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ANALYSIS OF HYPOTHETICAL LMFBR WHOLE-CORE ACCIDENTS IN THE USA


by

D. R. Ferguson, L. W. Deitrich,

N. W. Brown and A. E. Waltar





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# ANALYSIS OF HYPOTHETICAL LMFBR WHOLE-CORE ACCIDENTS IN THE USA*

D. R. Ferguson and L. W. Deitrich
Argonne National Laboratory

2695580

N. W. Brown
General Electric Company

A. E. Waltar
Hanford Engineering Development Laboratory

## I. INTRODUCTION

The issue of hypothetical whole-core accidents continues to play a significant role in assessment of the potential risk to the public associated with LMFBR operation in the USA. This paper briefly characterizes the changing nature of this role, with emphasis on the current risk-oriented perspective. It then describes the models and codes used for accident analysis in the USA which have been developed under DOE sponsorship and summarizes some specific applications of the codes to the current generation of fast reactors. An assessment of future trends in this area concludes the paper.

## II. THE NEED FOR ANALYSIS OF HYPOTHETICAL WHOLE-CORE ACCIDENTS

Historically, the hypothetical LMFBR whole-core accident has been an important part of fast reactor safety evaluation. This importance reflects a recognition of the theoretical possibility of a core compaction leading to significant energy release and a desire to provide public protection against a broad category of extremely unlikely events of potentially high consequence. For early, small reactors, analysis of hypothetical whole-core accidents was primarily aimed at providing a design basis for containment design, since containment structures capable of accommodating energy release predicted by

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bounding-case calculations were feasible. Using this approach, a qualitative, but unmeasurable, gain in public protection was obtained.

More recent consideration of breeder reactor demonstration plants has included mechanistic accident analysis in an attempt to demonstrate that the expected energy release from a hypothetical accident would be within the energy accommodation capability of a practical containment. Much of the analytical and experimental work done to the present has been directed at this eventual goal. However, the economic penalties associated with this approach could be severe and difficult to balance against qualitative gains in public safety.

It now appears necessary to extend the considerations of hypothetical accidents to include a more definitive determination of risk from such accidents as well as a deterministic statement of consequences. The concept of "Lines of Assurance" [1] with the associated goals stated in terms of smaller and smaller probabilities of more and more severe consequences reflects the increasing emphasis in the USA on quantitative risk assessment. The burdens placed by such an approach on the accident analysis capabilities, both analytical and experimental, are severe, since it is necessary to define quantitative probabilities at each accident sequence branch point. Accident analysis models and codes, supported by an extensive phenomenological and integral experimental data base, must be the major tool through which the risk assessments are made and supported through the design and licensing process.

## III. ANALYSIS METHODS AND COMPUTER CODE DEVELOPMENT

### A. Models of Material Behavior and Accident Phenomenology

Modeling of material behavior during hypothetical whole-core accidents has a principal focus in the area of fission gas behavior. The FRAS2 code [2] implements the latest modeling of intragranular fission-gas behavior. Like its predecessors, FRAS2 computes gas bubble migration and coalescense, from which is derived the gas release to grain boundaries and intragranular swelling.

Consideration of non-equilibrium bubbles, of the dynamics of bubble growth following coalescense, and of the work available from intragranular gas is included [3]. Redistribution of fission gas within networks of connected porosity is considered in the POROUS code [4], in which fuel is modeled as a porous medium. Pressure gradients available to produce net forces on fuel can be computed. Effects of intergranular gases on fuel microstructure during transients have been considered. A simple model for prediction of a threshold for intergranular void formation and consequent fuel swelling has been completed [5]. These microscopic-scale models and codes, coupled with a thermal-mechanical analysis of the fuel, will be used to predict the nature and timing of fuel pin disruption leading to early fuel dispersal, a phenomenon of great importance in mitigating the energetics of a loss-of-flow accident in a reactor like CRBR. Experimental support of fission gas behavior and fuel disruption modeling has come from fission-gas-release (FGR) [6], direct-electrical-heating (DEH), [7] and in-pile experiments [8]. Gas release fractions, bubble sizes, and microstructural changes are observations of interest. Experimental results suggest that the timing and rate of fuel motion depends on the transient thermal history and the fuel character.

Fuel-pin-mechanics models are required for evaluation of time and location of cladding failure in overpower situations in both transient overpower (TOP) and loss-of-flow (LOF) accidents. Several detailed codes are available [9], each of which incorporates its own special features and ideas. Generally, agreement with the current experiments is good, although extension of the data base to lower ramp rates, prototypic fluence-to-burnup ratio, and loss-of-flow-driven TOPs is needed. Very detailed codes are generally not suitable for accident analysis, so the concepts of a pressurized molten-fuel zone and strengthless solid fuel are used in SAS3D [10], and an empirical correlation is used in MELT-IIIA [11]. While satisfactory results may be attained using these methods for overpower transients at full flow, more detail may be required for analysis of the LOF-driven overpower situation, due to more complex thermal history.

Sodium voiding and cladding relocation have received much attention due to the importance of these phenomena in the LOF accident initiating phase. The present one-dimensional SAS3D modeling of sodium boiling and voiding produces excellent results when applied to a 37-pin bundle not having excessive radial heat loss [12]. Large radial heat loss will retard voiding compared to the one-dimensional prediction. However, present one-dimensional cladding relocation models are thought to predict substantially too much upwards motion. In a large pin bundle, net cladding motion is expected to be slow. Two-dimensional flow diversion effects, not modeled in present accident analysis codes, appear to be important.

Fuel motion within coolant channels following pin failure has also received considerable attention. Key issues include the extent of fuel motion prior to channel plugging and the extent to which fuel and sodium move out of the core together. Present analytical models, such as SAS/FCI [13] and PLUTO [14], treat the intrapin fuel motion and fuel-coolant interaction in one dimension. Application of these models to a full subassembly involves considerable uncertainty due to two-dimensional effects introduced by incoherence in pin failure. Such effects appear to be of considerable importance to fuel motion within channels and in evaluation of post-transient coolability.

Termination of hypothetical accidents with limited core damage may be possible in some cases (TOP, LOPI) with present designs, while other cases (LOF) may require special design features or modifications. All of the above models and codes address problems of relevance in determining the success of limited-core-damage termination, as well as in determination of initiating-phase energetics. If the accident is not terminated with limited core damage, fuel assembly disruption and whole-core involvement will follow. While direct disassembly from the initiating phase may obtain for certain conditions, a slow meltdown is the expected outcome. The critical issue in the meltdown process is demonstrating that the transition from a fuel-pin-and-duct geometry to a fully disrupted core takes place without fuel compaction leading to recriticality and potentially large energy release. Such matters as penetration of molten fuel through above-core structures are under active investigation to define the probable state of the disrupted core and potential fuel dispersal paths.

B. Whole-Core Accident Analysis Codes

The complex nature of hypothetical whole-core accidents necessitates the use of large-scale computer codes if the accident scenarios are to be followed in any detail. With increased emphasis in the US program being placed on examining the potential for terminating the progression of such accidents short of whole-core involvement (LOA2 termination), the need for these codes becomes even greater. Thus substantial efforts are being devoted in the US to the enhancement of existing whole-core accident analysis codes and to the development of new codes.

The best-known and widely-used of the US accident analysis tools is the SAS code family. The current production version is SAS3D, a streamlined version of SAS3A [15] which includes the capability of treating a large number of channels (up to 34). SAS3D includes models which allow an integral treatment of both LOF and TOP accident sequences through fuel pin disruption and up to the point of subassembly disruption. In addition to models for fuel characterization after steady-state irradiation, steady-state and transient heat transfer and coolant flow, and space-independent reactor kinetics, SAS3D includes models for sodium boiling and voiding with a moving liquid film, molten cladding relocation (CLAZAS) [16] and fuel motion (SLUMPY) [17] in voided subassemblies, and pin failure and fuel motion in unvoided subassemblies (SAS/FCI) [13]. It interfaces through a series of binary files with the DIF3DS two- and three-dimensional multigroup neutron diffusion/perturbation code. An iterative procedure built into the two codes provides a highly automated capability for achieving a consistent steady-state neutronic and thermal-hydraulic description of the LMFBR. DIF3DS is a derivative of the DIF3D code [18].

The SAS4A code [19], currently undergoing development, will provide substantially more powerful models for fuel pin dynamics, fuel pin disruption, and fuel and cladding relocation. Its SSFUEL/DEFORM model will provide a detailed description of the thermal and mechanical state of fresh and irradiated fuel pins at transient initiation. The PLUTO2 [20] model, already in operation in a stand-alone version, will treat both the intrapin and coolant channel fuel relocation following cladding failure in unvoided or partially-voided subassemblies. The LEVITATE model extends the PLUTO2 modeling concept to treat material relocation in voided regions of subassemblies where the cladding offers no radial restraint. The neutronics treatment in SAS4A will be based on two- or three-dimensional time-dependent multigroup diffusion theory and a variation of the improved quasistatic solution method.

The MELT-IIIA code [21] is the most recent production version of a coupled neutronics, thermal-hydraulics computational system for analysis of protected and unprotected TOP accidents. It has proved particularly useful for analyzing mixed-oxide-fueled LMFBR transient response, because of its direct link to the experimentally-based SIEX [22] steady-state fuel performance code. This emphasis on experimental observations has allowed the code to be used with considerable confidence for both protected and unprotected TOP sequences.

In the area of LOF transition-phase modeling, the TRANSIT-HYDRO code, latest member of the TRANSIT code family [23], is undergoing pre-release testing. It provides a mechanistic, subassembly-by-subassembly, analysis of the core from the time SAS models break down until permanent subcriticality has been reached. Molten region propagation, blockage formation, and fuel removal from the core are modeled. Reentry of core material as a result of melting of above-core structure can be identified. The effect of neighboring disrupted subassemblies on voiding and disruption of intact fuel subassemblies is incorporated. A quasi-three-dimensional hydrodynamics model in TRANSIT-HYDRO follows the independent axial motion of three components - fuel and steel vapor, molten fuel, and molten steel - in each of the one or more molten regions.

Assessment of the potential for energetic recriticality during the early portion of the transition phase is addressed by the FUSS [24] system (FUMO/SAS3A/NUS). This system links SAS3A with the FUMO [25] fuel hydrodynamics model, to allow the initiating phase analysis to be extended past early fuel disruption up to duct-wall melt-in. A Neutronics Update System provides corrections to the nominal point-kinetics routine for disconfigured cores.

VENUS-II [26], the current primary tool for analysis of energetic hydrodynamic disassembly, provides a refined model of the basic Bethe-Tait concept. It includes a two-dimensional Lagrangian hydrodynamics formulation, an adiabatic energy deposition model, an energy-density-dependent equation-of-state, and a point-kinetics treatment of the neutron kinetics. The soon-to-be-released FX2/VENUS-III code [27] offers additional capabilities, including a multi-component, two-field, Eulerian hydrodynamics model, a model for energy and mass transfer between fields, and space-time neutron kinetics.

An even more sophisticated extended-material-motion disassembly and transition-phase analysis capability is offered by the SIMMER code family [28], of which SIMMER-II is the current version. It is being described in detail in other papers being presented at this conference and will not be discussed further here.

As each of the above-mentioned codes - SAS4A, MELT-IIIA, TRANSIT-HYDRO, FX2/VENUS-III, and SIMMER-II - are being completed, emphasis is being shifted from starting yet another round of development of even more sophisticated models and codes (as has been the case in the past) to initiating a rather long-term program of code and model validation by extensive analysis of existing experiments and cooperative planning of additional relevant in-pile and out-of-pile tests. This shift in direction arises from the recognition that, in a licensing arena, a computer code is only as good as its experimental basis and that the models in the above-mentioned codes have out-stripped the present experimental base in key areas, particular in post-pin-failure fuel and steel relocation dynamics.

## IV. APPLICATIONS OF CODES AND METHODS

Over the past few years, selected hypothetical whole-core accidents in both the FFTF reactor and Clinch River Breeder Reactor Plant (CRBRP) have been analyzed using the methods and codes just described. Primary attention has been focused on the unprotected LOF and TOP accidents.

For the FFTF reactor, with its overall negative sodium-void reactivity worth, it has been concluded from studies with SAS3A that an unprotected LOF accident inevitably leads to whole-core voiding and fuel melting. However, incoherency in fuel melting limits the reactivity addition rate should molten fuel slump, thus largely precluding an early energetic disassembly [29]. Rather, a gradual meltdown of the core known as the transition phase follows. Consideration of the dispersive nature of a boiling oxide-fuel/steel mixture has provided a strong argument against the possible occurrence of energetic recriticalities during this transition phase [30], leaving post-accident debris accommodation as the principal concern.

Analyses of the LOF accident in CRBRP with SAS3A and SAS3D indicate a greater potential for an energetic initiating phase termination. Voiding in the lead subassemblies causes a power increase to 10-30 times nominal before fuel motion begins because of the positive sodium-void reactivity worth in CRBRP. Under these conditions, there seems to be a potential for sufficiently prompt and rapid fuel dispersal to preclude early energetic disassembly [31]. However, if the voiding-driven power burst causes pin failures in unvoided subassemblies, the potential exists for rapid FCI-induced voiding to take the power even higher (the LOF-driven TOP). The potential for energetic initiating phase termination is judged to be greater for larger LMFBRs which would have larger sodium void worths than the CRBRP.

This concern over the possibility of increased LOF accident energetics potential of larger reactors with several dollars of positive sodium void worth has spawned efforts in the US along several paths, all aimed at reducing this concern for large LMFBRs. Phenomenological modeling and experimental efforts discussed in Section III.A are aimed at determining the extent to which the prompt release of fission gas could result in sufficient early fuel dispersal to cancel out the positive reactivity insertion due to voiding. Should an LOF-driven TOP event lead to a hydrodynamic disassembly of the core, studies like those being done with the SIMMER code [32] point out that a good portion of the energy contained in the molten and vaporized core material at burst termination may be harmlessly dissipated through various mechanisms as it expands from the core region. Attention has also been focused on low-void-worth heterogeneous core designs as it has been shown [33] that such cores may have significantly reduced energetics potential. Finally, efforts are being initiated under the LOA2 program in the US to explore various core and primary system design alternatives which might result in an increased potential for LOF accident termination short of whole-core involvment.

It has been recognized that an unprotected TOP accident would not necessarily lead to total core involvement, with the subsequent questions of gross core rearrangement, recriticality, and/or core disassembly. Rather, analysis and in-pile testing have confirmed that pin failures during TOP transients will likely be well above the core midplane, such that molten fuel discharge into the coolant channel will constitute a negative reactivity feedback to arrest the accident. There remains a substantial question, however, on the ultimate disposition of the once-molten fuel. If it is preferentially swept out of the core via the hydraulic coolant forces, the fuel motion negative reactivity feedback is considerably strengthened and long term cooling is guaranteed. However, if the fuel should solidify near the rupture location, coolability becomes more difficult to demonstrate. The data on blockage potential are mixed. Hence, analytical efforts have been directed towards a more refined predictive capability for pin failure patterns within the subassemblies. Both statistical analyses of time and location of pin failure and studies which account for local power/flow variations within a subassembly [34] provide strong support for the belief that pin failures will occur incoherently. Pins near the high power/flow regions will be the earliest pins to fail. Even if local blockage should occur, the reactivity loss due to fuel motion will reduce the power such that coolant diverted around the blockage should prevent failure of all the pins. Recent experimental results [35] indicate a propensity for partial fuel sweepout. Analytical efforts are underway [36] to provide a better understanding of the events surrounding a partial blockage condition.

## V . DIRECTION OF FUTURE EFFORTS

Development of reliable accident analysis models, methods, and codes, backed by integral and separate-effects data from in-pile and laboratory experiments continues to be a key effort. Certain areas, such as sodium boiling, appear to be satisfactory at present. Others, such as early fuel dispersal due to fission gas effects, fuel failure and motion in LOF-driven TOP, and the details of transition from intact to disrupted subassembly geometry require substantial additional modeling and experimental effort. Potential introduction of alternate fuel materials and reactor design features aimed at termination with limited core damage emphasizes the need for improved understanding of basic phenomena so that confident analytical evaluations can be made and experiment programs designed for maximum benefit in verifying accident scenarios.

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