



THE DOE TECHNOLOGY DEVELOPMENT PROGRAMME ON SEVERE ACCIDENT MANAGEMENT

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Abstract

The US Department of Energy (DOE) is sponsoring a program in technology development aimed at resolving the technical issues in severe accident management strategies for advanced and evolutionary light water reactors (LWRs). The key objective of this effort is to achieve a robust defense-in-depth at the interface between prevention and mitigation of severe accidents. The approach taken towards this goal is based on the Risk Oriented Accident Analysis Methodology (ROAAM). Applications of ROAAM to the severe accident management strategy for the US AP600 advanced LWR have been effective both in enhancing the design and in achieving acceptance of the conclusions and base technology developed in the course of the work. This paper presents an overview of that effort and its key technical elements.

1. INTRODUCTION

A comprehensive severe accident management strategy which integrates safety goals and methodology of assessment into a unified tool for ensuring defense-in-depth and resolving safety issues has been developed. Known as the Risk-Oriented Accident Analysis Methodology [1], it provides rational approaches to focusing research and development efforts to obtain answers needed for commercial nuclear plant licensing.

The basis for development of ROAAM is found in philosophical and practical difficulties in quantifying likelihood in presence of large uncertainties in knowledge (epistemic uncertainty) [1]. Specifically, such difficulties have been encountered in addressing certain containment challenge mechanisms that generically became known as "severe accident issues," and ROAAM was specifically developed as a tool to facilitate their resolution. In this **issue resolution context**, ROAAM provides a synergistic collaboration among experts (nationally and internationally) on a particular issue; and facilitates that collaboration by a technical and procedural framework whose key elements are problem decomposition and explicit identification and treatment of any part that cannot be approached in a demonstrably quantifiable fashion ("intangibles"). Such an approach has been shown to be uniquely suited to achieving resolution.

However, resolution of individual issues is not particularly useful unless the results can help establish unambiguously whether an adequate level of safety has been achieved. The acceptance criteria must be derived from philosophically sound safety goals, and the path to closure must be **clear, consistent, and complete**. Accordingly, the methodology was also developed towards satisfying the needs in this direction. It is known as the **Integrated ROAAM** [1, 2, 8].

Key components of the Integrated ROAAM are: explicit *a priori* integration of probabilistic and deterministic elements, consistency among them, and utilization of this duality to achieve and demonstrate defense-in-depth. This leads to the safety goal that containment failure is “physically unreasonable” for all accidents that are not “remote and speculative”. The term “remote and speculative” refers to frequencies based on reliability considerations, and “physically unreasonable” refers to verified applications of the basic laws of physics as implemented by ROAAM in its issue resolution context. Adoption of this evaluation technique leads to a way to evaluate rare, high-consequence hazards, which are difficult to quantify in a probabilistic manner and hence even more difficult to evaluate in a regulatory context based solely on a probabilistic risk assessment.

In its implementation, the integrated ROAAM begins with a complete systems analysis along the lines of a Level 1 Probabilistic Risk Assessment. This is used to define major accident classes and associated plant damage states, and to compute respective frequencies. A quantitative definition of a remote and speculative level is then made, and the resultant “screening frequency” is used to identify those accident classes which must be considered. For these classes, containment failure must be shown to be “physically unreasonable.” This is defined as the severe accident management, or mitigation, window. The strategy can then be optimized by deriving the effect of system changes on the accident content within the window, and of containment hardware as they affect the physics of mitigation [2].

Considerable experience with ROAAM has been accumulated. Applications in its issue resolution context include the α -mode (steam-explosion-induced) containment failure [3], the Mark-I liner attack problem which is relevant to core melt accidents in boiling water reactors with a Mark-I containment configuration [4], and the direct containment heating (DCH) issue [5]. Resolutions have been obtained in all three areas [3, 4, 5, 24]. Further, in its integrated context, ROAAM has been applied to severe accident assessment and management for the Loviisa plant in Finland [8] and more recently to the AP600 design [2]. Current work includes evaluations of lower head integrity under thermal loads (referred to as “in-vessel retention”) and under dynamic loads (referred to as “in-vessel explosions”).

1.1 The use of ROAAM in the DOE approach to Severe Accident Management (SAM)

Using ROAAM in its integrated context, the DOE approach that has been adopted for the SAM strategy for advanced LWR designs is to focus on areas where a reliability-type approach is appropriate, and others where the phenomenology of the event itself is the central issue. This is done in terms of the prevention-mitigation interplay evident in the safety goal stated above. That is, on the one hand aiming to eliminate inherently uncertain scenarios, so as to allow the “physically unreasonable” to be clearly demonstrable, while on the other, subjecting the equipment procedures necessary for such elimination to the criteria that failure is “remote and speculative” (the “screening frequency level”) subject to technological constraints of the specific reliability achievable. Using the Westinghouse AP600 design [6] as an example, this approach leads principally to three main reliability components, and two mainly phenomenological ones. They are:

Reliability

- Depressurize the reactor pressure vessel (prevent high pressure core melt ejection)
- Cool the containment - external spray cooling
- Control hydrogen gas buildup by passive autocatalytic recombiners

Phenomenology

- External coolability of reactor vessel containing relocated core melt
- Maintaining lower head integrity under steam explosions loading

The DOE program is focusing research efforts on opportunities for major advances in these two phenomenological areas. The objective is to provide rational approaches to addressing them, within accident scenarios relevant to particular reactor designs, and the data bases and computer codes needed to fully support these approaches. Emphasis is placed on focusing research to be efficient and effective, and on procedural approaches to involve the international expert community toward convergence and resolutions of the issues. For this, ROAAM is used in the issue resolution context for the two main research areas of **in-vessel core melt retention** and **in-vessel steam explosions**.

2. IN-VESSEL RETENTION (IVR)

The work on IVR involves two major experiments, ULPU and ACOPO, thermal and structural models of the debris and lower head, and an integration approach for assessing likelihood of failure. The basic document on this issue [7] was prepared and sent for a ROAAM review to 18 experts at the end of 1994. Resolution was reached after iterations with the authors, and the result is being used to support the AP600 treatment of debris coolability in the vendor's design certification application to the US Nuclear Regulatory Commission (NRC). The ULPU and ACOPO experiments described below played a key part in the resolution of this issue.

2.1 ULPU-2000 Experiment

The ULPU experiment [7] is a full-scale simulation of a nuclear reactor pressure vessel lower head, heated internally and submerged in a pool of water. The simulation is made in a vertical "slice" geometry, which allows a direct visualization of the heat transfer phenomena (Figs 1a, 1b). The experiment provides information on the coolability limits (critical heat flux, or CHF) as a function of distance along the arc length from the bottom to the upper edge positions on the hemisphere.

2.1.1. ULPU-2000 Test Facility Design

This experiment evolved from the original ULPU work which modeled phenomena characteristic of the Loviisa reactor. In that experiment the heater was limited by design and power to 1400 kW/m², and CHF could not be reached for the range of conditions investigated. In the present experiment, both the power and heater designs were upgraded to allow a peak heat flux of 2000 kW/m², and the test section represents a downward-facing hemisphere.

Electric heater assemblies are mounted on the inner radius of the test section to simulate the internal decay heating from a molten pool of corium. The assemblies are made of individual copper block segments with embedded resistance heaters. Power shaping of individual heaters is used to simulate the axisymmetric geometry of the reactor lower head, and the instrumentation included surface microthermocouples. The walls of the vessel which confine the water are equipped with viewing ports through which the boiling and the vapor generated can be observed and recorded.

To understand the effects of local subcooling and vapor condensation effects which influence the CHF, experiments are conducted in two different modes. One mode of operation is to obtain lower bounds for the effects by running experiments at saturated, pool boiling conditions. The other is to allow for a natural convection flow loop in which the water in the downcomer (see Fig 1b), while saturated at the top of the facility, attains a subcooling equivalent to the gravitational head at the bottom of the test section.

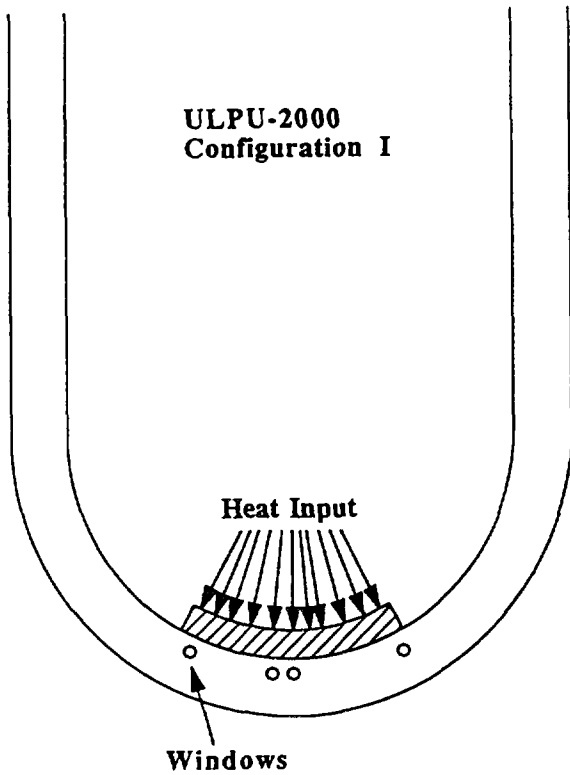


Figure 1a Schematic of Configuration I in ULPU-2000. The heater blocks extend over the region $-30^\circ < \theta < 30^\circ$.

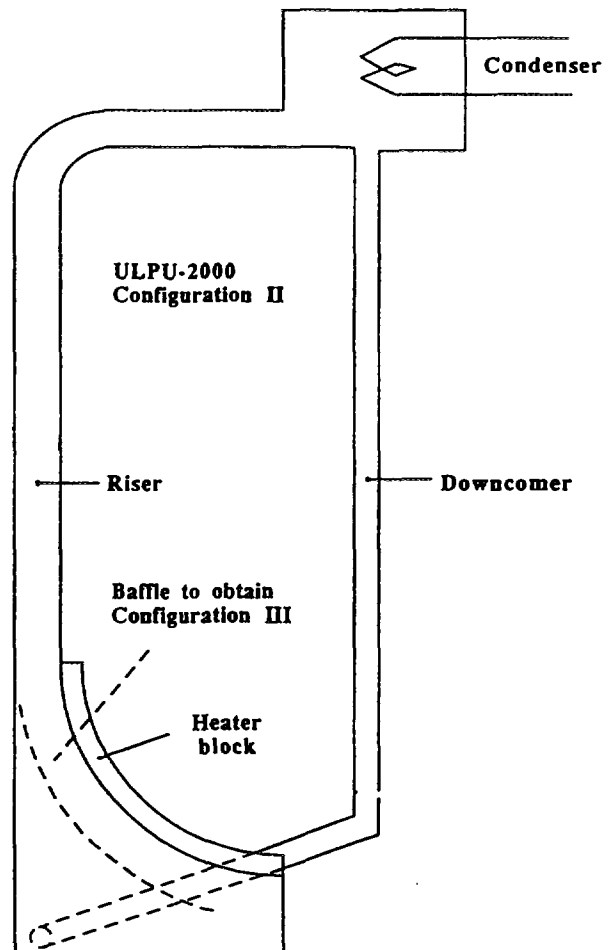


Figure 1b Schematic of Configurations II and III in ULPU-2000. The heater blocks extend over the region $0^\circ < \theta < 90^\circ$.

In relation to previous work on CHF, the present problem involves two unique aspects. One is that the vapor generated by boiling remains confined by gravity within a two-phase boundary layer all along an extended (large scale) heating surface. Within this boundary layer, flow velocities and phase distribution depend heavily on the local surface orientation to the gravity vector and on the cumulative quantities of steam generated in all upstream positions. The other unique aspect is that the thick heating surface (up to ~15 cm) has a very large thermal inertia. The ULPU-2000 was specifically conceived to represent these unique features.

2.1.2. ULPU-2000 Test Programs

The test programs were run in several configurations (see Figs 1a and 1b). Configuration 1 (C1) testing studied saturated pool boiling in $-30^\circ < \theta < 30^\circ$, and especially around $\theta = 0^\circ$. Configuration 2 (C2) simulated the complete geometry (a full quarter-circle) under both loop flow or pool boiling conditions. C2 represented an open-to-the-cavity geometry. Configuration 3 (C3) had a baffle to represent the thermal insulation that surrounds the reactor vessel, and the resultant channel-like geometry.

Most experimental runs were carried out in C2 to determine CHF as a function of power shaping, water subcooling, and recirculation flow rates. Additional tests examined the effect of surface wettability changes due to "aging."

2.1.3 Conclusions Drawn from the Experiments

An important first conclusion to be drawn from the experiments is that reliable full-scale simulations of the CHF distribution on the lower head of a reactor vessel submerged in water have been efficiently obtained with the ULPU-2000 facility. Secondly, the tests that were run indicate that the margins between predicted thermal load distributions and CHF are very large.

More recent experiments utilizing microthermocouples and high-speed videotaping have identified the presence of a new boiling transition regime with significant coupling between the overall system dynamics and the microphenomena [9]. These experiments lead to the conclusion that the coolability of the curved, inverted surface is controlled by microlayer evaporation, and by the time available between successive liquid contacts as dictated by system pulsations.

2.2 Axisymmetric Corium Pool (ACOPO) Experiment

The ACOPO experiment is a half-scale simulation of a nuclear reactor pressure vessel lower head in hemispherical geometry [10], as in the lower head of the US AP600 design. The test is designed to provide information on natural convection heat transfer (the thermal loading), from a simulated pool of molten corium in the lower head, over the range of prototypic Rayleigh numbers from $\sim 10^{15}$ to 10^{16} . The ACOPO test is the successor to the COPO experiment [12], which provided data in the Rayleigh number range of interest but in a two-dimensional slice geometry, and to the mini-ACOPO, a one-eighth scale proof-of-concept for ACOPO. Previous notable work in this area was the pioneering research of Mayinger at the University of Hanover [14,23] and Dhir at the University of California-Los Angeles [11].

2.2.1. ACOPO Test Facility Design

The ACOPO facility is designed to help overcome difficulties in reaching the range of Rayleigh numbers of interest in the axisymmetric geometry. These difficulties arise from a strong dependence of the Rayleigh number on the characteristic length scale, and from the need to provide

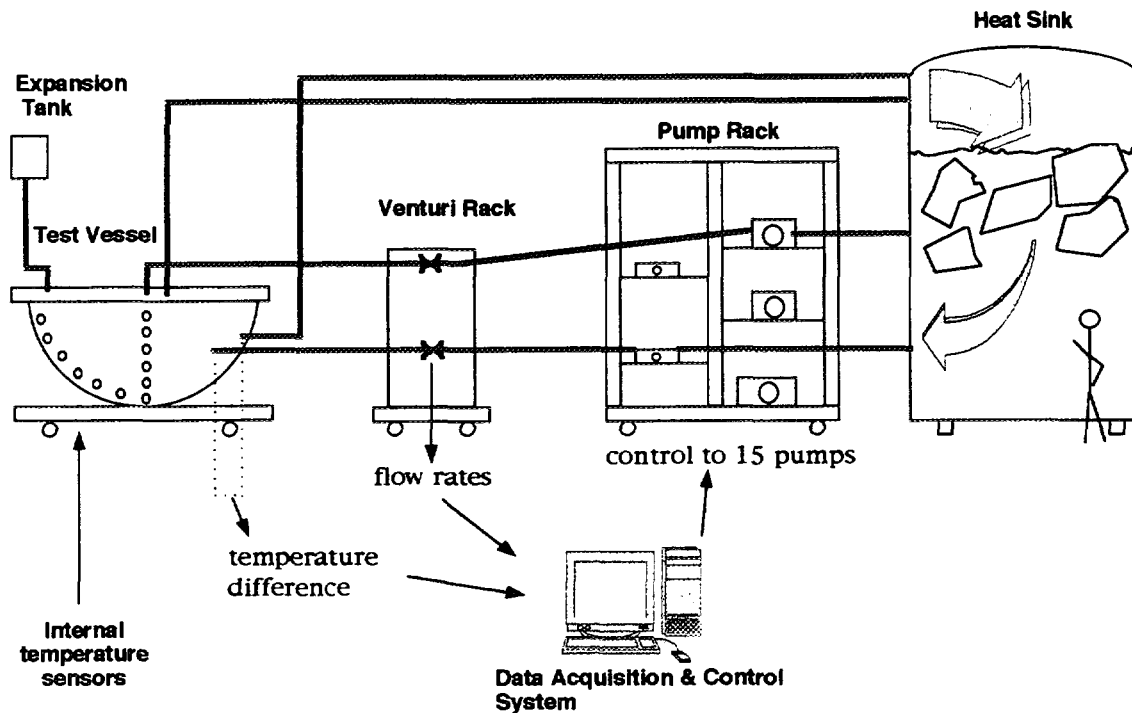


Figure 2a The ACOPO Half-Scale Facility

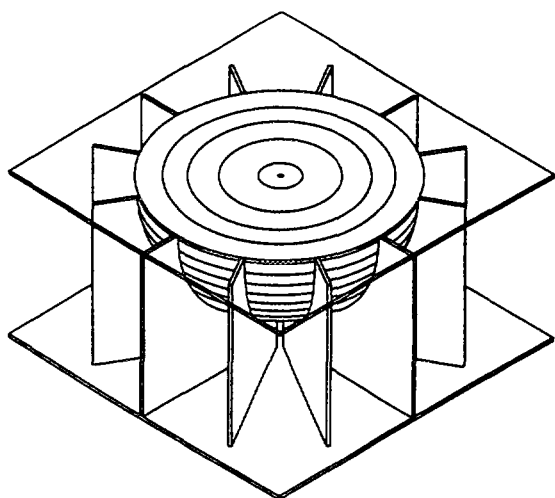


Figure 2b Schematic of ACOPO Test Vessel

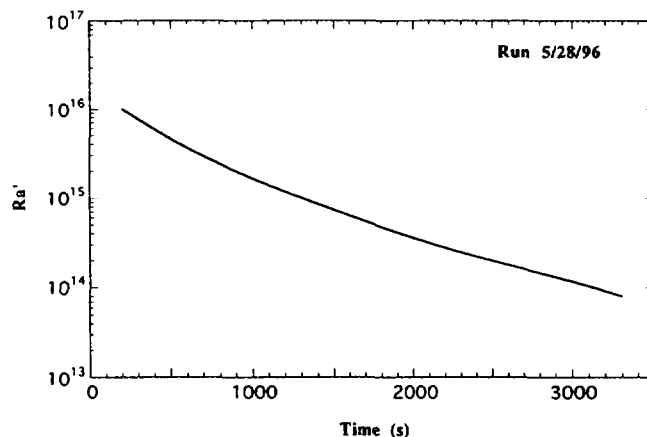


Figure 2c Rayleigh Number Transient in ACOPO Run

a uniform volumetric heating on a large scale and in a hemispherical geometry. The ACOPO design resolves these problems by the large scale of the test section and by using the internal energy of a preheated fluid to simulate volumetric heating. A rapid cooling of the boundaries of the hemisphere containing the heated fluid then provides a transient cooldown, which is interpreted as a sequence of quasi-stationary natural convection states. The earlier experiments with mini-ACOPO demonstrated and confirmed this approach.

The test section is a large (2 m) diameter hemisphere fabricated of square copper tubing cooling coils which are individually served by cooling units (Fig 2a). There are ten cooling zones on the hemispherical structure and five on the lid of the vessel (Fig 2b). Chilled water is used as the

circulating fluid in the cooling coils; it is circulated through an ice-filled external steel tank to maintain cooling water temperature near 0° C. Individual cooling coils are controlled for flow rate and monitored for flow and temperature by venturis and thermistors respectively.

The copper hemisphere is filled with water and the contents are heated to ~ 95°C by recirculating through an external heater. Then, a topping procedure is carried out with heated and degassed water to ensure that no air is trapped under the vessel lid.

The apparatus is insulated on the outside, and special care was taken to avoid external heat flow paths to the cooling units. As a check, the energy balance between the calculated energy loss from the vessel contents and that transferred to the cooling units was found to be well within 10%.

2.2.2. ACOPO Test Program

Experiments are run in the ACOPO by heating the vessel contents slowly with the external heater and final steam injection. The cooling circuits are then switched on to start the cooldown. Typically the experiment is continued for ~ 1 h.

As of the date of this presentation, five experiments have been conducted with highly reproducible results. The range of Rayleigh numbers from a typical run (dated 5/28/96) is shown in Fig 2c.

Analysis of the experiments provides upward and downward heat transfer data which are then compared with correlations based on the Nusselt number (Nu). The results of the ACOPO test runs are well correlated by

$$\text{upward heat transfer:} \quad Nu_{\text{upwd}} = 1.95 Ra^{0.18} \quad (1)$$

$$\text{downward heat transfer:} \quad Nu_{\text{down}} = 0.3 Ra^{0.22} \quad (2)$$

In the range $10^{15} < Ra < 10^{16}$, Eq. (1) is slightly lower than the well-known Steinberner-Reinecke correlation [14], and Eq. (2) is slightly above the Mayinger and mini-ACOPO [13] correlations. The heat flux shape was found to be in excellent agreement with that obtained in the mini-ACOPO.

The current work is oriented to more fundamental aspects, such as internal flow structure and its relation to the local heat transfer behavior.

2.2.3. Conclusions drawn from the experiments

The in-vessel retention analysis for the AP600 design was based on the Steinberner-Reinecke correlation and the mini-ACOPO correlation for upward and downward heat transfer, respectively. The difference between using these early correlations and the above experimental correlations shows that previously, the upward flux was underestimated by less than ~10%, while the downward flux was overestimated by less than ~6%. These variations are negligible in the context of the analysis and the margins to failure reported. *These experiments provide important confirmatory support to the validity of the in-vessel retention severe accident management strategy for AP600-type designs.*

3. IN-VESSEL STEAM EXPLOSIONS (IVE)

This work involves two major experiments, MAGICO and SIGMA, and two computer codes, PM-ALPHA and ESPROSE.m, to address the premixing and propagation, respectively, of steam

explosions. Application of these tools were integrated (under ROAAM) with the melt relocation physics and the structural response of the lower head under impulsive loads to assess the likelihood of failure. This is the second component (IVR being the first) of a comprehensive SAM scheme based on lower head integrity. The basic document on this issue [15] was prepared and sent for a ROAAM review to 18 international experts in mid 1996. Supporting documentation on verification of the two codes was recently completed. The MAGICO and SIGMA experiments played a key role in the verification task, and the SIGMA experiment demonstrated the microinteractions concept, which is the key idea for ESPROSE.m. These experiments are described briefly below.

3.1 MAGICO-2000/PM-ALPHA

The MAGICO-2000 experiment [16], and the multifield code, PM-ALPHA [17], are the primary tools for studying the premixing phase of steam explosions (the multiphase transient obtained during contact of a high-temperature melt with a liquid coolant). The experiment involves well-characterized, high-temperature particle clouds mixing with water, and detailed measurements on both the external and internal characteristics of the mixing zone. The PM-ALPHA code, which is intended to simulate the thermalhydraulic transient, is used to aid in interpreting the experimental results with good predictive capabilities. The code results which are of most interest are the mixing zone compositions and associated length scales. These compositions are expressed as volume fraction distribution maps, evolving in time, which can be then used in a propagation code to compute a steam explosion.

3.1.1. *MAGICO-2000 Experimental Apparatus and Program.*

The experimental objective is to generate a uniform cloud of particles at temperatures of $\sim 2000^{\circ}\text{C}$ which are released simultaneously into a water pool instrumented with thermocouples, videotape capability, and X-radiography. Particle clouds of ZrO_2 , Al_2O_3 , and steel were utilized in the experiments. A machined and drilled graphite heating block, electrically heated, contains and heats the particles before the drop into the water pool below. The overall arrangement of the heating block, and of the water pool, are shown in Figs 3a and 3b. Electrical resistance heating of the graphite block raises the particle temperature to the 1500 to 2000°C range in 7 to 10 h.

Experiments were conducted in two series: one, which addressed momentum interactions, was at room temperature ("cold" pours), while the second series was carried out with high temperature particles ("hot" pours) for phase change effects. Additionally, single-particle runs were done to test the techniques. Interactions during the pours were recorded on videotape; the particle cloud position and the liquid region interior boundaries were determined by flash radiographs. Measurements of the liquid swell, the particle cloud density, local void fraction, the water "hole" produced by the particle cloud in cold pours, and the height of the spray dome were all analyzed and compared with the predictions of the PM-ALPHA code.

3.1.2. *Conclusions Drawn from the Experiments*

The MAGICO-2000 facility provides a unique capability to produce uniform particle clouds at high temperatures for study of detailed interaction with water pools. Interesting phenomena identified with the cold pours have included the formation of densely packed regions at the penetration front, formation of a cavity behind the dense particle clouds, and development of fingerlike instabilities at the penetration front. Hot pour phenomena which have been quantified include local voiding in the mixing zone, global voiding through the level swell, and the effects of pool water subcooling on the above. The use of the PM-ALPHA code has helped to interpret these

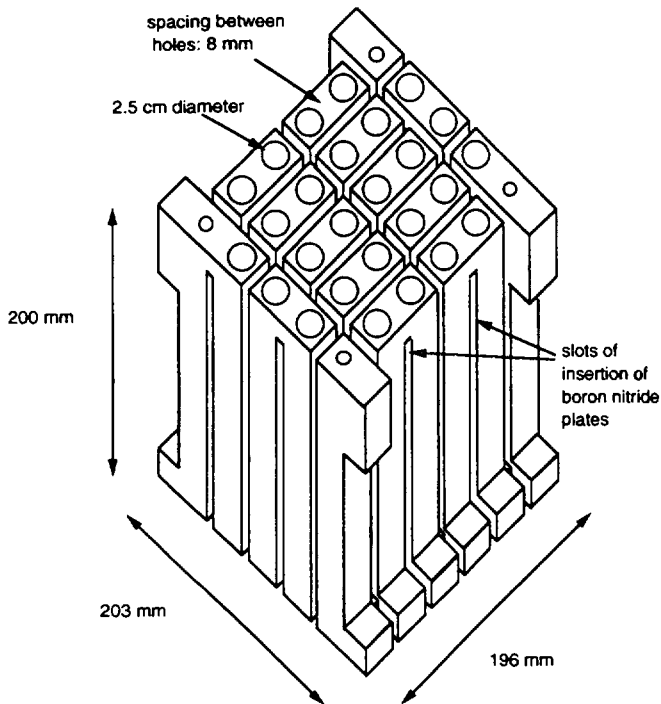


Figure 3a Heating element in MAGICO-2000

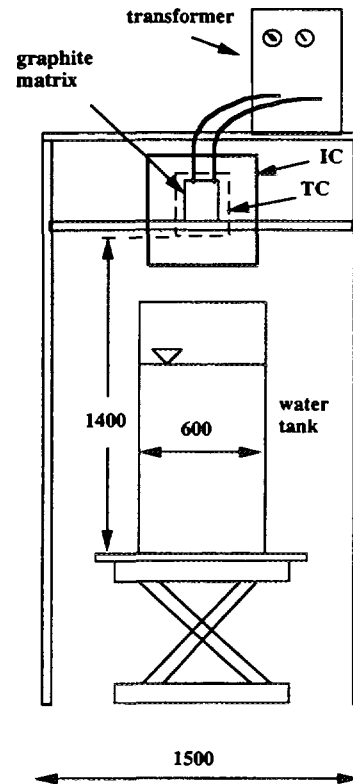


Figure 3b Schematic of MAGICO-2000. All dimensions in mm.

phenomena, and to establish a good predictive capability of the multifield aspects of the premixing phenomena. For the treatment of melt breakup and integral aspects of premixing with PM-ALPHA, see Ref. 25.

3.2 SIGMA/ESPROSE.m

The SIGMA facility [18] continues the investigation of steam explosion phenomena by experimental quantifications of the recently-introduced concept of microinteractions. In this approach, the fragmentation kinetics of hot liquid drops (e.g., molten corium) in another liquid (e.g., reactor coolant) under the influence of sustained pressure pulses are observed in the SIGMA hydrodynamic shock tube experiments. The underlying physics of the concept postulate that only a small quantity of coolant around each premixed melt mass sees the fragmenting debris coming from it (as opposed to a concept of homogeneous mixing of the fragmented debris with the coolant). The study of this in a controlled and quantifiable way is done by subjecting single drops of melt to a simulated explosion environment in the SIGMA facility.

The analytical formulation of this process is implemented in the ESPROSE.m code [19]. This code is designed to simulate the escalation and propagation of steam explosions, i.e., the wave dynamics through a given premixture and the surrounding medium, following an applied triggering pressure pulse. The outputs of the code are the pressure loads on adjacent structural boundaries, and kinetic energies of any mobile masses subjected to the explosion. The pressure loads can then be used in a structural analysis code to assess potential for failure; the kinetic energies are utilized for addressing mechanical damage from mobile object collisions.

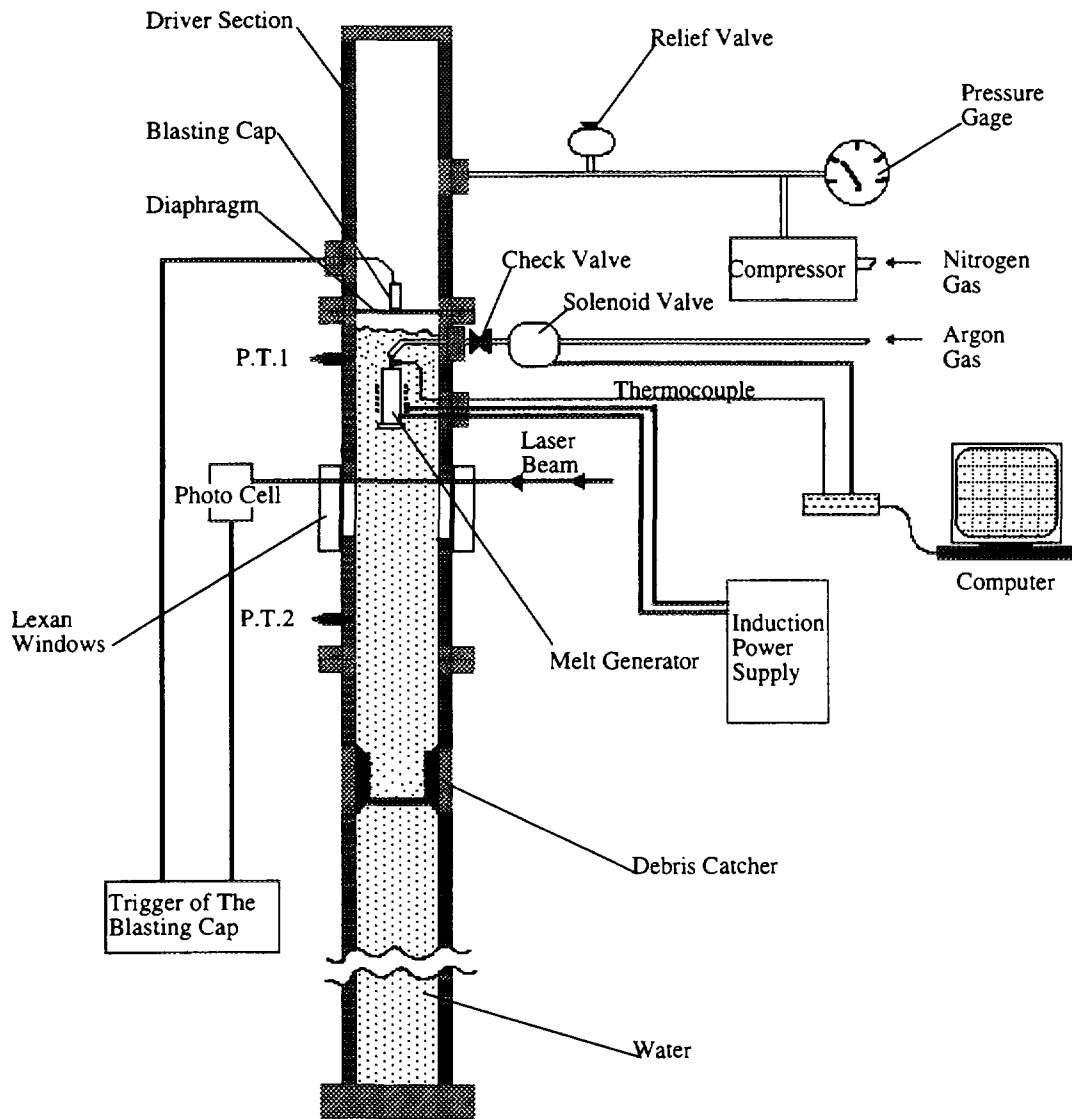


Figure 4 Schematic of the SIGMA-2000 facility

3.2.1. SIGMA-2000 Experimental Apparatus and Program

The basic component of the SIGMA facility is a shock tube with a 1 m long driver section and a 2 m long expansion section (see Fig 4). The design pressure is 1000 bar; with water in the expansion section, a pressure pulse of up to ~ 2 ms can be achieved before reflection of the shock wave from the tube end arrives back to the interaction location. The shock wave is initiated by rupturing a pre-scored diaphragm pressure boundary with an explosive cap.

A melt generator which is designed to melt and reproducibly release single drops of test material at temperatures up to 1800°C is placed above the viewing window. An induction coil heats a graphite crucible containing the test material. At the desired temperature, a single molten drop is released into the water-filled expansion section, and the explosive cap is triggered. The experiment is timed by means of a laser beam which detects the molten drop and initiates the firing sequence. High-speed movie and x-ray images provide information on the evolution of the microinteraction region, while transient pressures in the shock tube are measured with quartz piezoelectric transducers.

To date, the experimental program has utilized materials ranging from gallium and tin to aluminum and steel oxides, with varying melt and water temperatures, at shock pressures ranging from 68 bar to ~ 280 bar. Future experiments are planned with zirconium and uranium oxides.

3.2.2. Conclusions Drawn from the Experiments

The SIGMA-2000 facility allows the observation of exploding molten drops under conditions resembling those in larger scale detonations. These experimental observations have provided input to the microinteraction models used to compute the propagation phase of steam explosions with ESPROSE.m. This derived capability to compute steam explosions has been verified by comparison to integral experiments, properly interpreting results ranging from weak propagations to strong detonations [26].

4.0 RELATED PROGRAMMATIC WORK

In addition to the experimental programs described above, the US DOE provides support to, or liaison with, other programs in severe accident research. These are the MACE program being managed by EPRI with ANL as the executing agency, and the RASPLAV program sponsored by the OECD-NEA with the Kurchatov Institute in Moscow as the executing agency. In addition, a contingency experiment in molten core retention (PACOPO) has been designed as a potential follow-on to the ACOPO test. All of these tests involve the use of prototypical corium mixtures as molten material.

4.1 Melt Attack and Coolability Experiment (MACE)

The MACE experiment [20], for which the US DOE is a major co-sponsor, is being conducted at ANL in cooperation with an EPRI-led international consortium which earlier was responsible for performance of the Advanced Containment Experiments (ACE). The MACE program addresses the use of water to terminate and stabilize a core melt accident at the ex-vessel stage, wherein molten corium is attacking basemat concrete. The M3b test is currently being prepared for execution in December 1996. In this experiment, a molten core/concrete interaction (MCCI) is created involving 2000 kg of molten corium (with a nominal composition of 57 UO₂/29 ZrO₂/6 Cr). The corium is generated at ~ 2500K initial temperature via an exothermic chemical reaction which takes place in ~ 30 s. The melt is thereafter internally heated at 300 W/kg UO₂ by use of direct electrical heating. The corium attacks and decomposes the underlying concrete basemat (which is composed of limestone and common sand for test M3b). At an ablation depth of 5 mm, water is flooded in to the test section. The M3b test is a repeat of a previously unsuccessful test, M3, which experienced moisture contamination in the corium reactants. Measures to strictly control and detect moisture contamination have been instituted for M3b.

4.2 The RASPLAV Program

The RASPLAV program being conducted at the Russian Research Center-Kurchatov Institute (RRC-KI) is sponsored by the OECD-NEA as a multinationally-supported project. [21]. The US NRC is the US participant in the program. At the request of the NRC program management, and with the approval of RRC-KI, DOE has supplied a contractor (University of California-Santa Barbara) to assist in the planning, evaluation and review of the program. The main experiment of this program involves 200 kg of prototypic melt contained by an externally cooled steel wall. The experiment is being conducted in "slice" geometry using molten corium at ~ 3000 K which is heated by induction through the parallel semicircular sides of the apparatus. The first such test has just been successfully completed. Other important aspects of this program are measurements of thermophysical properties of corium melts and natural convection studies using molten salts.

4.3 Prototypic Axisymmetric Corium Pool (PACOPO) Experiment

In late 1994 the idea was proposed for a DOE-sponsored prototypical material experiment to address heat transfer in an in-vessel molten corium pool by using the ACOPO experiment principle [22]. The experiment would utilize ~ 2000 kg of prototypic thermic materials with an initial superheat of ~ 250 K and would be conducted in the same facility at ANL as the MACE test. A peer review of the ANL proposed design was undertaken in late 1995 which provided valuable guidance. However, budgetary restrictions and the imminence of results from the RASPLAV experimental program led to deferral of further work at this time on this concept.

5.0 CONCLUDING REMARKS

The work described above has been planned and executed with the intention of achieving closure to the concerns that arise in severe accident issues. We believe that significant progress has been made towards this goal, as evidenced by the wide acceptance of in-vessel retention as a key severe accident management strategy. This adds to the previous ROAAM results on α -mode failure, Mark-I liner attack, and DCH, which were carried out under the sponsorship of the Nuclear Regulatory Commission, and helps to establish a new approach in addressing defense-in-depth at the containment integrity level. Special efforts have been made to involve the international safety community in this work and to synergize with it towards effective and widely accepted severe accident assessments and management approaches. We have confidence that the management goals can be efficiently achieved for advanced reactors with closure close at hand. Overall, this work is believed to be very beneficial for the future of nuclear power by removing uncertainties and thus increasing public acceptance. This work, focused so far on advanced reactors, could also benefit existing reactors, and we believe we can have an overall beneficial effect on the future of nuclear power.

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