# Experimental Thermal-Fluids Program In Support of Reactor Operations

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## Abstract

Production at the Savannah River Site (SRS) was authorized in 1950 to produce nuclear materials for weapons. The Cold War was in full swing, and the Soviets had developed thermo-nuclear weapons. Our goal was to safely produce as much material as possible as quickly as possible. The mission was clear.

A key link in that long chain of events required to maximize production was to increase reactor power. To achieve this, a good understanding of the controlling phenomena was needed as well as a strong technical basis. The experience at Hanford provided some of the information needed to design the reactors. But these reactors were totally new and large machines that needed on-going technical support. Accordingly, a research effort was started (initially at Columbia University before any facilities were available at SRS). The research and experimentation were soon moved to SRS as the plant was constructed. The heat transfer and fluid mechanics work was assigned to "Pile Engineering" in the Savannah River Laboratory. ("Pile" was the original name for "reactor". For many a young engineer on his first assignment, the name conjured all kinds of meanings.) The research effort, along with hardware changes in the reactors, allowed the power to be increased by more than a factor of six. This document describes some of the key technical advances stemming from the experimental effort along with some anecdotal accounts by some of the people involved.

## First Experiments

The first experimental facility at SRS was CMX (Corrosion Mockup eXperimental). It was built on the banks of the Savannah River to determine what water treatment was needed to prevent fouling the reactor heat exchanger tubes. The reactors were cooled with heavy water, which dumped its heat to the river water through large heat exchangers. Frequently, heat exchanger tubes will foul when cooled by river water. The reactors were initially designed with large water treatment plants to thoroughly treat all the river water used for cooling (approximately 90,000 gpm per reactor). This was a large and expensive treatment process, and it was desired to know the minimum treatment required to reduce cost. Accordingly, heat exchanger fouling experiments were initiated at CMX. Within a few months it was determined that no treatment was necessary. The abrasive characteristics of the river water scoured the tube surface and prevented significant fouling. Any fouling that did occur could be cleaned with oxalic acid. As a direct result to the CMX tests, one of the largest cost savings ever (for the era) was achieved.

A key element in achieving high reactor power (production) stemmed from the reactor design, which provided much of the instrumentation and to monitor the performance of every reactor fuel and target assembly. The design of the initial fuel assembly consisted of four slug columns in ribbed channels in one housing unit called a "quatrefoil" (Mark I assembly). The assemblies were installed in the reactor and rested, covering a monitor pin in the bottom of the reactor. The monitor pin contained four pressure taps (pressure averaged) to indicate assembly flow and four thermocouples to measure the effluent temperature from each slug column. When R Reactor was started (the first), it was found that the flow and temperature monitoring was erratic. Methods were proposed to improve the monitoring, but they had to be tested before making any changes to reactor hardware. Test results did show that hardware changes could correct the problem. Subsequently, all new reactor assembly designs (and there were many) were tested to ensure accurate monitoring of each reactor assembly. Accurate flow and temperature measurements were critical, especially as the power went up, because the signals were monitored continually to maintain safety margins and maximize production.

## Vibration Wear Problems

Cladding failure of the slugs was an early reactor problem. Because of clearances between the channel ribs and the slugs, which were required to load and unload the slugs in the channels, slugs vibrated against the ribs. If the vibration were severe enough, the slug cladding could be penetrated. The hot uranium and hot coolant could react to produce uranium oxide. Two immediate problems were associated with cladding failure. First, the slug would swell because of the increased volume of oxide, such that channel flow would decrease and possibly overheat other slugs in the channel. Second, fission products would be released into the bulk reactor coolant/moderator, causing excessive radiation in external piping and penetrating the first barrier to the environment.

The vibration problems were reduced, but never eliminated, by reducing clearances between ribs and slugs and by reducing inlet flow velocities at the tops of the slug columns. The CMX work consisted of long-term flow testing to characterize any wear and the effect of changes in assembly design.

## Moderator Circulation Problems

Reactor instrumentation indicated that "hot spots" or areas of poor circulation occurred in the bulk moderator inside the reactor tank. A full-size mockup of one-sixth of a reactor tank, called a crossflow tank, was constructed to study the circulation problems. One wall of the crossflow tank was constructed to allow visual observation inside the tank. Dye could be injected into the flowing water at the point of interest, and the flow path of the dye could be observed and its velocity determined by time of transport.

A large area of poor circulation was found in the tests. A special assembly called a jet-tube sparger (sparjet) was designed and installed in the crossflow tank to improve circulation in this area. The "jet-tube" portion of the assembly consisted of coolant flowing downward as in a fuel assembly but discharged upward through nozzles at various elevations along the length of the assembly. The upward, high-velocity flow promoted circulation in the "dead" spot. Six of the assemblies were placed in each reactor in the symmetric location determined by the single assembly in the crossflow tank. The sparger portion of the assembly was also used to provide the means to inject nuclear poison into the moderator for rapid reactor shutdown in the event of failure of the normal and emergency shutdown systems.

## Increased Heat Transfer Surface Area

Reactor power (production) was severely limited by the then allowable maximum surface temperature and surface area available for heat transfer within the fuel elements. Programs were implemented to increase each and provide a sound technical basis. One of the first steps was to make the slugs hollow instead of solid. This step doubled the number of coolant flow channels. Previously the flow in each channel had been calculated. But with the increased complexity and higher assembly power, it was necessary to measure the flow distribution. This would allow reduction in the uncertainty allowances used to calculate the power limits and to establish a better technical basis.

Methods were devised to install pressure taps near the top and bottom of each coolant channel. By restricting the flow to one channel at a time, the flow and differential pressure relationship could be established. Then when the full assembly was operated, the flow split among the channels was determined. This information was needed input for the power limits calculation.

Failure is a natural part of success. In another attempt to increase the surface area of fuel elements, a bundle of five thin flat plates with ribs was designed and built. Hydraulic tests at CMX were required before the element could go into the reactor. While attempting to load the plates into a special housing tube, the plates stuck and would not go in or out of the housing. The design was subsequently abandoned; it was commented that we were trying to put a square peg in a round hole.

The next attempt at fuel elements with extended surface areas was the concentric tubular assembly (nested tubes). This design was the result of a newly developed co-extrusion process. The design was so superior that it was used from then on. The technology for measuring the flow distribution and temperature monitoring was already developed, so the transition to the new design was comparatively easy. The rest is history.

## Two-Phase Flow

Early on, management wanted to know about the conditions necessary for the onset of flow instability. (Flow instability is an excursive process whereby excessive power of an assembly causes two-phase flow to develop and the flow rate to drop quickly to near zero. Once the flow decreases, overheating and melting can occur if the power is not immediately reduced.) That initial investigation was the start of an enlightening experience in two-phase flow of steam-water mixtures for many engineers. Some quick and dirty experiments were devised. A 14-foot-long tube was heated by two Lincoln welding generators hooked in parallel and connected for direct resistance heating. Flow through the tube mocked up the flow in one channel of the fuel assembly. CMX was chosen as the test site.

The tube was brought up to power, and then the flow was gradually throttled until the outlet temperature of the coolant reached the boiling point. The flow then decreased catastrophically as the pressure drop for two-phase flow was significantly greater than for single-phase flow. We had achieved our first flow instability. Power was maintained on the tube, and it became glowing red hot and warped and wiggled like a snake. If that tube had been a fuel assembly, it would have melted and released fission products to the moderator coolant and possibly to the environment. The major concern was reactor safety and melting of the fuel. Such tests were used to help determine reactor power limits.

## Heat Transfer Laboratory

The nameplate rating of the reactors was 378 MW. This was based on a Mark I fuel assembly with a maximum surface temperature of the aluminum cladding of $80^{0}$C. We carefully calculated power limits based on not exceeding an $80^{0}$C surface temperature, including hot spots on the surface of the aluminum cladding and a maximum central metal temperature of the uranium of less than $600^{0}$C. These limits were based on concerns about aluminum corrosion and uranium swelling from phase changes.

New corrosion studies showed that in the high-purity coolant, the temperature was very conservative. We set out to find out what the limit on the heat removal capability of the fuel was, since this was directly proportional to producing nuclear materials. A Heat Transfer Laboratory was built in Building 773-A to study assembly cooling phenomena and power limits. We determined that surface temperatures of the aluminum could be allowed to be consistent with nucleate boiling. However, we were defining the maximum heat fluxes from the assemblies without getting film boiling. Film boiling was referred to as the hot stove effect. If film boiling occurred, the surface would be blanketed with steam essentially resulting in adiabatic heating of the fuel and subsequent fuel melting. Several rigs were built to determine the maximum heat flux without incurring film boiling as a function of pressure, subcooling, and coolant velocity. Film boiling would essentially lead to melting the fuel surface and burnout. A burnout safety factor was born and referred to as the Burnout Safety Factor (BOSF), the ratio of burnout heat flux to operating heat flux. The data were correlated in an equation using the power of a slide rule, now practically extinct.

In one experiment while trying to get to the burnout heat flux, we kept reducing the subcooling and unknowingly got two-phase flow in the downstream piping. The glass tube that formed the annulus around the heated tube, which was glowing red-hot, ruptured, and, although it was encased in a "Plexiglass" housing, allowed steam to escape. We were almost trampled by one of our larger managers who was escaping from the "steam explosion". At the next management meeting, the Laboratory Director pointed out with some indignation that unusual occurrences sometimes happen in the presence of the foreman.

We developed a correlation for the effect of spacer ribs to be used in conjunction with our burnout correlation. In an attempt to improve the precision of the heat transfer correlation, subsequent researchers developed a modified correlation. Unfortunately, this correlation was used with the earlier correlation on the effect of ribs on the burnout heat flux, which eliminated conservatism in the analyses. In some subsequent very-high-flux charges, some slight melting occurred along some ribs. Fortunately, the fuel exposure was very low, and no significant radioactivity was released.

The heat transfer experiments had established higher operating limits for the fuel. Engineering developments and design had significantly increased the assembly surface area. Together they provided higher assembly power by more than a factor of six. Reactor modifications were then able to take advantage of the higher power limit.

Other activities as an extension of the Heat Transfer Laboratory were related to PAR pond studies. A small lake behind C Reactor was built to determine the optimum way to introduce hot water into PAR Pond, which was then in the planning stage. Management wanted to know the effect of the depth of draw-off on the outlet temperature on an urgent basis as design of the lake was proceeding. The engineer and technician went to Central Shops salvage yard and found some galvanized ventilation pipe and two swivel joints and headed out to the pond. They took off most of their clothes, went into the pond, installed a variable draw-off and were collecting data the next day. A diesel generator and pump were operated to support the experiment. They requested equipment operators to watch it that night. The next day the operators said alligators were crawling out of the lake and over the dam. It was jointly decided that the equipment didn't need surveillance at night.

## Airborne Activity Confinement System (AACS)

When the reactors were designed and built, they contained a once-through ventilation system with essentially no filtration of the exhaust air. This meant that in the event of a significant reactor accident with fuel melting, large amounts of radiation could escape the building and contaminate the surrounding area. Calculations showed that such a postulated accident could also release large quantities of steam. A filtration system was wanted for the reactor ventilation system to significantly reduce radioactive releases in the event of a reactor accident. Tests of standard filtration systems showed that particulate filters could in fact remove more than 99.9% of all contaminants except iodine, tritium, and noble gases.

However, they could not survive for long in the presence of wet steam. Further testing established that a demister immediately upstream of the particulate filter would knock out enough water particles to prevent failure of the particulate filters. A large part of the required filtration system was thus defined.

No industrial process was available at the time to capture tritium or noble gases. However, it was known that iodine could be captured on activated carbon. Testing commercially available carbon beds showed that none of them could meet the requirement for 99.9% removal efficiency. Testing was undertaken to determine the cause of the low efficiency and to correct the deficiency. It was determined that a one-inch-thick bed of coconut-shell-activated carbon could meet the requirement. It was also determined that the iodine would pass along the interface between the carbon and the metal container wall with somewhat less adsorption efficiency. Redesign of the container frame to include baffles at the interface solved the problem. We then had a workable design for the reactor exhaust gas confinement system (AACS). The system was then designed and installed in all reactors.

But just because the filters were designed to remove more that 99.9% of the particulate and iodine, that doesn't mean that errors in the installation process couldn't defeat the objective. In-place testing was needed to verify the system performance. A standard test called a DOP test was available for particulate filters, and this was adapted for use in the exhaust gas confinement system.

A program was initiated to develop an applicable in-place test for the carbon beds. A substitute tracer material was needed because using iodine as a tracer would deplete the capacity of the carbon for the iodine and defeat the purpose of the test. A tracer material was needed that would adsorb, then desorb from the carbon but pause long enough to measure any leakage flow. Chlorinated hydrocarbons were proposed because they were expected to meet the requirements. Testing was initiated at CMX to verify the strategy and hypothesis. Several chlorinated hydrocarbons would meet the requirements, and the best was selected and demonstrated in a reactor. A 0.05% leak path was installed in one of the reactor filter compartments, and the test was used to determine if the known leak could be detected. The results clearly showed the leak path and the ability to detect leaks of less than 0.01%. We then had an exhaust gas confinement system that could meet requirements, and two in-place tests that would verify the fact.

The activated carbon bed testing method was totally new and was found applicable to the nuclear industry. The test was standardized for use in nuclear power plants and is still in use today.

## Flow Oscillations

Very-high-neutron-flux charges were designed for the reactors in the latter part of the 1960s. This required fewer assemblies with much higher flow (>1000 gpm). The fewer assemblies were clustered near the center of the reactor and resulted in much higher velocities of coolant through the plenum. When the charge was operated, there were dramatic changes with time in the assembly flow rates. The flow varied up and down (oscillated) for no apparent reason and with a variable period. This required de-rating the design reactor power, and the reasons for the variation were investigated.

A full-scale mockup of the reactor plenum was built at CMX to investigate the cause of the flow oscillations. Cooling water was admitted to the assemblies through slots in the plenum sleeve. It was determined that the plenum was behaving like a large fluidic amplifier. That is, a small perturbation in one region of the plenum caused dramatic changes in fluid velocity in other parts of the plenum without changing the total reactor flow rate. The suspected cause of the small perturbation was vortex shedding, but this was never confirmed.

The assembly flow was sensitive to plenum local velocity because the inlet slots were located in the narrowest passage between plenum tube positions and, of course, this was the region of highest velocity. Hence, an increase in velocity perpendicular to an inlet slot caused reduced flow to the assembly. In fact, study of the data from previous operations revealed the same phenomenon had existed, only on a much smaller scale.

To reduce the effect of flow oscillations, several new potential designs for the inlet to the universal sleeve housing (USH) tube were tested. The USH passed through the plenum tube and held the fuel and target assemblies in the reactor. The old USH design contained slots that aligned with plenum sleeve slots. As a result of the testing, a new USH was designed that used 270 one-fourth-inch-diameter holes spaced uniformly around the sleeve. This had the effect of damping the variations in assembly flow with changes in fluid velocity in the plenum. That design was used permanently thereafter.

## Starved Pump Test

The postulated loss of coolant accident (LOCA) was studied in the late 1960s. The LOCA is a very low probably accident characterized by an instantaneous pipe break with unimpeded discharge from both pipe ends. This accident would initiate emergency cooling (ECS) water injection into the reactor at considerably reduced flow to each assembly. However, the reactor would be shut down by then, so that only decay heat would need be removed.

Analysis showed that higher reactor powers could be achieved if credit could be taken for pump re-circulation. That is, given a LOCA and an empty reactor tank, the ECS may not be the only source of water. It was estimated that the pumps would return must of the ECS water to the plenum if they would in fact operate with what was termed a starved suction (an air-water mixture flowing into the pumps from the reactor).

Preliminary tests were performed at CMX to characterize pump behavior under such conditions. Then, a reactor test was conducted in P Reactor to verify the expected performance. A great deal of instrumentation was installed to measure the system flows, pressures, and vibration at critical locations.

The primary test was started with five pumps operating at full power and one pump shut down to represent the pumping system with the broken line. The moderator level in the reactor tank was gradually lowered until the pumps began to aspirate air at which time the pumping systems began to make some noise. Typically, you could hear a quiet hum in the control room when the reactors were operating. As the level was lowered further, more and more air was aspirated into the pumps, and the noise became louder and louder. As the level reached the point of maximum air entrainment (just before loss of suction), it sounded like large rocks were being pumped in the system. The experienced operators were walking around the control room as on egg shells. Very little was said. The piping system was vibrating beyond anything anyone had seen, but it did not reach what was previously determined as critical. Once the pump suction was lost, the extreme noise and vibration vanished. The pumps then just re-circulated the water that ran into them by gravity flow and operated quietly. That was the result we had expected and wanted. It proved we could take credit for pump re-circulation in the event of a LOCA.

## The New Heat Transfer Laboratory

The early 1970s brought more attention to severe postulated accidents. A new Heat Transfer Laboratory was built, and one of the first phenomena to be studied in more detail was flow instability. The new Heat Transfer Laboratory had 3 MW of installed and rectified power that could be used for electrical resistance heating of mockup fuel tubes, far more than was previously available with welding generators. This meant that full-scale mockups of one or two channels of a fuel assembly could be achieved with prototypic heat flux.

One of the first accidents to be studied was the pump shaft break accident (very low probability) in which there would have been a sudden reduction in coolant flow, which could drive the assemblies into flow instability if they were operating at too high power. The onset of flow instability would mean that fuel melting would very quickly follow if the power were not reduced immediately. This is a very fast accident that could quickly get out of control so that the reactor charge was operated such that flow instability would not happen even if a pump shaft break did occur. Experiments were initiated to better determine the power levels at which flow instability would be initiated given the accident.

The first experiment to study flow instability used a single-heater tube with an inter housing rod of fiberglass to form a single heated annulus. Electrical power at 120 volts and up to 30,000 amps was passed through the outer stainless steel tube to create the heat. The first test did not use full power but was sufficient to generate steam. The people involved had lots of experience with experimentation but no previous experience with flow instability. Precautions were taken to protect personnel against the unexpected.

The test was started, and the assembly brought up to power. When steady-state test conditions were established, a quick-opening valve was activated that caused a sudden reduction in coolant flow. Flow instability was immediately triggered. The engineers understood the flow instability process, but had little appreciation for its speed or violence. After the fact, it was recognized that the flow instability created extremely high pressures in the bottom of the assembly, which provided very large lifting forces on the inter-housing rod. The result was the rod broke its restraints and came out of the test assembly like a rocket. It rose to the ceiling, hit a wide-flange I-beam, was deflected, and created just a small dent in the roof.

No one was hurt because the personnel were all in a shielded control room. However, the incident brought a new respect for high-powered experiments and pointed out the need to calculate expected results and forces to predict the consequences of a test. It was something that was done for the reactors, but now must be done for the experiments. It was called a learning experience.

## Before and After the Three Mile Island Accident

The accident at Three Mile Island (TMI) marked a turning point in much of the history of nuclear reactors. In the late 1970s at SRS and before TMI, there was a drift away from experimental programs (because of their expense) and toward computational solutions. Experiments and calculations have always gone hand in hand. One cannot flourish without the other. But in a mature industry, it was believed that the experimental basis had been adequately established, and calculations could handle most new situations.

TMI changed that thinking. As the analysis of TMI evolved, more attention was given to low-probability accidents. Experiments were started in the mid 1980s to study the Loss of Coolant Accident (LOCA). The initial LOCA experiments studied the heat-removal ability at various flow rates down one channel of typical dimensions. The flow splits among the various channels of an assembly had previously been measured along with a conservative estimate of ECS cooling water flow rates. When this information was put together in a mathematical model that accounted for heat splits, the results showed there would be some melting of reactor assemblies in the event of an ultimate LOCA. This of course was not acceptable, and the reactor power was reduced about 25%.

The National Academy of Science was asked to review the results, and they wanted a more conservative power level, so the reactor power level was again reduced to about 50% of initial power. Furthermore, at about the same time, there developed a procedural issue for the startup of P Reactor after an extended shutdown. These conditions, along with an abundant supply of weapons material and the end of the cold war in sight, resulted in the shutdown of all the reactors to further study low probability accidents.

Multiple committees were established to review the limits system, and critics came from everywhere. Reviewers experienced with power reactors maintained that critical heat flux (film boiling and burnout on surfaces while maintaining flow) would be our limitation at high power and not excursive flow instability as our experience indicated. Hence, major programs were established to study film boiling and flow instability as well as ECS cooling conditions. The experimental programs were far more than could be accomplished in the Heat Transfer Laboratory, so contracts were let to B&W (Alliance, Ohio), Creare (Lebanon, New Hampshire), and Columbia University (New York City) for additional experimental work. Combustion Engineering was also contracted to produce a prototypic assembly for testing in the Heat Transfer Laboratory.

After about six years, the end results of the experimental and computational programs were verification of the flow instability criteria and establishment of new, more-conservative limits for LOCA where ECS cooling would be required. In the mean time, however, the Cold War came to an end, and DOE decided to shut down the reactors permanently.

Thus, a long history of experimentation in support of reactor operations came to an abrupt end. The program had provided direct support to both reactor operations and the technical basis for the computational program and the limits system. In the scramble to find other work or be shut down too, Heat Transfer Laboratory personnel scoured the plantsite looking for a new role. The laboratory name was changed to Thermal-Fluids Laboratory (TFL) to provide a more descriptive name for potential customers. Both large and small jobs were found for the transition, and now the laboratory is once again thriving with a broad base of customers. It is a tribute to the long history of excellence in the experimental program at SRS.

The success of the experimental thermal-fluid program was a direct result of the many excellent people who supported it. Credit belongs not only to the engineers, but also to the technicians, secretaries, and manager as well as other support services. The authors also acknowledge assistance in the creation of this paper from George Richardson, Jim Smith, John Steimke, and Vince Walker.

# Biographies

## David Muhlbaier

David Muhlbaier has a BSME degree from Drexel University and arrived at SRS in 1961. He worked for many years at CMX and was responsible for various experiments, including the development of the carbon bed leak test. He then became involved in two-phase flow experiments after the new Heat Transfer Laboratory was built. In 1980, he was assigned to the L-Reactor restart project and subsequently to Reactor Technology. He concluded his career as manager of the new Heat Transfer Laboratory beginning shortly after the initial shutdown of the reactors and during the time of extensive internal and external experiments designed to enable reactor restart.

## Sam Mirshak

Sam Mirshak has an MSChE degree from Northwestern University and came to work at the Savannah River Site in 1953. He held progressive positions in research and development and management leading to Works Technical General Superintendent and Production General Superintendent for SRS. He was appointed Director of Research of the Savannah River Laboratory. In 1984, he was assigned Associate Technical Director for the Atomic Energy Division in Wilmington, Delaware. From 1987 to his retirement from Du Pont in 1989, he also served as Director of Nuclear Safety and Quality. Subsequently, he was a consultant for the Department of Energy and their national laboratories. Mr. Mirshak was the principal contributor to the heat transfer correlation that allowed a large increase in reactor power with demonstrated safety margin.

## Vascoe Whatley

Vascoe Whatley has a BSChE degree from Clemson University and came to work at SRS Operations in 1956 after several short stints in SRS Construction. He worked much of his career at CMX in the experimental arena. He was the principal contributor to the hydraulic studies of new or proposed fuel and target assemblies and to the long-term vibration and wear issues with production assemblies. His experience included in-reactor testing. He also worked in isotope applications and then returned to CMX and subsequently to the new Heat Transfer Laboratory.

## Elwyn Wingo

Elwyn Wingo has an MSME degree from Georgia Tech. He came to work at SRS in 1955 and worked as a test engineer at CMX. He was responsible for much of the experimental work that lead to the solution for inadequate moderator circulation and assembly flow oscillations, as well as various other experiments. He also managed work in the computational effort, principally in the arena of heat transfer and fluid mechanics. Other assignments included Reactor Technology, the Heat Transfer Laboratory, Long Range Planning, and Probabilistic Safety Analysis.

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