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Safety Related Physics Activities in the DEBENE-FBR-Project

Present Status, Needs and Perspectives

R. Fröhlich, F. Helm, G. Keßler, E. Kiefhaber
Institut für Neutronenphysik und Reaktortechnik
Projekt Schneller Brüter

Kernforschungszentrum Karlsruhe

KERNFORSCHUNGSZENTRUM KARLSRUHE

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ABSTRACT

The physics research and development activities in the DEBENE-Fast-Breeder-Projekt related to core disruptive accidents are discussed. These activities are placed into the context of an overall plant safety concept for a future commercial size LMFBR plant, which is strongly influenced by construction and licensing of the prototype reactor SNR-300. This paper starts with a presentation of the overall LMFBR (e.g. SNR-300) safety design concept, reviews the present core disruptive accident analysis procedure for SNR-300 and describes the present status of physics design and analysis for the future commercial-sized SNR-1300. Based on this review key issues of physics related safety research for core disruptive accidents are discussed and the necessity of more research and development effort to resolve these issues for commercial-sized plants is indicated.

Sicherheitsrelevante physikalische Aktivitäten im DEBENE-Schnell-Brüter-Projekt;

gegenwärtiger Stand, Erfordernisse und Ausblick

ZUSAMMENFASSUNG

Es werden die physikalischen Forschungs- und Entwicklungsarbeiten des DEBENE-Schnell-Brüter-Projektes, kritisch erörtert, die im Zusammenhang mit schweren hypothetischen Störfällen stehen. Diese Arbeiten werden vor dem Hintergrund eines generellen Sicherheitskonzeptes für einen zukünftigen kommerziellen schnellen Brutreaktor dargestellt, wobei anzumerken ist, daß ein solches Sicherheitskonzept sehr wesentlich durch den Bau- und das Genehmigungsverfahren des SNR-300 geprägt wird.

Dieser Beitrag beginnt mit einer Darstellung des generellen Sicherheitskonzeptes eines schnellen Brutreaktores (erläutert am Beispiel des SNR 300); dann folgt eine kritische Erläuterung der gegenwärtigen Vorgehensweise zur Analyse schwerer hypothetischer Störfälle für den SNR-300; außerdem wird der gegenwärtige Stand der physikalischen Auslegung und Analyse eines zukünftigen kommerziellen schnellen Brutreaktors beschrieben. Ausgehend hiervon werden schließlich Schlüsselprobleme der physikalischen Sicherheitsforschung für schwere hypothetische Störfälle diskutiert und es wird die Notwendigkeit zusätzlicher Forschungs- und Entwicklungsarbeiten zur Lösung dieser Schlüsselprobleme für große kommerzielle Brutreaktoren aufgezeigt.

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1. INTRODUCTION

Construction and licensing of the sodium cooled prototype fast reactor SNR-300 is the main basis for a licensable safety concept of a future commercial size LMFBR plant SNR-1300 in the FRG. This safety design and licensing concept strongly influences the related R&D-programmes. Any discussion of the present status needs and future trends of safety R&D therefore must be seen in the frame of the overall plant safety concept. This paper deals only with safety and physics R&D activities related to core disruptive accidents with extremely low probability of occurence.

2. LMFBR SAFETY DESIGN CONCEPT

The safety concept of SNR-300 and a future commercial size plant SNR-1300 is based on the multiple barrier concept and on several levels of safety design /1,2,3/. The first level of safety design prevents the occurence of fault initiation by application of inherent design characteristics, the design principles of diversity and redundancy as well as inservice inspection of important safety components and quality assurance during design and construction of the plant. This assures low probability of occurence for accident initiations. The second level of safety design prevents the propagation of faults into serious core accidents by maintaining integrity of the core and providing reliable reactivity control. The third level assures adequate mechanical response of the primary coolant systems, as well as the inner and outer containment against extremely unlikely severe accidents, thereby limiting the release of radioactivity to acceptable levels (barrier concept).

2.1 Accident Prevention

On the first and second level of safety design a criterion of less than 10^{-6} per year for loss of coolable core geometry is the design objective. Loss of coolable core geometry would lead to core melt down or core destruction. This may occur if the plant protection system together with the redundant shut down systems or the decay heat removal systems fail on demand /2,3/. It has been shown for SNR-300 that this design objective can be met by using a plant protection system with two diversely and independently acting shut down systems /4,5/. The same shut down system design will also be used for SNR-1300.

The primary shut down system drops absorber rods into the core from above by gravity forces. The secondary shut down system pulls by use of a precompressed spring a flexible absorber chain into the core from below. Both systems have diverse electronic channels.

In case of a commercial size LMFBR plant SNR-1300 four main primary and secondary heat transfer chains provide for safe decay heat removal. In addition a second diverse decay heat removal system is provided by a sodium air emergency core cooling system which extracts heat from the reactor outlet plenum by immersed heat exchangers in the reactor tank. Immersed heat exchangers will also be installed in the intermediate heat exchangers. In case of SNR-300 similar diverse and redundant decay heat removal systems attain a probability of shutdown heat removal breakdown of less than 10^{-6} per demand in case of the steam generator failure and, together with the fact, that the steam generator failure rate is less than 1 per year, one obtains a probability of shutdown heat removal breakdown of less than 10^{-6} per reactor year /6,7/. Even the complete loss of all active systems would not lead to core damages. The decay heat would be transfered to the sodium and will penetrate through the insulation of the still intact primary coolant pipe into the inner containment /7,8/.

2.2 Protection Against Fault Propagation

In addition to the required high reliability of the safety shut down systems it must also be assured that initiation of faults which could develop from local blockages in fuel elements, are counteracted directly by the plant protection system to avoid the possibility of fault propagation and global disturbances. E.g. local blockages in fuel elements and fuel pin failures are detected by individual subassembly instrumentation (thermocouples) and delayed neutron monitors /5/.

2.3 Primary System and Containment Design Basis

Under the above conditions severe core melt down or core disassembly accidents for the SNR-300 can only be initiated if the plant protection system with both shut down systems fail on demand. Consequently their probability of occurence would be less than 10^{-6} per year for the SNR-300; and this will also be a design requirement for the SNR-1300. Despite of this low probability of occurence such core melt down accidents have to be considered as a design basis for the reactor tank and the inner and outer containment system /1,2,3/. The reactor tank, the primary coolant and inner containment system must withstand certain pressures and heat loads defined from the analysis of such severe core accidents. The structural requirements for the outer containment building are given by the design basis for earthquakes, airplane crashes and gas cloud explosions /1,3/. The leak tightness of the inner and outer containments is determined by the radioactive source term characteristics of the low probability severe core melt down accident. The outer containment consists therefore of a relatively leak proof steel containment surrounding a reinforced concrete containment for protection against external impacts. In case of SNR-300, the outer containment is equipped with a ventilation system to keep a certain underpressure in the gap between the steel and the concrete containment /3,8/.

Prior to a discussion of the present status of the core design and safety analysis of a commercial size plant SNR-1300 and a subsequent discussion of key safety phenomena it seems appropriate to present first the approach taken for the CDA analysis of SNR-300.

3. CORE DISRUPTIVE ACCIDENT ANALYSIS PROCEDURE FOR THE SNR-300

A detailed discussion of potential initiators for core disruptive accidents of the SNR-300 leads to the conclusion that primary pump coastdown with simultaneous failure of both independent shutdown systems - the so called LOF (loss of flow) accident - requires the most careful attention. The removal of all 9 control rods (of the primary shutdown system) with maximum speed (reactivity ramp of less than 4 ¢/sec) and the subsequent failure of both independent shutdown systems - a so called low ramp rate TOP (transient overpower) accident - has a much lower probability of occurance than the LOF-accident. The following discussion will therefore focus attention on the LOF-accident and only a few remarks will be made about the low ramp rate TOP-accident.

3.1 Accident Path Structure and Key Phenomena

The possible accident paths and accident phases are indicated in Fig. 1. They are defined in a similar way as discussed by Jackson et al /26/. It will not be possible here to discuss in detail all of the numerous phenomena in the various phases which are presently responsible for uncertainties in LMFBR accident analysis; only some of the key phenomena will be briefly discussed in section 5. These key phenomena determine whether the accident develops a mild or energetic behavior and whether the core debris can be contained within the primary system. For the SNR-300 these uncertainties have been taken care of by using pessimistic assumptions (see the energetic accident paths in section 3.2).

3.2 Analysis of the Loss of Flow Accident for the SNR-300

The analysis concentrates on the end of life core of the Mark 1A reactor core design of the SNR-300, which has a larger energetics potential than the fresh core /27/.

Best Estimate LOF Accident Path

By using best estimate parameters and modelling for the LOF analysis of the SNR-300 (e.g. fuel axial expansion, fuel dispersal by fission gas and sodium vapor) the accident proceeds (see the big black arrows in Fig. 1) through a relatively mild initiating phase into a transitory phase /27/. During the transitory phase and the phase of integral material movement secondary

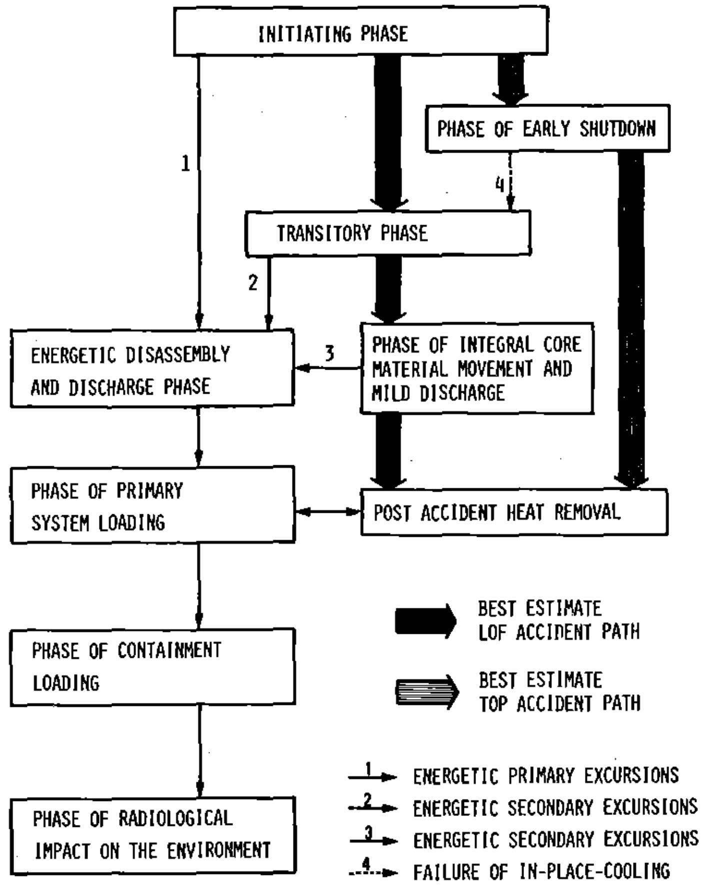

Fig. 1: Phase Diagram for Core Disruptive Accidents (SNR-300)

excursions cannot be excluded but if they occur they are expected to be mild (based on dispersive properties of core material and on incoherency arguments for fuel movement during the transitory phase). They will promote a mild discharge process of core material predominatly into the upper plenum of the reactor vessel /34/. It has been shown /29/ that the core material debris is permanently coolable within the reactor vessel. The mechanical load of the reactor vessel and of the primary circuits is negligible. The same conclusion holds for a range of parameters around the best estimate values.

Energetic LOF Accident Paths

Because of still existing uncertainties for some of the key phenomena in theoretical understanding, theoretical modelling, and experimental verification of these models it has been considered prudent and necessary to use pessimistic assumptions for the accident development (energetics) during the various accident phases. For the initiating phase a limiting energetic accident path (see the small black arrow 1 in Fig. 1) has been derived by making several rather pessimistic assumptions (e.g. no axial fuel expansion reactivity feedback; no fuel dispersal by fission gases, sodium vapor, and steel vapor; instantaneous fragmentation and mixing for fuel/sodium heat transfer with a small fuel particle radius). This accident path leads into a LOF driven TOP event /27,30/, which was mechanistically simulated by using the SAS3D code system /31/. In addition energetic limiting accident paths due to secondary excursions have been derived out of the transitory phase and the phase of integral core material movement (see the small black arrows 2 & 3 in Fig. 1) by using pessimistic assumptions. For the transitory phase rather coherent slumping of intact fuel pin segments into non boiling regions has been assumed to obtain a limiting energetic secondary excursion, which was mechanistically simulated by using the SAS3D code system /31/. For the phase of integral core material movement coherent recompaction processes of boiling pools and coherent slumping of remolten fuel slugs (from upper core structure blockages) into a boiling pool have been considered /28,30/ to get limiting energetic secondary excursions. The initial and boundary conditions for these latter cases were derived from best estimate SAS3D calculations and the energetic excursions were simulated by KADIS /32/. The SIMMER-II code system /33/ although available in Karlsruhe for a mechanistic analysis of the integral core material movement phase /34/ was intentionally not applied for SNR-300 licensing purposes because experimental testing of SIMMER-II for the complex phenomena involved is still not sufficient.

At the end of these energetic neutronic excursions the thermal energy content of the molten fuel (energy above the liquidus point) is up to 5500 MJ. But one has to realize, that the other conditions (sodium content of the core, steel temperatures, etc.) are quite different for the different cases. The process of transforming thermal into mechanical energy is strongly dependent on these conditions. The mechanical energy potential of the hot molten fuel masses has been conservatively estimated by taking these conditions into account and by considering the effect of other (partly more effective) working fluids.

In fact, several combinations of hot fuel and one additional working fluid (sodium, steel, fission products) have been evaluated with respect to their mechanical energy potential /30/. In addition first mechanistic simulations of the energetic expansion and discharge phase for SNR-type reactors with SIMMER-II /35/, taking into account the complex processes involved, show that the mechanical energy (kinetic energy) generated is much less (almost one order of magnitude) than the isentropic fuel work potential. But due to insufficient experimental testing of the SIMMER-II code for some of the phenomena involved, these SIMMER results have not been introduced into the licensing procedure of SNR-300. One can conclude that the mechanical energy potential for all of these energetic accident paths is reasonably well bounded by the isentropic fuel work potential, which yields maximum values around 100 MJ, which is well below the design load limit of 370 MJ for the SNR-300, which was set by the licensing authorities as early as 1972.

In spite of the conclusion that the primary vessel and the circuits will withstand the mechanical load generated by these energetic excursions, two other problems need carefull evaluation:

1.) Core debris distribution and long time coolability within the reactor vessel,

2.) Containment loading by leaking core material (through un-tight seals and valves).

The first problem has been discussed in some detail /28/ and it has been shown that long time coolability can be established within the reactor vessel. Thus, the external core catcher of the SNR-300 has therefore only a backup function.

The analysis of the second problem shows that even for these severe accidents the release of radioactive material will lead to dose values below the maximum values permitted by law /9,10/.

3.3 Some Remarks about the Low Ramp Rate Transient Overpower Accident

The low ramp rate TOP accident has been analysed taking into account incoherence effects of pin failure in different fuel subassemblies /36/. The best estimate path leads through a relatively mild initiating phase into a phase of early shutdown with in-place coolability of the reator core (see the big hatched arrows in Fig. 1). Some uncertainties related to fuel freezing processes and blockage formation make it necessary to consider also an accident path leading into whole core involvement (see arrow 4 in Fig. 1). The conditions reached have some similarity with the transitory phase of an LOF accident.

4. PRESENT STATUS OF PHYSICS DESIGN AND ANALYSIS FOR SNR-1300

Two alternative core designs are studied in parallel for SNR-1300: a conventional core with 2 radial enrichment zones and a heterogeneous core. The design and constraints for both core types are almost identical (e.g. max. nom. lin. pin power: 415 W/cm, total power: 3420 MW(th) equivalent to about 1300 MW(e), pin and fuel element dimensions for the fissile and fertile regions, external circumference of the radial blanket). More design details and results of the core analysis may be found in /14,15/.

Possible advantages of heterogeneous versus conventional cores are e.g. a larger breeding ratio, a smaller neutron fluence, improvements concerning the power peaking factor and its variation during burnup, lower control and shut down reactivity requirements and especially a reduction of the sodium void reactivity and its possibly favorable influence on the safety behavior of the core. More or less severe disadvantages could be a higher fissile enrichment and inventory, larger core dimensions, difficulties with the mechanical and thermohydraulic behavior of the internal fertile regions during burnup (bowing, fixed or variable orificing, thermal stripes etc.), and a smaller Doppler reactivity effect. Often the performance of such cores is burdened by a so called looser neutronic coupling of spatially separated core regions (for more details see /11,12,13/).

Some integral values characterizing safety related nuclear properties of these two cores are summarized in Table I. The results shown have been derived on the basis of diffusion calculations using a Δ-Z model for the heterogeneous and an R-Z-model (confirmed by a few ΔZ model results) for the homogeneous core.

	Quantity	Dimension	Homog. reactor	Heterog. reactor
Fissile Region (Without Axial Blankets)	Max.Pos.Circulating Sodium Void Reactivity[1]	\$[2]	4.55	2.45
	Total Circulating Sodium Void Reactivity[1]	\$	4.16	2.09
	Total Clad Steel Loss Reactivity	\$	6.03	4.43
	Doppler Constant wet at 3000 K	10^{-3} Tdk/dT	-5.80	-3.91
	Doppler Constant dry at 3000 K	10^{-3} Tdk/dT	-4.32	-2.90
Internal Fertile Region (Including Axial Blankets)	Max. Pos. Circ.Sodium Void Reactivity	\$	-	0.90
	Total Circ. Sodium Void Reactivity	\$	-	0.82
	Total Clad Steel Loss Reactivity	\$	-	1.42
	Doppler Constant wet at 3000 K	10^{-3} Tdk/dT	-	-2.46
	Doppler Constant dry at 3000 K	10^{-3} Tdk/dT	-	-2.06

[1] excluding stagnant sodium between wrappers

[2] 1 \$ ≡ $3.80\cdot10^{-3}$ Δk/k and ≡ $3.88\cdot10^{-3}$ Δk/k for the hom. and het. reactor, respectively

Tab. 1: Important Safety Related Nuclear Characteristics of a Homogeneous and a Heterogeneous 1300 MW(e) LMFBR Design (at End of Equilibrium Cycle 2)

Basically, the lower sodium void reactivity of a heterogeneous design may be attributed to the increase in enrichment which is necessary in order to maintain criticality of the core. The higher enrichment and the associated increased neutron leakage into the internal fertile (blanket) regions leads to a less steep increase of the neutron importance with increasing energy in the region above some 100 keV. Thus, the flatter adjoint function together with a more negative leakage term lead to a considerably less positive sodium void reactivity for a loss of coolant perturbation in the fissile region of a heterogeneous core. Naturally, the reduction of the sodium void coefficient depends on the degree and kind of heterogeneity chosen and has to be determined for each design individually. Designs which at first glance may look quite similar may exhibit quite different sodium void reactivities. This fact together with the usually weaker neutronic coupling and the pronounced influence of perturbations e.g. by control rod movements stresses the need to perform detailed studies in critical assemblies for these fairly complicated configurations. Heterogeneous core arrangements similar to the SNR-1300 heterogeneous core design were studied within the frame of the BIZET-programme at Winfrith. The programme was terminated mid 1980. A detailed analysis of the experiments is still underway. Table II shows the present state of art within the SNR-project for calculated/measured integral parameters of large LMFBR cores.

C/E	SNR-300 KNK-II	Large LMFBR's (1300 MWe)	
		homogeneous	heterogeneous
K_{eff}	1.005-1.011	0.999-1.005	analysis underway (BIZET, RACINE)
Power distribution within the core	1.0 ± 0.04	1.0 ± 0.05	
Breeding ratio	1.02 ± 0.06	1.04 ± 0.06	
Absorber rod reactivity	0.81-1.21	0.86-1.17	
Na-void Effect in core center	0.75-1.25	0.9-1.1	
Doppler-effect	0.82 ± 0.03	(0.82 ± 0.1)	

Tab. 2: Ratios of Calculation/Experiment for Reactor Physics Parameters in Various Types of Sodium Cooled Fast Reactors

4.1 Calculational Tools for Heterogeneous Cores

Improved calculational tools have to be used for the analysis of heterogeneous cores. Because of their specific complicated geometry leading to the presence of many internal boundaries within the core region the application of diffusion approximation is sometimes not very suitable and may be a too crude method. Thus, transport theory may be the proper method to choose and this choice may sometimes be mandatory. In addition, the 2-dimensional cyclindrical R-Z-models which are usually sufficient for the calculation of conventional cores are only adequate for screening studies of heterogeneous cores and possibly for the determination of their fissile inventory, their breeding properties and rough calculations of the sodium void reactivity. They have to be replaced by 3-dimensional hexagonal-z models to obtain more accurate values especially for the power shape, control rod worths and sodium-void coefficients. But presently the application of 3-d methods is usually restricted to diffusion theory. Even then the computer time required for accurate Na-void calculations using a sufficient number of energy groups may be quite considerable since the convergence behavior for heterogeneous cores is usually worse than for homogeneous ones. Due to the increased occurence of interfaces between regions of different material composition and the presence of these boundary surfaces in regions of high neutron flux and importance, the validity of the usual conventional homogenization procedures has to be considered more carefully for a proper transition from the heterogeneous cell arrangement to a homogenized material composition which is often used in global reactor calculations.

4.2 Safety Analysis of SNR-1300 Cores

Preliminary results of safety analysis for the conventional core design has been obtained completed recently. It was performed in a manner similar to earlier studies /16,17/ for a 2000 MW(e) LMFBR. Preliminary results are available /18/ for the CDA initiated by a pump coast down followed by a failure of both shut down systems. Similar studies for the heterogeneous core design have been started but results are not available yet. A fair copmparison is therefore not possible yet.

Since the safety aspects of a reactor design depend in a fairly complex and complicated way on its neutronic, thermohydraulic and mechanical characteristics it is necessary to perform a detailed safety analysis for both alternative core designs. Only a quantitative safety analysis of the complete accident progression for representative accidents can show the merits of design differences and thus allows reliable conclusions on the safety performance of the reactor. Furthermore, one should be aware that an accident analysis approach which uses pessimistic assumptions to cover the uncertainties in the data and theoretical modelling might not provide a realistic comparison of homogeneous versus heterogeneous core accident behavior and this approach is, therefore, of limited usefulness.

4.3 Role of Sodium Reactivity Effect in the Accident Analysis

Based on the fact that the sodium void reactivity is one of the most important reactivity coefficients which influence the accident sequence - at least during the initial phase of LOF initiated CDA - there is a tendency to reduce its magnitude by design modifications. One aims at coming down from about 4-5 $ for conventional large LMFBR cores to values close to 1 or 2 $ for heterogeneous cores. It is difficult to achieve this restriction with reasonable design modifications for radially heterogeneous cores /12/. However, there exist possibilities for heterogeneous core arrangements to attain values between 2 and 3 $. Such sodium void reactivities should also be acceptable, because the importance of the sodium void reactivity for the core safety behavior should not be overestimated. Other effects, e.g. positive reactivity effects by voiding of internal fertile subassemblies, steel relocation, fuel slumping and the associated phenomena of freezing and blockage of coolant channels have a dominant influence in later stages of the accident transient. These predominantly positive reactivity effects will still play their dominant role even if the core will be designed for very small positive - or even zero - sodium void reactivity coefficients. In addition, the small positive sodium void coefficient of heterogeneous cores is usually combined with a reduced Doppler reactivity of the fissile region. The Doppler feedback reactivity associated with the temperature increase of the fuel, however, makes a strong, negative contribution to the power-coefficient and mainly determines (for a given reactivity ramp) the energy release during the disassembly phase.

5. ON SOME KEY ISSUES OF SAFETY RESEARCH FOR CORE DISRUPTIVE ACCIDENTS

In section 3 and 4 the current accident analysis procedure for the SNR-300 and the present status of analyses for SNR-1300 have been explained. Uncertainties for some of the key phenomena with respect to accident energetics in the SNR-300 safety analysis have been treated by using pessimistic assumptions for the accident simulation. This procedure is practicable for the SNR-300. For large commercial LMFBR's like SNR-1300 a similar procedure could lead to a very expensive plant design. Therefore it is highly desirable to develop a better understanding of the key phenomena.

In this section the influence of some of the key phenomena on accident progression will be briefly discussed and it will be explained which R&D efforts are required for a better understanding of these phenomena.

5.1 Neutron Physics Evaluation of Accident Relevant Material Configurations

In the course of the CDA analysis the reactivity and power distribution must be determined for material configurations which deviate significantly from that of the reactor in normal operation. These configurations are characterized by

- cavities and streaming channels resulting from melting, relocation and expulsion of some or all materials from portions of the core
- compacted fuel and structural material, resulting from the movement of molten materials into neighbouring parts of the reactor or from lateral compaction of fuel elements.

Experiments to verify the calculational methods for this type of configurations and in particular for the reactivity effects of fuel slumping were performed in ZPPR 5 - a mock-up of the Clinch River Breeder Reactor /19/ -, and in an assembly in ZPR 9 which was especially designed to test safety related physics parameters /20/. Evaluation with both diffusion and transport theory have shown considerable discrepancies between measurements and calculations by non negligible amounts.

In SNEAK (critical facility at the nuclear research center Karlsruhe, Germany), an extended series of experiments on the effect of material movements in accidents is presently being performed (SNEAK 12), covering a wide variety of configurations for the first time, this type of experiments will be performed in both plate and pin geometry (Fig. 2).

The core of SNEAK 12A is constructed of plates and uses 20% and 35% enriched uranium metal plates as fuel. The experimental program is being performed mainly in the central 16 elements of the assembly. It comprises the following steps:

1) Introduction of cavities of varying sizes up to 8000 cm^3
2) Introduction of streaming channels through the core and through core and blanket in one and four SNEAK-elements.
3) Slumping and compaction of steel in core and blanket
4) Slumping and compaction of fuel in core and blanket
5) Simulation of molten pools in which fuel and steel are mixed or vertically separated.

The use of SNEAK plates in the test zone allows independent changes of concentration for fissile and fertile material, structural material and sodium. The streaming effects for plate cells are not typical for fast breeder pin lattices, their order of magnitude, however, can be tested by using horizontal and vertical plate orientations.

The heterogeneity and also the composition of a fast breeder lattice is better simulated in SNEAK 12B where the testzone consists of a PuO_2UO_2 pin lattice (15% Pu enrichment, 8.2 mm pin diameter). As of present planning the rods are placed into calandrias, assuming a condition during which the sodium voiding has already taken place. In order to allow a simulation of various types of material movement, the pins in the 16 central elements are vertically subdivided into 7 core sections and a top and bottom blanket section. Pin sections can be taken out of selected regions and be compacted in other ones, reaching 1.5 times the original material concentrations thus simulating material movements in vertical and horizontal directions during slumping processes and the formation of cavitites. According to preliminary estimates, changes of neutron streaming through the pin lattice caused by vertical dislocation of pin sections in the envisaged experiments will bring about only a relatively small fraction of the total reactivity effect.

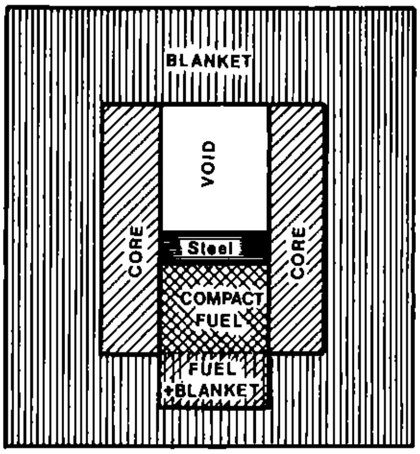

Fig. 2a: SNEAK 12A MOLTEN POOL CONFIGURATION

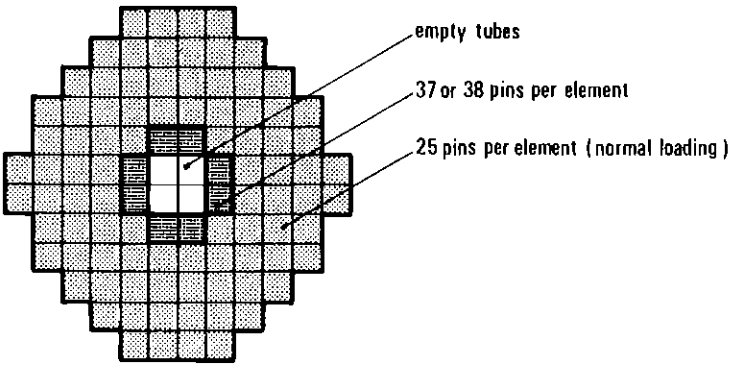

Fig. 2b: SNEAK 12B TESTZONE AFTER RADIAL DISLOCATION OF CENTRAL PINS

However, corresponding changes of the streaming properties for the radial direction are expected to constitute the main component of the effect for horizontal fuel movement. Therefore the use of pins for this type of experiments is essential.

The evaluation of the experiments asks for the application of specific transport theoretical methods. A number of methods which are important in this context were developed at KFK in recent years. These include: One and two dimensional transport perturbation codes /21/, collision probability methods for plate and pin cells /22/, a cell code taking into account the conditions at region interfaces /23/, and a new method to treat streaming through empty channels /24/. These methods will be further developed, particularly in order to allow the treatment of pin geometry and of anisotropic diffusion in all types of calculations which so far is possible only partially. In addition, the currently used KFKINR group constant set /25/ will be replaced by an updated, improved version taking into account presently available evaluations of nuclear cross section data.

The test and validation of the applied methods and data in the analysis of the critical experiments described here will establish a more accurate and reliable basis for the nuclear part of LMFBR safety analysis and help to identify the needs where improvements in data and methods could reduce the uncertainty margins in the prediction of the LMFBR core safety behaviour.

5.2 Other Key Phenomena Requiring more R&D-effort

Apart from the mainly neutronic aspects of future safety related research activities mentioned in section 5.1, Table 3 shows a list of several physical key phenomena which require intense future R&D efforts.

Table 3: Key Phenomena Requiring More R&D Effort

PHASE OF INTEGRAL CORE MATERIAL MOVEMENT AND MILD DISCHARGE:	● TRANSIENT BEHAVIOR OF BOILING FUEL/STEEL POOLS ● SECONDARY EXCURSIONS (SPACE DEPENDENT KINETICS) ● STEEL VAPOR DRIVEN DISCHARGE OF CORE MATERIAL (CONTACT WITH SODIUM)
ENERGETIC DISASSEMBLY AND DISCHARGE PHASE:	● CORE MATERIAL EXPANSION AND DISCHARGE UNDER THE PRESENCE OF SODIUM ● BEHAVIOR OF UPPER CORE STRUCTURES DURING ENERGETIC DISCHARGE OF CORE MATERIAL
POST ACCIDENT HEAT REMOVAL PHASE:	● MELT-THROUGH OF STRUCTURES ● BEHAVIOR AND COOLING OF FUEL/STEEL PARTICULATE BEDS IN SODIUM

Table 3: Continued

INITIATING PHASE:	● AXIAL FUEL EXPANSION ● CLAD MATERIAL MELTING, MOVEMENT, AND FREEZING ● EARLY FUEL DISPERSAL IN SODIUM VOIDED FUEL ELEMENTS ● FUEL PIN FAILURE, FAILURE PROPAGATION AND FUEL MOVEMENT IN PARTLY VOIDED FUEL ELEMENTS (DURING LOF DRIVEN TOPs)
TRANSITORY PHASE:	● FUEL/STEEL PENETRATION INTO AXIAL AND RADIAL STRUCTURES (BLOCKAGE FORMATION, REMELTING) ● MELTTHROUGH OR MECHANICAL FAILURE OF HEXCAN WALLS

The axial fuel expansion due to fuel temperature increase has - if fully effective - a large negative reactivity effect. The CABRI Program /38/ is expected to provide the required experimental data.

Clad material melting and movement, occuring separately from fuel melting and movement, is especially important for reactor core designs with a low maximum positive sodium void reactivity effect (e.g. heterogeneous core designs). This clad material movement could cause positive reactivity effects and/or blockages above or below the active core region. The latter effect could lead to a bottled-up pool configuration and thus complicate - inspite of a relatively mild primary excursion - this later phase of the accident (potential of secondary excursions). Out of pile experiments /39,40/ at KfK and in-pile experiments /41,42,43/ have already and will be extremely useful. In addition, improved theoretical simulation models have been developed /44/ at KfK and were incorporated into the SAS3D code system /45,46/.

Early fuel dispersal in sodium-voided fuel-elements is extremely important for assuring a mild primary excursion, i.e. for avoiding a LOF driven TOP situation. In-pile experiments /37,38,46/ contributed already to a better understanding of this phenomenon. Additional experiments are planned within a cooperative effort between SANDIA and KfK. Theoretical simulation methods for fission gas behavior and fuel breakup/motion in sodium voided subassemblies are under development at KfK /47,48/.

Fuel pin failure, failure propagation and fuel movement in partly sodium voided fuel elements during LOF driven TOPs will strongly influence the energetics of LOF driven TOP events /49/. In-pile experiments in CABRI should provide more relevant information for these phenomena in the future.

Fuel/steel penetration into axial and radial structures together with a possible incoherent slumping of upper core and blanket subasembly stubs determine the accident development during the transitory phase; especially, these phenomena determine whether or not a bottled-up pool situation will develop. The results of reactor material freezing tests[50] indicate that the data base is incomplete and that a more unified theoretical modeling method is needed. A strong experimental and theoretical effort is planned at KfK.

Meltthrough or mechanical failure of hexcan-walls are also important for the evolution of the accident during the transitory phase because blockage formation, fuel dispersal phenomena, and failure of hexcan-walls (also for control rod subassemblies) are competing effects and the path of the accident depends on the outcome of this competition. Experimental and theoretical studies with reactor-prototypic boundary conditions are needed.

Transient behavior of boiling fuel/steel pools is another key phenomenon. Only recently a potentially adequate theoretical tool, i.e. SIMMER-II, was developed at Los Alamos Scientific Laboratory /33/ and was adopted at KfK. Scoping calculations by Bohl /34/ have shown an energetics potential for bottled-up pool situations. It is necessary to evaluate critically whether these - partly assumed - initial and boundary conditions are possible during accident evolution of a specific reactor design. These conditions are characterized by a bottled-up pool situation with a high inventory of mobile fuel and the potential of coherent material movement. If such situations cannot be avoided they have to be investigated experimentally and theoretically.

SIMMER analysis of energetic expansion and discharge processes /35,51/ indicate that the transformation of thermal into mechanical energy yields dramatically reduced mechanical energies (by an order of magnitude) if compared to the isentropic fuel work potential (isentropic expansion to the cover gas volume). This is very encouraging, but the presence of sodium and the question whether the upper core structures stay intact during the expansion process need more attention. An experimental testing program was recently initiated at KfK.

If in-vessel cooling of fuel debris should be accomplished for large commercial LMFBR's one needs more information on penetration and meltthrough of structures and on the behavior of fuel/steel particulate beds in sodium. This is so because the fuel debris distribution process within the reactor vessel depends very much on the time needed for penetration and meltthrough of structures and on the coolability of particulate beds in sodium. Some remarkable progress has been achieved on the particulate bed cooling issue by some in-pile experiments at SANDIA /52/. A combined in-pile and out-of-pile effort was recently initiated by SANDIA and KfK.

In all accident phases a good knowledge of thermodynamic data for reactor mterials (fuel, steel, sodium, fission products) is necessary, partly up to relatively high temperatures (e.g. 5000 K). At KfK much effort has been put into experimental and theoretical work for equation of state data of fuel, steel and sodium /53,54,55/.

6. RESULTS AND CONCLUSIONS

6.1 The accident analysis must consider the complete path of an accident starting with an initiating event, leading through several phases up to mechanical and/or thermal loading of the vessel, piping and containment systems and up to permanent cooling of core-debris and possibly release of radioactivity. Separate limiting cases must be considered for mechanical loading and for thermal loading of the primary vessel and piping system.

6.2 Some key phenomena of the later accident phases especially phenomena which are crucial for the occurance of secondary excursions need more attention of the experimental and theoretical safety analysists in order to have a more balanced understanding of the whole accident path (a benign behavior for the primary excursion could lead to a more complicated and energetic situation for the later accident phases).

6.3 Cores with a low maximum void reactivity effect (e.g. heterogeneous cores) should not be judged on the basis of limiting energetics accident behavior of the primary excursion alone. The merits of different designs can be judged reliably only on the basis of a complete, thorough and realistic safety analysis. Especially, a heterogeneous core with a low void reactivity worth might encounter a relatively mild primary excursion, but clad material movement, which in low void worth cores precedes the fuel movement, could form axial blockages which might in the later accident phases lead to more complicated and possibly more energetic situations (secondary criticality potential of bottled-up pools).

6.4 Structural behavior is not only an issue for the mechanical loading of the vessel and piping system at the end of an excursion, it is important in all accident phases and should receive more attention during the accident progression period.

6.5 There is still a large number of challenging problems to be solved for comercially sized LMFBRs, which include for example: neutron physics evaluation of accident relevant configurations, effective equations of state for irradiated fuel, multifield and multicomponent fluiddynamics coupled with space dependent neutron kinetics for analysis of secondary excursions, structural behavior during expansion of extremely hot core-materials, entrainment of sodium into a hot expanding two phase bubble, aerosol behavior and development of filter techniques.

6.6 A strong international cooperation in the experimental and theoretical solution of design and safety problems for commercial-size LMFBRs would be extremely valuable. Besides the BIZET- and RACINE-programs, anotherrecent example is SIMMER experimental testing and application to the later accident phases where international cooperation received more momentum during the last year. SIMMER is considered in Germany as an extremely flexible and usefull tool which should be further developed and validated to improve our understanding of key phenomena and to resolve some of the remaining important issues of core disruptive accidents.

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