected to the primary coolant system or a spurious failure of a relief system train. Using this definition, it can be shown that the radiological consequences of a postulated design basis depressurization accident for a large HTGR do not exceed acceptable guidelines for most sites, even if the FSV concept of containment is utilized. It should be noted, however, that while the aspects of a conventional containment are not required to mitigate the radiological effects of a DBDA, the building enclosing the PCRV would have to withstand all postulated natural phenomena and to act as a barrier to all potential missiles including aircraft if the probability of a crash is not deemed negligible.
4. RADIOLOGICAL CONSEQUENCES OF DBDA AND MHFPR*

The off-site doses following a DBDA for a 3000 MW(t) HTGR are shown in Figure 2 through 5 for a variety of containment schemes, coolant circuit activity levels and blowdown areas. For those cases identified as "vented containment," it was conservatively assumed that the entire inventory of circulating activity and the fraction of plateout activity estimated to be lifted-off for the selected blowdown area are instantaneously released to the environment through a vent system without credit for any filtration, plateout, fallout, or decay in the reactor building. It is expected that a large fraction of the radioactive iodine will plateout in the blowdown circuit and on the walls of the reactor building. In addition, calculations indicate that a substantial fraction of helium (and hence gasborne activity) remains in the building after venting ceases. Those fission products which remain in the building after venting will be subjected to radioactive decay and atmospheric clean-up before leaking to the environment.

For comparison, the results for a conventional high pressure, low-leakage (0.1%/day for the first day and 0.05%/day thereafter) and an unlined conventional containment (assumed constant leakage of 2.5%/day) are also included in the figures. For the conventional containments, it was conservatively assumed that there was no cleanup of the containment atmosphere at any time during the 30 day period of interest. Atmospheric dilution factor used in the analyses were taken from NRC Regulatory Guide 1.3 using building wake factors where appropriate.

* Maximum Hypothetical Fission Product Release
does on the whole body dose. This effect is caused by the dependency of iodine lift-off on the maximum shear forces which are attained during the blowdown and these in turn are strongly dependent on the blowdown area. However, even for the worst case assumptions and a vented containment the thyroid doses at distances as short as 500 meters meet guideline requirements. If expected activity levels are used in the calculations, the dose levels associated with the vented containment are significantly lower than guidelines and with conventional containment they are, for all practical purposes, insignificant.

In order to demonstrate the acceptability of a proposed power reactor site in the United States it is necessary to hypothesize a fission product release (MHFPR) from the reactor which results in potential hazards not exceeded by those from any accident considered credible. For light water reactors, site and containment qualifications are traditionally based on the non-mechanistic assumptions which were first identified in TID-14844(4). This so-called TID-release assumes an instantaneous release of a large fraction of the core fission products to the containment. This release is assumed to approximate an instantaneous, hypothetical meltdown of the core.

Because of its inherent features, it is not physically possible to experience a release from an HTGR that is analogous to the instantaneous "TID-release". For an HTGR, the event is characterized by a time-dependent release of fission products to the containment and is discussed in detail in another paper(5) presented at this conference.

The curves in Figures 6 and 7 depict the 0-30 day whole body and thyroid doses following an MHFPR for both the expected low leakage rates from a conventional containment building and the relatively high leakage rate (2.5%/day) which could be expected for an unlined or vented containment structure. Because of the time dependent release of fission products
5. CONCLUSIONS

Based on the off-site doses presented in the figures, it is apparent that for many sites, a conventional containment is not required by an HTGR power plant to meet US Regulatory Guidelines. Furthermore, a single state-of-the-art conventional containment restricts off-site doses to levels which are so far below the guidelines that they are insignificant and hence double containment is not required for those sites near urban population centers.

Regardless of the type of containment, the structure must provide adequate protection against potential missiles and natural phenomena.

It should be noted that the conclusions reached in this paper were formulated in the context of current US regulatory practice. To speculate on future changes in this practice and to form conclusions based on these changes was considered to go beyond the intended scope of this paper.
Fig. 1. Large HTGR general arrangement
Fig. 3. 0-30 day whole body dose following a DBDA using Regulatory Guide 1.3 meteorology (ground level release)
Fig. 5. 0-30 day thyroid inhalation dose following a DBDA using Regulatory Guide 1.3 meteorology (ground level release)
Fig. 7. 0-30 day thyroid inhalation dose for a MFPR using Regulatory Guide 1.3 meteorology (ground level release)

Dose (REMS)

Distance (Meters)

Unlined or Vented Containment

Low-Leakage Containment

Conventional Low-Leakage Containment

10 CFR 20 GUIDELINE FOR THYROID DOSE (300 REM)
8. REFERENCES


TEMPERATURE BEHAVIOUR OF HIGH-TEMPERATURE REACTORS
IN THE CASE OF
HYPOTHETICAL ACCIDENTS

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1 INTRODUCTION AND GOAL

Heat removal after reactor shut-down is performed today in all reactor types by the main cooling cycle or by special cycles of after-heat removal. These systems have a multiply redundant lay-out. Hence, a total fall-out is regarded as a hypothetical accident and excluded therefore.

Despite of that it should be studied here the behaviour of a high-temperature reactor for an extraordinary situation as described before. As has been shown by preliminary estimations, a destruction of fuel elements is not possible with an appropriate lay-out. Following to this, more precise two-dimensional calculations have been performed. Also we will investigate if and to which extent a natural convection of the coolant gas will arise because of the temperature-dependent differences in it's density, and of which effects the cooling of core installations on reactor temperature behaviour is.

2 BRIEF DESCRIPTION OF THE CONSIDERED REACTOR TYPES

For the purpose of a preliminary analysis are considered the process heat reactors PR 500 and PR 3000 developed in the Nuclear Research Center Juelich. They have a thermal power of 500 MW resp. 3000 MW and an average power density of 5 MW/m$^3$ /1,2/. They are helium-cooled high-temperature reactors with spherical fuel elements, and once-through feed system. The construction of the one type is similar to that of the other. Fig. 1 shows the construction scheme of PR 500.

The pebble-bed is surrounded at first by a graphite reflector of 50 cm in thickness, followed by a carbon stone isolation of 1 m in thickness. The total core is included by a pre-stressed concrete pressure vessel, the inside walls of which
are covered by the metallic liner. On the concrete side the liner is cooled by water to 50°C. The heat produced in the reactor core is released to steam generators via primary circuits in which helium is circulated under a pressure of 40 bar. The steam generators (3 for PR 500 and 12 for PR 3000) and their blowers are situated outside the pressure vessel and are connected to the reactor core by coaxial tubes. The whole reactor is arranged in a safety vessel which is filled by nitrogen to avoid air flowing into the reactor core in the case of a leakage of the primary system. In the case of a rupture of the primary circuit, the safety vessel could admit the arising mixed pressure (appr. 3 bar).

3 ASSUMED ACCIDENTS

It is assumed a simultaneous fall-out of all blowers of the primary circuit. The follow of this is an immediate quick reactor shut-down. An after-heat removal by forced circulation cooling does not occur, so that core and installations heat themselves. The arising temporal temperature distribution is calculated for the whole reactor in two-dimensional cylinder geometry.

4 HEAT CONDUCTION IN THE PEBBLE-BED

The heat conduction in the pebble-bed is of particular
importance for the reactor temperature behaviour. On the one hand, neglecting the convective heat transfer, the transport mechanism is based on the heat conduction in the fuel balls and the surrounding gas, and, on the other hand, on the radiation exchange between the fuel balls. Fig. 2 shows the effective coefficient of thermal conductivity for the pebble-bed, calculated by different authors (LEYERS resp. ZEHNER and SCHLÜNDEL /3,4/). $\lambda_K$ is here the heat conductivity of the graphite matrix, $\lambda_L$ the portion of pure heat conduction. As can be seen from this figure, the heat conduction in the pebble-bed is strong temperature-dependent because of the high portion of radiation.

The lower effective coefficient of thermal conductivity of ZEHNER and SCHLÜNDEL is basically used for the following temperature calculations, so that pessimistic values result for temperatures in the center of the pebble-bed.

5 RESULTS
5.1 Stagnant Cooling Gas

At first the cooling gas is assumed to stagnate in the reactor, and after-heat transported by only heat conduction and heat radiation to the installings. In addition to the blowers, another fall-out of the liner cooling will occur. This extraordinary case would lead to the highest possible temperature increase of the reactor.

Fig. 3 shows for PR 500 the temperature curve, plotted over the radius, at the hottest place of the pebble-bed at differ-
ent times. The pebble-bed achieves a maximum value of 2300°C after 34 hours. Then temperature is slowly compensated with simultaneously heating the prestressed concrete pressure vessel.

The corresponding axial temperature curves in the center of the reactor are shown in Fig. 4. In consequence of the once-through feed system, the high after-heat power density in the upper part of the pebble-bed causes a strong temperature increase in narrow limited zones.

Fig. 5 shows the radial temperature profile calculated for PR 3000. The maximum value of 3200°C is achieved after 300 hours only. Compared to the after-heat to be admitted, the available heat capacities represented by the installings are lower by the factor 2.7
than those of PR 500; consequently PR 3000 heats itself much stronger.

Fig. 6 finally shows the axial temperature curve in pebble bed and installings of PR 3000. After 1152 hours corresponding to 48 days, high temperature gradients still arise in concrete only, which is a bad heat conductor.

A well-operating liner cooling is of nearly no influence on the temperature of the pebble-bed. This is due to the insulating effect of carbon walls. The reactor core up to the reflector/carbon boundary shows the same temperatures as without liner cooling. A strong temperature decrease can be found in carbon only to the given liner temperature.

In contrary to that, the choice of the average power density is of high importance for the instationary temperature behaviour. Fig. 7 shows for PR 3000 the temporal temperature curve at the hottest place of the pebble-bed for different power densities. The corresponding temperature curve for PR 500 is given for comparison.

A decrease of the average power density to 3 MW/m³ in PR 3000 leads to 2700°C after 400 hours, an increase to 10 MW/m³ leads to approximately 4000°C after 96 hours.
5.2 Natural Convection in the Primary System

If one starts from a fall-out of the circulating blowers only, with primary circuit, steam generators and coaxial conducts remaining undamaged, a natural convection will occur due to the temperature differences. Within one minute after the fall-out of the blowers, the cooling gas flows in reverse direction. Helium flows upwards in the hot core, while flowing downwards in the steam generators because of cooling down.

If the complete cooling gas pressure of 40 bar is remained, nearly all the after-heat can be released from the reactor core. Fig. 8 shows for PR 500 the radial temperature profile at the hottest place. The fuel element temperatures increase to 1200°C after 100 minutes only, and suddenly decrease as a follow of convection flow cooling. The flat radial temperature profile in the pebble-bed is remained because of the equal heat removal in the total cross section.

The axial temperature curves in Fig. 9 show the shifting of heat from the lower into the upper region of the reactor core because of gas flow. The hottest place is densely below the surface of the pebble-bed.

During reverse flow all components conducting cold gas under normal operation will transport hot gas then. Thus the top reflector
heats itself after 3 hours to 800°C, the external coaxial tubes to 700°C also after 3 hours.

If one assumes a leakage in the primary system arising simultaneously with the fall-out of the forced circulation cooling, e.g. rupture of a coaxial conduct, the helium flows into the safety vessel within a short time and is mixed with nitrogen. The gas mixture enters the primary system through the leakage, where it is cooled by the remaining and not separated steam generators. However, since gas-throughput and thereby heat-removal from the reactor core decrease by decreasing pressure, only a small portion of after-heat power is released.

At a pressure of the helium/nitrogen mixture of 3 bar, the maximum temperatures remain below 2100°C in PR 500, below 2900°C in PR 3000. The top reflector in this case heats itself especially quickly because the convection is indeed strong enough to conduct heat upwards from the pebble-bed, but is not sufficient to limit decisively the temperature increase in the pebble-bed. With the system pressure of 1 bar, the temperatures are hardly below those of stagnant helium, since after-heat removal still achieves 1% of after-heat power only.

5.3 Circulation Flow in the Pebble-Bed

If coaxial conducts are closed after the accident had occurred, a circulation flow arises because of the radial temperature profile in the pebble-bed. Gas flows upwards in a relatively broad hot inner zone and flows downwards again in the colder outer zone. Thereby is achieved a much better thermal conduction to the side installings than with stagnant gas. Preliminary calculations which have not yet been finished give reason to expect that this effect will cause a limitation in temperature increase in the
pebble-bed to 1600°C with a helium pressure of 40 bar and an intact liner cooling, if core installations are loaded correspondingly higher.

5.4 Cooling of Core Installations

In all cases regarded until here, reactor components like top reflector and metallic cooling gas conducts were heated thus they are endangered. A counter measurement could be to cool these components directly after the accident had occurred. Without treating in more detail the associated constructive and safety-technical aspects, it should be considered here the effects on reactor temperature behaviour only. Therefore the cooled installations are assumed to be kept on a temperature of 100°C and that cooling gas stagnates.

A cooling at the external boundary of the side-reflector decreases the core center temperature in PR 500 by 100 K, in PR 3000 by only 60 K. This low temperature decrease can be explained by the mentioned strong temperature dependence of the effective heat conductivity of the pebble bed. Due to the cooling, a relatively cold zone of 600 to 800°C arises in the region close to the wall of the pebble-bed, which mostly insulates the hot inner zone.

Another alternative exists by cooling the top reflector. With a total thickness of 1 m, the reflector efficiency is limited to the first 30 cm; thus 70 cm are available to admit cooling tubes.

Axial temperature distributions for PR 500 result under this assumption as plotted in Fig. 10. Maximum fuel element temperatures are decreased by top reflector cooling by only about 100 K. The temperature curve drawn in dotted lines results without top reflector cooling. A comparison with the undotted one shows that temperatures at the surface of the pebble-bed can indeed be limited effectively, but the bad heat conductivity in the pebble-bed impedes a continuous temperature decrease up to the hot place.
Only with higher temperature gradients the heat transport from the hot place of the pebble-bed to its surface is that good that the heat removal at the top reflector is as large as that of after-heat production. This equilibrium is achieved in PR 3000 with a maximum fuel element temperature of 2560°C after 78 hours, as Fig. 11 shows. The hot place is shifted more downwards by time because of the top reflector cooling. If natural convection will occur, the cooled top reflector, however, could represent an effective heat sink by which even with large reactors a considerable portion of after-heat could be released. This possibility is presently studied in more detail.

6 CONCLUSION

As has been shown theoretically, the maximum core temperatures do not exceed 2300°C in PR 500, and not 3200°C in PR 3000, not even under most unfavourable assumptions.

A destruction of fuel elements can thereby be excluded in any
case, for the graphite structure material sublimes at 3650°C only. Core installations, however, are loaded far beyond the permissible limit. The top reflector, for example, hung at steel bolts, is endangered to subsidence.

This does not even change by considering the arising natural convection. It is indeed strong enough with intact primary systems and full cooling-gas pressure to prevent an important excess of operation temperatures in the fuel elements. Because of the reverse flow of cooling gas, however, all components transporting cold gas under normal operation are leading hot gas and are heated unpermissibly high. Reactor emergency cooling by natural convection without additional measurements is therefore only a theoretical alternative.

The cooling of core installations prevents their superheating; under the assumption of stagnant cooling gas, however, it may not decisively reduce the fuel element temperature. In PR 3000 only the maximum temperature can be limited by a top reflector cooling up to below 2600°C.

The smaller PR 500 shows in general a more favourable temperature behaviour than PR 3000. They have in common the temperature increase in pebble-bed and installings due to the comparatively low power density and the high heat capacity running that slow that a few hours remain for counter measurements after the accident had occurred.

By appropriate construction it seems feasible to develop a high-temperature reactor system by which a danger for the environment is excluded even under the hypothetical assumption of a fall-out of all blowers and of the after-heat removal systems.
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CONSEQUENCES OF UNRESTRICTED CORE HEAT-UP EVENT

by

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CONTENTS

1. SUMMARY 1-1
2. INTRODUCTION 2-1
3. MHFPR ANALYSIS 3-1
   3.1 OBJECTIVE 3-1
   3.2 BASES 3-3
   3.3 EVENT PROGRESSION 3-5
   3.4 RESULTS 3-7
4. UNRESTRICTED CORE HEATUP 4-1
   4.1 OBJECTIVE 4-1
   4.2 SEQUENCE DIAGRAM 4-1
   4.3 RISK ASSESSMENT 4-2
5. CONCLUSIONS 5-1
6. FIGURES 6-1
7. REFERENCES 7-1
1. SUMMARY

The Maximum Hypothetical Fission Product Release, which is postulated in a non-mechanistic manner for the purposes of evaluating ultimate safety of a proposed HTGR site, illustrates safety advantages unique to the HTGR concept. The bounding event for siting is shown to consist of a time-dependent release which offers a wide choice in containment design alternatives. The mechanistic equivalent of the MHFPR is also considered to assess the risk associated with an HTGR unrestricted core heatup event and to illustrate large margins of safety.
2. INTRODUCTION

In order to evaluate the ultimate safety of a proposed nuclear power plant site in the U.S., a fission product release from the core has to be postulated "that would result in potential hazards not exceeded by those from any accident considered credible" (Ref. 1). For the high-temperature gas-cooled reactor (HTGR) this "Maximum Hypothetical Fission Product Release" (Ref. 2, 3) bounds postulateable HTGR accidents. The MHFPR is intended to be analogous to the TID-14844 interpretation of the loss of coolant accident (LOCA) used to evaluate the engineered safety features and the sites for light water reactors (LWR). An event in the HTGR that could conceivably lead to a source term of the magnitude postulated for an LWR is a hypothetical loss of all forced circulation including loss of the core auxiliary cooling system (Ref. 4). Section 3 of this paper discusses the objective, bases, analysis, and results associated with this hypothetical event.

In addition to the analysis of the MHFPR for site evaluation purposes, there is motivation to consider permanent loss of forced circulation from a mechanistic viewpoint in order to assess risk to the public arising from this event. Risk assessment consists of evaluation of probability of occurrence as well as consequences to the public. This subject is discussed in Section 4. Conclusions for the MHFPR as utilized in the U.S. for siting and for the probabilistic assessment of the unrestricted core heatup risk, discussed to illustrate margins of safety, are presented in Section 5.
3. MHFPR ANALYSIS

3.1 OBJECTIVE

The guidelines of 10CFR100 (Ref. 1) require for evaluation of a proposed site that, given the conservative design value of containment leak rate and the conservative meteorological conditions pertinent to the proposed site, the applicant should determine the total whole body and thyroid doses such that:

1. An individual located at any point at the exclusion area boundary "for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure."

2. An individual located at any point at the low population zone boundary "who is exposed to the radioactive cloud resulting from the postulated release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure."

Site qualification and evaluation of the effectiveness of containment systems for light water reactors are traditionally assessed by the non-mechanistic assumption first postulated in TID-14844 (Ref. 5) that an instantaneous gross release of fission products to the contain-
ment follow the design basis loss of coolant accident. Subsequently, the assumptions to be used for evaluating the potential radiological consequences of a loss of coolant accident for boiling water and pressurized water reactors were stated in Regulatory Guides 1.3 and 1.4, respectively.

Due to the fundamental inherent differences in reactor concepts it is not physically possible to experience a release in the HTGR that is analogous to the instantaneous release postulated in TID-14844. However, in order to preserve the precedent of the conservative posture toward the assessment of nuclear power plant safety in the U.S., HTGRs are postulated to undergo a MHFPR to the containment where the time-dependent release is limited only by fundamental physical laws and no credit is taken for engineered safety features in limiting the release from the core.

The specific objectives of this approach are:

(a) to assure a considerable margin of safety in the containment functional integrity throughout the duration of all postulated accidents.

(b) to assure preservation of fission product removal systems.

(c) to assure an appropriate margin of safety in relation to site related characteristics.

(d) to assure a margin of safety in design and analyses for uncertainties in the data base and analytical techniques.
3.2 BASES

The MHFPR is postulated to occur as a result of an unrestricted core heatup process resulting in failure of fuel particle coatings and transport of fission products from the fuel to the containment atmosphere without restraint by the primary coolant pressure boundary. The MHFPR is considered only as a non-mechanistic source term; it is not the result of any credible sequence of events. Therefore, it is necessary to precisely define the basis for the release in order to establish that it exceeds any fission product release resulting from a credible sequence of events. The bases for the MHFPR can be defined in terms of initial conditions at the onset of the MHFPR and assumptions governing the state of the plant and the transport processes during the course of event.

The initial conditions which establish the starting points of each of the variables affecting transient core temperatures and fission product transport processes are:

1. Initial spatial fuel, moderator, reflector, and primary system component temperatures are those which correspond to the condition immediately following reactor trip from 105% power.

2. The inventory of fission products in the fuel corresponds to the value calculated for end of cycle shutdown of the equilibrium core operating at full power at 80% plant load factor. The quarters of the fuel loading are one, two, three, and four years old.

3. The core afterheat rate is based on four year operation of the entire core at 105% power at 80% load factor.
4. The primary system pressure is in equilibrium with the containment pressure.

5. The DBDA source terms are postulated to be released at time zero. (For the 3800 MW(t) plant, this additional source will not be assumed (Ref. 6)).

The basic assumptions constrain the state of the plant and the evaluation of the plant variables during the release process. There are two types of basic assumptions which are relevant to the MHFPR, mechanistic assumptions and non-mechanistic assumptions.

The mechanistic assumptions for the MHFPR include fundamental physical laws, empirical correlations, and primary system properties relevant to the release process. Conservative representation is taken for the following principles and properties:

1. Core afterheat generation in the HTGR core.
2. Aggregate core thermal properties
3. Heat conduction and radiative heat transfer in the core.
4. Fuel failure correlation with time, temperature, and particle type.
5. Time-dependent release from intact and failed fuel particles.
7. Correlation of evaporation and condensation of chemical elements with temperature and vapor pressure.
8. Free convection transport of chemical elements in core coolant channels.

The non-mechanistic assumptions include specifications of the state of plant components and structures, irrespective of any cause, and specification of physical properties of principles which override mechanistic representations. The following non-mechanistic assumptions
are applied to the MHFPR:

1. No heat is removed from the core by forced or natural convection of primary coolant.

2. No delay in transport to the containment of gas-borne fission products due to retention within the primary coolant pressure boundary is assumed.

3. The containment remains intact and retains its qualified leakage characteristics and the containment cleanup system becomes operative.

4. The accumulated release of gas-borne fission products to the containment at any time is limited to 100% of the noble gases, 25% of the halogens, and 1% of the solids.

3.3 EVENT PROGRESSION

The initial conditions and assumptions specified above define the event which is characterized by a time-dependent release of fission products from the fuel to the containment. The driving potential for the gradual transport of the fission products is the slow heatup of the core.

Following reactor trip, the temperature of the fuel decreases as it comes into equilibrium with the local moderator temperature. Because of the assumption that no core heat is removed by convection of primary coolant, the fuel and moderator gradually begin to heat up due to fission product decay. Heat transfer to the PCRV liner cooling system results in heat removal. However, this does not occur until the reflector surface temperatures are sufficiently high to get appreciable radiation heat transfer.
As core temperatures increase, the fuel particle coatings begin to fail due to temperature and pressure induced stresses. The fission products entrained within the fuel kernels then begin to diffuse through the failed coating into the matrix bedding of the fuel rod. From the matrix bedding the nuclides migrate or diffuse across the bedding/graphite webbing interface and through the graphite webbing toward the graphite/coolant channel surface.

When fission products reach the graphite/coolant channel boundary, evaporation into the gas phase occurs and mass transport across the surface boundary layer takes place due to gas phase diffusion and vapor transport. Small convection currents upward through the core hot regions carry the fission products which reach the gas phase into cooler core regions where condensation occurs, depositing the material on other graphite surfaces. Some fission products escape completely the core and reflectors and are available for deposition on cooler primary system surfaces such as the PCRV thermal barrier. Iodine, in particular, exhibits a strong affinity for metallic surfaces.

Gas-borne fission products which are not plated out in the PCRV are transported to the containment. Particulates and halogens in the containment are depleted by the containment recirculation cleanup system and also by plateout on containment surfaces. However, noble gases and undepleted gas-borne halogens and particulates are available for release to the atmosphere via containment leakage.

3.4 SIMULATION

The time-dependent evaluation of temperatures throughout the core and interior regions of the PCRV core cavity is performed using the CORCON (Ref. 7) computer program. CORCON utilizes finite difference integration technique to solve the transient heat transfer equations
in two-dimensional geometry. Several models are contained in the code to simulate the heat transfer processes, the heat generation and migration of heat generating radionuclides, and the heat exchange across open core plenums.

The time-dependent core temperature distributions generated by the CORCON code are used as input to the SORS (Ref. 8) fission release and transport programs which calculate the transient distribution of radionuclides in the fuel particles and fuel element graphite during the course of the event. The SORS codes solve a coupled set of ordinary differential equations that describe the decay and transport behavior of the 64 chains of the 234 nuclides. The fuel performance model (Ref. 9) within the SORS codes utilizes the transient temperature distributions to determine the fractional amounts of intact and failed coatings at any time and location (fuel age). The fractional fission product release from both intact and failed coatings is shown in Fig. 1 as a function of temperature. These fractional release characteristics are currently under revision to reflect a statistical analysis of all available data.

Thus, the combined use of CORCON and SORS programs results in a prediction of a time-dependent release of reactor core fission products which properly accounts for the core heat capacity and thermal response, the transport properties of fission products, and the release and holdup characteristics of HTGR fuel materials. Both CORCON and SORS have been verified by independent analyses (Ref. 10, 11) at Oak Ridge National Laboratory and Los Alamos Scientific Laboratory.

3.5 RESULTS

Temperature distributions throughout the active core and the reflector regions calculated by the CORCON code for a 3000 MW(t) HTGR (Ref. 12) are illustrated in schematic diagrams in Fig. 2. Isotherms
are shown in core axial cross sections at 2 and 22 hours. The isotherms are centered just above the axial midplane of the reactor core and spread upward with time more rapidly than downward. The maximum local temperature and the volume average active core temperature are presented and compared in Fig. 3.

Results of time-dependent fission product release for the 3000 MW(t) MHFPR case are presented in Fig. 4, which shows the release fraction of the total inventory existing at any given time accounting for decay. These release fractions represent the airborne activity in the containment. Each of the curves in Fig. 4 for a class of nuclides is indicative of the behavior of the highest released isotope in that class.

The results show that at two hours, temperatures in the active core reach a maximum of 2900° and an average of 2400°F. Corresponding fractions of initial activity released at two hours are less than 0.01% for noble gases, less than 0.001% for iodines, and essentially zero for particulates. Noble gas and iodine nuclides are calculated to be completely released from the core in approximately 40 hours. Particulate nuclide release is delayed in the graphite. Essentially all strontium activity is released from the fuel in 50 hours and from the graphite in 100 hours.

With breathing rates, dilution factors, leak rates and other necessary parameters of Ref. 13 in compliance with Reg. Guide assumptions, doses as a function of exclusion area boundary (EAB) and low population zone (LPZ) are shown in Fig. 5. As seen from the figure, doses are well below those allowed by 10CFR100, particularly at 2 hours for the EAB. These results are for a conventional low-leakage containment (other choices are discussed in another GA paper presented at this conference (Ref. 13)). The small fractions of the 10CFR100 guidelines even at distances of 800 meters or less can be reduced further by increasing the containment clean up rate.
4. UNRESTRICTED CORE HEATUP

4.1 OBJECTIVE

The objective of this section is to consider, mechanistically, an unrestricted core heatup in the HTGR, for the purposes of assessing the risk to the public, which is the product of the probability of the occurrence and the consequences.

The primary barrier to the release of HTGR core activity is the fuel particle coating. The main loop cooling system or the Core Auxiliary Cooling System (CACS) maintain core temperatures well below particle failure threshold. In order to experience an unrestricted core heatup event, a complete failure of both cooling systems for an extended period of time has to occur. Loss of offsite power (LOSP) followed by subsequent turbine trip has been studied since these two combined events may impose relatively frequent demands on the CACS, in contrast to other events of lower frequency of occurrence such as total loss of feedwater flow, and hence constitute a representative initiating event that may conceivably propagate to an unrestricted core heatup.

4.2 SEQUENCE DIAGRAM

For the initiating event of a complete loss of all offsite power, a sequence diagram may be constructed to assess possible paths that the course of events may follow. The expected, highest probability response to the LOSP establishes 25% flow in the secondary coolant system, operates the main turbine generator with house load, and maintains core cooling with the main loops until off-site power is restored. The transient
is terminated by a return to full power. In addition to the expected sequence, a complete event sequence diagram not described in this paper, encompasses other less probable sequences; several lead to a loss of forced circulation. Of the paths in the event sequence diagram that lead to an unrestricted core heatup and a subsequent core activity release, the enveloping (highest risk) sequence is shown in Fig. 6. This sequence consists of a loss of all offsite power, turbine trip, and no subsequent core auxiliary cooling. This sequence envelopes all others in that its risk bounds that determined from the probabilities and consequences of other sequences.

4.3 RISK ASSESSMENT

The LOFC sequence in Fig. 6 has the highest risk to the public; that is, others may be postulated with greater probability consequence, but no other with a greater value of the product of probability and consequence. The probability for an entire event chain is the product of the individual probabilities. Preliminary estimates for the sequence shown in Fig. 6 are on the order of $10^{-7}$ per year. In this analysis, the basis of consequence was chosen as the whole body dose at a low population zone after 30 days. For a low population zone of 2400 meters, the dose in this case is approximately one rem. Therefore, the preliminary assessment of the risk to the public from an HTGR restricted core heatup is approximately $10^{-7}$ rem/year.

It should be noted that this whole body dose per annum is orders of magnitude lower than the $5 \times 10^{-3}$ rem/year allowed for normal operation at the EAB as prescribed in 10CFR50 Appendix I (Ref. 14). If the risk from an LOFC initiated from an LOFC initiated with a LOSP were assessed at the EAB rather than the LPZ the result would be approximately an order of magnitude greater, but still several orders of magnitude less than that in 10CFR50. In addition to comparison with 10CFR50, this whole body dose of $10^{-7}$ should also be compared with the site's natural background level.
5. CONCLUSIONS

In summary, in regards to the MHFPR, the time-dependent release of the HTGRs offers a considerable U.S. siting advantage over an instantaneous release that is consistently postulated for LWRs. The HTGR's graphite moderator, as well as the fuel itself, afford a large time delay before consequences of the event propagate. The graphite's heat capacity slows the temperature transient, while the pyrocarbon and silicon-carbide fuel kernels provide a fission product barrier with retention characteristics even above coating failure temperatures (Ref. 9). The time-dependent release has the effect of allowing a wide latitude in the choice of containment recirculation rates, containment leak rates, and site boundaries.

For the probabilistic assessment of the risk due to an unrestricted core heatup, the loss of offsite power was chosen as the initiating event. The sequence of events leading to a loss of forced circulation that had potentially the highest risk was examined. The risk in terms of whole body dose per annum for this enveloping sequence was found to be orders of magnitude less than specified by regulations for normal operation and background radiation level.
Fig. 1. Fission product release versus temperature for intact and failed fuel particles during accident conditions. (Ref. 8)
Fig. 2. CORCON Spatial Dependence of the Core Temperature at 2 and 12 hours.
6-2
Fig. 3. MHFPR 3000 MW(t) Temperature Transient

6-3
Fig. 4. Time-Dependent MHFPR for the Three Nuclide Groups
Fig. 5. 2 Hour and 30 Day Thyroid, Whole Body, and Bone Dose from the MHFPR as a Function of Distance from the Reactor Containment.
Fig. 6. Enveloping Sequence for Fission Product Release

PROB (THIS SEQUENCE) = 10^-7/ YEAR
30 DAY LPZ WHOLE BODY DOSE = 1 REM
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7-1
Safety Problems arising from Process Heat Applications of

High Temperature Gas-cooled Reactors

by

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SAFETY PROBLEMS ARISING FROM PROCESS HEAT APPLICATIONS OF HIGH TEMPERATURE GAS-COOLED REACTORS

INTRODUCTION

1. There is little doubt that as world resources of fossil fuels diminish, nuclear sources will play an ever increasing part in supplying the demand for energy. Initially, we are seeing a rapidly expanding use of nuclear reactors for electric energy generation via what is essentially fairly conventional steam turbine plant. Inevitably it will become desirable to apply nuclear energy more directly to a wide range of processes without the intermediate use of electricity.

2. The range of temperatures used for process purposes is large, for instance production of hot water in the range of about 80°-200°C is adequate for desalination and district heating applications whilst for cement and glass manufacture temperatures in excess of about 1400°C are necessary. At the lower end of the process heat temperature range a number of reactor systems are potentially capable of supplying the heat. Further up the range only gas-cooled reactors can meet the requirements higher still the choice is restricted to HTR, and for the top 600°C or so of the range quoted an advanced or developed HTR would be necessary. Present HTR technology is limited because of materials properties to gas outlet temperatures which make it unlikely that 800°C can be exceeded in the first process fluid.

3. In the long term, process heat reactors will be required to provide heat directly to processes which previously relied on combustion of fossil fuels for this purpose. In the shorter term there could be some applications of nuclear heat sources to the conversion of the most plentiful fossil fuel (coal) into a more flexible secondary fuel. The basic concept here would be that the energy required for the conversion would be provided by an HTR rather than by combustion of some of the coal.

4. The discussion in this paper is limited to those applications requiring temperatures available only from HTR or developed HTR and of necessity the primary coolant of such reactors is helium. The first fluid to receive reactor heat for use in the associated process is then, in normal operation, no nearer to the reactor than the reactor secondary circuit. This fluid (the first process fluid) may in some instances be heated in a tertiary circuit, the secondary (intermediate) circuit in such cases also being assumed to contain helium.

5. Examples of processes for which HTR's might provide the heat source include:

(a) production of aromatic hydrocarbons by catalytic reforming
(b) production of vinyl chloride
(c) production of ethylene
(d) production of olefines
(e) iron ore reduction
(f) ammonia production
(g) lignite gasification
(h) thermo-chemical production of hydrogen as a basic secondary chemical fuel to replace dwindling fossil fuel reserves.
6. It is to be noted that not all the individual processes mentioned require large quantities of energy, even on national scales. The energy consumption of a given manufacturing unit will in some cases be small compared to the heat output of, even a single HTR of practical economic size so that it would become desirable to combine energy provision, both thermal and electrical, to a complex of plants. A further important point arising from a survey of the most promising processes is that many of them require hydrogen manufacture at some stage. Hydrogen production to serve either one or a variety of plants, therefore becomes a prime candidate for early application of HTR's. The hydrogen could be distributed locally as a hot gas, or further afield as a cold gas. The possibility of distributing the hydrogen by pipeline leads directly to the concept of a non-integrated plant, i.e. the consuming complexes could be located some distance from the reactor, in fact they could be many kilometres away. Whilst no insuperable safety problems are foreseen in the use of reactors within an integrated chemical complex it is obvious that physical separation of the chemical plant and the reactor does reduce the possibility of interaction. The concept of a non-integrated system is not restricted to the production of hydrogen by nuclear process plant, but the hydrogen producer does serve a useful illustrative purpose; particularly as such applications are perhaps the most likely in the early stages of commercial nuclear process development. Interaction problems in integrated chemical plant will be considered more fully later in the paper.

7. An important characteristic of many of the processes considered for the application of nuclear heat sources is that the energy required must be supplied within a narrow temperature band just below the maximum. For hydrogen production, for instance, depending on the particular process adopted, energy available below say 750°-900°C is of very little use for the process. Practical and economic considerations demand that the remaining energy output of the reactor (say to reduce the reactor gas inlet temperature to about 300°C) be used in some other way. Possibilities here include electricity production, either for export, for use in the process, or for both; direct utilisation of low temperature heat in later process stages, or utilisation in one or more different lower temperature processes.

8. The recognition that use must be found for the low temperature heat in many applications leads to the necessity of a design choice with immediate implications for the safety analysis. If the choice for use of the low temperature heat is either electricity production or wide scale distribution of hot water by pipeline, and if the primary purpose of the reactor is production of a 'transmittable' process fluid, such as hydrogen, then siting options are available, i.e. the plant can be integrated or non-integrated. If, on the other hand, the low temperature heat is to be utilised directly in the same process or in others in the vicinity, then irrespective of primary purpose, the nuclear unit must be either integrated with the process plants or sited within a few kilometres of them.

9. To exceed 800°C in the first process fluid would require considerable materials and fuel development. To achieve 800°C in this fluid requires gas outlet temperatures to be increased somewhat beyond this figure and demands at the very least, some re-optimisation of core parameters as compared with the steam cycle power reactor. Basically, the mean temperature of operation of the fuel in the reactor must be increased as compared with the steam cycle plant. For early process applications it would be desirable to attain this end without exceeding the peak particle temperatures utilised in steam cycle HTR's so that fuel operating conditions remain within the range already proven. There are several ways in which mean fuel temperatures can be increased without increasing the peak temperature and these include:

(a) axial flattening
(b) development of a fuel element with more effective radial heat transfer
(c) down rating of the fuel
It is possible that these techniques could be used singly or in combination. The relative merits of the techniques are outside the scope of the present paper, they are noted here simply to set the scene for the subsequent safety discussion.

10. Much attention has been given in the past to the safety and siting problems of steam cycle nuclear power stations, it is clear that the basic principles established for such reactors must also apply to reactors used for process purposes. There are in addition a few other key aspects which require consideration for a process application. These may be summarised under three broad headings:

(a) the possibility of hazards caused by faults on the process plant affecting the reactor and vice versa

(b) possible modifications to the basic siting policy adopted for steam cycle nuclear power stations resulting from (a)

(c) the prevention of radioactive contamination of process products

11. This paper discusses some of the problems arising under (a) and (c) and attempts to highlight those areas needing further work. Siting policy (b) is properly the responsibility of national licensing authorities, but clearly before any supportable policy can be laid down much analysis is required under (a) and often this will be very specific to a particular plant and process. Depending on the process involved, the analysis would have to cover a range of fault situations; typical conditions are considered later in the paper. The analysis will be required at the initial design stage as the results could have a marked influence on a number of major decisions, for instance number of intermediate circuits (if any), choice of secondary (and possibly tertiary) circuit working fluid, overall plant layout, control requirements and containment of the reactor and its essential systems. As far as siting (b) is concerned this paper merely draws attention to some of the problems that will need to be taken into account and, it is hoped, will suggest some fruitful lines of thought.

12. Decay heat removal is of fundamental concern in any reactor system. In the case of an HTR used for process applications only, it may not be possible to rely on the normal secondary circuits for removal of such heat even though the primary circuit has, itself, guaranteed circulation. In these instances it is conceivable that such process reactors would require separate decay heat removal loops, probably water cooled. For reactors utilising all or part of the low temperature heat to generate electricity, the normal steam generator loops might suffice for this purport.

STANDARDS OF PROTECTION

13. In recent years there has been a growing interest in the quantitative assessment of the risks involved in operating nuclear power reactors, and this has necessarily involved the assessment of accident frequencies and the consequential release of radioactive material. In the UKAEA, it has also been deemed desirable to express the safety requirement quantitatively, to enable both designer and safety assessor to identify the dominant aspects of safety in a particular system and, so to concentrate attention on the most important issues. The target line which has been adopted recently is shown in Fig 1(1).

14. Similar methods of analysis have subsequently been employed in the UK study of water reactors by Rasmussen and Levine(2). Here a detailed analysis was made of a particular BWR and PWR and the consequences to the surrounding population evaluated and compared with other risks, both man-made and arising from natural background. The accident frequencies and corresponding release (in terms of iodine 131) derived by Rasmussen are also shown in Fig 1.
15. Figures 2 and 3 show the results of risk evaluations based on the UKAEA criterion and on the Rasmussen water reactor evaluation. In each case a comparison is made between the nuclear and some non-nuclear risks.

16. Despite the relative safety advantages of nuclear power shown by these comparisons there has been no suggestion that it would be permissible to reduce existing standards. No doubt this partly reflects a reluctance to rely entirely upon a theoretical synthesis of accident behaviour, although for very infrequent accidents, operating experience should not contribute any significant statistics. It also involves the belief, supported by experience, that the risks peculiar to nuclear energy are less acceptable to the public than other risks which they are better able to comprehend.

17. Without being too specific, then, about the detailed nature of the safety target, there appears to be a consensus of opinion that 'severe' accidents, involving 'large' releases of activity, should occur no more frequently than about $10^{-5}$-$10^{-7}$ per reactor operating year.

18. This range has been derived from studies of accidents originating within a nuclear power reactor complex. Events of origin beyond the station boundary, may also endanger the reactor and recognition of this has led to studies of such hazards, and in particular, protection against aircraft crashes. Estimates of the frequency of potentially damaging crashes onto reactors have been made in various countries, and generally lie within the range $10^{-6}$-$10^{-8}$ per year for most sites away from the direct vicinity of airports(34,5). Some countries are requiring design features to protect against crashes at the more frequent end of this range, but do not include the large commercial aircraft, which are generally assessed as lying in the range $10^{-7}$-$10^{-8}$ per year.

19. It therefore appears that, in broad terms, any event likely to present the risk of a serious accident to a nuclear plant, and which has a frequency in the range $10^{-6}$-$10^{-7}$ per year must be taken into account. This is true both for accidents originating within the plant and for risks of external origin, such as those arising from adjacent industrial installations of a non-nuclear nature.

**PROCESS REACTOR SAFETY**

**The Nature of Circuit Activity in a Process Heat HTR during Normal Operation**

20. The use of particle type fuel in an HTR implies a small release of fission products to the coolant during normal operation, as it is not possible with present manufacturing and quality control techniques to ensure that no particles with damaged coatings occur in the finished fuel bodies. This release is likely to change only slowly with time. Certain of the released species remain gas-borne whilst others may plate out to a greater or lesser degree on the cooler parts of primary circuit surfaces.

21. In normal operation it is possible to keep the continuous emission of fission products from the fuel at a very low level provided that fuel manufacturing standards are defined with a limiting release in mind and rigorously maintained thereafter. The initial level of release (ie the level before any in-service particle failure occurs) is predictable from operating conditions and fuel quality control data.

22. It is to be expected that the fission product release rates from a core of given power with defined particle breakage or damage fraction will be greater for a process reactor than for an electrical power reactor because of the influence of the higher temperatures on fission product retention and transport properties.
As temperatures are increased more dependence will be placed on the retentive capability of the particle coating since delays caused by other components of the fuel element will become relatively less important. Particular importance will attach to the possibility of chemical attack by fission product metals on the silicon carbide coating at elevated temperatures. Much further work remains to be done on these topics, and it could be that for a given limiting release the process HTR will require a tighter fuel specification than a straightforward electrical producer of the same thermal power.

23. If fuel particle failures occur during service there will clearly be an increased fission product release rate to the coolant and whether the operating conditions of a process heat HTR make this more likely to occur, or more onerous if it does occur, than in a steam cycle power plant, are matters requiring some detailed attention.

24. The nature of the resulting circuit activity in a steam cycle HTR is now fairly well established from both reactor experiments (principally DRAGON\textsuperscript{6,7,8}) and several design studies\textsuperscript{(9,10,11)} for steam cycle power reactors using realistic temperature, power and fuel particle damage distributions together with the available fission product release and transport data. These studies have shown that the most significant species escaping from broken or defective fuel are the rare gases, caesium, strontium, iodine and silver. Tritium is also likely to be present in the coolant, this arising partly from fission but mainly from irradiation of the lithium 6 impurity in graphite and from the \((n, p)\) reaction on helium 3. Sulphur 35 is produced from sulphur 34 and chlorine 35 impurities in the graphite, but unless a fair amount of graphite corrosion takes place the sulphur is very unlikely to be released to the coolant.

25. Whilst the amount of tritium in the coolant circuit is likely to be relatively low at any time and is unlikely to present a hazard due to normal coolant leakage, there is considerable evidence to show that tritium appears in the secondary circuit of most gas-cooled power reactors and also in the DRAGON secondary circuit. In general this is not likely to be due to primary-secondary circuit leakage as the gas-cooled power reactors, unlike DRAGON, have pressures on the water-side considerably higher than on the gas-side, and so the mechanism is presumably hydrogen diffusion through steel. Tritium levels in secondary circuits may well become important in a process heat system if no intermediate circuit is used. Other active species are not likely to be present in the secondary circuit in any significant quantity provided that the circuits in contact with the reactor primary coolant are built to a sufficiently high standard of leak tightness. This may require welds of a higher standard than current general practice in the chemical industry, and in fact than currently achieved in steam cycle nuclear power plants.

26. For a process heat HTR operating so as to give first process fluid temperature up to 800°C, it is unlikely that the radiologically significant species will be different from those already identified for the steam cycle reactor, and support for this view can be found in the results of DRAGON fuel irradiation experiments performed in the relevant temperature range. For reactors achieving higher temperatures further experimental work under appropriate conditions will be required. At the present time, work on the addition of gettering agents\textsuperscript{(12)} to the kernel to reduce the mobility of fission product metals at elevated temperatures is underway, and this could be a useful line of development of particular relevance to process HTRs.

27. Under 'normal operation' must be included also those fairly frequent operation transients which cannot be classed as faults, but are practically inseparable from the running of any large plant. If, as seems highly likely in some applications, the process reactor is also used to produce electricity, grid disturbances have the potential for reflecting back onto the reactor. Attention is drawn to the fact that
the more onerous temperature conditions of a process HTR may lead to a higher probability of particle failure under such operational regimes. Increased circuit activity due to any such failure would require careful assessment, the results obtained could well have implications on allowable fuel dwell times and fuel development programmes.

Safety Considerations during Normal Operation of a Process Heat HTR

28. Major process plant faults which could affect the reactor whilst it is operating normally are dealt with later in the paper.

29. Apart from such major faults, normal operational transients on the process plant have the potential for reflecting back onto the reactor, and must receive careful attention for each particular proposal so that any safety implications can be appreciated at the design stage and appropriate action taken. For dual cycle plant producing electricity as well as process heat the decoupling of process plant and grid induced disturbances from each other is an interesting control problem which may have safety implications and would clearly require detailed study for each proposed combination of uses.

30. Three further safety aspects, all related to circuit activity levels, must be considered under normal operational conditions. These are:

(a) possible hazards caused by leakage of coolant to atmosphere
(b) maintenance problems due to deposition of active species on circuit components
(c) potential contamination of process products

(a) and (b) are common to both power and process HTR's although there may be a difference in degree due to the different temperature and power distributions within a process reactor. For reactors operating up to first process temperatures of around 800°C both (a) and (b) should be controllable within acceptable limits using current technology. For the developed HTR, operating well above this temperature range, it is not possible to comment sensibly at this time, because as yet there is little data under appropriate conditions.

31. For a nuclear process plant the degree of radioactive contamination of the product must be very small. For example if the primary purpose is to produce hydrogen as a secondary fuel which is then distributed to off site consumers, perhaps by pipeline, a very small amount of contamination indeed will be tolerable. It is expected on present evidence that the most likely contaminant of secondary circuit hydrogenous fluids will be tritium, and only detailed studies could show just what levels might be reached. A considerable amount of work has been done in the US in connection with 'Project Gasbuggy' on the commercial and domestic utilisation of natural gas which could be tritium contaminated(13), and this may provide a starting point for further studies in connection with tritium contamination of hydrogen. The results of such studies might well be a determining factor in the use or otherwise of intermediate circuits.

FAULT BEHAVIOUR OF A PROCESS HEAT HTR

General

32. In this section we review those basic features of HTR fault behaviour which have been identified from previous detailed studies of steam cycle power production designs and from work on experimental reactors. The relevance or otherwise of these features to process HTR's is discussed and the differences imposed by the process application which lead to a need for further investigation are identified,
33. Previous experience of gas-cooled reactor fault analysis has shown that the classic fault conditions may be broadly summarised in six main categories. These are reactivity accidents, leakage of secondary circuit coolant into primary circuit, loss of forced circulation in primary circuit, depressurisation, loss of forced circulation in secondary or later coolant circuits and local coolant blockage. The last of these, local coolant blockage, will not be considered further here as it is similar for process reactors and power reactors. The remaining categories are grouped under three headings:

(a) Reactivity and Control Considerations

(b) Chemical Reactions of Secondary and Subsequent Coolants

(c) Loss of Coolant

34. Before considering these categories of fault, it is worth drawing attention to a general point related to the nature of the fuel in an HTR, and to the expected fuel temperature distribution within a process reactor. In an HTR the use of particle type fuel with a large number of small particles means that sudden and catastrophic fuel failure would not be expected. For a steam cycle HTR, the operating temperature of the fuel is such that there is a fairly large over-temperature margin which reduces the possibility of large fission product releases in transition of short duration. It must be noted that in process applications, particularly in those identified as suitable for the developed HTR, quite large inroads could be made into this margin. For process applications requiring temperatures at the upper end of the range achievable with existing HTR technology it is almost certain (depending on the design solution adopted) that although no particles will be operating at higher than peak steam cycle reactor temperatures, the temperature distribution in the fuel will be such as to mean that a much larger fraction of the particles will in fact be at the higher temperatures than in the steam cycle design. Even here, therefore, fewer particles have as large a margin to failure as in the steam cycle power reactor. The implications of this in fault conditions are, at least qualitatively, fairly clear. Increases in the number of particles failing compared to the steam cycle power reactor case could be expected for a given temperature increment. This would give rise to an immediate increase in fission gas and iodine content in the coolant; the effect on metallic fission product release to the coolant would depend markedly on the time for which the increased temperatures persist. The effect of transient changes of conditions on fission product release could as usual, only be assessed quantitatively for a given design of reactor. However, it is not possible at the moment, even for steam cycle plant, as insufficient information exists as to the fission product release behaviour in HTR's under fault conditions. Reliable information is also lacking on the effect of transient situations, particularly water ingress and depressurisation, on active material deposited on primary circuit surfaces. These are clearly some of the important gaps in our knowledge and require carefully planned and executed programmes of research and development work to support HTR safety assessments.

Reactivity and Control Considerations

35. HTR's, with either low enriched uranium or uranium-thorium fuel cycles, have negative fuel temperature coefficients of reactivity, the value for the low enriched cycle being markedly more negative than is the value for the thorium cycle. In common with other gas-cooled systems there are no reactivity problems due to coolant voids or loss of coolant.
36. With the low enriched design the very high negative value of the fuel temperature coefficient leads to a high degree of temperature stability during power and flow transients at power, at least on the short time scale of several tens of seconds which is adequate for normal control action. It also leads to a high negative reactivity investment in control rods in order to cope with the large changes associated with bringing the reactor from cold shutdown to hot operational conditions. This can require very high reactivity rates if fast start-up is to be achieved and consequently control rod withdrawal faults at start-up can give rise to very fast power transients as temperature reactivity feedback is initially minimal under such conditions. This effect is quite marked even with the steam cycle HTR; in the case of process heat reactors and particularly with developed HTR's for very high temperatures, the higher average fuel temperatures imply somewhat larger rod investments still, and the capability for rapid shutdown on reactivity faults will be necessary.

37. The graphite moderator adds considerable thermal capacity to the system and on loss of coolant flow faults at power the combination of this effect and the high negative Doppler coefficient in the low enriched design means that rapid trip action on loss of forced circulation is not necessary. Under such circumstances the reactor is set back in power due to reactivity feedback so that fuel temperature rise only slowly, the high thermal inertia of the moderator prevents its temperature changing rapidly and reactor trip action is not essential for core protection purposes for many tens of seconds. Necessary fast control action under such circumstances could well be determined by process plant characteristics rather than by reactor properties.

38. Detailed kinetic studies of combined reactor and chemical plant characteristics may reveal that for certain process applications of HTR's fairly complex shutdown sequences will be necessary to ensure overall plant safety. These sequences may in fact be different for different initiating faults. Should such suggestions be supported by the analytical results, that part of the chemical plant's control circuitry associated with the shutdown sequences could be required to be of reactor safety circuit standard.

Chemical Reactions of Secondary and Subsequent Coolants (includes first process fluid)

39. In a steam cycle HTR, water-side pressure in the boilers is higher than the helium pressure and the principal problems from secondary-primary circuit leakage arise from the chemical reaction of water or steam with graphitic components. In a process HTR, it is possible to conceive of a number of other fluids which could leak from the secondary to the primary circuit depending both on the process and on the detailed plant design. Close attention will have to be given to all such possibilities and a study made of the potential chemical reactions and reaction products to see if any additional hazards arise. Such leakages, if not taken up, may be of particular importance if they occur just before, at, or just subsequent to, a depressurisation when the possibility arises of air mixing with the reaction product to form an explosive mixture.

40. The analysis must take account of the fact that most process heat reactors will probably also have secondary circuits either to produce electricity or to remove shutdown heat, or both. In one proposed design (15) (Fig 4), the high temperature heat is used for process purposes, the primary coolant then supplies helium gas turbines which drive the compressors responsible for coolant circulation, and finally the low temperature heat is used either for a low temperature process or rejected via a precooler. Thus, even in this case, water may be present in a secondary circuit.
41. The designer and the safety analyst may, therefore, be faced with both the water gas problem and with other in-core chemical reactions. It may well be that such considerations alone will necessitate adoption of an intermediate helium circuit in certain circumstances.

42. It is assumed throughout this paper that any HTR built for process heat purposes will follow modern gas-cooled reactor practice and utilise a prestressed concrete pressure vessel with its attendant safety advantages. In common with the practice adopted for steam cycle nuclear power plants utilising concrete pressure vessels, it is assumed that any heat exchangers, or process plant such as reformers, which take reactor primary circuit gas, will be located within pods in the main vessel walls so that primary circuit coolant (apart from any small leaks) is always contained within the main vessel. This procedure, unless an intermediate circuit is used, will involve PCRV penetrations carrying process fluids and the safety implications of such an arrangement will need careful analysis. Depending on the nature of the process fluid and its possible reactions with oxygen there could be some argument for inerting (ie blanketing with nitrogen say) cavities within the vessel containing process plant.

43. In some cases this might not be considered adequate protection, particularly if the first process fluid was of a flammable nature, as risk of chemical explosion within cavities of the main concrete pressure vessel cannot be accepted because of the problems associated with depressurisation. It is suggested that the risk of a depressurisation incident for a process reactor should be no higher than the similar risk for a power reactor, ie that the presence of the chemical/industrial complex should not contribute significantly to the overall risk of depressurisation.

44. Adoption of an intermediate helium circuit between the primary circuit and the first process fluid gives the possibility of removing process plant components from main vessel cavities and considerably reduces the risk of damage to the reactor pressure vessel by chemical plant induced explosions. This could well prove to be one of the strongest arguments for use of an intermediate circuit, but, even with such provision, there could still be cases where special conditions or restrictions would be necessary for certain maintenance procedures on the process plant. A good example of a situation where such procedures would be necessary occurs in hydrogen production via the gas reforming method (see the section of this paper dealing with explosions).

Loss of Coolant

45. Depressurisation, with or without loss of blowers, and loss of blower supply at power, have some similarities in their requirements for corrective action and it will be convenient to discuss them together.

46. Although it has been fairly clearly established for steam cycle HTR’s operating on the low enriched fuel cycle that rapid trip action on loss of coolant flow fault is not essential for core protection purposes, it has to be borne in mind that smaller margins to fuel failure will exist in a process reactor and shutdown may be required somewhat earlier than for the steam cycle case. It is envisaged, subject to the results of detailed studies, that several tens of seconds will still be available in the process case for the pressurised fault, other considerations may reduce this time for the depressurisation incident.

47. In the slightly longer term of course, whether the loss of flow is due to loss of blower power with the reactor pressurised or due to a depressurisation incident, the reactor must be shut down and cooling guaranteed. A number of schemes for auxiliary core cooling have been proposed for steam cycle HTR’s, including alternative sources of electrical supply to blower motors, and schemes utilising auxiliary cooling loops in which the helium may be circulated by either electrical...
driven blowers fed from multiple independent electrical supplies or for short times, by electrically driven blowers fed from essential turbo-alternators taking steam from boilers located in the auxiliary loops themselves. In the latter case, the essential turbo-alternators are continuously running and, with the reactor pressurised, are capable of operating for several hours on steam generated from decay heat. Any of these schemes appear applicable to a pressurised process reactor, particularly perhaps the last one if the low temperature heat is used for power generation. In the case of the self-powered gas turbine driven compressor scheme described earlier primary coolant circulation during reactor shutdown when pressurised is by use of the starter motor on the turbine shaft and primary coolant heat rejection is via the low temperature process plant or pre-cooler. The water cooling capacity of such a pre-cooler would be more than adequate for decay heat rejection provided both primary and secondary coolant flow is guaranteed. The guaranteeing of primary coolant flow in such a system is a matter requiring further detailed examination, in particular, restoration of normal compressor drive would not be possible following a depressurisation. To cover the situation following a depressurisation incident on a process reactor it seems fairly clear that, whatever the basic operational mode of primary coolant circulation, there will be no alternative but to provide an electrically driven system with multiple independent power supplies.

48. Most, if not all, current HTR designs for power plant utilise downflow cores. This feature could well give rise to lack of, or unpredictable, natural circulation characteristics. It will be shown later that there may be occasions where it is desirable after a process plant fault that an associated reactor should be able to survive in a safe condition for a number of hours with minimal outside assistance and/or power. Under such conditions, the capability for sustained natural circulation in a pressurised primary circuit could be of inestimable value. Detailed consideration is, therefore, necessary to determine whether downflow cores are such a pre-requisite of HTR design as seems to be presently accepted.

49. The guaranteeing of forced circulation in secondary (or later) circuits in general is again very specific to the process. If the reactor is not a dual cycle system (ie does not utilise the low temperature heat for power production) the nature of the fluid in the first process circuit, and the way in which it loses its heat in the process clearly have a great bearing on the problem. If the process is such that shutdown is required immediately on loss of primary heat source (the reactor) it may be that the effectiveness of the process loops for heat rejection purposes is lost. In such a case, the reactor would have to be designed with auxiliary cooling loops, possibly water-cooled, and an essential boiler/essential turbo-alternator scheme fed from decay heat, similar to that proposed by EDO:16) could be very attractive from the safety viewpoint, as secondary circuit pumps could also be powered from this source, and the reactor is then virtually capable of 'standing on its own feet' after an incident in which it remains pressurised.

**NATURE OF RISK FROM ADJACENT INDUSTRIAL PLANT**

**General**

50. In the absence of a specific proposal it is only possible to generalise about the risks posed to a nuclear reactor from adjacent plant, but the main risks may be categorised as follows:-

(a) Explosions on the Industrial Site  
(b) Missiles  
(c) Release of Flammable Vapour or Gas  
(d) Release of Toxic Gases or Vapour  
(e) Fires
51. Following an incident on a process plant associated with a nuclear reactor it is conceivable that the normal access route to the reactor could become impassable because of wreckage, smoke or fire. Provision of alternative access routes to facilitate approach to the reactor by emergency vehicles under such circumstances should be a design requirement.

52. A further risk applicable to coastal sites is that associated with the passage of large tankers close inshore, particularly if the industrial complex associated with the reactor is serviced by such vessels. Under these circumstances the possibilities arising from a collision between ships must be considered at the design stage. Typical examples of the conditions to be taken into account are explosion fires and entry of fuel into circulating water intakes.

The Explosion Hazard

Explosions Outside the Reactor Site

53. Explosions on the industrial complex can be envisaged from a number of causes and are particularly liable to occur as a result of operations involving highly flammable substances, as in oil refineries or the petro-chemical industry. Such explosions are by no means unlikely, and indeed one or two such events are reported each year somewhere in the world. It is difficult to ascribe a frequency without taking account of the nature of a particular plant, but for industries dealing with petroleum products or materials of similar characteristics, it would be unwise to expect a lower occurrence than about $10^{-3}$/year per plant.

54. The magnitude of the explosion, expressed in TNT equivalent may lie in the range 1-20 Te. For example, the explosion at the oil refinery at Pernis, Holland in 1968 produced damage equivalent to between 5 and 10 tonnes of TNT, and the explosion at the Nypro plant at Flixborough, England in 1974 was equivalent to between 10 and 20 tonnes of TNT. Such explosions would produce overpressures of a few psi (0.1-0.2 bar) within the range of 200-400 m approximately.

Explosions Directly Affecting Reactor Secondary Circuit

55. In an earlier section of this paper attention was drawn to the possible use of an intermediate coolant circuit in order to isolate process plant from main pressure vessel cavities. Even with such provision of an intermediate circuit, it was stressed that special maintenance procedures could be required on the process plant for some applications. A particular case where such procedures would be necessary is that where a reactor supplies heat to a number of reformers producing hydrogen. Periodically it is necessary to change the catalyst in such reformers. To ensure continuity of hydrogen supply it would be convenient to take one circuit out of action, leaving the reactor at power with the remainder of the reformers still in production. There could be a danger period when an explosion might occur and a high pressure wave could be transmitted through the intermediate (secondary circuit to the primary-secondary heat exchanger, possibly causing multiple tube failure and rapid depressurisation of the primary circuit. Such explosions are not unknown, and indeed there have been press reports recently (April 1975) of an explosion causing several fatalities in South Africa during reactivation of a gas reformer used for hydrogen production from methane. Preferably, design action should be taken to remove such possibilities, but if this is not practicable very strictly controlled operational procedures would be required.

Protection Against Explosions

56. Protection must be provided so that no serious consequences to the reactor would ensue following an explosion. This could be accomplished by a suitable combination of distance and design. It should be noted that if the explosion is assessed at $10^{-3}$ then the safeguards provided, together with the assessment of the magnitude of the explosion, must provide a factor of $10^{-3}$-10^{-4} against a large release of activity, if the safety target of $10^{-6}$-$10^{-7}$/y is to be achieved.
Missiles

57. Missiles may be generated either from explosions originating within chemical process plant or may be generated by failure of pressure vessels containing supersaturated liquids or gases.

58. As an example of missiles generated by explosions, the case has been reported in which a fragment of a butadiene refining vessel weighing 800 lb (about 400 kg) was propelled a distance of 3000 ft (about 1000 m)(17).

59. The velocity attained by a missile formed from the failure of a vessel depends upon:-

(a) The work done by the fluid in the vessel in expansion

(b) The mass and surface area of the missile

(c) The manner in which the fragments separate and permit the expanding fluid to escape.

60. A mathematical model for calculating the final velocity of such a missile has been developed and is being tested against an experimental programme. The results of calculations using a simplified model with propane as the pressurised fluid are presented in Fig 5(18). Here the dimensional group MV/U (M = mass, U = velocity of missile and V = volume of fluid contained) is plotted against the initial temperature of the fluid.

61. For a range of 1 mile (1.6 km) it has been estimated that an initial velocity of at least 500 ft/s (150 m/s) would be required for a missile such as a large section of a thick walled pressure vessel, the impact velocity would then be about 350 ft/s (100 m/s). For a vessel containing propane the value of MV/U for this missile would be at least \( 10^4 \) lb/ft² sec corresponding to an initial temperature of 75°C. Correlations with experiment suggest that the estimates given here, based on the theoretical model, may somewhat over-estimate the velocity attainable, but not excessively.

62. The vessel failure may be caused by structural failure or by excessive conditions of temperature or pressure, and many volatile materials could be expected to show behaviour similar to that of propane. Thus it may be concluded that the generation of missiles with a range of up to 1 mile, and possibly somewhat further cannot be excluded from consideration for plants processing volatile materials under pressure.

63. If the missile is generated and propelled in a random direction, and if the probability of the maximum range is uniform between zero and the maximum possible, the probability of it landing in a target area can be calculated and is shown in Fig 6. For example, the ballistic probability of hitting a target of \( 10^{-6} \) at a distance of 1 km where the maximum range is 1.5 km would be about \( 2 \times 10^{-2} \) and at 500 m about \( 10^{-2} \).

64. A single missile would not usually be capable of creating serious conditions on a reactor site, although there may be localised areas of particular susceptibility and these must be identified (for example, the fuel charging area could be particularly sensitive). When this is done, the probability of serious damage depends upon the product of the probability of generation of the missile and the ballistic probability of striking a vital area. Multiple missiles can usually be ruled out as even less probable.
Release of Flammable Vapour

65. As noted earlier, the release of flammable hydro-carbons followed by local ignition and subsequent explosion is not infrequent in the oil industry. Releases without explosion are probably more frequent, but information on this is not readily available. It is clearly possible for releases of flammable materials to occur and to be carried some distance before ignition occurs, possibly onto an adjacent nuclear site where there are plenty of ignition sources.

66. The explosion of a gas-cloud on a reactor site could cause serious damage leading to release of activity from the core. The nature of the damage would depend upon the detailed design, but loss of emergency services, incoming grid, standby diesels etc can be envisaged. In addition, the station staff would be endangered and some may be killed.

67. The further danger exists that ventilation systems, by drawing in gas, could produce explosive gas mixtures within buildings, and ignition, again likely, could cause incalculable harm to plant and personnel. Diesel engines may over-speed even in relatively weak mixtures, and if suitable over-speed protection is fitted may become unavailable for emergency supplies.

68. The distance over which hazard may exist for releases of flammable gases depends upon the size or rate of release, the conditions for dispersion and the density of the gas. Means for calculating dispersion for heavy gases are not well established although some work is being done on this problem (19, 20).

69. The probability of release of a potentially hazardous quantity of flammable gas depends upon an examination of the quantities and processes involved. For large petro-chemical or oil refinery plants it is suggested that the probability of release probably lies in the range 10^{-2} - 10^{-3}/year. On most occasions when the gas is ignited it will be on the original site, but if explosions on the reactor site are to be avoided, an additional factor of probability of 10^{-4} - 10^{-5} has to be found to meet the safety standards suggested. Some of this may be derived from the wind and weather factors controlling dispersion, but at short distances these are not likely to be found adequate. Additional safety features may then be sought by design, by ensuring ignition at the nuclear site boundary and by monitoring for hazardous concentrations at the reactors so that reactor and ventilation plant may be shutdown at agreed concentrations.

Release of Corrosive or Toxic Material

70. The release of material of a corrosive nature would require consideration during design to ensure that the reactor plant could operate without risk to reliability or to life-span. Acute safety problems should not arise from such a cause.

71. The release of quantities of airborne toxic material could hazard the ability of operators to superintend the shutdown of the reactor if drawn in through ventilation systems and, if prolonged, make conditions on the site untenable. It is impossible to generalise about the nature and risk of this hazard without specific proposals to consider. Similarly it is difficult to assign a frequency to such an event, but there have been press reports in the UK in 1974 about a release of potentially lethal gas which required 65 workers to be treated in hospital and an accident in which a release of brominated chemicals necessitated the evacuation of 400 people within a distance of 1 mile.
72. For purposes of assessment it would appear unwise to assume that potentially serious leaks of toxic gases occur less frequently than about $10^{-2}$-$10^{-3}$ per year from plants handling large quantities of such material. The effect of the reactor is indirect as indicated above, but suggests that the system should be capable of remaining in a safe condition after shutdown with the minimum of attention.

**Fires**

73. Fire is a major hazard in the oil industry, and experience has shown that major fires in bulk terminals and pipeline stations lie in the range $10^{-4}$-$5 \times 10^{-4}$ per year (see Fig 7). The probability of a storage tank fire has been assessed by an international oil company as $3 \times 10^{-4}$ per tank per annum. Most fires are small and soon extinguished but a small fraction, perhaps $10\%$, may result in a large conflagration. A figure of about $3 \times 10^{-5}$ per tank per year is thus not inconsistent with the results shown in Fig 7. In process areas dealing with similar flammable materials, the incidence of serious fires would be expected to be higher, a range of $10^{-2}$-$10^{-3}$/year is suggested for such events.

74. Many chemical processes involve the use of flammable materials, and it seems unlikely that outbreaks of fire will be much less frequent than those quoted above for the oil industry, but clearly this will depend upon the particular materials, the nature of the process and the size of the installation.

75. Fires involving oil product are characterised by the evolution of large volumes of black smoke which may be acrid and corrosive if the sulphur content is high. There would be little serious hazard to people at distances greater than about 300m, although discomfort could be experienced. Ventilation system intake filters may be blocked by carbon which may also deposit on insulators, switchgear and other electrical apparatus and cause short circuits if the fire is prolonged. Grid disconnection could, therefore, occur, and emergency electric supplies could be at hazard, making it difficult to provide for continued supply of feed water to boilers and power to gas circulators.

**SOME GENERAL CONCLUSIONS ABOUT SAFETY PROBLEMS**

76. Table I attempts to summarise the frequency, range and nature of the risk to the reactor arising from the hazards considered in the previous section. It should be noted that additional factors must be taken into account in assessing the risk to the reactor system which are not generally given in the Table. For example, hazards from fires and the release of toxic substances would be of concern only if the wind direction carried the smoke or noxious material from the industrial area to the reactor site.

77. However, even though there is uncertainty in frequencies of the events considered, it is apparent that a considerable additional factor has usually to be found if the reactor safety is to reach the standard of $10^{-2}$-$10^{-3}$ for serious accidents corrected earlier. Most of the hazards considered have an originating frequency in the range $10^{-2}$-$10^{-4}$/year requiring factors of $10^{-3}$-$10^{-5}$ to be available from wind, atmospheric dispersion, distance, or the safeguards built into the reactor against these risks.

78. The need, not only to be able to shut the reactor down safely but to maintain it safely in the shutdown state for a considerable period when the conditions on the site may be chaotic, is of paramount importance. It is possible to envisage conditions in which operator control would be minimal, or even lost altogether and, ideally, the plant should be capable of surviving a period of many hours after shutdown with little or no operator action.

79. There appears to be no reason why an HTR could not be developed to acceptable standards of safety for use with process plant, but much detailed work remains to be done.
<table>
<thead>
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<th>EVENT</th>
<th>SUGGESTED YEARLY FREQUENCY</th>
<th>MAIN CONSEQUENCE</th>
<th>RANGE OF HAZARD</th>
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<tr>
<td>Explosions on Industrial Site</td>
<td>$10^{-3}$ for processes dealing with inflammable materials. Other processes require specific assessment.</td>
<td>Interference to ability to maintain emergency supplies</td>
<td>Up to a few hundred metres</td>
</tr>
<tr>
<td>Missiles generated in plant or process vessel failure</td>
<td>Missile generation probably in range $10^{-2}$-$10^{-4}$ depending on process. Ballistic probability has also to be taken into account.</td>
<td>Severe damage to limited areas, eg pile-cap, charge machine or boiler feed pipes</td>
<td>In the worst circumstances up to about 2 km but depends upon the fluids and temperature of the vessel prior to failure</td>
</tr>
<tr>
<td>Gas clouds arriving on reactor site in hazardous concentrations</td>
<td>$10^{-2}$-$10^{-3}$ for release from oil industry complex. Other factors available from wind direction and atmospheric dispersion. Latter not well understood for heavy gases.</td>
<td>Intake to buildings via ventilation systems and subsequent explosion. Explosion in the open close to reactor and auxiliary buildings. Damage to emergency supplies. Loss of operators.</td>
<td>Could be several kilometres depending on nature and quantity of gas</td>
</tr>
<tr>
<td>Release of Toxic Material</td>
<td>Difficult to assess without a specific assessment, where toxic material is handled in quantity could be in range $10^{-3}$-$10^{-4}$.</td>
<td>Hazard to operators. Difficult to maintain supervision of reactor if the release is prolonged.</td>
<td>Depends on quantity and nature of toxic material</td>
</tr>
<tr>
<td>Serious fires at Industrial Site</td>
<td>$10^{-2}$-$10^{-3}$ for processes involving large amounts of flammable material</td>
<td>Damage to electrical equipment by carbon deposition. Difficult conditions for operators.</td>
<td>Depends on size and nature of fire, but could be 1-2 km</td>
</tr>
</tbody>
</table>
ACKNOWLEDGEMENTS

The authors are grateful to many colleagues in the UKAFA for helpful discussion and comment; in particular, thanks are due to Mr F Abbey (Culcheth), Mr R L Faircloth (Harwell), Mr G Coast, Mr E C Heath and Mr A N Knowles (Risley).

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FIG. 1 FARMER CRITERION & RASMUSSEN LWR ASSESSMENT
**FIG. 2** COMPARATIVE RISK TO POPULATIONS FROM AIRCRAFT, METEORITES AND NUCLEAR REACTORS (U.K.A.E.A. CRITERION)
FIG. 3 FREQUENCY OF FATALITIES DUE TO MAN-CAUSED EVENTS
(RASMUSSEN LIGHT WATER REACTOR EVALUATION)
FIG. 4 FLOW DIAGRAM OF PROPOSED HELIUM GAS TURBINE DRIVEN COMPRESSOR SCHEME FOR USE WITH A PROCESS HEAT REACTOR
FIG. 5 VELOCITY OF MISSILE FROM RUPTURED VESSEL
FIG. 6 PROBABILITY OF A MISSILE LANDING IN A GIVEN AREA
FIG. 7 SIZE OF OIL FIRE Vs. FREQUENCY OF OCCURRENCE
R. Schulten, K. Kugeler, C.B. von der Decken, R. Hecker

STUDIES FOR INTERMEDIATE HEAT TRANSFER LOOPS IN PROCESS HEAT REACTORS

Paper presented at Specialists Meeting

HIGH TEMPERATURE GAS COOLED REACTOR SAFETY

Current Status and Perspective

Petten, 13th-15th May, 1975
1. Introduction

There are some reasons and considerations why intermediate heat exchangers (IHX) could be necessary for nuclear process heat applications. Some arguments which are often used are the following:

- The permeation of hydrogen from the process to the reactor coolant can be reduced.

- The permeation of tritium from the reactor circuit to the process gas can be lowered.

- The contamination of the parts of the process heat exchanger which must be accessible is avoided.

- A complete separation of the nuclear island and of the chemical plant is attainable.

- The inleakage of water to the core in the case of accidents is prevented.

- For some special processes (steam gasification of coal, water splitting processes) the process heat exchangers should be outside the containment, if solid materials are handled in the process.

The purpose of this paper is to study some special aspects of the question of IHX including the technical feasibility of these heat exchangers. The following points are discussed in more detail:

- H₂- and T-permeation
- Flow sheets for IHX-systems
- Media for IHX
- Arrangements of IHX, steam reformer and steam generator
- Data and design of IHX
- Materials for heat transfer
- Safety considerations
- Advantages and disadvantages of introduction of IHX
- Design steps and test facilities for IHX
2. Hydrogen- and Tritium Permeation

Large quantities of hydrogen which permeate from the process side through the wall of the heat exchanger (steam reformer or coal gasifier) to the primary circuit of the nuclear reactor would cause high corrosion rates on the fuel elements and on the structural graphite \((C + 2H_2 \rightarrow CH_4)\). Furthermore carbon deposition on the cold parts of the helium circuit and changes in high temperature material structure in the process heat exchanger could take place.

Large quantities of tritium coming from the helium circuit to the process gas by permeation would cause a high intolerable value of radioactivity.

Therefore these two questions seem to require an IHX. The state of knowledge today about permeation rates and their importance is as follows:

In the last years the permeation rates of hydrogen through the walls of a high temperature reformer tube have been measured depending on the wall temperature, the \(H_2\)-partial pressure, the \(H_2O/H_2\) ratio and type of materials. The result is as follows:

Pure hydrogen permeates at a very high rate, the permeation rate of a mixture of hydrogen and steam is much smaller caused by inner oxyde layers which act as diffusion barriers. The following figures 1, 2, 3 show this special behaviour for Incoloy 800. Using these results the amount of hydrogen entering the helium circuit of a 3000 MW\(_{th}\) plant for steam reforming will be in the order of \(\approx 0.5 \text{Nm}^2\text{H}_2/\text{h}\) (see table 1) which can be tolerated. This corresponds to an impurity level for hydrogen in the helium circuit of 3.. 5 vpm which is comparable to the AVR reactor. Following these considerations an IHX is not needed. However, before a real decision can be made, the behaviour of these oxyde layers during instationary operations and in a big facility (30 tube bundle experiment) should be tested. Furthermore promising possibilities to overcoat the tubes with special layers as an additional barrier should be developed and tested.

Regarding the question of tritium the following ideas are obtained today:

If all the tritium which is released in a high temperature reactor (3000 MW\(_{th}\)) to the helium circuit and which can permeate to
the outside (30 Ci/d), is distributed to the product gas of a SNG-plant (∼ 3 · 10^7 Nm^3/d), than the tritium concentration will be in the order of 1 μCi/Nm^3 gas which is within the values allowed today.

Normally the gas is burnt and an additional distribution of the activity occurs.

Furthermore it is reasonable to assume that there is a similar reduction in permeation as in the case of hydrogen permeation. This would cause another reduction of more than a factor of 10 of specific activity for the product gas. It should be, however, developed a method to remove the tritium from helium circuits by getting in special material like tritium.

Fig. 1: H₂-Permeation for Incoloy 800

Fig. 2: H₂-Permeation for Incoloy 800

Fig. 3: H₂-Permeation for Incoloy 800

Table 1: Estimation of H₂-Permeation

\[ N_{th} = 3000 \text{ MW} \]
\[ F = 3000 \text{ m}^2 \quad (T > 870 \text{ °C}) \text{ assumed} \]
\[ K \sim 30 \text{ Ncm}^3 H_2/h \quad (P_{H_2} = 1 \text{ bar}, T = 900 \text{ °C}, S = 3.3 \text{ mm}) \]
\[ K^* \sim 150 \text{ Ncm}^3 H_2/h \quad (P_{H_2} = 20 \text{ bar}, T = 900 \text{ °C}, S^* = 15 \text{ mm}) \]
\[ K^* \sim K \cdot \rho / (S^* / S) \]
\[ \dot{m}_{H_2} < 0.5 \text{ Nm}^3/h \]
3. Flow Sheets for IHX

The following figure 4 shows some basic arrangements:

- Steam reformer and steam generator are directly integrated into the primary circuit of the nuclear reactor (1).
- The steam reformer is heated by an IHX, the steam generator is directly heated by the primary helium circuit. There are three main possibilities which are different from the view of amount of process heat, heat fluxes in all exchangers and energy demand for the blowers.
  - steam generator in IHX-circuit (2a)
  - hot blower in IHX (2b)
  - recuperator and cold blower in IHX (2c)
- Both reformer and steam generators are heated by an IHX, all reactor heat is transferred through the IHX (3).

In table 2 there are some typical numbers for the heat transfer, the amount of high temperature process heat and of energy demand for the blowers. Others IHX-type systems can be realized, if steam is used as heat carriers (see fig. 4).

Steam could be an interesting medium if condensation in the circuit occurs, because the power to pump the feed water is very small in comparison to that for gases which must be compressed. A further possibility is to work with a high steam/methane-ratio ($H_2O/CH_4 \sim 8/1$) and to condense the steam after expansion in a steam turbine and then to separate the product gas from the water (5).
4. Media for IHX

In addition to helium, carbon dioxide, nitrogen, steam and other noble gases like neon and argon, could be used as intermediate system fluids. Hydrogen cannot be used because of its permeation into the primary circuit and safety reasons. An analysis of the required heat transfer area as well as of the necessary pumping energy in the intermediate system, shows helium to be the economically most interesting medium (see fig. 5). CO₂, N₂ and steam incur very high pressure drops. Steam can be used as medium only when it is condensed in the cycle and then repressurized in the liquid state as in flow sheets shown in chapter 3.

Fig. 5: Heat Transfer and Pressure Drop for some typical IHX-Media dependent on Velocity in Heat Exchanger Bundle o Primary Side (v₂).
5. Arrangement of IHX, Steam Reformer, Steam Generator

The question how to arrange the different heat transfer areas requires new considerations and solutions for nuclear process heat applications. The main reason is that the heat fluxes and the power densities in process heat exchangers (steam reformers, IHX) is much smaller than in steam generators. Depending on the temperature split in the reactor circuit the value is 1 ... 1.5 MW/m$^3$ instead of 5 ... 6 MW/m$^3$ in steam generators of HTR. Fig. 6 shows the consequence of this typical difference for a 3000 MW$_{th}$-plant in which the temperature difference 950 ... 700 °C is used for steam reforming and 700 ... 250 °C for steam generation (without IHX).

Either the volume of the reactor vessel should be increased in size in integrated designs or more loops should be used in loop designs.

If an IHX is used in principle the following solutions (see fig. 7) are reasonable:
- IHX in reactor vessel, SR and SG outside containment. Fast acting hot valves (T > 900 °C) must be available, gas pipes for long distances for the hot helium are needed (1).
- IHX in reactor vessel, SR and SG inside containment. In this case the valves are not required, but the required separation between the nuclear part and the chemical plant is not attained (2).
- IHX in reactor vessel, SR and SG inside containment. The parts of the containment which contains the heat exchangers are inertised with N$_2$ to prevent explosive gas mixtures (3).
- IHX, SR and SG are arranged in different cavities of the reactor vessel. A very large reactor vessel is needed due to this principle (pod diameter in the order of 6 m instead of 4.5 m) (4).

![Fig. 6: Vessel Dimension for 3000 MW$_{th}$-Plant](image-url)
Fig. 7: Principal Solutions for the Arrangement of IHX

- IHX, SR and SG are integrated in the same cavities in the reactor vessel. A double walled type tube element is used for the steam reformer heat exchanger. The requirement of space is nearly the same as in case of solutions without IHX (5).
- In the case of loop designs the reactor vessel itself in all the arrangements is the same, the other qualities are similar to the cases discussed before nr. 6 shows a loop arrangement corresponding to nr. 1.

Perhaps from the point of view of costs, feasibility and knowledge about components (hot gas pipes, bendings, penetrations, sampling systems for IHX-circuits) a solution similar to nr. 5 including an additional inertisation of the containment could be a good solution.

6. Data and Design of IHX-Systems

To give an impression and to show the feasibility of an IHX some data and figures are shown now:

It is assumed that all the reactor heat is transferred in 12 loops of a 3000 MWth plant. The IHX is divided in two parts: a heat exchanger for the upper temperature level of helium (970 ... 550 °C) \( \equiv \) high temperature IHX (HT - IHX) and a heat exchanger for the lower temperature level of helium (550 ... 270 °C) \( \equiv \) low temperature IHX (LT - IHX) (see fig. 8).
Both heat exchanger parts should consist of module type of elements as fig. 8 shows.

The data of the IHX-system can be taken from table 3. Fig.9,10 show the arrangements of the HT- and LT heat exchangers inside a loop. The hot helium coming from the nuclear reactor through a coaxial pipe is ducted to the inner part of the heat exchanger containing the coaxial tubes of the HT-IHX. From here the primary helium is led to the surrounding LT-IHX which consists of coaxial tubes too. All tubes are supported by a gas tight thick walled plate made for a temperature of 500 °C. Above this support there are two separated rooms for the secondary helium. One is made for an inlet temperature for the LT-IHX (200 °C) and for the HT-IHX (480 °C). For the hot and cold secondary helium internal or external sampling systems should be used.

Each loop contains a circulator for the primary helium with a power of 3.5 MWel. The diameter (5.0 m) and the height (15 m) of the heat exchangers are such that they can be integrated to podboiler vessels.
### Data

- **Reactor power**: 3000 MW
- **Reactor heat transferred by IHX**: 3000 MW
- **Number of loops**: 12

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Dimension</th>
<th>HT-IHX</th>
<th>LT-IHX</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Power</strong></td>
<td>MW</td>
<td>150</td>
<td>100</td>
</tr>
<tr>
<td><strong>Inlet temperature (prim.)</strong></td>
<td>°C</td>
<td>970</td>
<td>550</td>
</tr>
<tr>
<td><strong>Outlet temperature (&quot; &quot;)</strong></td>
<td>°C</td>
<td>550</td>
<td>270</td>
</tr>
<tr>
<td><strong>Inlet pressure (&quot; &quot;)</strong></td>
<td>bar</td>
<td>40</td>
<td>39.8</td>
</tr>
<tr>
<td><strong>Outlet pressure (&quot; &quot;)</strong></td>
<td>bar</td>
<td>39.8</td>
<td>39.7</td>
</tr>
<tr>
<td><strong>Mass flow (&quot; &quot;)</strong></td>
<td>kg/sec</td>
<td>68</td>
<td>68</td>
</tr>
<tr>
<td><strong>Heat transfer number (&quot; &quot;)</strong></td>
<td>kcal/m²K</td>
<td>1380</td>
<td>1230</td>
</tr>
<tr>
<td><strong>Inlet temperature (second.)</strong></td>
<td>°C</td>
<td>480</td>
<td>200</td>
</tr>
<tr>
<td><strong>Outlet temperature (&quot; &quot;)</strong></td>
<td>°C</td>
<td>900</td>
<td>480</td>
</tr>
<tr>
<td><strong>Inlet pressure (&quot; &quot;)</strong></td>
<td>bar</td>
<td>39.19</td>
<td>40</td>
</tr>
<tr>
<td><strong>Outlet pressure (&quot; &quot;)</strong></td>
<td>bar</td>
<td>37.95</td>
<td>39.19</td>
</tr>
<tr>
<td><strong>Mass flow (&quot; &quot;)</strong></td>
<td>kg/sec</td>
<td>68</td>
<td>68</td>
</tr>
<tr>
<td><strong>Heat transfer number (&quot; &quot;)</strong></td>
<td>kcal/m²K</td>
<td>1550</td>
<td>1460</td>
</tr>
<tr>
<td><strong>Overall heat transfer number</strong></td>
<td>kcal/m²K</td>
<td>560</td>
<td>520</td>
</tr>
<tr>
<td><strong>Log. temp. Difference</strong></td>
<td>°C</td>
<td>70</td>
<td>70</td>
</tr>
<tr>
<td><strong>Heat flux</strong></td>
<td>kcal/m²h</td>
<td>38280</td>
<td>36300</td>
</tr>
<tr>
<td><strong>Area</strong></td>
<td>m²</td>
<td>3371</td>
<td>2368</td>
</tr>
<tr>
<td><strong>Number of tubes</strong></td>
<td></td>
<td>2000</td>
<td>2000</td>
</tr>
<tr>
<td><strong>Dimension of tubes (outer diameter)</strong></td>
<td>cm</td>
<td>5.9</td>
<td>5.9</td>
</tr>
<tr>
<td><strong>Distance of tubes</strong></td>
<td>cm</td>
<td>0.8</td>
<td>0.8</td>
</tr>
<tr>
<td><strong>Length of tubes</strong></td>
<td>m</td>
<td>9.1</td>
<td>6.39</td>
</tr>
<tr>
<td><strong>Diameter of bundle</strong></td>
<td>m</td>
<td>~3.3</td>
<td>~4.7</td>
</tr>
</tbody>
</table>
IHX-High Temperature Section

- Arrangement of the High Temperature Modules -
7. Materials

Four main ways to transfer heat from the HTR-helium circuit to the chemical process (see table 4) are now in discussion.

- Coal gasification with steam
  \[ C + H_2O \rightarrow CO + H_2 \text{ (28 kcal/mole)} \]
- Steam reforming of methane
  \[ CH_4 + H_2O \rightarrow CO + 3H_2 \text{ (49 kcal/mole)} \]
- Olefin production from gasoline
  \[ C_{n}H_{2n} - C_{2}H_{4} + \text{heat} \]
- Water splitting by nuclear heat
  \[ H_2O \rightarrow H_2 + \frac{3}{2}O \text{ (heat in several steps)} \]

Table 4: Main Processes

<table>
<thead>
<tr>
<th>Process</th>
<th>Average Temperature</th>
<th>Helium temperature without IHX</th>
<th>Helium temperature with IHX</th>
</tr>
</thead>
<tbody>
<tr>
<td>Steam Reforming</td>
<td>900...1200 °C</td>
<td>950 °C</td>
<td>1650 °C</td>
</tr>
<tr>
<td>Coal gasification of hard coal</td>
<td>1000...1200 °C</td>
<td>1050 °C</td>
<td>1050 °C</td>
</tr>
<tr>
<td>Methanol steam</td>
<td>750...950 °C</td>
<td>950 °C</td>
<td>950...1000 °C</td>
</tr>
<tr>
<td>Olefin from gasoline</td>
<td>750...850 °C</td>
<td>950 °C</td>
<td>1000 °C</td>
</tr>
<tr>
<td>Water splitting</td>
<td>800...900 °C</td>
<td>1000 °C</td>
<td>1050 °C</td>
</tr>
</tbody>
</table>

Table 5: Extreme Temperature Requirements

Table 5 shows the extreme requirements in reactor outlet temperature for the different processes without and with IHX if temperature differences which are sufficient for effective heat transfer are included.

These temperature specifications of the helium require the following temperatures for the IHX: as an example for the steam reforming process an IHX with an maximum helium temperature of the secondary helium of 900 °C has a wall temperature of 940 °C. This temperature required is consistent with materials well known today from the conventional steam reformer technology as fig. 14 shows.

These strength values are competitive with an IHX design for a lifetime of $10^5$ h if the pressure difference across the wall in the hot parts of the tubes is nearly zero. Following the usual methods of calculations for the thick walled (5 mm) IHX tubes a safety factor in the order of more than 10 is included. Above all temperatures of 1000 °C new materials are needed. Even the steam gasification of hard coal could be done with helium temperatures of 900 °C in the gasifier (naturally with smaller throughput), therefore, it should be discussed very carefully whether hydrogen temperatures are of advantage.

Fig. 11: Strength of High Temperature Alloys for $10^5$ h
8. Safety Considerations

In fig. 12 and fig. 13 there are simple flow sheets for safety consideration of steam reforming systems without IHX. As preliminary analysis shows, the accidents of burst of steam reformer tubes and burst of steam generator tubes are no major problems for the plant. The reason is that the amounts of gas and water coming to the reactor core and to the containment after an accident are limited to small values, because the production of each reformer tube is below 400 Nm$^3$H$_2$/h and because the steam generator consists of a lot of parallel tubes with limited throughput. All accidents inside the containment by broken pipes or sampling systems can be tolerated easily if the part of the containment which contains the heat exchanger system is filled with an inert gas.

If the steam reformers are outside the containment, a big new problem of safety for IHX systems arises: Fast acting hot valves for the big pipes (1 m diameter), which penetrate the containment must be used, to avoid loss of radioactive helium to the environment during accidents.

![Flow Sheet for Steam Reformer without IHX](image)

**Fig. 12: Flow Sheet for Steam Reformer without IHX**
9. Discussion of the Advantages and Disadvantages of IHX-Systems

For the special process of steam reforming the following advantages of the intermediate system can be seen:

a) There is an effective separation between the nuclear plant and the process plant. Therefore inflammable gases (H₂ and CO) are not brought into the containment.

b) Deposition of radioactive fission products on the reformer tubes is avoided. The removal and replacement of catalyst and of tubes are thus not hindered by radioactivity.

c) There is possibly more scope in the choice of pressure for the separate circuits.

d) There are additional possibilities of isolation after accidents. A further barrier against the release of radioactivity into the air thus exists.

e) Hydrogen permeation into the reactor coolant circuit can be prevented more easily. The tritium which possibly permeates in the other direction, perhaps, may be removed more easily from the intermediate system than from the primary helium or from the product gas.

f) The possibility of corrosion of core components and fuel elements by steam and hydrogen is decreased. There is also no reactivity ramp after water leakage into the core. The specification of the surface temperature of the fuel elements (Tₚ ≥ 1050 °C) can be left out.

g) It may not be necessary to make the containment atmosphere inert.
The following are the disadvantages of an intermediate system:

a) Additional heat exchangers, ducts, blowers, and penetrations are necessary.

b) The net efficiency of a plant with an intermediate system is reduced because of the necessary pumping energy required for the intermediate fluid.

c) The maximum helium temperature available for the process is about 50 - 100 °C under the mean nuclear reactor outlet gas temperature. Either a higher nuclear reactor temperature is required (connected with higher fission product release), or the reforming temperature in the steam reforming furnace is lower. This leads to a lower methane conversion. A further possibility is to reduce the heat flux in the reformer tubes.

d) The operation of a heat exchanger with very hot tube walls (T ∼ 900 °C) is necessary.

e) The heat fluxes in the intermediate system heat exchangers are relatively low (k ∼ 500 kcal/m²h°C, \( \bar{q} \sim 25.000-35.000 \) kcal/m²h) in order to avoid excessive pumping power. This low heat flux leads to large-volume heat exchangers, (1 ... 1.5 MW/m³), which are expensive and difficult to accommodate in the integrated pod boiler system.

f) The penetration of helium ducts at T ∼ 900 °C through the top of the heat exchanger of the primary circuit as well as the containment is technically problematic.

g) Decay heat dissipation requires separate coolant circuits.

h) Hot header systems, valves and ducts are required for the intermediate system. The length of typical pipes is in the order of 40 m for some arrangements, which includes bending and compensations.

i) The addition of extra loops and components may possibly lead to a greater susceptibility of the overall system to failure.

j) Tritium removal can also be done in the primary circuit, if it is possible to do so in the intermediate system (for instance with titanium oxide).

k) The permeation of hydrogen does not seem to be a decisive criterion for an intermediate system according to the present results, if the steam fraction is high enough (H₂O/H₂ ≥ 1/1) in the reformer tube (as it normally must be to avoid carbon deposition on the catalyst).

l) Impurities in helium (CO, CO₂, H₂, H₂O, CH₄), which could at high temperatures and long operation life lead to structural
changes in the tube walls, will certainly cause less trouble
in thick-walled reformer tubes (15 mm wall thickness) than in
intermediate heat exchanger walls (~ 4 ... 5 mm wall thickness).
m) Eventual temperature losses on the long pipes between IHX and
steam reformer (~ 40 m) occur.
n) Overall costs for nuclear heat are about 20% higher when
intermediate systems are present.

Considering all of these arguments, there appears to be no com-
pelling need to introduce the extra complication and expense of
the intermediate heat exchanger, for the steam reforming process
with helium temperatures of 950 °C at reactor outlet. Perhaps a
very good solution would be to use double walled reformer tubes
to obtain an additional barrier for fission products. If there
are other reasons to use an IHX, the flow sheets 2a, 2b, 2c of
fig. 4 can be used as fall-back positions. There will not be
need to change the reactor designs for systems without IHX. The
steam reformer bundles would be substituted by IHX bundles. This
can be done because the power density in SR and IHX has nearly
the same value. As far as coal gasification is considered perhaps
there are other overriding considerations in favour of using an
intermediate heat exchanger:

a) The handling of coal and ash within the containment building
would be very difficult and undesirable.
b) The operation life and need for repairs of the gasifier are
more difficult to predict as compared to steam reformers.
Therefore these components should be placed outside the pri-
mary circuit, and arranged so that they can be independently
turned off.
10. Design Steps and Test Facilities for IHX

The development of IHX heat exchangers for steam gasification of coal or as fall back position for steam reformer application requires a broad activity in the future:

- Engineering and design work
- Material qualification for wall temperatures of $T_W < 920^\circ C$
- Development and qualification of materials for $T_W > 920^\circ C$
- Test on IHX bundles in different steps (modules with $\sim 100$ kW, bundles with a power of 1 MW, 10 MW)
- Test on special components of IHX (pipes with 1 m diameter, insulations, valves, bendings, penetrations, sampling systems).
- Safety analysis and licensing for IHX-systems.

For the tests on components in the KFA Jülich will be in the next years the HHV-plant ($N \sim 45$ MW, $T_{\text{Helium}} > 850^\circ C$, $\dot{m}_{\text{He}} \sim 150$ kg/sec, $p > 40$ b) available.

If the module type IHX-system will be accepted one can start to test the coaxial tubes in the following facilities:

- the 100 kg C/h-gasification plant of Bergbauforschung ESSEN contains a recuperator with a power of $\sim 0.5$ MW and a temperature of max. $900^\circ C$
- in the EVA plant of the KFA Jülich (1 MW, $T_{\text{He}}^{\text{max}} \sim 1000^\circ C$, $p \sim 50$ b) some tubes could be tested.
- In the future in the HHV-plant in Jülich a large bundle of tubes could be tested, if the helium temperature of the facility will be $950^\circ C$
- Additionally a facility with 10 MW power similar to the SUPER-EVA-plant for steam reforming could be constructed including an IHX in connection with the 1 t C/h gasification plant of Bergbauforschung/ESSEN, which will be built after 1977,

Material tests, especially corrosion tests, can be made in a lot of existing facilities: DRAGON, RISU, TURIN, KFA Jülich. Additional facilities for creep tests in helium atmosphere, burst tests, low and high cycle fatigue, must be installed.
SAFETY ASPECTS OF THE PR 500 FOR THE COMBINED PRODUCTION OF ELECTRICITY
AND DISTRICT HEAT

by

and G. Schroeder

1. PR 500 - A Small HTR for the Combined Production of Electricity and
District Heat

The companies Arbeitsgemeinschaft-Versuchsreaktor GmbH (AVR) (Joint
Experimental Reactor Company), Ruhrkohle AG and STEAG AG have jointly
established the Studiengesellschaft für Nukleare Fernwärme mbH (Association
for Studies on Nuclear District Heating) located in Essen. It is the
objective of this company to examine the technical, economic and legal
possibilities in connection with the use of high-temperature reactors
for district heating and, in particular, to carry out the necessary
preparatory and planning work.

As a basis of the present investigations a study is used (lit. 1) which
was completed in 1973 and which shows that the PR 500 station for the
generation of industrial process steam is almost competitive to fossil-
fired plants.

In consequence of the energy crisis, HTR heat is likely to be even more
attractive today. It is obvious, therefore, to consider the use of the
PR 500 for district heating as well. On account of the relatively high
heating load per unit involved, however, sufficiently high connecting
capacities are required for an efficient use. In this connection, it is
of interest to know that, as a first step, the German regional heat
transfer line, called Fernwärmeschiene Ruhr, with a connecting capacity

+ KFA, Julich, Germany
++ Arbeitsgemeinschaft Versuchsreaktor GmbH, Germany
+++ G. Schroeder, Steag, Germany
of about 810 MJ/s will be under construction by Steag, soon (Lit. 2). In the completed form, as discussed at present, the heat of up to 15 PR 500 could be fed in there.

Figure 1 represents a schematical diagram of the nuclear district heating system: The primary and secondary circuit are shown in the upper part of the drawing. A variable part of steam is tapped out of a condensation turbine and passed through a heating condenser, from which the nuclear heat transfer line feeds into the regional heat transfer line. By means of several heat transfer stations and the district heating line, the energy is brought to the consumers. Peak demand is covered by local, fossil fuel fired heating power stations.

A high-temperature reactor with a thermal reactor power of 500 MW is thus capable of an electrical net output of approx. 170 MW in pure condensation operation. The maximum possible heating load is approx. 295 MJ/s. Consequently, there is a higher efficiency in using such a multi-purpose plant.

On the whole, after a realisation of the above described system we expect:
- profitability
- security of energy supplies and
- the reduction of the environmental impact due to smoke and dust emission as well as waste heat.

2. **PR 500 Plant - General Safety Aspects**

The rentability of this reactor system is essentially governed 1. by the capital expenditures for the reactor, where the non-integrated construction system causes essentially lower container costs on account of the small reactor size, and

2. by the heat transport distances, where the siting of a reactor in the vicinity of urban distribution networks, i.e. in the vicinity of densely populated areas, provides the most economical solution.
This special situation gives rise to a number of safety aspects, some of which shall be discussed in the following.

Figure 2 shows the layout of the PR 500 plant. It is essentially characterized by 3 features:

1. the use of a containment, which is divided into three categories of compartments: the operators' room, the condensation cooler and the safety vessel which surrounds the entire primary circuit and which is filled with inert gas to prevent an inflow of air.

2. by the external placing of the 3 steam generators, where the cooling gas connection to the core is formed by 3 coaxial ducts, and

3. by a pebble bed core with OTTO-fuel-cycle.

The after-heat removal system makes use of the main steam generators which have a redundancy of 3 times 100 %.

An examination as to whether or not a separated after-heat removal system would be advantageous, has not been completed yet.

In order to be able to assess the risks involved in a non-integrated arrangement of the PR 500, a number of investigations relating to fracture mechanics have been carried out for the coaxial line and the steam generator pressure vessel. The essential results obtained are as follows:

In the case of both components, the criterium of 'leakage before fracture' is met even at room temperature, i.e. the failure of a component is announced by detectable leakage early enough to permit a shut-down of the reactor.

An investigation into the subcritical fracture growth conditional upon starting and shut-down operations shows a more than 60-fold
safety of the operating load cycle rate as against the critical load cycle rate even on the basis of extremely pessimistic assumptions.
But in spite of the extremely small probability of a failure, the consequences have to be examined in detail.

3. The Coaxial Duct of the PR 500

As to the consequences of such a failure, particular significance must be attributed to the coaxial ducts for two reasons:

1. they provide the cooling gas connection between core and heat sink and
2. a fracture would cause the maximum possible pressure transient in the primary circuit.

In fig. 3 a preliminary layout of the coaxial duct is shown: The hot gas (40.5 bar; 850 °C) flows through the inner tube to the steam generator; the cold gas (40 bar; 250 °C) is led back through the interspace between the hot gas tube and the outer pressure tube. The pressure tube is water-cooled and insulated by metal foils. For the hot gas tube a carbon insulation is provided, which reduces its temperature to about 260 °C. Although in normal operation there only occurs a radial pressure difference of about 1 bar, this hot gas tube is rated for a system pressure of 40 bar.

In addition, the following properties are relevant with regard to accident prevention:
- Interchangability of the entire line and all of the individual components
- possibility of inspecting the hot gas collecting chamber after flanging off the duct and steam generators.

To reduce the effects of a coaxial duct damage an additional construction can be provided, which limits the gas flow in case of an accident and simultaneously serves as a splinter protection. Two designs seem to be feasible:
An arrangement outside the coaxial pipe or a second steel tube arranged inside the pressure tube which, under normal operational conditions, is not pressurized.

4. The Transient Pressure Behaviour after a Coaxial Pipe Rupture

Although the probability of a failure of the coaxial pipe is very low, the consequences of a guillotine-type fracture on the reactor are being studied.

For the dynamic pressure reduction operations in the system a digital multi-point model has been developed (lit. 3). This computer program permits the calculation of pressure equalisation processes between random intercoupled volumes with a random number of gas components. The conservation laws of mass and energy are applied to each volume, and the momentum theorem is applied to the flow parts.

By means of this code the pressure transients and the time-dependent pressure differences at various locations of the primary circuit have been evaluated which result of a guillotine fracture of the coaxial pipe.

Figure 4 gives an idea of the time-dependent pressure distribution along one of the two hot gas pipes that have remained intact: an oscillation of the radial pressure difference with a max. amplitude of less than 3 bar takes place, which results in an alternately internal and external pressurisation of the hot gas duct and which is finished after about 1/4 sec.

In the light of the calculations so far carried out, there is every reason to suppose that the other core internals of the reactor are not damaged either by the effects of the local pressure differences.
Although investigations are still under way, the results obtained so far are, to say the least, not indicative of any major disadvantage involved in the non-integrated type of construction.

I would now like to report on two investigations which do not exclusively relate to the non-integrated type of construction.

5. Transient Core Behaviour after an Ingress of Water

This accident has been analysed by means of a computer code that solves the one-dimensional, two-group diffusion equation in axial direction.

For the present analysis a pipe failure in one of the three steam generators during normal power operations has been assumed. Previous calculations have shown that at the worst a total of about 1700 kg water will leak into the primary circuit within 50 sec, if none of those countermeasures are taken which are provided to limit the amount of ingressing water; in this case and provided that the safety valves do not respond, the total pressure will be raised to 53 bar.

Fig. 5 gives the time-dependent behaviour of the total reactor power P(t), normalized to the steady-state value of 500 MW.

Curve "a" describes the case when no countermeasures are taken. The power rises as long as the steam content in the primary loop increases. After 50 sec, when the ingress is completed, the temperature feedback causes a sharp reduction and the reactor stabilises itself on a new power level.

The dotted curve "a" shows the behaviour of the total power if a scram is triggered after 15 sec; the prompt decrease clearly demonstrates that the shutdown system copes with these presupposed accidents.

Curve "b" in this figure gives the time-dependent behaviour of the maximum fuel element temperature, if no scram is considered. The peak temperature of 1400 °C lies considerably above the specified
limit of 1300 °C; but this value is reached only in a relatively small part of the core and could be endured for at least some hours in the unlikely event of an additional failure of the shut-down system.
If a scram will be triggered, a maximum temperature of roughly 1000 °C is reached.

6. Core Flow-State during Non-Power Operation

In conclusion, the results of an investigation into the transient core temperature behaviour and time-dependent cooling gas flow will be dealt with.
For this purpose, a two-dimensional computer programme system has been developed, permitting the calculation of all flow states between a pure natural convection and a pure forced convection as well as their impact on the transient temperature behaviour of the reactor. Particular emphasis has been placed on the determination of the core-internal circulation flow.

Fig. 6 shows the relationship between the present core design and the axial-symmetrical computation mesh.

Two typical flow states after the shutdown of the reactor will be briefly described:

1. the natural convection without any blower operation
2. low blower operation (1 %) for after-heat removal.

The value of the mass flow moved by natural convection is highly dependent on the system pressure.

Fig. 7 shows the computer-generated flow pattern of the core in the case of a system pressure of 40 bar for the pure core-internal circulation-flow as well as for a cooling gas transport via the steam generators.
In terms of thermodynamics, there will result a heating of the upper core areas and a cooling of the lower ones as well as a reduction of radial temperature differences in the core in conjunction with an elimination of temperature peaks. Thus, a considerable portion of after-heat generated in the core can be removed through the core walls or via the steam generators.

Fig. 8 illustrates the mass flow through the core during the application of the after-heat removal system. On the right-hand side, where the flow diagram for a 3 bar equilibration pressure is shown, the cooling gas moves through the pebble bed, thus providing a very smooth cooling. Because of the decreasing temperature level of the upper core areas, the control rods have to be inserted. On the left-hand side, the 1% after-heat removal at full system pressure is shown. The mass flow supplied from the blowers is forced to the outside of the core by the circulative convection which still increases with time.

The resulting thermodynamic effects are as follows:
- The core walls and the entire gas circulation are cooled very effectively.
- Owing to this circulation flow and the upward heat transport involved, a heating of the core bottom is effectively avoided.
- Despite the cold downward cooling gas flow, there is a rise in the mean temperature of the upper areas of the pebble bed.

This fact is of great significance, for it points to a possibility of keeping the reactor subcritical during longer periods of time even without inserted shutdown rods by a simple throttling of the cooling gas blowers.
In summary the following statement can be made:

Although the results of a few major investigations, such as gas-dynamic studies on the hot gas chamber or clarification as to fission product transport into the containment after a depressurization accident are only at a preliminary stage, we see a chance to get a license for building the PR 500 in the closer vicinity of populated areas.

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THE STATUS OF INFORMATION REQUIRED FOR AN INDEPENDENT
ANALYSIS OF RISKS ASSOCIATED WITH AN HTR

by

F.P O. Ashworth and H.J. de Nordwall

ABSTRACT

The information and modelling capability required to assess, independently of a vendor, the risk associated with an HTR sited in Europe is reviewed. Directions for future technological development, system validation and performance surveillance are suggested.

OECF HIGH TEMPERATURE REACTOR PROJECT
DRAGON,
1. INTRODUCTION

The large high temperature reactor awaits commercial acceptance and licensing in Europe, i.e. neither the financial nor the radiological risk has been fully accepted.

In the United States a substantial commercial commitment has been made by utilities and licensing of the first large plant has proceeded to the point where the National Regulatory Commission (NRC) has issued a favourable report[1] which awaits review by the Advisory Committee on Reactor Safeguards. Fort St Vrain has received licensing clearances, but is experiencing prolonged commissioning problems; other gas cooled reactors are experiencing discouraging construction delays.

From a technical viewpoint we have a reactor system with excellent, well-accepted, inherent safety features,[2] whose engineered safeguards, particularly containment, auxiliary cooling and accident control systems await final acceptance under U.S. licensing rules. However, issuance of a license only implies a judgement that the Public beyond the site boundary are adequately protected from radiation, with the implicit caveat that the single failure criterion and other concepts used to limit the scope of accident analysis are in fact adequate.

No wider ranging risk analysis of the type performed recently for the light water cooled reactor by Rasmussen [3] has been published, perhaps because the detailed plant-specific information required is not available for the large incompletely designed plants and because the basis for extrapolating the technology from present experience is not clearly validated (This is not to imply that this latter situation was any different for the LWR's when they were at a similar stage of commercial development.)

The HTR's built clearly provide proof of important principles relating to containment, fuel performance, control and materials, but they are not prototypes in the sense that future plants will be built from identical units.[4]

To fill these gaps constructively and better define commercial and ecological risks we need to organise expertise and data acquisition with the objective of justifying the extrapolation of existing knowledge to large plants.

Such a study should be independent of the vendors' judgements but should certainly not spurn his expensive generated facts. Furthermore, we are confident that such an independent study will enhance the vendors' position, as has been the case with the Rasmussen study, rather than damage it, and assist him in formulating commercial solutions to his problems.

These objectives could be met in the short term by attempting a wide ranging risk analysis following the formats used by Rasmussen and the Safeguards Division of the UKAEA[5] which would serve to focus attention on where future Rand D funds should be spent to reduce risks.

In the longer term, emphasis needs to be on reactor performance surveillance, aimed at increasing our confidence in the system, for which Europe would need a prototype HTR. This should be built on a grid where the demand for power is low enough to permit more frequent and more extensive inspections, and even experimentation, than would be desirable for a base load station.
The construction of a prototype or lead station would force the safety project to give attention to practical commercial issues. For example, during the design stage attention might be focussed on component performance limits, system analysis, trade-off's between engineered safeguards and cost, the design of surveillance instrumentation and the analysis of causes of delay in other plants.

Later, attention would shift to relating quality, cost and risk, ensuring compliance with quality standards, qualifying the contractors and demonstrating that the construction plan was viable.

Finally, surveillance over the reactor's working life-time from commissioning to burial will document performance, generate confidence and alert the industry to any weaknesses, such as long term material or design problems ahead of the base load stations built in its footsteps.

The type of organisation needed is shown in Fig. 1. The key functional unit is the risk analysis unit since it continuously compares externally defined risk goals to evaluated expected risks in order to determine the need and direction of further work and when to quit. Perhaps the second most important function is the Information Centre's. This unit is concerned with marketing the project's output, and maintaining a depository library of reports, published design data and computer codes.

With such a wide market among utilities, vendors, bankers, national licensing organisations - particularly those that cannot afford supporting national laboratories and environmental groups - the programme's management will need considerable feed-back from users in relation to the depth and precision of information required about a particular topic. If governments supporting the programme do not provide realistic goals as distinct from exhortations to minimise all risks simultaneously, the programme will never end. Hence the user's advisory committee, which is outside the programme, has been charged with formulating the complete set of risk goals needed to ensure a balanced programme. It must contain utility representation.

Technology development and validation would be shared with promotional agencies, and organised strictly on a component basis, viz Fig. 2, the safety project's role being confined to ensuring that the data and technology bases are wide enough to permit accident analysis at meaningful confidence levels and, if necessary, sponsoring additional validation experiments or full scale tests of novel equipment to reduce first-of-a-kind risks.

The Demonstration Reactor Surveillance Task would have to be shared with a consortium of utilities and governments. It would not be a substitute for component test beds, where necessary, since they would be altogether more flexible.

The duration of the Programme would be determined by the quality of the technology and its rate of advancement.

The U.S. draft plan for an HTGR related safety programme [5] arbitrarily selected a ten year timespan, without specifying the conditions for termination and withdrawal of government support, or the transfer of effort to more advanced systems.

We would hesitate to guess life-times for the functional units in Fig. 1; 10 years may well be adequate; but substantial progress should be possible in a shorter time given a viable-sized group.
Surveillance programmes must last beyond 1985, since as a minimum, fuel performance in a prototype power reactor must be followed until recycle fuel has been irradiated for its full life-time and the major structural units and the steam generators have run for the plant life-time, i.e. 30-40 years.

During this period the Safety Programme may fluctuate in size and in emphasis; it could also expand its scope into advanced systems, but it must not be disbanded or the benefits of the valuable long-term effort will be lost.

On the economic side, we believe that in later years the coordinating risk analysis group could become quite small and that many of them exist already. Furthermore, in the important area of the PFRV the performance of the Magnox and AGR vessels will provide a 20 year lead in time.

ORNL estimated that their accident analysis unit would cost $8,000,000 over 10 years at 1975 prices. Surveillance and validation activities in the technology area and the maintenance of useful reporting, based again on the ORNL estimates, require $19,000,000 for the first 10 years. The annual cost can be expected to fall quite rapidly once the fuel has been proven.

The cost of large scale test facilities and the demonstration reactor are not included.

We believe that when national HTR development plans have been fully formulated and costed a degree of internationalisation will be found to be desirable. European countries acting individually would not be able to support the degree of overlap inherent in completely separate promotional and critical organisations on the ERDA-NRC model, and since fission products know no frontier, we favour an international approach to risk control.

We shall now discuss how work done to date could be applied to producing an overall risk analysis.

2. STATUS OF HTR ACCIDENT ANALYSIS

The functions to be reviewed in this chapter are shown diagrammatically in Fig. 3. External functions are those carried out by sponsoring agencies working through the programme advisory committee, or by information users for their own benefit.

2.1 Formulation of Acceptable Risk Criteria

Criteria aimed at the control of radioactivity are currently formulated in terms of permissible doses. These criteria are set by balancing what is achievable with today's technology against a desire to reduce discharges of man-made radioactivity to a level that is demonstrably negligible when compared with the natural background. Some relaxation is permitted in licensing accidents. The single failure criterion has caused published accident analyses to rely heavily on the low probability of event sequences involving multiple failures, without encouraging definition of acceptable performance criteria for the relevant engineered safeguards.
Any agency sponsoring a wide range risk analysis programme will have to face the need to formulate risks in terms of probabilities if it is to appear responsible and be capable of guiding and terminating its programmes logically. Criteria for acceptable financial risks have been given little public attention in atomic energy literature.

2.2 Accident Identification

The faults most commonly considered in HTR safety analyses relate to containment failures, loss of cooling capacity and water entering the core, though the analysis of many other situations is required for licensing. [7] Published analyses of these accidents and the arguments supporting choices of accident end-point may be read in the Preliminary Safety Analysis Reports for the Summit [8] and Fulton [9] Plants. More recently GAC has produced a generic safety analysis report for use by utilities purchasing their reactors.

A list of postulated major HTR accidents is given for convenience in Table 1.

The following additional accidents have also been examined in some detail:

(a) Delayed initiation of emergency cooling. [10, 11]
(b) Total loss of cooling. [12, 13]
(c) Failure of a steam generator caused by a depressurisation. [14]

In all these accidents the reactor is assumed to shut-down safely.

Recently in Europe more attention has been focussed on man-made external events rather than with climatic freaks such as tornadoes and earthquakes, or the loss of off-site power for unstated reasons. For example, accidents initiated by crashing aircraft, exploding liquid petroleum gas clouds from passing tankers and terrorism have been suggested.

What is needed now is a systematic way of documenting the decisions about accident sequence that have led to the rejection of a large number of event sequences on the grounds of triviality, i.e. negligible consequence; similarity to other analyses already carried out, or low probability. While the expert may accept many such judgements a priori the layman is disinclined to, and a lack of confidence can result. Broadening of the scope of analysis to include financial risk may also change one's views about the significance of an event that causes a month's loss of power generation, but has no other environmental impact.

Attempts have been made by GAC, [15] and others to use accident frequency estimates [16, 17] to justify accident end-point selection and to guide technology programmes. Rubel of ORNL [10] proposed a method of choosing accident sequences that could be convincing to a layman and at the same time permit a large accident analysis programme to be closely documented. It also permits consequences of time delays of different duration to receive the emphasis they deserve in HTGR technology.
A number of computer programmes using Monte Carlo or synthesis methods have been developed to determine the probable outcome signals entering the network of switches and delays used to represent decision or fault trees. [18, 19] Noted, the programme used by SRD, has been applied to a reliability evaluation of the Fort St Vrain emergency diesel engine generator system. [20]

2.3 **Plant Dynamic Response Analysis**

The step following potential accident recognition is the determination of the plants' responses to the initial perturbation in terms of temperature, pressure, coolant composition and stresses in components. This analysis defines the working conditions of components and the parameters to be used in calculating discharges of coolants, heat and radioactivity to the environment.

The precision required for this calculation is far greater than for the calculation of fission product discharges, firstly because many of the quantities derived using its output are strongly temperature-dependent, and secondly because the margins to failure in the primary circuit are believed to be smaller than in the graphite core. This has led to the development of very sophisticated proprietary computer codes by reactor designers. The development effort in generating a comparable independent analytical capability to TAP [21] for HTR system analysis or trying to marry existing codes for individual component response analysis is daunting to the accident analyst, who seeks a high degree of spatial resolution at a limited cost and wishes to be sure he has not missed an important region of the core or the steam generator. Brookhaven National Laboratory is modifying RELAP 3 [21] for HTR systems analysis since TAP is proprietary.

The Dragon Project, while opposed in principle to marrying complex computer codes, is combining the core heat transfer and neutronics code CONSTANZA with TUBER, the code developed to calculate carbon removal, in order to gain a necessary increase in the precision with which the consequences of a severe steam accident can be computed.

In the analysis of gas turbine circuit transients, an analogue model in which individual elements have been represented by digitally computed transfer functions has been used successfully. [23]

Validation of predictions of plant performance under accident conditions is difficult. But even more difficult is ensuring that a demonstration experiment models the most important event sequence.

Numerical accuracy can be checked by inter-comparisons in which control over a proprietary code is never relinquished - if results agree.

Validation of the physical model is more difficult. Given an experimental reactor like Dragon with its computerised logging system, limited transients may be studied experimentally, as described by Ashworth. [24] A prototype would of course be more convincing. Validation of plant responses to larger transients clearly requires special test facilities which will almost certainly be confined to single components or sub-systems. Examples of existing facilities are the thermal cylinder PCRV model at ORNL, the large shaker tables.
used for core seismic testing, the insulation testing facilities in France, and the large heat transfer rigs at KFA and in the CEA laboratories at Saclay.

In Paper 1 [24] of this conference a boiler-circulator module test facility, which could be used to simulate depressurisation and tube failures in addition to its prime purpose of proving steady state performance, and a high pressure steam loop for the simulation of water ingress accidents, are proposed.

Codes developed for other reactor systems and published can be usefully adapted to HTR problems. This is true of many physics codes, codes addressing heat and fission product behaviour in tight secondary containments and codes describing fission product and heat dispersion beyond the reactor.

2.4 Component Response Analysis

The major question being posed in the early stage of the accident analysis is: will the key components survive the transient. A quantitative answer involves an assessment of three points, the probability and mode of failure during the transient and any repair period in which a safeguard is still in use, the probability of random failure and the probability that external influences or the mal-operation of other components involved in the accident will cause failure.

From the point of view of a status report the important question is: what sources of performance and reliability data and engineering judgement are available to the analysis team since setting up a data collection, storage and review system geared to dealing with new technology represents a major long term investment that should be shared between as many industries as possible.

In the U.S. the acquisition and collection of HTR related data is being carried out by GAC for the Government following discussions about power industry sponsored and national laboratory based systems. In Europe, Systems Reliability Services, an organisation sponsored by the UKAEA, is available to carry out data collection, storage and critical review on a contract basis and has published specific assessments of component reliability. In Germany, the Institute of Reactor Safety has estimated reliabilities of key water reactor components [17, 25] from field observations and other sources. Given these sources of data and care in extrapolating reliability data to new conditions we feel that a European programme can be well served in this field. We would not propose funding methodology development in this area, we would rather expect those offering service to do this as a capital investment.

2.5 Summary

We believe that the means to make a useful probabilistic study of the accident response of a steam-cycle HTR independently of commercial vendors exist.

3. STATUS OF THE COMPONENT TECHNOLOGY REQUIRED FOR RISK ANALYSIS

In this section we shall discuss how the goals of an overall risk analysis and the properties of components of the reactor and fission products are related. In doing so we will concentrate on identifying areas where effort appears to be lacking and reasonably short term benefits such
as easier licensing and commercial acceptance can be expected. The long
term benefits are almost entirely those associated with proving the
technology by monitoring the performance of a prototype or an early power
plant.

The further constraint that the programme cost should be minimised
will also be recognised. In this context this implies that one should
not compute the consequences of improbable events too precisely. It also
implies that one's risk goals must be defined before one starts; the
excuse that this is difficult and politically sensitive is no reason for
not making the attempt; politically responsive bodies able to do so
exist already (e.g. IAEA).

The most efficient analysis would probably be one that proceeded
infinitesimally slowly, but the need to generate power means that we must
use some intuition to determine what will be required in later iterations
of the sequence shown in Fig. 3.

The bases of our recommendations is a family of tables like
Tables 2, 3 and 4.

In column 1 we have listed systematically what can happen, without
regard to its probability, to three major components, the core, the steam
generator and the secondary containment.

Potential causes and consequences are listed in column 2 on the same
basis.

Columns 2 and 3 will be formally represented by fault and decision
trees respectively when priorities have to be considered.

Clearly if several coincidental events or a long sequence of causally
related events is required to reach a given accident end-point one would
not assign high priorities to frequency or consequence analyses in that area.

The information needed to quantify the accident frequency analysis
using the method outlined in Section 2 above is listed in column 4. That
required for consequence analysis is given in column 5.

Finally column 6 gives an opinion of current priorities based upon
our assessment of the technology, including that in the USAEC's HTGR Safety
Program Guide.[6]

In assigning priorities for development work aimed at improving
accident descriptions we have been careful to bear in mind the long time
constants of the high temperature gas-cooled reactor with regard to its
thermal responses and fission product transport within its core.

The lists of accidents in Tables 3-5 are not claimed to be exhaustive,
though in the final analysis one must do one's best to make a complete list.
The diversity introduced by looking at accidents from different standpoints
(Tables 3, 4 and 5) creates confidence that an important sequence has not
been missed. The fact that the same components and perturbations occur in
many different accident sequences permits a substantial reduction in the
number of accident analyses needed.

Very significant differences in accident consequence analysis
priority arise when the options for fission product control available to
the HTGR designer are considered. In our analysis we have assumed that a
programme purporting to address potential users who have not made their choice would automatically include consideration of options such as vented versus tight containment, maximum practicable confinement of fission products in the core and different approaches to core auxiliary cooling.

An examination of Tables 3-5 and the GAC's analyses of HTR accidents indicates very clearly that rather long sequences of unfavourable events are required to create consequences that approach U.S. Government Guidelines. Damage to the plant is also relatively minor. However, some degree of damage analysis is needed to define even the smallest risk to help the plant designer set reliability goals for engineered safeguards such as auxiliary cooling systems. Open-ended risks encourage open-ended speculation.

With respect to the primary containment, which includes the PCRV liner, its tendons and its penetrations, we have accepted the recommendations for further work made by ORNL and GAC[6] namely that emphasis should be on:

(a) Improving descriptions of the mechanical properties of concrete over long periods and at higher temperatures;
(b) Developing a general failure criterion for concrete for use in analysing the performance of vessels with complex stress patterns;
(c) Formulating and testing accepted design bases for penetration closure and throttle designs;
(d) Development of material with larger margins to thermal failure for use as hot duct and plenum insulation and liners;
(e) Formulation of standards for protecting tendons from corrosion; and
(f) Long term surveillance.

No firm recommendation was made for further large-scale testing of PCRV heads to failure, but there was a feeling that one would not easily be accused of being excessively cautious.

In the instrumentation area the highest priorities were given to:

(a) Further development of water and in core neutron flux monitors;
(b) Thermometers for use in controlling restoration of helium flow following interruption of cooling; and
(c) Lower cost fuel failure location systems.

A need for incipient failure detection was recognised.

4. CONCLUSIONS

4.1 Programme Feasibility and Scope

Europe is able to mount a useful collaborative HTR risk analysis programme whose management could be modelled on the familiar Dragon Reactor Development Project provided that:

(a) Objectives, markets, scope and work distribution are defined rapidly in European terms. Special care needs to be taken to avoid translation errors in adopting foreign guidelines and procedures.
(b) Specific plants can be designated for study.

(c) Long term funding and collaborative agreements on at least a 5 year rolling budget basis can be guaranteed to justify set-up costs and generate confidence that surveillance will be maintained.

A budget is not being recommended in this paper since policies for industry assistance must be formulated first. Certain facilities such as the Dragon reactor, the SYREL data bank, planned and existing loops and the Ispra computer code library would be expected to play important roles.

4.2 Technical Programme Initiation and Content

Further programme plans and status reports are not necessary since the U.S. ERDA has commissioned a series of need assessment reports that, together with GAC's licensing topical reports series, should give an adequate overview of non-proprietary 1975 HTR technology.

What is needed are reference manuals for designers and safety analysts. These will only be produced following attempts to use the technology, for example in analysing risks associated with a European prototype. In addition we feel that a library of HTR computer codes and an HTR Information Service to keep track of the fragmented technology would repay immediate investment.

Based upon existing programme plans available to Dragon we recommend the following topics for study during the first programme iteration. From then on the Programme should generate its own work plan and termination signals.

4.2.1 Accident Analysis

The consequences of HTGR design basis accidents are relatively mild if the system responds as designed. The programme's initial thrust in the accident analysis area should therefore be as follows:

(1) A careful examination of the logic that limits current reference accidents, following the patterns set by ORNL, Rasmussen and SRD.

(2) Analysis of the reliability of components whose malfunction could cause the accident to proceed to a significantly more dangerous conclusion. Relevant items would seem to be the secondary containment, the control systems of the main and auxiliary heat exchangers and the reactor control systems. Analysis of the boiler control system is justified specifically by the concern that its maloperation could cause a rapid depressurisation to be followed by water ingress, the others by the need to define ultimate accident consequences further.

(3) The plant dynamic response group would be advised to work initially on what temperatures would be experienced by various primary circuit components following delayed attempts to restore circulation, the plant's response to failure of all cooling systems and the more extreme consequences of steam ingress.
4.2.2 Technology Development

The primary objective must be generation of commercial confidence by experimental programmes aimed at reducing financial risks. The following long range tasks should be put in hand immediately:

(a) In-reactor validation of specific quality and performance prediction for the fuel under normal conditions. The fuel performance parameters $\phi$ and $\phi'$ directly define the principal accident source term, and circuit accessibility for inspection and maintenance. Indirectly they contribute to containment and siting evaluations.

(b) Verification that the reference steam generator is able to withstand stresses induced by the coolant under normal conditions and certain transients.

Item (a) would be fulfilled by the operation of a European prototype of Dragon with instrumentation capable of monitoring fully individual fuel zone performance. On-going work at Peach Bottom and surveillance being planned for the Fort St Vrain core will supply generic support.

The facility required for item (b) has been discussed in Paper SNI 4/1 by Ashworth et al. Again actual prototype operation would generate additional confidence.

The secondary objective should be to begin the new co-ordinated experimental work that can be anticipated to be needed in the next few years. Suggested high priority tasks have been listed below:

(a) Validation of predicted fuel failure modes and criteria using reference fuels under accident conditions. Fuel failure and the subsequent irreversible rise in the primary circuit inventory of long-lived nuclides such as $^{137}$Cs is believed to be more important than the actual fission product releases during most HTGR transients because the inherent delay time of the HTGR acts to minimise transient severity at the same time as small diffusion coefficients slow down release. Experimental observations of fuel behaviour to date have been conflicting for reasons that were not always clear. [26-31]

(b) Simulation of the reaction between part of a block element and steam should be carried out at realistic pressures and steam concentrations as a validation of damage predictions.

(c) The high temperature thermodynamic and transport data for U, B, Th, Pa and fission products available for the analysis of a situation following loss of cooling or failure to trip on demand should be reviewed.

(d) Consideration should be given to post accident control of slow fission product release from a core by using the time available between loss of cooling capability and
fuel failure to restrict the path of fission products through failed penetrations with local fission product adsorbers.

On-going work on the following topics should be maintained:

(a) Experiments aimed at establishing the 40 year capacities of the primary circuit for iodine and caesium under normal conditions. A failure to decontaminate the coolant would have a large effect on normal discharges of these nuclides if no other attenuator were added to the release chain.

(b) The kinetics of iodine displacement from steel surfaces by steam. These experiments seek to determine whether the prolonged contact experienced in certain postulated accidents could result in larger quantities of iodine entering the secondary containment if relief valves lift.

(c) Definition of the conditions under which aerosols could play an important role in fission product redistribution following a depressurisation. This analysis is needed before embarking on large scale experiments aimed at justifying very small re-entrainment fractions.

(d) Experiments in Dragon aimed at explanation of unexpectedly high corrosion rates at high helium flow rates and low water concentrations.

(e) In-reactor and laboratory work on caesium and silver transport in graphite.

5. ACKNOWLEDGMENT

The authors are indebted to their colleagues in the U.S.A. and Europe for advice on what is needed to launch an HTR in Europe. The answer was always the same - money.

Finally they wish to thank the Chief Executive and Board of Management of the Dragon Project for their permission to publish this paper.
6. REFERENCES


<table>
<thead>
<tr>
<th>Accident Type and Initiating Event</th>
<th>Hypothetical End-Point(^8)</th>
<th>Changes in Fission-Product Distribution</th>
<th>Fission-Product Release Routes</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Primary Containment Failure (PCHR)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Failure of penetration, PCHR liner, or safety relief system.</td>
<td>Loss of coolant pressure.</td>
<td>Primary coolant and associated gas- borne fission inventory released to secondary containment.</td>
<td>Release via leaks in secondary containment and filtered effluent through stack; or Release through containment vent valve and stack.</td>
</tr>
<tr>
<td></td>
<td>Auxiliary heat removal system being used to remove decay heat. Main system available.</td>
<td>Fraction of deposited fission products swept into secondary containment.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Secondary containment filled with helium containing fission products.</td>
<td>No change in distribution of fission products in core.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>No mechanical damage to components, e.g. no associated steam leak.</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Primary Cooling System Failure</strong></td>
<td>Reactor cooled by auxiliary system, which is assumed to operate before core becomes so hot as to cause coolant circulation to be a hazard.</td>
<td>None expected.</td>
<td>As in normally operating plant.</td>
</tr>
<tr>
<td>Interruption of forced circulation.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Core Channel Blockage</strong></td>
<td>Assumed blockage detected, located, and removed if necessary.</td>
<td></td>
<td>As in normally operating plant.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Reactivity Excursion</strong></td>
<td>Reactor shut down, with normal cooling available.</td>
<td>Some fuel damage.</td>
<td>As in normally operating plant.</td>
</tr>
<tr>
<td>Accelerated control rod withdrawal.</td>
<td></td>
<td>Some loss of fission products from core to primary coolant circuit.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Secondary Coolant System Failure</strong></td>
<td>Steam enters reactor.</td>
<td>Certain fission products in primary circuit may become volatile in steam.</td>
<td>As in normally operating plant, or by PCHR relief valve it if opens.</td>
</tr>
<tr>
<td>Failure of tube in high-pressure section of steam generator.</td>
<td>Steam reacts with core to produce CO + H(_2).</td>
<td>No fuel damage expected.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Reactor cools, thermal reactions stop. No loss of containment, i.e. no associated breach of primary containment (PCHR).</td>
<td></td>
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<td></td>
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</tr>
<tr>
<td>Steam ingress with water monitor failure in leaking loop.</td>
<td>Relief valve lifts and closes.</td>
<td>Partial loss of fission products from primary circuit to containment.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Reactor cooled using CACS.</td>
<td></td>
<td>Release via leaks in secondary containment and filtered effluent through stack; or</td>
</tr>
<tr>
<td></td>
<td>Steam reacts with core to produce H(_2) + CO in containment.</td>
<td></td>
<td>Release through containment vent valve and stack.</td>
</tr>
<tr>
<td></td>
<td>Containment isolated.</td>
<td></td>
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</tr>
</tbody>
</table>

\(^8\) Reactor always assumed to shut down though not necessarily in response to first demand. All systems assumed adequate for any seismic perturbations encountered either by the primary design system or a back-up system.
<table>
<thead>
<tr>
<th>Accident Type and Initiating Event</th>
<th>Hypothetical End-Point</th>
<th>Changes in Fission-Product Distribution</th>
<th>Fission-Product Release Routes</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reheater failure.</td>
<td>Primary coolant enters turbine, some steam enters reactor.</td>
<td>Gas-borne fission products in secondary coolant system.</td>
<td>Associated with leakage from turbine condenser via air ejector to stack.</td>
</tr>
<tr>
<td>Steam loop pipe rupture.</td>
<td>Steam in turbine hall or in secondary containment.</td>
<td>Tritium in secondary coolant discharged to secondary containment or turbine hall.</td>
<td>For break within containment building, tritium lost via stack, leakage, and controlled-liquid-disposal system. For breach in turbine hall, tritium lost directly to atmosphere and via controlled disposal system.</td>
</tr>
</tbody>
</table>

* Reactor always assumed to shut down though not necessarily in response to first demand. All systems assumed adequate for any seismic perturbations encountered either by the primary design system or a back-up system.
<table>
<thead>
<tr>
<th>Event/Failure</th>
<th>Potential Causes</th>
<th>Potential Consequences</th>
<th>Information Needed for Probability Assessment</th>
<th>Information Needed for Consequence Assessment</th>
<th>Priority</th>
</tr>
</thead>
<tbody>
<tr>
<td>Control rods.</td>
<td>Unexpected variability in material properties and service conditions.</td>
<td>Excess reactivity, power and temperature. Damage to FCRV.</td>
<td>Control mechanism reliability.</td>
<td>Thermal and neutronic analysis. Failure criteria. Fission product transport analysis.</td>
<td>3</td>
</tr>
<tr>
<td>Event/Failure</td>
<td>Potential Causes</td>
<td>Potential Consequences</td>
<td>Information Needed for Probability Assessment</td>
<td>Information Needed for Consequence Assessment</td>
<td>Priority</td>
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<tr>
<td>Failure in Normal Service</td>
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</tr>
<tr>
<td>Tube sheet failure.</td>
<td>Unexpected material degradation and variability in service conditions.</td>
<td>Steam in primary circuit.</td>
<td>Material variability.</td>
<td>Validation of failure criteria with emphasis on creep and failure.</td>
<td>1</td>
</tr>
<tr>
<td>Tube failure.</td>
<td>He discharged to environment through turbine.</td>
<td>Fission product to containment or environment.</td>
<td>Reliability of water detection and reactor shutdown systems.</td>
<td>Component test facility.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Fission product to containment or environment.</td>
<td></td>
<td>Metal-coolant compatibility.</td>
<td>Fission product displacement, evaporation and adsorption data.</td>
<td></td>
</tr>
<tr>
<td>Accidents</td>
<td></td>
<td></td>
<td></td>
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</tr>
<tr>
<td>Heating-cooling mismatch.</td>
<td>Reactor power transient.</td>
<td>Tube, tubesheet failure.</td>
<td>Initiating event frequency.</td>
<td>Failure criteria in HTR environment.</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>Loss of feedwater supply.</td>
<td>Evaporation of deposited fission products.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Heating-cooling mismatch.</td>
<td>Excessive feedwater supply following reactor trip.</td>
<td>Tube, tubesheet cracking, due to thermal and mechanical stress.</td>
<td>Initiating event frequency.</td>
<td>Failure criteria in HTR environment.</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>Flap valve closed accidentally.</td>
<td>Feedwater control system reliability.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Loss of coolant.</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fast depressurisation.</td>
<td>PCCV or penetration leak or failure.</td>
<td>Demand for CACS*.</td>
<td>Initiating event frequency.</td>
<td>Fission product transport data - wet re-entrainment and evaporation.</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Release of fission product to containment.</td>
<td>Probability of detecting a.g. degradation before accident.</td>
<td>Stress analysis model and inspection criteria.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td>Damage to steam generator.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td>Steam in core with breached primary circuit.</td>
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</tr>
<tr>
<td>Low frequency vibrations.</td>
<td>Earthquake.</td>
<td>Tube, tubesheet damage.</td>
<td></td>
<td></td>
<td>4</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Steam in core.</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* Core Auxiliary Cooling System
<table>
<thead>
<tr>
<th>Event/Failure</th>
<th>Potential Causes</th>
<th>Potential Consequences</th>
<th>Information Needed For Probability Assessment</th>
<th>Information Needed For Consequence Assessment</th>
<th>Priority</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core support structure.</td>
<td>Small steam generator or circulator bearing leaks.</td>
<td>Loss of compressive strength leading to core segment collapse etc.</td>
<td>Experience with high quality quality boliers, and circulators with water bearings from Fort St Vrain and THTR.</td>
<td>Failure criteria, Corrosion model validation in reactor, Definition of material variability, More relevant rate constants, Coolant composition models.</td>
<td>2</td>
</tr>
<tr>
<td><strong>Accidents</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Temperature excursion at normal power</td>
<td>Coolant channel blockage.</td>
<td>Enhanced fission product release and fuel failure.</td>
<td>Initiating event probability.</td>
<td>Validation of computed fuel and graphite failure criteria, Location of damaged fuel, Additional graphite physical property data.</td>
<td>1</td>
</tr>
<tr>
<td></td>
<td>Fuel matrix oxidation (only fuel overheats, not graphite).</td>
<td>Higher subsequent normal radioactive discharges.</td>
<td></td>
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<tr>
<td></td>
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<td>Changed stress pattern in block.</td>
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</tr>
<tr>
<td>Temperature excursion at decay heat power</td>
<td>CACS* delayed.</td>
<td>Mechanical damage to circuit components and containment.</td>
<td>Initiating event probability.</td>
<td>Validation of computed fuel and graphite failure criteria, Location of damaged fuel, Additional graphite physical property data, Component failure criteria, Thermal analysis, Fission product, B, U, Th transport data &gt; 2500°C, Containment response.</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td>CACS falls on demand or later.</td>
<td>Redistribution of B, U, Th, fission product in core.</td>
<td>Fault detection reliability, CACS performance and reliability.</td>
<td></td>
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</tr>
<tr>
<td></td>
<td></td>
<td>Local criticality.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td>Graphite sublimation.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Temperature excursion at decay heat power + steam</td>
<td>CACS* delayed.</td>
<td>Mechanical damage to circuit components and containment.</td>
<td>Initiating event probability.</td>
<td>B migration in steam.</td>
<td>3</td>
</tr>
<tr>
<td></td>
<td>CACS falls on demand or later.</td>
<td>Redistribution of B, U, Th, fission product in core.</td>
<td>Fault detection reliability, CACS performance and reliability.</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Steam generator, circulator bearing or liner failure.</td>
<td>Local criticality.</td>
<td></td>
<td></td>
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</tr>
<tr>
<td></td>
<td></td>
<td>Graphite sublimation.</td>
<td></td>
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</tr>
<tr>
<td></td>
<td></td>
<td>B migration to cool regions.</td>
<td></td>
<td></td>
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</tr>
<tr>
<td>Power excursion.</td>
<td>Control rod maloperation.</td>
<td>Enhanced fission product release and fuel failure leading to higher subsequent primary circuit inventory.</td>
<td>Initiating event probability.</td>
<td>Experimental validation of reference fuels' response to fast transients.</td>
<td>3</td>
</tr>
</tbody>
</table>

* Core Auxiliary Cooling Systems
<table>
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<tr>
<th>Event/Failure</th>
<th>Potential Causes</th>
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<th>Information Needed for Consequence Assessment</th>
<th>Priority</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rapid steam leak into primary circuit.</td>
<td>Steam generator failure.</td>
<td>Fuel coating corrosion, fission product release and subsequently enhanced fission product primary circuit inventory.</td>
<td>Initiating event frequency, detector reliability, boiler control system response.</td>
<td>Failure criteria, Accident model validation in high pressure high flow rig, Improved reaction rate data.</td>
<td>2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Fuel matrix corrosion.</td>
<td>锅炉控制系统可靠性。</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Block corrosion, structural weakening leading to unloading problems and fuel in coolant.</td>
<td>Boiler control system reliability.</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Core support block corrosion.</td>
<td>Fission product displacement from circuit surfaces by prolonged exposure to steam. Review (H₂ + CO) flammability in air + Re.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure reduction – steam generator failure.</td>
<td></td>
<td>PCHR relief valve lifts fission product and combustible oxidation products in containment.</td>
<td>Efficiency of detector of steam generator corrosion or reaction with coolant.</td>
<td>Fission product displacement from circuit surfaces by prolonged exposure to steam. Review (H₂ + CO) flammability in air + Re.</td>
<td>1</td>
</tr>
<tr>
<td>Mechanical damage to core.</td>
<td>Earthquake.</td>
<td>Damage to core support, loss of cooling, misalignment of control rod holes.</td>
<td>European earthquake frequency versus intensity spectra.</td>
<td>Specific core dynamic responses. Analysis of damage consequence if significant damage.</td>
<td>4</td>
</tr>
<tr>
<td>Event/Failure</td>
<td>Potential Causes</td>
<td>Potential Consequences</td>
<td>Information Needed for Probability Assessment</td>
<td>Information Needed for Consequence Assessment</td>
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</tr>
<tr>
<td>Helium + fission products in containment</td>
<td>Penetration or PCRV failure</td>
<td>Fission product release to environment, Damage to containment building and instrumentation, Interruption of reactor cooling due to inaccessibility of essential services.</td>
<td>Initiating event frequency, Accident sensor reliability, Isolation valve or vent valve reliability, Containment purification system reliability, Probability of men in containment during accident.</td>
<td>Failure criteria for cabling and instrumentation, Purification system efficiency, Containment leak rate experience, Containment test procedures, Ability to predict amounts of fission product entering containment.</td>
<td>1</td>
</tr>
<tr>
<td>Total loss of reactor cooling capability</td>
<td></td>
<td>Fission product release to environment and instrumentation, Damage to containment building, Interruption of reactor cooling due to inaccessibility of essential services.</td>
<td>Initiating event frequency, Accident sensor reliability, Isolation valve or vent valve reliability, Containment purification system reliability, Probability of men in containment during accident.</td>
<td>Specification of need for access to maintain essential services, Temperature and pressure analysis.</td>
<td>2</td>
</tr>
<tr>
<td>Helium + fission products + CO + H2 + steam in containment</td>
<td>Primary containment failure accompanied by steam generator or steam line failure</td>
<td>Fission product release to environment and instrumentation, Damage to containment building, Interruption of reactor cooling due to inaccessibility of essential services, CO + H2 combustion in containment.</td>
<td>Initiating event frequency, Accident sensor reliability, Isolation valve or vent valve reliability, Containment purification system reliability, Probability of men in containment during accident.</td>
<td>Specification of need for access to maintain essential services, Temperature and pressure analysis, Ability to predict composition of gas entering containment, Ability to predict probability and consequence of water gas combustion.</td>
<td>1</td>
</tr>
<tr>
<td>Steam in containment</td>
<td>Steam line failure</td>
<td>Damage to instrumentation by steam, 3H release to environment.</td>
<td>Initiating event frequency, Probability of men in containment during accident.</td>
<td>Failure criteria for containment systems in steam.</td>
<td>4</td>
</tr>
<tr>
<td>External impacts</td>
<td>Storms, crashing aircraft</td>
<td>Breach of containment</td>
<td>Initiating event frequency, Chance of coincidence with demand for containment isolation.</td>
<td>Failure criteria for containment systems in steam.</td>
<td>3</td>
</tr>
<tr>
<td>Low frequency vibrations</td>
<td>Earthquake</td>
<td>Breach of containment</td>
<td>Initiating event frequency, Chance of coincidence with demand for containment isolation.</td>
<td>Failure criteria for containment systems in steam.</td>
<td>4</td>
</tr>
</tbody>
</table>
**FIG. 1 AN INDEPENDENT SAFETY PROGRAMME ORGANISATION**

- **USER'S ADVISORY COMMITTEE**
  - Specification of Goals
  - Liaison with National Programmes and Industry

- **PROJECT DIRECTION**

- **RISK ANALYSIS UNIT**
  - Accident Identification and Delineation
  - Plant Dynamic Response Analysis
  - Component/System Performance/Reliability

- **TECHNOLOGY DEVELOPMENT**
  - Industry and Government funded HTGR Development

- **TECHNOLOGY DEVELOPMENT**
  - Project Sponsored R & D on Component Performance
    - Description
    - Modelling
    - Validation

- **DEMONSTRATION REACTOR SPONSORS**

- **DEMONSTRATION REACTORS**
  - Project Sponsored Surveillance in an European Prototype HTR and Dragon

- **HTR INFORMATION CENTRE**
  - Collection and Distribution of Information
  - Production of Reference Manuals

- **PROTOTYPE REACTOR RISK ANALYSIS**

- **LIAISON WITH U.S.**
  - Peach Bottom
  - Fort St. Vrain
Fig. 2. Technology Work Areas.

- Need assessment
  Definition of goals within work area, reporting to technology users.
- Information for design and analysis.
- Physical and mathematical models
- Design and analysis verification.
- Power reactor surveillance

- Fuel and graphite
- Control absorbers
- Coolants
- Heat removal systems
- Containments
- Instrumentation
FIG. 3 INFORMATION FLOW WITHIN RISK ANALYSIS UNIT
50. One final point is that since at least one application of very high temperature reactors is to supply process heat to industrial processes, it is highly likely that the plant will be associated with large chemical process installations or oil refineries. Economic considerations are likely to force close location of the reactor and the consumer of process heat. However, chemical and similar installations are likely to include large stores of dangerous chemicals even the process inventory could be substantial. In para 40, the present position taken by NII in respect of gas cloud explosions is seen to be that the only acceptable solution at present is separation. This clearly introduces a conflict between safety requirements and what almost certainly will be indicated economically. While no solution to this question can be put forward, it would seem that this is a matter to which the protagonists of the process plant application of VHTR should give serious considerations.

51. This paper has been prepared with the intention of using experience on CO₂ cooled reactors in the United Kingdom to illustrate certain issues affecting the safety case for HTR. In many respects HTR is similar to CO₂ systems however the differences which stem from the nature of the fuel and somewhat higher temperature are important. The resistance of the fuel to excessive temperature is a feature greatly in favour of HTR as is the negative temperature coefficient which will tend to stabilise many faults. However, the possibility of a fission product burden in the primary circuit, material performance uncertainties, boiler and pressure vessel integrity are aspects requiring careful attention.

52. It is a matter of fact that the NII has already reviewed one commercial proposal for HTR and although no formal position was stated at the conclusion of that review it is expected that an HTR could be designed to be acceptable for siting in the UK on a similar basis to other gas cooled systems. For at least the first plant programme, granting of the first licence would depend on through examination of the proposed design and the safety case put forward for that system,
46. The added uncertainties of the higher temperature environment may bring new problems, but it can at least be stated that those issues already referred to in relation to other gas cooled reactors may occur in a more acute form and therefore, appropriate action to ensure plant integrity will have to be taken to demonstrate safety with this in mind.

FUEL

47. Because higher gas temperatures are proposed it is to be expected that fuel will also operate at higher temperature conditions so that fuel deterioration may be more severe. Higher fission product burdens in the primary circuit may arise and this would lead to enhanced difficulty in the areas referred to above in paras 28/29 such as maintenance and inspection etc. The question may be asked as to the proposals for design and operation of advanced high temperature reactors which have a bearing on such matters.

IMPLICATIONS OF STEAM RAISING BY VHTR

48. The provision of a steam raising capability in an advanced HTR raises the question of boiler integrity and the consequences of moisture ingress in normal and fault conditions with higher graphite temperatures. Would boiler failure be acceptable in these conditions and what specific steps would be required to deal with such occurrences?

49. The proposal to produce process steam from HTR heat introduces the problem of ensuring that the recipients of the process steam do not also have radioactive material delivered with it. Effective barriers will have to be provided to ensure that no such accident can occur, which of course will have to take account of the use to which process heat is put.
apart from the reactor, such items as essential services, control points, all items
essential to removal of heat to the ultimate heat sink including the sink itself.
Each of these services must be considered individually and together as a system
for the purpose of assessing the effect of any external hazard and the measures
required to deal with it.

VERY HIGH TEMPERATURE REACTORS

43. The views of the NII are necessarily limited in respect of advanced versions
of HTR. No studies have been carried out by NII on the concept and no proposals have
been made to NII in respect of such a system. The Inspectorate would be interested
seeing even an outline safety case including a few representative fault studies.
It is only possible at present therefore for NII to make a few general observations
and possibly raise a few questions.

MATERIALS

44. The development of advanced designs of HTR, in which higher gas outlet
temperature is proposed, raises many questions from the safety point of view.
Most of the issues relevant to gas reactor safety referred to in this paper stem
almost exclusively from the behaviour of materials whether it be in structural
applications, the materials comprising the fuel, core or various components within
the primary circuit.

45. As temperatures increase so material properties tend to deteriorate and their
vulnerability to aggressive environments increases, the changes occurring frequently
at an increasing rate with temperature. Further, to combat the effects of temperatu
more sophisticated materials are likely to be brought into use for which experience
and data is scarce particularly when those materials are exposed to the special
conditions of the helium coolant.
however be taken into account. In dealing with the effect of aircraft impact the

effect of a fuel fire should not be overlooked.

40. Gas cloud explosions are best dealt with by separation of reactors and plant
containing large inventories of explosive substances, 5 miles appears to be a
suitable minimum. Research into gas cloud explosions and their effects may produce
data which would permit a reduction of this distance. The passage of vehicles
and ships carrying dangerous substances and the routing of pipelines must also
be considered. In general steps can usually be taken to plan safe routes or to set
special conditions for journeys if passage near a nuclear plant cannot be avoided.
Pipe lines should be routed to avoid their passing close to a nuclear plant. Other
effects which may require attention due to the location of industrial facilities near to
a nuclear plant include the possible release of toxic vapours or gases or the
release of substances which could damage essential safety equipment on the reactor
site or cause it to malfunction.

41. Natural phenomena such as rain, snow, ice, wind and storm must all be taken into
account. The general conclusion reached by the recent review was that for all
those natural effects considered, the magnitude judged appropriate for the United
Kingdom was such that nuclear plant built to current standards had a sufficient
degree of integrity. Certain specific features were, however, found to be vulnerable
such as overhead pipe bridges. Attention is currently being paid to the establishment
of design basis and safe shut-down earthquakes.

42. In determining design action to safeguard the plant against external hazards
it is necessary to identify all those items of plant and services on site upon
which the safety of the plant depends. The list will be substantial and will include
will, it is expected, provide a sound and tangible basis for safety judgement and action which will be complemented by such techniques as reliability analysis. In judging a system the NII give considerable credit to inherent processes which may contribute to the suppression or interception of faults, any move towards utilising such characteristics in the interest of safety is to be encouraged. Reliability analysis will presumably be capable of recognising the difference between engineered safeguards that operate continuously and can be shown to run through a transient undisturbed to provide their protective action and those which need to be initiated in response to a fault. A further aspect of safeguard reliability which is not always fully recognised is that it is not only necessary to evaluate the reliability with which a protective system or component will perform its initial action but where continued application of a safeguard function is necessary, possibly for a considerable period following a fault, so it is equally important to examine the reliability with which this continuing protection can be assured.

EXTERNAL EFFECTS.

38. Experience with Magnox and AGR design and siting has revealed the need to consider and codify precautions to be taken to safeguard commercial nuclear plant against various forms of external hazard. Examples of such hazards are aircraft impact, gas cloud explosion, natural phenomena etc. Reference to NII activities in this field is made in para 19.

39. The position with regard to future plant commissioned in the UK is as follows. For aircraft impact, provided nuclear installations and civil airfields are separated by at least 5 miles the risk from impact by commercial passenger and cargo jet is sufficiently small as to be neglected. Impact by private light aircraft and executive jets and also impact by military aircraft of the MRCA type should
SAFETY ANALYSIS TECHNIQUES

35. The methodology employed in assessing reactor system safety has evolved in the UK as elsewhere. Much philosophical discussion is engaged in on this topic and much has been and no doubt will be written on the matter. It is not the purpose of this paper to add significantly to the discussion but consideration of the approach the UK regulatory body would adopt in the light of its experience on other gas cooled reactor systems would be incomplete with no reference to this matter.

36. The Rasmussen report dealing with light water reactors has illustrated the use that can be made of logical analysis employing an attempt to quantify reliability and to synthesise overall risk. The UK regulatory position is that such analyses are a useful and important contribution to safety analysis particularly where complex systems with many branching fault sequences can be postulated. It is seen as a useful means of inducing discipline into the analysis of a system and might be expected to indicate design changes which would be beneficial in the interest of safety or conversely show up areas which were overprotected. It is not however, the position of the UK regulatory body that analyses of the kind exemplified by the Rasmussen study should become a primary factor in the determination of safety policy nor would the method lead to means of expressing policy. It is not regarded by the UK regulatory group as realistic to consider the results of such analyses as having absolute meaning but it is accepted that the techniques when applied on the basis of mutually agreed ground rules and consistent data can as indicated above be useful. It follows therefore that the safety case for HTR should, where meaningful information can be generated by the application of such methods as reliability analysis, include the results obtained from such studies.

37. Other aspects of methodology include the development and application of criteria, codes and etc, this form of codification of the regulatory position allied to guides
33. Increasing attention has been given in recent years to the need for diversity in protective systems in order to overcome reliability deficiencies arising from common mode failures. Shut down systems have probably received the greatest attention and in the case of gas cooled reactors specific action has been taken to deal with perceived common mode effects. It is to be expected that where a mechanism which might cause a common mode failure can be foreseen that the design should, as far as possible, be arranged to eliminate the possibility. However, there will be practical difficulties in some circumstances and furthermore, unrevealed common mode effects will exist and, the extent to which protective reliability can be demonstrated will be limited either because of the cost and time required to conduct a very large number of tests or the difficulty of conducting an adequate series of tests on a complete system under the correct conditions.

34. Considerations of the kind outlined above and past experience lead to the conclusion that where very high reliability of protection is required, diversity of safeguard action should be provided. Examples of circumstances where such a requirement might arise are faults or initiating events or fault sequences which are judged to occur in the medium to high probability range and which, if not controlled by effective engineered safeguards could lead to serious damage to the core and a large radioactive release. In the terminology used in para 19, medium to high probability range, means once in a reactor life-time and many cases occur once in a reactor programme. It should perhaps be emphasised that these considerations are not confined in their application to reactor shut down but are of equal importance in relation to any engineered safeguard which plays an essential part in the interception of a fault, its control or the supression of any release due to a fault should this occur.
HTR PRESSURE VESSEL

30. High temperature reactors for commercial application have invariably been designed with a prestressed concrete pressure vessel. These vessels are similar in many respects to those developed for AGR. The reactor core is located in a central vault while boilers are located in pods in the vertical cylindrical wall. Pressures are however, higher than for Magnox and AGR while the cross sectional area of the vault is rather less in proportion to wall area than in other gas cooled systems.

31. The gas temperature to which most of the internal surface of the pressure vessel is exposed will probably be limited to core gas inlet temperature. That part of the surface which has high temperature outlet gas in contact with it will require special attention and in this context the remarks made in paras 21 to 24 are relevant. Furthermore, the possibility of failure of thermal barriers in the high temperature region would have to be considered along with the possible effects on the liner and concrete. Insulation performance, bearing in mind the particular difficulties in Magnox and the influence of induced vibration arising from coolant circulation will be factors to be examined closely, particularly since the structural characteristics of the insulation is likely to differ from that employed in CO2 cooled systems.

32. The PGRV in an HTR context can be expected to be rather more sensitive to insulation and liner failure than is thought to be the case for Magnox and AGR. Liner defects which allow coolant gas to penetrate and pressurise cracks or pores in the concrete could lead, if the penetration is extensive, to a significant change in distribution of forces in the PGRV system. In the extreme this may lead to an unacceptable reduction in the margin to failure of the vessel. Consideration should be given to the provision of positive means of removing this particular cause of concern, in this context it is noted that proposals have been made to provide ventilation of the concrete immediately behind the liner in some designs.
fuel behaviour has confirmed the expectation that these reactors will operate with essentially clean circuits. HTR fuel is expected to start life with a small but significant fraction of leaky particles. Thus it is expected that there will be a standing level of fission products in the system during normal operation which may be increased due to minor transients or fluctuations in fuel performance. It may also be that the loss of fission products from fuel will be less than anticipated. Whatever the actual behaviour, a new situation in gas reactor safety arises which must be taken into account, and the main issues are as follows.

CIRCUIT CONTAMINATION

28. Sound knowledge of fission product mobility within the primary system is important. Both from the point of view of estimating the standing quantities in the gas phase also the fraction fixed on surfaces and adsorbed in graphite. This knowledge is necessary to provide a basis for design action in relation to releases to the environment during normal operation and the provision of facilities for in-service inspection and maintenance within the circuit and on components removed from it. There is thus an important connection between fuel performance and the issues raised in para 22 above.

29. Depressurisation accidents even at a relatively low rate will produce some level of radioactive release along with the primary coolant owing to the activity within the primary circuit either in the gas phase or on surfaces. The extent to which all fission products in the primary circuit are available for release will depend on the extent of lift off during or following the fault. This will be influenced by the circuit conditions such as the extent of moisture ingress from faulty boiler tubes. Means may have to be provided to limit the magnitude of release to the environment and in this context the criteria set out in 19 would be relevant and for this purpose it would be necessary to assume that small scale breaches of connections to the primary circuit, such as for example impulse lines, would occur more frequently than once in a reactor life-time.
developing coolant ingress can only be accurately assessed by observation of in-service performance of components particularly graphite. Corrosion of graphite will be affected by a wide variety of factors each of which will vary from point to point in the core. Clearly, analysis of this effect can only be carried out in relation to local conditions and there will be a considerable statistical element involved. Once again, notwithstanding the confidence which can be accumulated from experiment etc; prudence would indicate that core integrity would be better assured if all graphite components affecting core stability and coolant flow could be replaced and or replacement be available for inspection. Deterioration of the core structure, should this occur, should be detectable by some form of monitoring probably indirect, perhaps changes in core structure which will most strongly affect fuel cooling will most readily be detected by changes in local core gas outlet temperature or fission product monitoring systems. Obstruction of control rod movement by the same cause will also require means of detection.

26. Boiler tube failure during a fault may involve coolant ingress to the primary circuit on a much larger scale. Failure of boiler tubes coincident with a fault would not be considered as a chance occurrence but as a direct consequence of substantial changes in the boiler operating conditions. Combined effects of this kind are in general not regarded as important in Magnox and AGR depressurisation faults owing to the lower operating temperatures of the graphite. However, loss of boiler availability would be a matter for concern, and this aspect must be taken into account in considering post fault heat removal as would also be the case for HTR.

FUEL

27. Fuel behaviour in HTR is generally recognised as being different from that in other gas cooled systems. Operational experience with AGR fuel is, of course, not yet available on commercial plant. However, as has already been mentioned Magnox
THE HTR ENVIRONMENT

22. The helium environment will, of course, not be inert with respect to structural materials. It is to be expected that the state of surface oxidisation will be different from that in which most materials have been proven. In addition, trace elements in the He coolant will have an important effect on material behaviour and corrosion type processes may occur. Since the trace element content will vary during reactor life according mainly to moisture ingress, so the oxidising condition may change with respect to various circuit structural materials.

23. The Magnox experience on corrosion is directly relevant in that corrosion processes were shown to be highly specific to materials and conditions, so in HTR consideration of the generality of the situation may be misleading for there may be certain specific situations depending on temperature, materials, fluid flow conditions etc, within the circuit which might encourage local but, nonetheless, important deterioration.

PRIMARY CIRCUIT COMPONENTS

24. Material behaviour is, therefore, a central issue. Notwithstanding all the experimentation and operating experience in pilot plant there is a strong case to design the plant so that internal components can be inspected or remotely monitored and also be replaced. Where practical considerations preclude either or both of these features the safety of the plant should be shown to be insensitive to the failure or malfunction of the particular component.

25. Boiler behaviour must be assumed to include leakage. Small scale and slowly developing faults should be demonstrably the mode of deterioration in normal conditions. In faults, however, such a pattern may not be a valid model of events if the boiler, as a result of the fault, is exposed to thermal and pressure transients and has at time already been subject to operational deterioration. The effect of small gradual
or more separate areas have been identified for which safety criteria reports will be prepared; among the next to be completed will be those dealing with protective instrumentation, diversity of protection, containment, fuel integrity and general criteria.

SUMMARY OF GAS REACTOR EXPERIENCE

20. The above is a somewhat brief and incomplete summary of some of the matters which have attracted the attention of NII as a result of experience with the gas cooled reactors in the United Kingdom. The issues referred to can be seen to have parallels in the High Temperature Gas Cooled Reactor and while of course no specific solutions will be apparent to HTGR problems this experience brings into focus some of the issues which would require consideration by the Regulatory Body in relation to licensing of that class of reactor.

21. Probably the most important single item of all the experience gained from the UK gas cooled reactor programme is that concerned with the behaviour of materials in the reactor environment. Magnox reactors operate at comparatively low core gas outlet temperatures compared to HTR and AGR will have higher temperatures at outlet than Magnox but still less than HTR. In general there is little to learn from Magnox in relation to the long term mechanical properties of materials. Mild steel is the predominate material in Magnox and there is no reason to expect creep to occur nor has there to date been any strong evidence of a fatigue problem. The high temperature reactor is, however, likely to have materials at the core outlet and boiler inlet region in the creep regime and there may be also fatigue to be considered. Since the materials to be used in the high temperature region will be special alloys there may be a lack of data on time dependant mechanical properties particularly in the helium environment. Acquisition of such data, being a lengthy process, may not be completed in time to allow design and operating decisions to be made on anything other than information obtained from extrapolation of short or medium time creep and fatigue da
in the UK has not relied on written codes or criteria as guides to safety assessment. Because of the particular situation in the UK it has been found possible to operate quite satisfactorily so far, although from time to time there have been calls for explicit statements to be made by NII as to the criteria which should be met. Siting is perhaps one exception where written guides have been issued. There have of course existed unwritten, as far as NII is concerned, criteria etc usually of a fairly general kind and there has been an acceptance of certain national or international design codes etc, where these are relevant. A substantial effort is being mounted to produce a set of working criteria specific to nuclear reactor safety which will eventually be published in full. In preparing these criteria full use is being made of current experience on the Gas cooled reactors in the UK and other systems elsewhere. Careful study is also being carried out of codes and criteria employed in the Nuclear Safety field elsewhere.

19. Work in this area is not yet complete but certain projects have been completed. For example, following consideration of external hazards ie, earthquakes, flood, aircraft impact and the like, a report setting out the basis for safeguarding action has been prepared. A copy of this document has already been communicated to the secretariate of CSNI. We have completed work on radiological protection in both normal and abnormal conditions. This is a statement of the exposure limits to be adopted for design and operational purposes. The radiological criteria for fault conditions recognise a crude relationship between maximum permissible exposure and frequency. For frequencies of less than once in a reactor life time (about once in 30 years) the upper bound of dose is 1/20 of DWL, below this frequency and down to once in a reactor programme about 1 in 3 x 10^4 the exposure limit as a result of a fault is 1 DWL and for any credible fault the limit is 1 ERL below the once in a reactor programme frequency. These numbers are upperbound figures which, of course do not remove the obligation to minimise exposure where practical. In all some 20
Boiler tube failures are generally dealt with by plugging although
in some cases specific design features which were identified as inducing tube
failure have been rectified satisfactorily.

ENGINEERED SAFEGUARDS

16. Protection of the reactor during and following a transient depends not only on
high reliability shut-down but also on the removal to a guaranteed sink of sufficient
sensible heat and decay heat to preserve fuel integrity so as to prevent a serious
release of radioactive matter. It is therefore essential that the systems provided to
support the heat removal process should, when taken as a whole, perform this process
with a reliability comparable to that expected of other engineered safeguards such as
shut down systems.

17. In some circumstances, conflict of requirements can arise, between the need to
prevent undue thermal shock or other large and possibly damaging departures from the
specified mode of operation as a result of a fault and the need to maintain the
plant in operation to minimise the effect of a transient. An example arises in
straight through boilers where the need to prevent water flooding up to the
superheater region can most effectively be achieved by dumping, thus removing
coolant medium from the system which could be used to remove heat subsequently.
While such circumstances can be dealt with by the provision of automatic controls
of various kinds, the added complexity could lead to reduced reliability. Experience
suggests that much more attention needs to be given to the reliability of heat
removal systems in relation to safety, particularly in the context of fault situations

CRITERIA AND CODES

18. For most of the period during which Magnox reactors were being brought into
operation and AGRs were being constructed the Nuclear Installations Inspectorate
13. Study of the effects of corrosion induced component failure has led to the conclusion that a common mode failure might be induced in a depressurisation fault which would prevent a significant number of shut down adsorbers entering the core because of graphite displacement. This has led to the decision that alternative shut down systems should be installed which would not be vulnerable to movements of graphite core blocks and which would thus ensure that the reactor could, should such a fault occur, be safely shut down.

FUEL

14. Magnox fuel has performed with a remarkably high level of integrity. Certain classes of failure are removed when observed other less serious failures can be left in pile. This good performance coupled with the high degree of subdivision of the fuel and on load refuelling has led to very low activity levels in the primary circuit. As a result, handling and maintenance of many components removed from the reactor and access to certain parts of the primary circuit has been accomplished with relative ease and quite modest radiation dose to personnel concerned. Compare with experience elsewhere total man rems acquired on each twin reactor site is also reasonably low.

Because circuit contamination is low the contribution to radioactive gaseous discharges to the atmosphere is small so that control of coolant loss is essential of economic importance.

15. BOILERS

Failure of boiler tubes leading to water ingress has occurred in Magnox reactors apart from a limited number of cases the amount of water entering the circuit has been small. Graphite temperatures in Magnox are too low for serious corrosion problems to arise because of moisture in the coolant although the presence of water in the primary circuit creates an undesirable environment and should substantial quantities enter the vessel thermal insulation may be affected.
10. A further interesting aspect of this particular experience is the specific nature of the phenomena its localised effect and particular mechanism. No generalised consideration of reactor conditions or material performance would have led one to think from consideration of the design before the events that these effects could occur. Further an environment which in general appeared relatively harmless proved to be quite aggressive given favourable but unexpected combination of circumstances.

11. The Magnox reactors of course are operating and will continue to do so. In fact, these reactors have exhibited a remarkable availability despite the corrosion problem. As a result of corrosion certain components exposed to the reactor coolant have deteriorated and the general condition has been established by various techniques of internal inspection and monitoring. An acceptably safe mode of operation has been defined and accepted which is essentially a modest reduction in core gas outlet temperature. This, in itself, illustrates a simple but fundamental point which is, the extreme non linearity of such effects as functions of temperature as might be expected from consideration of basic phenomena.

12. The mechanical effect of corrosion in the Magnox reactors has been to reduce the integrity of certain mechanisms and structural components. This deterioration has been dealt with in a variety of ways albeit with considerable difficulty and expense. Had access to the vessels been possible and had all components been readily accessible for direct or remote viewing and repair the situation would have been more readily recovered. A more important facility would have been the ability to remove essential components from within the vessel for repair or replacement. Design could perhaps in some cases have been contrived so as to reduce the sensitivity of the system to corrosion induced component failure.
effects observed are, in brief, a corrosion mechanism in CO₂ of certain specific grades of mild steel, which in certain circumstances, leads to failure of bolted components or seizure of components intended to act as parts of mechanisms. All the elements of this phenomenon were known in one way or another when the designs were executed. Nonetheless, the effect was not recognised probably because the overall behaviour had not, for a number of reasons, been previously observed in practice before. With hindsight it might be argued that the effect should have been foreseen, on the other hand, one wonders how many mechanisms would have had to have been postulated, using basic information, before the particular set of circumstances was hit upon. Perhaps one message is that, with all the resources at our disposal, there are always many possibilities which are missed simply because of the sheer complexity of the problems, and in the absence of sufficient indicators from applications outside the nuclear field, it must be expected that a novel technology is inevitably going to produce a few surprises until substantial plant scale experience has been accumulated.

9. Cumulative damage of all kinds is a factor to be taken seriously in considering reactor plant integrity and the corrosion mechanism observed on Magnox plant is one example of such damage. It is seldom possible to adequately reproduce reactor conditions in rig experiments sufficiently faithfully to be assured that experiments out of reactor can reveal such effects. Even if the appropriate conditions can be produced the need to carry out accelerated testing, if results are to be obtained in time to be of use, may itself introduce changes in basic mechanisms. Nor is it possible to be sure that deterioration is necessarily linear so that extrapolation from short term tests may prove misleading.
6. The general arrangement of the liner cooling system, liner and insulation is well known. Foil type insulation is commonly employed for this purpose and in general operational experience has been satisfactory. Difficulties have arisen however in regions where complex arrangements of the foil insulation have been necessary owing to the particular local geometrical configuration. Concrete hot spots have been observed which are due to inadequate performance of insulation in these areas. There are good reasons to believe that such small localised imperfections would not impair vessel integrity even if, as a result, the concrete properties in the locality deteriorate.

7. Apart from the small scale defects referred to above, in service performance of liners and insulation appears to be satisfactory. It is expected that before any significant change in vessel integrity could possibly occur due to insulation failure, ample evidence would become available either from increased heat release to the liner cooling system or by the presence of loose insulation in the primary coolant circuit. There is an awareness of the fact that very large areas of this insulation are not accessible once the reactor has been brought to power, that the liner is not only inaccessible but not visible and that the facilities available to monitor the state of these items are few and indirect. It may be that exposure over the plant life to the conditions within the reactor including corrosion, thermal vibrational and other effects will induce failure at some stage, should this occur and if it is detected, repair may well not be possible and at best would be very difficult. It is perhaps fortunate that in both gas reactor systems currently in operation and under construction in the UK, the absolute safety of the vessel is probably relatively insensitive to quite substantial departures of the liner and insulation from its specified performance.

CORROSION

8. The experience in the Magnox reactors with corrosion has resulted in several important issues coming to be recognised as being of primary importance. The actual
justified a relaxation in siting criteria. The original basis for the perceived improvements in safety remains as valid today as was originally the case, this being a combination of high redundancy in the main load carrying members, the facility to inspect and the expected gradually developing failure in the event of a pressure overload. The overall behaviour of these vessels has been up to expectations but a few issues have emerged in the course of design and construction which are of interest in the context of safety.

4. The concrete monolith is a relatively simple structure which can be analysed by modern methods with reasonable ease. The monolith must, however, be penetrated to allow access for various services and components etc. The closures of these penetrations can be structures having a variety of services etc, passing through them and must be designed to provide a high integrity barrier between the high pressure hot gas inside the vessel and the environment.

5. It has been accepted as a basis for safety evaluation that the chance of a large failure of the monolith to allow depressurisation is sufficiently unlikely to allow that possibility, for this purpose, to be disregarded. The same view has not been taken with respect to the penetration closures or service connections passing through the concrete PCRV walls. The largest of these penetrations, for instance, the closures above pod boilers, are of such a size that failure of the closure to permit anything approaching full cross section flow would not be acceptable either on thermal or, mechanical grounds. In fact the design basis accident generally taken as the failure of a gas duct of about 100 square inches cross section. This means that special design provisions are required for larger penetrations which, in general, amounts to the load bearing functions of the closure members being performed by redundant and independent structural members.
INTRODUCTION

1. Having adopted gas cooled thermal reactors as the basis for the first and second commercial nuclear power programmes, the United Kingdom has acquired unrivalled experience in this particular class of reactor system. The original programme of Magnox reactors based on the Calder and Chapel Cross designs was initiated with Bradwell and Berkeley and reached its conclusion with the completion and putting to commercial use of the twin Reactors at Wylfa. Although no AGR station has yet fed power to the grid, the design and construction programmes of this batch of gas cooled reactors along with the supporting research and development work has added considerably to the general appreciation in the UK of the particular characteristics of gas cooled systems in general. This practical experience has been supplemented by an active interest in High Temperature systems which has been stimulated by design studies of commercial plant and the information which the OECD Dragon project at Winfrith Heath has generated in the fields of HTR. Research and development and allied pilot plant operation.

2. Much of the experience which has been acquired has, of course, been of particular interest to operators and designers for reasons other than safety, for instance, valuable information on availability and other aspects of plant economy has been generated. However, a considerable amount of information directly relevant to safety has been obtained although the actual safety record of those plant which have been in operation has been remarkably good as has been general plant availability and reliability.

PRESSURE VESSELS

3. It is well known that the earlier Magnox designs were based on steel pressure vessels with external boilers connected to the main pressure vessel by ducts. No evidence has so far come to light which would indicate any reason to question fundamentally the integrity of these vessels, however, the introduction of the prestressed concrete pressure vessel has been regarded as a safety bonus which
SYNOPSIS

The commercial design of High Temperature Reactor has many features in common with gas cooled reactors already licensed and operating in the United Kingdom. There are also some important differences which have a bearing on safety. The paper discusses the safety aspects of the commercial HTR in the light of UK experience on gas reactors and concludes that so far as the United Kingdom is concerned a commercial design of HTR suitable for siting in the UK is feasible.

The paper also reviews the basis of safety philosophy which would be applied in reviewing an HTR and the procedures which would be necessary for the clearance of the first off of a new design.

The place of safety criteria in overall safety evaluation is discussed and examples given of developments in this area currently underway within the UK Nuclear Installations Inspectorate.

Reference is made to the applicability of probability methods in safety evaluation.

In conclusion the paper considers certain specific issues relating to HTR safety.
OECD

CSNI SPECIALIST MEETING ON HTGR SAFETY

PETTEN 13 - 15 MAY, 1975

A review of certain gas cooled reactor safety issues.

by

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United Kingdom
so that a significant period of time would have to be allowed before the
licence for the first plant could be issued and a substantial amount of
information would be required.