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The SAS4A/SASSYS-1 Safety Analysis Code System, Version 5

Nuclear Engineering Division

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Chapter 1:
Introduction

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INTRODUCTION

The SAS4A/SASSYS-1 computer code is developed by Argonne National Laboratory for thermal, hydraulic, and neutronic analysis of power and flow transients in liquid-metal-cooled nuclear reactors (LMRs). SAS4A was developed to analyze severe core disruption accidents with coolant boiling and fuel melting and relocation, initiated by a very low probability coincidence of an accident precursor and failure of one or more safety systems. SASSYS-1, originally developed to address loss-of-decay-heat-removal accidents, has evolved into a tool for margin assessment in design basis accident (DBA) analysis and for consequence assessment in beyond-design-basis accident (BDBA) analysis.

SAS4A contains detailed, mechanistic models of transient thermal, hydraulic, neutronic, and mechanical phenomena to describe the response of the reactor core, its coolant, fuel elements, and structural members to accident conditions. The core channel models in SAS4A provide the capability to analyze the initial phase of core disruptive accidents, through coolant heat-up and boiling, fuel element failure, and fuel melting and relocation. Originally developed to analyze oxide fuel clad with stainless steel, the models in SAS4A have been extended and specialized to metallic fuel with advanced alloy cladding.

SASSYS-1 provides the capability to perform a detailed thermal/hydraulic simulation of the primary and secondary sodium coolant circuits and the balance-of-plant steam/water circuit. These sodium and steam circuit models include component models for heat exchangers, pumps, valves, turbines, and condensers, and thermal/hydraulic models of pipes and plena. SASSYS-1 also contains a plant protection and control system modeling capability, which provides digital representations of reactor, pump, and valve controllers and their response to input signal changes.

1.1 SAS4A/SASSYS-1 Background

In the late 1960s, the then U.S. Atomic Energy Commission gave development of a liquid-metal-cooled fast reactor (LMR) a high priority, and the development of the Fast Flux Test Facility (FFTF) became a cornerstone of that program. To provide adequate support for the FFTF and for the expected LMRs to follow, a major base technology program was established which provided a continuous stream of experimental information and design correlations. This experimental data would either confirm design choices or prove the need for design modifications. At the time, the “tremendous amount of data and experience pertaining to thermal design” of LMRs was recognized as providing the technical foundation for the future commercial development of LMRs.[1-1]

Along with the generation of experimental data came the development of safety analysis methods that used that data in correlations for mechanistic, probabilistic, or phenomenological models. These models were developed for a variety of needs ranging from individual components, such as heat exchangers, pumps, or containment barriers, to whole core or even whole-plant dynamics. A major portion of the overall technical

effort since that time has been allocated to safety considerations, and the SAS4A/SASSYS-1 safety analysis code is the result of that dedication.

Perhaps the strongest factor that influenced early fast reactor safety analysis was the concern over the possibility of core compaction followed by an energetic core disassembly — the so-called Bethe-Tait accident.[1-2] In the late 1960s, the Hanford Engineering Development Laboratory (HEDL) began developing the MELT code[1-3,1-4] to evaluate the initiating phase of hypothetical core disruption accidents (HCDA) as part of the FFTF project. The MELT series of codes has the capability to model the transient behavior of several representative fuel pins (channels) within a reactor core to allow for incoherency in the accident sequence. By 1978 MELT had evolved into the MELT-IIIB code.[1-4]

Around the same time that development on MELT began, Argonne National Laboratory began developing the SAS series of codes.[1-5–1-9] Like MELT, SAS has the capability to model the transient behavior of several representative channels to evaluate the initiating phase of HCDAs. SAS1A originated from a sodium boiling model and includes single- and two-phase coolant flow dynamics, fuel and cladding thermal expansion and deformation, molten fuel dynamics, and a point kinetics model with reactivity feedback. By 1974, SAS evolved to the SAS2A computer code[1-6] which included a detailed multiple slug and bubble coolant boiling model which greatly enhanced the ability to simulate the initiating phases of loss-of-flow (LOF) and transient overpower (TOP) accidents up to the point of cladding failure and fuel and cladding melting.

The SAS3A code [1-7] added mechanistic models of fuel and cladding melting and relocation. This version of the code was used extensively for analysis of accidents in the licensing of FFTF. In anticipation of LOF and TOP analysis requirements for licensing of the Clinch River Breeder Reactor Plant (CRBRP), new fuel element deformation, disruption, and material relocation models were written for the SAS4A version of the code,[1-8] which saw extensive validation against TREAT M-Series test data. In addition, a variant of SAS4A, named SASSYS-1, was developed with the capability to model ex-reactor coolant systems to permit the analysis of accident sequences involving or initiated by loss of heat removal or other coolant system events. This allows the simulation of whole-plant dynamics feedback for both shutdown and off-normal conditions, which have been validated against EBR-II Shutdown Heat Removal Test (SHRT) data and data from the FFTF LOF tests.

Although SAS4A and SASSYS-1 are generally portrayed as two computer codes, they have always shared a common code architecture, the same data management strategy, and the same core channel representation. Subsequently, the two code branches were merged into a single code referred to as SAS4A/SASSYS-1. Version 2.1 of the SAS4A/SASSYS-1 code [1-10,1-11] was distributed to Germany, France, and Japan in the late 1980s, and it serves as a common tool for international oxide fuel model developments.

Beyond the release of SAS4A/SASSYS-1 v 2.1, revisions to SAS4A/SASSYS-1 continued throughout the Integral Fast Reactor (IFR) program between 1984 and

1994,[1-12] culminating with the completion of SAS4A/SASSYS-1 v 3.0 in 1994.[1-13] During this time, the modeling emphasis shifted towards metallic fuel and accident prevention by means of inherent safety mechanisms. This resulted in 1) addition of new models and modification of existing models to treat metallic fuel, its properties, behavior, and accident phenomena, and 2) addition and validation of new capabilities for calculating whole-plant design basis transients, with emphasis on the EBR-II reactor and plant [1-14], the IFR prototype. The whole-plant dynamics capability of the SASSYS-1 component plays a vital role in predicting passive safety feedback. Without it, meaningful boundary conditions for the core channel models are not available, and accident progression is not reliably predicted.

By the mid 1990s, SAS4A/SASSYS-1 v 3.1 had been completed as a significant maintenance update, but it was not released until 2012.[1-15]

1.2 SAS4A/SASSYS-1 Version 5

In the time since the development of Version 3, several modeling additions and enhancements have been made to meet U.S. Department of Energy programmatic needs. Significant among these are

- Detailed sub-channel models for whole-core analyses to resolve intra-assembly temperature and flow distributions [1-9]
- 3D visualization capabilities for sub-channel results.
- Extended decay-heat models to support long-term transients and complex, actinide-bearing fuels.
- Support for coupling with external CFD simulations to resolve flow distribution and thermal stratification effects.
- Treatment of axial expansion feedback from assembly duct walls
- Support for spatial kinetics (requires DIF3D-K)
- Extension of the control-system model to include sinusoidal functions that can be used to represent seismic oscillation effects.
- Addition for heavy liquid-metal coolants (lead and lead-bismuth eutectic)
- Support for user-defined coolant properties
- Detailed steam-generator model updates
- Several bug fixes and other enhancements

In addition to the above, a major restructuring of the code has been completed to adapt all source files to free-form source format and new model developments are being implemented using modern object-oriented practices. Documentation Overview

1.3 Documentation Overview

The rest of this manual contains details of the modeling capabilities of SAS4A/SASSYS-1. The chapter organization shown in Table 1.3-1 reflects the major model delineations. Each chapter provides in-depth descriptions of the models, including model formulations, solution techniques, and input descriptions. It is critical that users understand the relationships between their model input and the model

formulations given in this manual. Failure to understand these relationships can result in broken models and misleading results.

SAS4A/SASSYS-1 provides a detailed, multiple-channel thermal/hydraulic treatment of the reactor core. Each channel represents a fuel pin, its cladding, the associated coolant, and a fraction of the subassembly duct wall. Other positioning hardware, such as wire wraps or grid spacers, is usually lumped into the structure field with the duct wall. Within a channel, the flow is assumed to be one-dimensional in the axial direction, and the temperature field in the fuel, cladding, coolant, and structure is assumed to be two-dimensional in the radial and axial directions. Usually, a channel represents an average fuel element in a subassembly or a group of subassemblies. A channel may also represent pins in blanket or control subassemblies. Alternately, a single channel may also be used to represent the hottest pin in an assembly, or any other subset of a subassembly. The axial extent of a channel covers the entire length of a subassembly, including the core, the axial blankets, the fission gas plenum and the spaces above and below the pin/cladding geometry. Different channels may be used to account for radial and azimuthal design geometry, power, coolant flow, and burnup variations within the reactor core.

Table 1.3-1: Organization of the SAS4A/SASSYS-1 Manual

Chapter	Subject
1	Introduction (this chapter)
2	SAS4A/SASSYS-1 User's Guide
3	Pin Heat Transfer and Single-Phase Coolant Thermal/Hydraulics Model
4	Reactor Point Kinetics and Reactivity Feedback Models
5	PRIMAR-4: Primary and Intermediate Loop Thermal/Hydraulics Model
6	Plant Control and Protection Systems Model
7	Balance-of-Plant Thermal Hydraulics Model
8	DEFORM-4 Oxide Fuel and Cladding Mechanics Model
9	DEFORM-5 Metal Fuel Cladding Mechanics Model
10	SSCOMP Metal Fuel Characterization Model
11	FPIN2 Metal Fuel and Cladding Mechanics Model
12	TSBOIL Two-Phase Coolant Thermal/Hydraulics Model
13	CLAP Molten Cladding Dynamics Model
14	PLUTO2 Fuel-Coolant Interaction Model
15	PINACLE In-Pin Fuel Relocation Model
16	LEVITATE Fuel Relocation Model

Chapter 2 contains a general user's guide for SAS4A/SASSYS-1, including a complete description of the standard input file. Although Chapter 2 includes a summary description of every input parameter, it is essential that users consult the relevant chapters to understand the relationship between the input and the model formulations.

Chapter 3 contains the description of the formulation for the SAS4A/SASSYS-1 pin heat transfer and single-phase coolant thermal/hydraulics model. The subassembly-to-subassembly heat transfer model has been improved, and axial conduction in the coolant has been added. A sub-channel model has been introduced to provide accurate predictions of intra-assembly temperature and flow distributions.[1-16] This modeling addition is being validated with results from the EBR-II Shutdown Heat Removal Tests [1-17] as part of an International Atomic Energy Agency Coordinated Research Project.[1-18]

Chapter 4 contains the description of the formulation for the SAS4A/SASSYS-1 reactor point kinetics, decay heat, and reactivity feedback models. A new addition to this module is the ability to represent more detailed decay heat characteristics in multiple regions of the core. This module provides the reactor power level to the core thermal/hydraulics models for determination of the heating rate in the fuel, and receives core materials temperature and geometry information to calculate the reactivity feedbacks employed in the solution of the point kinetics equations.

Chapter 5 presents a full description of the formulation for the PRIMAR-4 sodium loops thermal/hydraulic model. This model provides boundary coolant pressure and flow conditions for the core channel models, including transient heat losses through normal and emergency heat removal systems and the transient performance of pumps. PRIMAR-4 includes the option for multiple core inlet and outlet coolant plena, permitting exact representation of the actual EBR-II coolant systems geometry. Compressible volumes in PRIMAR-4 may also be coupled with external computational fluid dynamics simulations to better represent flow and temperature distributions during transients.

The plant control and protection system model described in Chapter 6 is mostly unchanged from prior versions of SASSYS-1, except for the addition a sinusoidal function to represent oscillations in control-system signals.

The balance-of-plant (BOP) model described in Chapter 7 was implemented to permit 1) improved simulation of EBR-II design basis transients, 2) whole-plant analysis of IFR designs for optimization of advanced reactor control system strategies, and 3) core temperature margin assessments in unprotected accident sequences (i.e. beyond design basis accidents (BDBA) and anticipated transients without scram (ATWS)). In these latter sequences, core response depends strongly upon the performance of the balance-of-plant, because the core neutronic and thermal/ hydraulic behavior is determined by the availability of heat sinks outside the core. The BOP model couples to PRIMAR-4 at the steam generator.

Chapter 8 provides a description of the DEFORM-4 fuel element behavior model for stainless steel-clad oxide fuel, which is unchanged from prior versions of SAS4A/SASSYS-1.

Chapter 9 contains the description of the DEFORM-5 model, which treats the transient behavior of stainless steel and advanced (HT-9) cladding for metal fuel elements. This model is aimed at predicting margin to cladding failure, and timing and location of failure in limiting transients. It includes physical phenomena unique to metallic fuel, such as fuel/cladding chemical interactions.

The SSCOMP model described in Chapter 10 reflects available metal fuel material properties evaluations recorded in the IFR Material Properties Handbook [1-19]. An efficient correlation technique has been implemented in all SAS4A/SASSYS-1 material properties routines that accurately generates the data from the IFR Handbook for use in all the modules of the code. It is planned to revise the material migration capability in SSCOMP for ternary fuel, to add models for fission gas generation and release, swelling, and all other phenomena needed to describe the transition from cold, clean, unirradiated conditions to hot irradiated conditions.

Chapter 11 contains the description of the FPIN2 metal fuel pin mechanics model [1-20]. FPIN2 is a validated model for metal fuel pin transient behavior. Unlike DEFORM-5, which treats only the cladding response, FPIN2 provides a finite-element solution of the fuel and cladding mechanics equations for the elastic/plastic response, including fission gas pressurization and migration, molten cavity formation and growth, and fuel/cladding chemical interaction and cladding thinning. The interface between SAS4A/SASSYS-1 and FPIN2 has been designed to permit stand-alone execution of FPIN2 for direct verification or to replace the FPIN2 thermal/hydraulics calculation with the SAS4A/SASSYS-1 counterparts for coupled calculations. The application for this model is design basis analysis of driver and experimental fuel elements in EBR-II for the purpose of margin-to-failure assessments.

The TSBOIL module for liquid metal coolant boiling and two-phase thermal/hydraulics calculations has been retained intact from previous versions of SAS4A/SASSYS-1. The current model includes a set of modifications to describe the sudden release of non-condensable fission gas from a cladding rupture in the upper fission gas plenum of metal fuel elements and the subsequent plenum blow-down and liquid coolant expulsion. This option has been used to assess the safety implications of long-term fuel element irradiations in EBR-II [1-21].

The CLAP and PLUTO2 models described in Chapters 13 and 14 are relevant only to oxide fuel, and have remained unchanged since the previous documentation.

The PINACLE model described in Chapter 15 and the LEVITATE model described in Chapter 16 have been upgraded for applications to metallic fuel [1-22]. The model enhancements added to PINACLE and LEVITATE for metal fuel include fuel/cladding and fuel/structure chemical interactions and fission gas generation and migration with fuel swelling. Preliminary analyses of TREAT M-Series in-pile metal fuel tests have been completed [1-23], and applications to severe accident sequences in metal-fueled IFR cores have been completed and documented [1-24].

REFERENCES

1-1. Y. S. Tang, R. D. Coffield, Jr., R. A. Markley, "Thermal Analysis of Liquid Metal Fast Breeder Reactors," Westinghouse Electric Corporation, Madison, Pennsylvania, American Nuclear Society Publications, 1978.

1-2. H. A. Bethe and J. H. Tait, "An Estimate of the Order of Magnitude of the Explosion When the Core of a Fast Reactor Collapses," RHM-56-113, April 1956.

1-3. A. E. Waltar, A. Padilla, Jr., and R. J. Shields, "MELT-II, A Two-Dimensional Neutronics-Heat Transfer Program for Fast Reactor Safety Analysis," Unpublished information, Hanford Engineering Development Laboratory, 1972.

1-4. K. K. Tabb, C. H. Lewis, L. D. O'Dell, A. Padilla, Jr., D. E. Smith, and N. P. Wilburn, "MELT-IIIB: An Updated Version of the MELT Code," Unpublished information, Hanford Engineering Development Laboratory, April 1979.

1-5. J. C. Carter, et al., "SAS1A, A Computer Code for the Analysis of Fast-Reactor Power and Flow Transients," ANL-7606, Argonne National Laboratory, October 1970.

1-6. F. E. Dunn, et al., "The SAS2A LMFBR Accident-Analysis Computer Code," ANL-8138, Argonne National Laboratory, October 1974.

1-7. M. G. Stevenson, W. R. Bohl, F. E. Dunn, T. J. Heames, G. Hoppner, and L. L. Smith, "Current Status and Experimental Basis of the SAS LMFBR Accident Analysis Code System," *Proc. Fast Reactor Safety Meeting*, American Nuclear Society, CONF-740401-P3, pp. 1303–3121, Beverly Hills, CA, April 2–4, 1974.

1-8. J. E. Cahalan, A. M. Tentner, and E. E. Morris, "Advanced LMR Safety Analysis Capabilities in the SASSYS-1 and SAS4A Computer Codes," *Proceedings of the International Topical Meeting on Advanced Reactors Safety*, American Nuclear Society, Vol. 2, pp. 1038–1045, Pittsburgh, PA, April 17–21, 1994.

1-9. F. E. Dunn, J. E. Cahalan, D. Hahn, H. Y. Jeong, "Whole Core Sub-Channel Analysis Verification with EBR-II SHRT-17 Test," *Proc. 2006 Int'l Congress on Advances in Nuclear Power Plants*, Reno, NV, June 6–8, 2006.

1-10. G. Birgersson, et al., Unpublished information, Argonne National Laboratory, 1988.

1-11. F. E. Dunn, et al., Unpublished information, Argonne National Laboratory, 1987.

1-12. C. E. Till, "Fast Reactor Directions: The LMR Integral Fast Reactor Program at Argonne," Proceedings of the International Fast Reactor Safety Meeting, Snowbird, Utah, August 12-16, 1990.

1-13. Argonne National Laboratory, Unpublished information, 1996.

1-14. J. Sackett, et al., "EBR-II Test Programs," Proceedings of the International Fast Reactor Safety Meeting, Snowbird, Utah, August 12-16, 1990.

1-15. T. H. Fanning, ed., "The SAS4A/SASSYS-1 Safety Analysis Code System," ANL/NE-12/4, Nuclear Engineering Division, Argonne National Laboratory, January 31, 2012.

1-16. F. E. Dunn, Unpublished information, Argonne National Laboratory, 1993.

1-17. D. Mohr, et al., "Loss-Of-Primary-Flow-Without-Scram Tests: Pretest Predictions and Preliminary Results," *Nuclear Engineering and Design*, **101**, 1987.

1-18. T. Sofu and L. L. Briggs, "Benchmark Specifications for EBR-II Shutdown Heat Removal Tests," *Proc. Int'l Congress on Advances in Nuclear Power Plants*, ICAPP 2012, Chicago, IL, June 24–28, 2012.

1-19. IFR Properties Evaluation Working Group, "Handbook of Properties of Metallic Fuels," Argonne National Laboratory, Idaho Falls, Idaho, June, 1988.

1-20. T. H. Hughes and J. M. Kramer, "The FPIN2 Code - An Application of the Finite Element Method to the Analysis of the Transient Response of Oxide and Metal Fuel Elements," Conference on the Science and Technology of Fast Reactor Safety, British Nuclear Energy Society, Guernsey, May 12-16, 1986.

1-21. J. P. Herzog and K. J. Miles, Unpublished information, Argonne National Laboratory, 1989.

1-22. A. M. Tentner, Unpublished information, Argonne National Laboratory, April, 1993.

1-23. A. M. Tentner and Kalimullah, Unpublished information, Argonne National Laboratory, 1993.

1-24. E. E. Morris and A. M Tentner, Unpublished information, Argonne National Laboratory, 1994.