APPENDIX D

PIRT PLANT AND SCENARIO DESCRIPTIONS

The PIRT library used to formulate the consolidated list of highly ranked PIRT processes/phenomena was discussed in Section 4.2. In this appendix, brief descriptions of each plant and accident scenario used in preparing the consolidated PIRT are provided.

D.1. Westinghouse AP600 LBLOCA (PWR)

D.1.1. Plant Description

As described in Ref. D-1, the AP600 is a two-loop design. Each loop contains one hot leg, one steam generator, two reactor coolant pumps, and two cold legs. A pressurizer is attached to one of the hot legs. The reactor coolant pumps are a canned-motor design and are attached directly to the steam generator. The loop seal is eliminated; an added safety feature in that core uncovery caused by the existence of water-filled loop seals is eliminated during a postulated small-break LOCA. The core is designed for a low power density and consists of 145 fuel assemblies with an active fuel length of 12 ft. The fuel assembly is a 17 x 17 array of fuel and control rods.

The AP600 incorporates passive safety systems that rely only on redundant and fail-safe valves, gravity, natural circulation, and compressed gas. There are no pumps, diesels, or other active machinery in these safety systems. During plant shutdown, all the passive safety features will be tested to demonstrate system readiness, flow, and heat removal performance. These systems are shown in an isometric cutaway view of the AP600 reactor design in Fig. D-1. Two Passive Safety Injection System (PSIS) trains, each with an accumulator, a Core Makeup Tank (CMT), and an injection line from the In-containment Refueling Water Storage Tank (IRWST) and sump are connected directly to the reactor-vessel downcomer via a direct vessel injection line.

Depressurization of the primary system is an essential process that is required to ensure long-term cooling of the AP600. For example, the accumulators inject coolant into the reactor coolant system only after the primary pressure has dropped to 700 psia. Coolant injection from large, safety-class water pools, specifically the IRWST and sump, can occur only after the reactor coolant system pressure decreases below the gravitational head of each pool. An Automatic Depressurization System (ADS) permits a controlled pressure reduction of the reactor coolant system. The ADS valves open in stages, based upon either reductions in CMT levels to a specified setpoint or elapsed time from a designated event.

After the accumulators and CMTs are depleted and the primary system has depressurized and approached the containment pressure, water injection is provided from the IRWST. This tank empties after several days. Provisions are also made for recirculating coolant from a sump. IRWST and sump recirculation may occur at the same time for some transients.

The AP600 containment plays an essential role in the long-term cooling of the primary via the Passive Containment Cooling System. Steam entering the containment, either through a break in the primary or through operation of the ADS, condenses on the inside of the steel containment shell. The condensate drains downward and a large fraction is delivered via gutters to either the IRWST or the sump. Heat transfer on the outside of the containment steel shell is by evaporation of liquid sprayed near the top of the steel reactor containment dome by the Passive Containment Cooling System, and by convection to an air stream induced by buoyancy-driven flow (unforced).

D.1.2. Scenario Description

As described in Ref. D-1, the LBLOCA scenario is subdivided into four time periods that characterize events during the sequence. These time periods, termed blowdown, refill, reflood, and long-term cooling are defined by the core and lower-plenum liquid-mass-fraction behaviors; the first three periods are shown in Fig. D-2. The scenario description that follows is largely based upon a TRAC-PF1/MOD2 calculation of an 80% DEGB in a single cold-leg pipe between the primary coolant pump and the connecting point for the CMT pressure balance line to the cold leg.[D-2]

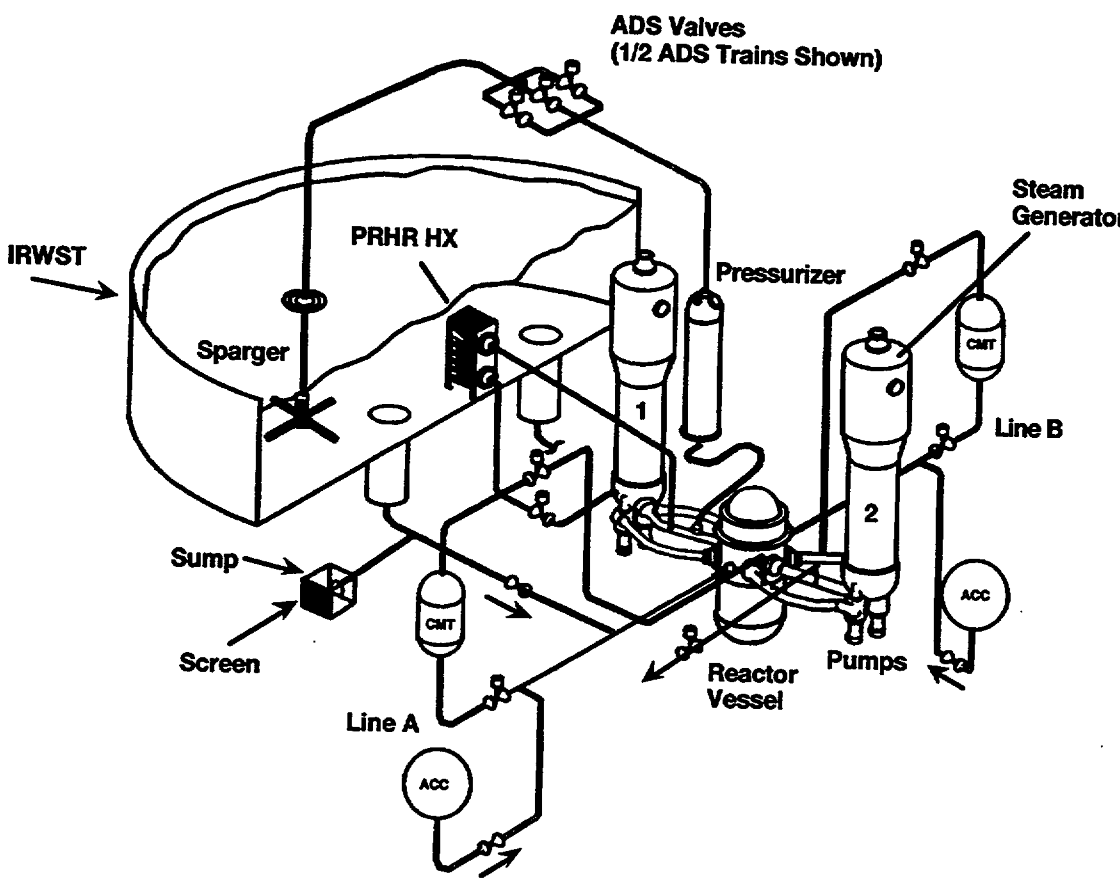

Fig. D-1. AP600 passive safety systems.

The *blowdown period* is the result of a break in the coolant system through which the primary coolant is expelled. Early blowdown physical phenomena include critical flow at the break, fluid flashing and depressurization, redistribution of fuel rod stored energy, and heating of the fuel rods due to degraded heat transfer. Later in the blowdown coolant reenters the core when the intact loop pump flows briefly exceed the break flows. Coolant also drains into the core during this period from the upper plenum. During blowdown, some components are affected more than others. In particular, the heat removal from the core results from the changing flow and heat transfer regimes in the core. The performance of the primary coolant pumps degrades as the coolant flashes. The steam generator heat transfer degrades after the steam-generator secondaries are isolated. The blowdown period ends when the intact-loop accumulator injection is initiated.

During *the refill period,* the reactor system starts to recover as the PSIS components (CMTs and accumulators) start to inject coolant into the primary system. The important refill components and phenomena concern the introduction of water into the reactor vessel downcomer and its subsequent distribution. Refill physical phenomena are the operation of the PSIS, including interactions between the accumulators and CMTs, bypassing injected water through the downcomer to the broken cold leg, and penetration of safety injection water into the lower plenum. The refill period ends when the mixture level in the lower plenum approaches the core inlet, and conditions are established for reflooding the core with coolant.

The *reflood period* begins once the lower plenum has refilled and the core liquid inventory enters a period of sustained recovery. The reflood process is highly oscillatory after the downcomer fills to the direct vessel injection line nozzle but the

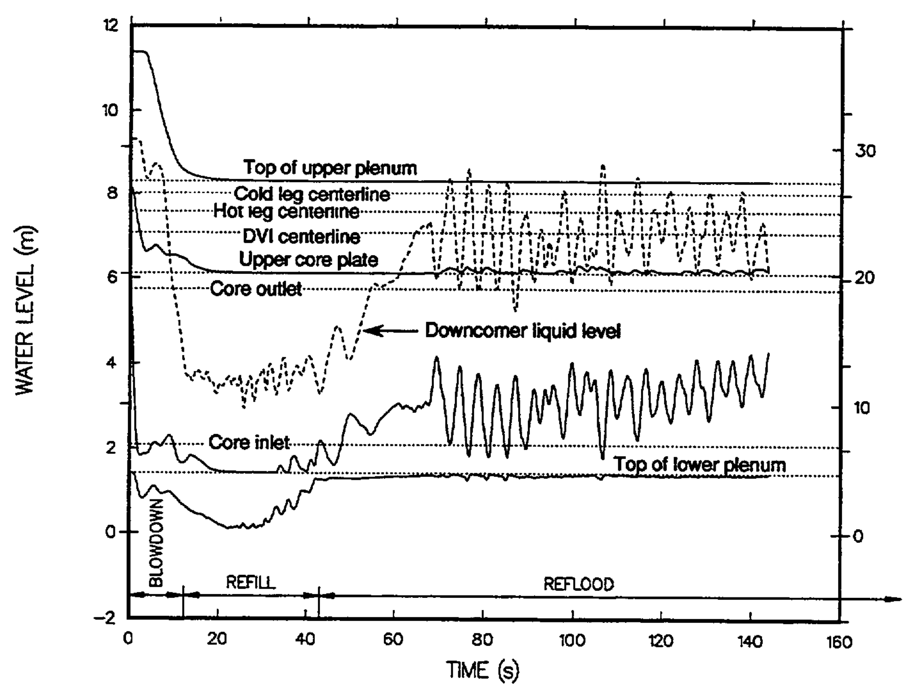

Fig. D-2. Vessel liquid volume fractions.

overall trend with increasing time is increasing core coolant inventory, i.e., a sustained recovery. Refilling of the core with coolant is well advanced by the end of the period. The reflood processes may be quite slow because much of the water is boiled and transported as steam and entrained droplets into the upper plenum and hot-leg piping. The reflood period ends when the entire core is quenched, that is, all fuel rod cladding temperatures are at or slightly above the coolant saturation temperature.

The *long-term cooling period* continues after the entire core quenches. At the time the fuel rod cladding is completely quenched, the core is only partially full. Accumulator discharge is still underway. After the accumulators empty, the CMTs resume draining their inventory into the primary. CMT draining leads to ADS actuation. IRWST injection is initiated when the primary pressure decreases to a level less than the static head in the IRWST. CMT and IRWST draining may occur simultaneously. Draining of the IRWST is expected to take several days, after which water from the sump is recirculated indefinitely. ADS stages 1–3 have an insignificant impact on the transient because the primary has largely depressurized to containment conditions before they open. After the inventory in one of the CMTs drops to 20% of its initial value, fourth stage ADS opens a direct path for release of core-generated steam to the containment. For many accident scenarios, the depressurization process must be assisted by operation of the ADS. However, the LBLOCA has sufficient area to depressurize the primary, even in the absence of ADS actuation.

D.2. Westinghouse 4-Loop Plant SBLOCA (PWR)

D.2.1. Plant Description

The following description is for the Callaway nuclear power plant.[D-3] The Westinghouse 4-Loop SBLOCA PIRT was based upon the Indian Point Unit 2 plant[D-4] but a description of that plant was not readily available.

The nuclear steam supply system consists of a reactor and four closed reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump and a steam generator (Fig. D-3). The nuclear steam supply system also contains an electrically heated pressurizer.

High-pressure water circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated water exits from the reactor vessel and passes via the coolant loop piping to the steam generators. Here it gives up its thermal energy to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel. The entire reactor coolant system is composed of leaktight components to ensure all fluids are confined to the system.

The reactor coolant pumps are Westinghouse vertical, single-stage, centrifugal pumps of the shaft seal type.

The steam generators are Westinghouse Model F vertical U–tube units that contain thermally treated Inconel tubes.

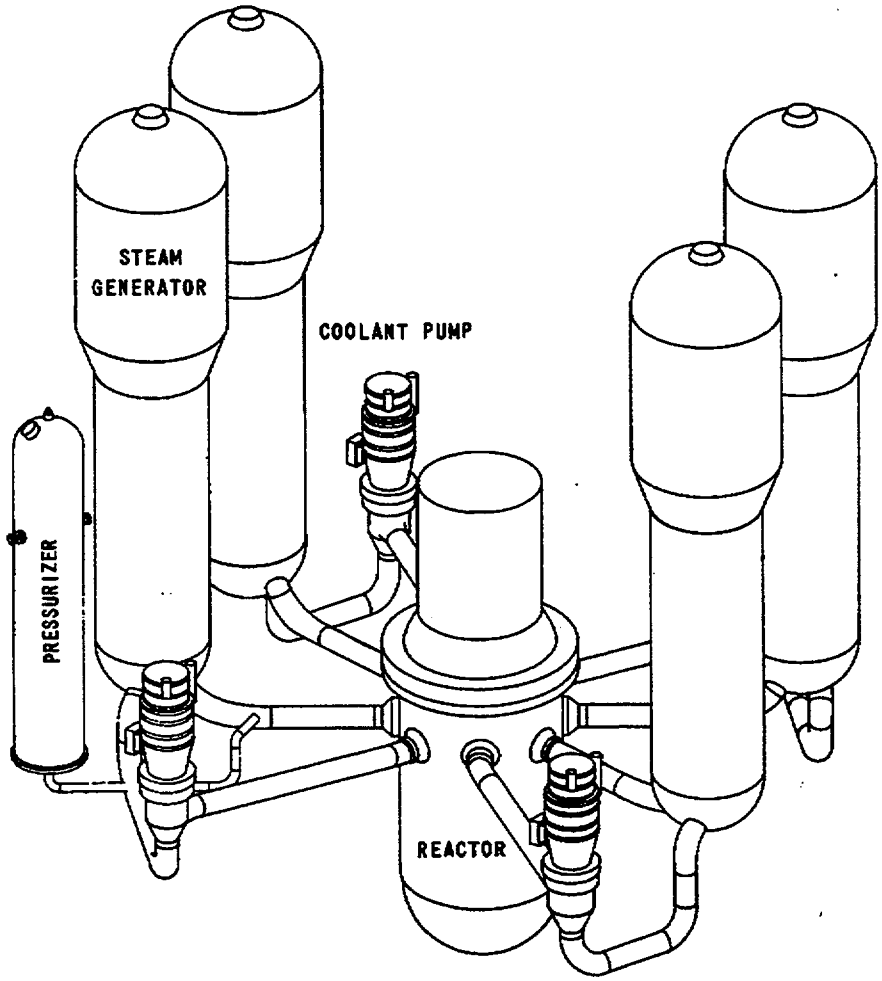

Fig. D-3. Simplified diagram of Westinghouse 4-loop nuclear steam supply system.

Essentially all of the metal surfaces in contact with the reactor water are stainless steel, except the steam generator tubes and the fuel rods which are Inconel and Zircaloy respectively.

An electrically heated pressurizer connected to one reactor coolant loop maintains reactor coolant system pressure during normal operation and limits pressure variations during plant load transients.

The ECCS injects borated water into the reactor coolant system following a LOCA. This limits damage to the fuel assemblies and limits metal–water reactions and fission product release. The ECCS also provides continuous long-term post-LOCA cooling of the core by recirculating borated water between the containment sumps and the reactor core.

D.2.2. Scenario Description

As described in Ref. D-4, the small-break transient is characterized by five periods: blowdown, natural circulation, loop seal clearance, boil-off, and core recovery. While the duration of each period is break-size-dependent, the small LOCA transient can be characterized as follows:

Blowdown: On initiation of the break, there is a rapid depressurization of the primary side of the reactor cooling system. Reactor trip is initiated on a low pressurizer pressure setpoint. Pump trip occurs either automatically at reactor trip (if the assumption is made that off-site power is lost coincident with reactor trip), or by the operators approximately 15-45 seconds following reactor trip if offsite power is available, based on plant Emergency Operating Procedures. Loss of condenser steam dump effectively isolates the steam generator secondary side, causing it to pressurize to the safety valve setpoints, and release steam through the safety valves. A safety injection signal occurs when the primary pressure decreases below the pressurizer low-low pressure setpoint and safety injection begins after a signal delay time. The reactor cooling system remains liquid solid for most of the blowdown period, with phase separation starting to occur in the upper head, upper plenum and hot legs near the end of this period. During the blowdown period, the break flow is single phase liquid only. Eventually, the rapid depressurization ends and the RCS reaches a pressure just above the steam-generator secondary-side pressure.

Natural Circulation: At the end of the blowdown period, the reactor cooling system reaches a quasi-equilibrium condition that can last for several hundred seconds, depending upon break size. During this period, the loops seals remain plugged and the system drains from the top down with voids beginning to form at the top of the steam generator tubes and continuing to form in the upper head and top of the upper plenum region. The steam generators remove decay heat during this time. Vapor generated in the core is trapped with the reactor cooling system by liquid plugs in the loop seals, and a low quality flow exits the break.

Loop Seal Clearance: The third period is the loop seal clearance period. When the liquid level in the downhill side of the steam generator is depressed to the elevation of the loop seal, steam previously trapped in the reactor cooling system can be vented to the break. The break flow, previously a low-quality mixture, transitions primarily to steam. Prior to loop seal venting, the inner vessel mixture level can drop rapidly, resulting in a deep but short core uncovery. Following loop seal venting, the core level recovers to about the cold leg elevation, as pressure imbalances throughout the reactor cooling system are relieved.

Boil-Off: Following loop seal venting, the vessel mixture level will decrease. In this period, the decrease is due to the gradual boil-off of the liquid inventory in the reactor vessel. The mixture level will reach a minimum, in some cases resulting in a deep core uncovery. The boil-off period ends when the core collapsed liquid level reaches a minimum. At this time, the reactor cooling system has depressurized to the accumulator setpoint, and the core boil-off rate matches the delivery of safety-injection to the vessel.

Core Recovery: The core recovery period extends from the time at which the inner vessel mixture level reaches a minimum in the boil-off period until all parts of the core quench and are covered by a low-quality mixture. The small-break LOCA is considered over, and the calculation is terminated once the entire core is quenched and the safety injection flow exceeds the break flow.

D.3. Babcock & Wilcox 2-x-4 Plant SBLOCA (PWR)

D.3.1. Plant Description

As described in Ref. D-5, the plant selected for the PIRT effort was a typical B&W lowered loop design (Fig. D-4). This design features two hot legs and four cold legs. The elevation of the lowest part of the cold leg is about 6 ft lower than the bottom of the reactor vessel, hence the name "lowered loop." The reactor coolant pumps are mounted such that the centerline of the discharge is 3.5 ft higher that the reactor vessel inlet piping. A section of the cold leg has an upward slope of 45 degrees to make up the elevation difference. One high-pressure injection line is connected to each of the cold legs on the side of this sloped section so that gravity will direct the high-pressure injection flow toward the reactor vessel.

A unique feature of the B&W vessel internals is the reactor vessel vent valves. These are circular flapper valves, hinged at the top, and are in the closed position held by gravity during normal operation. Eight of these valves are situated around the perimeter of the upper part of the downcomer and allow flow from the upper plenum to the downcomer. If the pressure in the upper plenum increases 0.1 psi greater than the pressure in the downcomer, the valves start to swing open, allowing mass flow from the upper plenum into the downcomer. The reactor vessel vent valves are fully open at 0.25 psid. Thus, the reactor vessel vent valves limit the possibility of pressure building in the upper plenum and depressing the core level.

The two steam generators of B&W design are once through, counter-current-flow heat exchangers. The primary coolant flows vertically downwards, between two plenums, through about 15,500 52-ft-long tubes. Because the primary coolant enters the steam generators at the top, the hot leg must rise up past the top of the steam generators and bend down to connect to the upper plenum. The characteristic inverted U-bend shape gives the hot lag a candy cane appearance. The uppermost part of this hot leg U-bend is a potential location for accumulation of vapor that can block the primary flow path. If the hot leg should drain such that the level falls below the U-bend, primary coolant flow will be interrupted. The U-bend has a small vent valve that can be opened by the operator to vent any bubbles that may have collected at the top.

In the secondary side of the steam generators, subcooled feedwater, preheated before it enters the steam generator, comes in through several nozzles located around the perimeter of the generator about midway between the bottom and top. The feedwater flows through an annular downcomer to the lower plenum and upward through the center of the steam generator, on the outside of the tubes. As the feedwater enters the downcomer, it mixes with saturated steam, which is pulled from the center of the steam generator through an aspirator. Sufficient steam mixes with the feedwater to produce saturated water at the bottom of the downcomer. Once in its upward path, the water

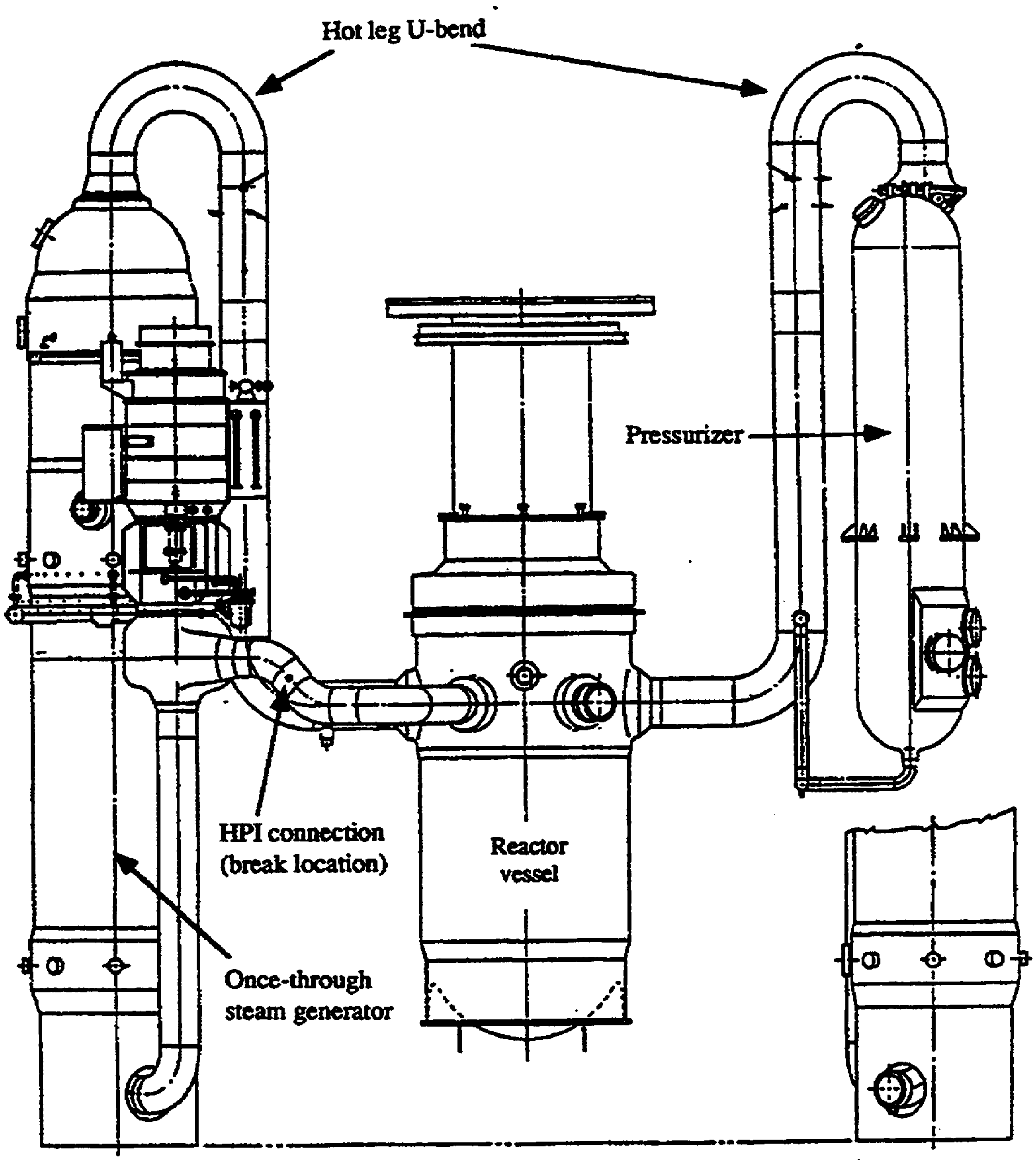

Fig. D-4. Typical B&W lowered-loop plant design.

boils and the generated steam superheats. As the water flows through the tube bundle, it is converted to steam, so that at the level of the aspirator, all of the liquid has been converted to saturated steam. The length of tubes remaining between the aspirator and the upper tube sheet then serve to superheat the steam. Steam superheated to about 33 K (60°F) leaves the generator through the steam annulus and into the steam line.

D.3.2. Scenario Description

As described in Ref. D-6, the SBLOCA scenario is subdivided into four time periods that characterize events during the sequence. The four time periods of the scenario are the following (Fig. D-5).

Blowdown Period: This phase begins with break initiation and ends with the end of reactor coolant pump coastdown. Following break initiation, the reactor begins a fast depressurization, which triggers the reactor trip. It is expected that flashing will start occurring throughout the hot path of the primary, as the primary begins to lose its subcooling margin. If sufficient flashing occurs, the depressurization may slow somewhat before the reactor trip occurs. Once the reactor trips, the heat source decreases rapidly and the rate of depressurization again increases. The operator becomes aware of the loss of subcooling margin and trips the reactor coolant pumps, as established by the emergency operating procedures for this plant. It is expected that at the end of this phase most of the primary side is single phase, conditions approach saturation, and the pump coastdown ends.

Saturation–Natural Circulation Period: This phase begins at the end of the pump coastdown and ends with the complete loss of natural circulation. The subcooling margin has been lost at the middle of this phase and the pressure remains on a plateau (saturation pressure) during this phase. The flow is becoming two-phase and a bubble begins to form at the top of the candy cane. As more and more steam is generated, it

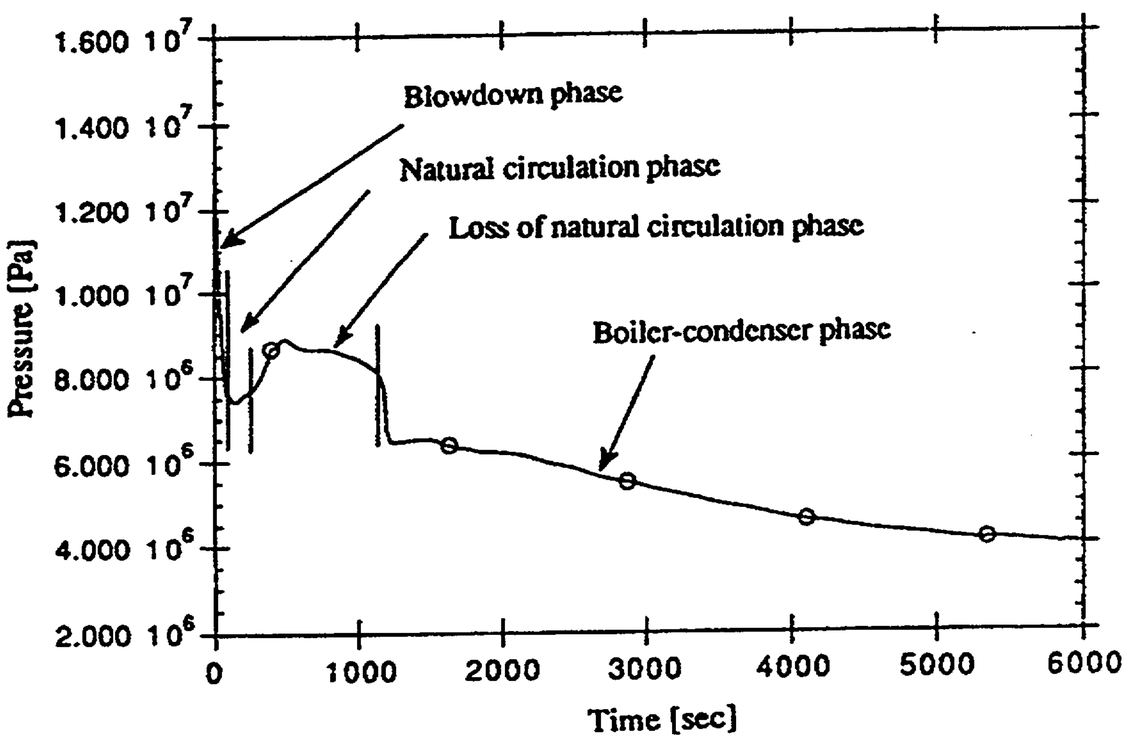

Fig. D-5. Scenario phases for B&W SBLOCA scenario.

becomes increasingly difficult for the natural circulation flow to sweep away the bubbles that now tend to accumulate at the top of the candy cane. A short intermittent mode is expected as the steam accumulates and the two-phase level recedes downward in the candy cane, thus momentarily interrupting the natural circulation. Once natural circulation is interrupted, the loss of the secondary heat sink results in repressurization of the primary. The pressure increase will compress the bubble on top of the candy cane, reestablishing the natural circulation. After a few cycles, the bubble will become too large to allow the liquid to rise to the inverted U-bend and the natural circulation will be interrupted permanently, thus ending this phase.

Loss of Natural Circulation Period: This phase begins with the loss of natural circulation and ends when the vessel steam begins to enter the candy cane. Having lost natural circulation, the pressure begins to increase once again. The main cooling mechanism for the core becomes internal vessel circulation. The reactor vessel vent valves open a flow path that allows the core outlet fluid into the downcomer where it can mix with the incoming high-pressure-injection coolant and recirculate through the core or communicate with the break. During the loss of natural circulation, the operator may decide to sequentially start, run, and stop the reactor coolant pumps, i.e., bump the pumps according to the emergency operating procedures. If the reactor coolant pumps are not bumped, the transient will eventually develop into the next phase, the boiler-condenser mode phase. The steam from the upper plenum begins to flow through the hot leg and find a condensing surface in the steam generator, thus removing decay heat.

Boiler-Condenser Mode Period: In this phase, the steam generated in the core condenses in the primary-side surface of the steam generator tubes and the secondary heat sink is reestablished. The pressure will drop as energy is removed by the boiler-condenser mode and through the break. This phase ends when ECCS begins to refill the primary and the plant enters a recovery phase.

D.3. General Electric LBLOCA (BWR/4)

D.3.1. Plant Description

A simplified diagram of the BWR/4 system configuration is presented in Fig. D-6. As described in Ref. D-7, the principal components of a BWR/4 system include the following.

- Reactor Vessel and Internals: Reactor pressure vessel, jet pumps, steam separators and dryers, core, and core support structures.
- Reactor Water Recirculation System: Pumps, valves, and piping used in providing and controlling flow.
- Main Steam Lines: Main steam valves, piping and pipe supports from reactor pressure vessel up to and including the isolation valves outside of the primary containment barrier.
- Control Rod Drive System: Control rods, control rod drive mechanisms and hydraulic system for insertion and withdrawal of the control rods.

- Nuclear Fuel and Instruments: The nuclear fuel (7 x 7) is located inside the core shroud.
- Engineering Safety Features: Pumps, valves, piping and water storage used to provide cooling and system inventory replacement during accident conditions. High Pressure Coolant Injection (HPCI), Low Pressure Core Spray (LPCS), Low Pressure Core Injection (LPCI), Automatic Depressurization System (ADS), and Residual Heat Removal (RHR).

The Reactor Vessel is divided into five regions: Lower Plenum, Core, Upper Plenum, Dome, and Downcomer region.

There are two external recirculation pumps and 20 internal jet pumps. Each recirculation line feeds five pairs of jet pumps, which are located outside the core shroud throughout

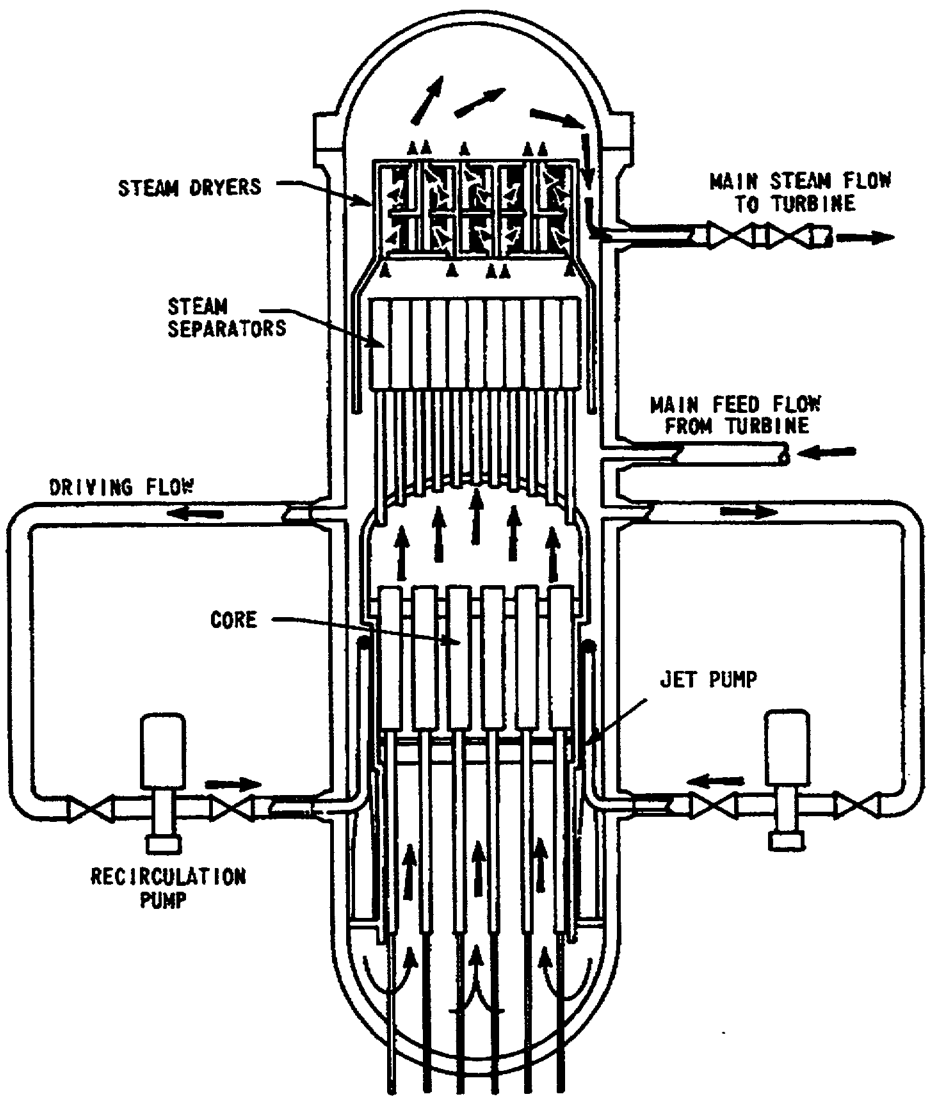

Fig. D-6. Simplified BWR/4 system illustration.

the perimeter of the reactor vessel. The jet pumps provide approximately two-thirds of the recirculation flow within the reactor vessel. Approximately one-third of the core flow is taken from the vessel through the two external recirculation loops. The external loop flow is pumped at a higher pressure, distributed through a manifold to which a number of riser pipes are connected, and returned to the vessel inlet nozzles. This flow is discharged from the jet pump throat where, due to a momentum exchange process, it induces the surrounding water in the downcomer region to be drawn into the jet pump throat. The two flows mix and then diffuse in the diffuser to be finally discharged into the lower plenum.

The BWR/4 power level is 3359 MWt, with a core consisting of 764 fuel assemblies. The steam separator and dryers are located above the core. This equipment is utilized to separate the steam from the liquid in order to avoid excessive rates of liquid in the steam supply system.

The control rods are utilized to effectively and rapidly reduce the power by absorption of neutrons. They are inserted from the bottom of the reactor vessel. There is one control rod for every four fuel assemblies in the core.

The ECCS for a BWR/4 consists of high-pressure coolant injection (HPCI), a low-pressure core spray system (LPCS), a low-pressure coolant injection system (LPCI), and an automatic depressurization system (ADS). The HPCI consists of a single motor driven pump and is designed to inject water into the vessel over the full range of operating pressures. The HPCI uses the condensate storage tank as the initial water supply and upon exhaustion of this source, the suppression pool provides water to this spray system. The injected coolant is injected into the vessel downcomer through the feedwater line.

The LPCS is a low-pressure core spray system. This low-pressure spray system is designed to provide injection for the larger breaks that result in rapid depressurizations of the vessel. The LPCS is also injected into the upper plenum through a circular sparger around the periphery of the core. The function of the LPCS is to limit the peak clad temperatures for intermediate and large breaks, whereas HPCI, along with ADS, is intended for core cooling following small breaks. The LPCS draws water from the suppression pool. The LPCI is capable of delivering large amounts of coolant to refill the vessel once the system depressurizes. The LPCI consists of three residual heat removal pumps, each of which injects coolant through separate piping into the recirculation loops.

The ADS activates about one-third of the safety relief valves in a BWR/4. These valves are opened to reduce the vessel pressure to mitigate the consequences of small breaks where depressurization is required to actuate the LPCI and LPCS.

D.3.2. Scenario Description

As described in Ref. D-7, a LOCA in a BWR is defined as an instantaneous break in the system with break sizes up to and including a double-ended severance of the recirculation loop piping. Recirculation line breaks produce the highest peak cladding temperatures in BWRs. As such, a double-ended guillotine break in the recirculation line for a BWR/4 with the unavailability of off-site power is chosen for the discussion below. LPCS, HPCI and LPCI are credited in the simulation.

As described in Ref. D-7, off-site power is assumed to be lost at the time of the break opening. Following reactor trip the core power decreases to the fission product decay heat.

Following opening of the break, the vessel pressure and core flow initially decrease. Because the energy expelled out the break and through the steam lines exceeds the energy deposited into the coolant from the core, the system depressurizes over the first few seconds. Very little mass is assumed to enter the system during this period because the feedwater flow is assumed to coast to a zero value in one second. At about 5 seconds, the main steam isolation valves are assumed to be completely closed, preventing steam from exiting the vessel. The closure of the main steam isolation valves causes the partial repressurization and the elevated system pressures during the first 10 seconds of the event.

The initial rapid loss in core flow is due to the opening of the break in the recirculation loop. However, the intact loop pump does coast down during the event and influences the core flow behavior during the early portion of the transient. When the suction to the jet pumps uncover, the core flow rapidly decreases. And, upon uncovery of the suction nozzle to the recirculation line, the volumetric flow rate through the break in this location increases significantly, causing an increase in the system depressurization rate. This increased depressurization after about 10 seconds causes the subcooled liquid in the lower plenum to eventually saturate and flash. Figure D-7 presents the lower plenum liquid mass and the decrease in inventory upon flashing at about 11 seconds into the transient. The flashing of the fluid in the lower plenum causes an associated increase in the core flow at about 11 seconds. The jet pump discharge mass flow rates display the early flow reversal on the broken loop side after the break opens. The downcomer liquid level rapidly decreases due to the opening of the break and the effect of lower plenum flashing at 11 seconds.

The break mass flow rate decreases as the suction line uncovers early in the event.

The clad temperature responses for the low, average and high power rods are given in Fig. D-8. The clad temperature is governed by the core flow early in the event as nucleate boiling governs the heat removal from the fuel rods during the initial portion of the event. As the core flow achieves a low flow condition, boiling transition develops as the core flow degrades and is a direct result of the uncovery of the jet pump discussed above. The heat transfer reaches film boiling, and with uncovery of the hot spot at about 25 seconds, the clad temperature for the high-powered rod begins to increase due to the low heat-transfer coefficients characteristic of steam cooling. The cladding temperature continues to increase at a rapid rate until the ECCS initiates injection into the reactor vessel initiating refill at about 46 seconds as noted in Fig. D-7.

Coolant enters the core peripheral bundles from the low-pressure core spray that condenses steam and pools in the upper plenum. The downflow of ECC injection (countercurrent flow) through the outer lower-power bundles initiates refill of the lower plenum. Because of the high steaming rates in the hotter fuel bundles, downflow

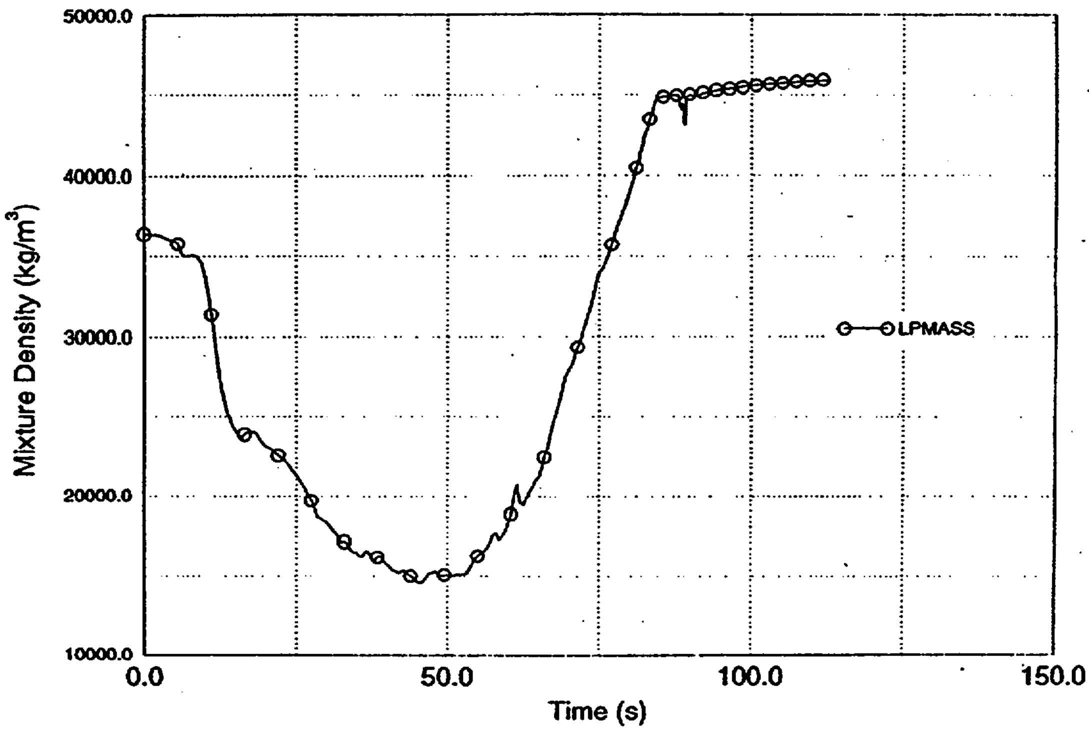

Fig. D-7. Lower-plenum fluid mass.

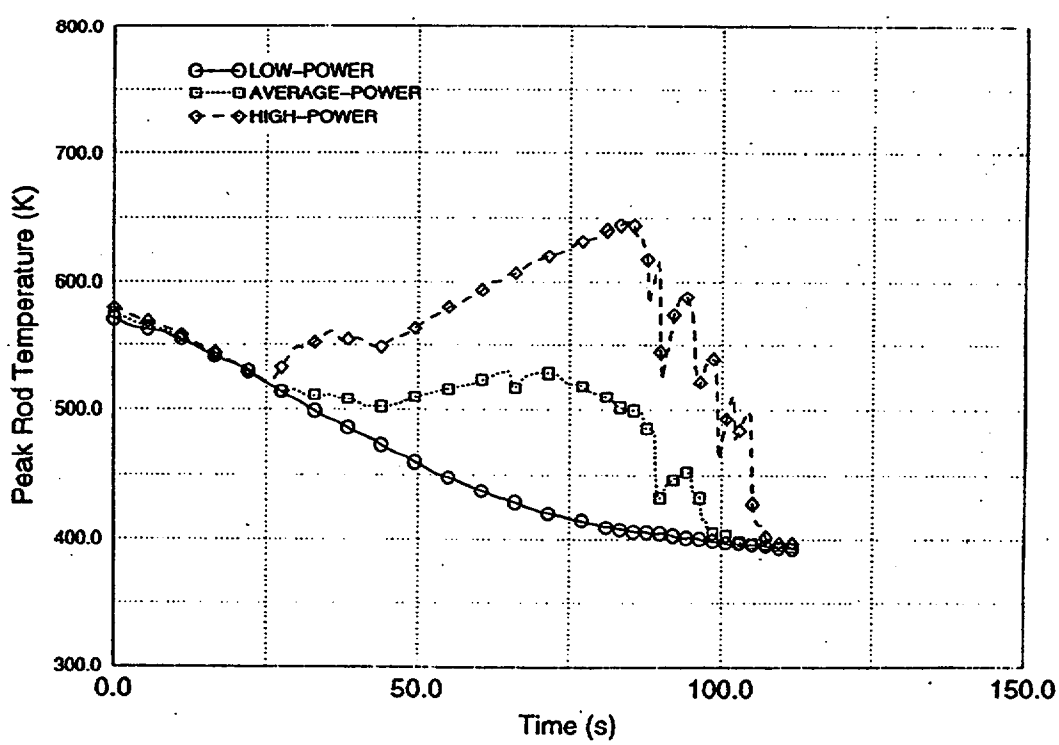

Fig. D-8. Peak cladding temperatures.

of liquid is precluded in the central core region. That is, the high steam velocities in these regions preclude counter-current flow through the upper tie plate and the channel inlet orifices. Once the lower plenum refills and flashing in this region subsides, reflood of the core central bundles begins at about 80 seconds in the core. However, until sufficient coolant enters the core, heat removal from the bundles in the interior of the core is controlled by forced convection to steam and thermal radiation. As sufficient coolant enters the core interior hot bundles, the droplets entrained in the steam eventually provide sufficient cooling to terminate the cladding heat-up as dispersed flow film boiling governs the heat removal from the upper portion of the fuel rods. As the coolant injection into the core continues during this reflood period, the core eventually quenches and the heat transfer returns to nucleate boiling, where the clad temperatures remain within several degrees of the coolant saturation temperature during the long term. Once sufficient coolant has entered the core's high-power region, the peak clad temperature is terminated and quench occurs at 107 seconds, as noted in Fig. D-8.

Early in the event, the two-phase level in the vessel remains at elevated values due to the early depressurization and attendant flashing of the liquid in the core. Following uncovery of the jet pump and the later lower-plenum flashing, the fluid lost through the break, along with the flashing and boiling in the core region, causes the upper portions of the fuel bundles to uncover. Following lower-plenum flashing and the continued depressurization of the system, the ECC is activated and coolant begins to enter and refill the vessel. Refill is initiated by the liquid downflow through the low-powered peripheral. The low- and average-powered core region bypass regions display this similar downward flow behavior. Reflood of the core begins after refill of the lower plenum and the clad temperature excursion is finally terminated at about 85 seconds into the event. Once sufficient coolant has entered the fuel bundles, fuel rod quenching is initiated. The heat transfer returns to nucleate boiling, which maintains the core in a cooled condition for the long term.

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APPENDIX E

OTHER STANDARD TEST PROBLEM SPECIFICATION EXAMPLES

Test problems developed by J. Mahaffy of Pennsylvania State University are summarized in this appendix. These problems illustrate several aspects of validation using standards other than those that employ experimental data as discussed in Sections 2 and 6 of the main report.

E.1. Static Vessel Test Problem

Purpose: The purpose of this problem is to test for anomalies in the 3D momentum transport terms that can result in spurious circulation patterns. It is an important test to qualify the code for use on passive reactors.

Success Metric: Fluid velocities should be observed at all positions in the final large edit and compared with the expected zero flow. In addition, the void fraction in level 11 should be compared with the expected value of 0.50.

Problem Description: This problem consists of a PWR vessel connected via short, single-cell pipes to zero flow boundary conditions on the cold legs, and constant atmospheric pressure conditions on the hot legs. All temperatures in the system are set to 300 K. All pressures are initialized to 0.1 MPa. The hot- and cold-leg pipes are initially full of air, and the vessel contains water up through the midpoint of the second level below the cold (or hot) legs. The upper vessel is filled with air.

Under ideal conditions, this problem undergoes a brief transient to adjust the pressures to appropriate hydrostatic values and then settles into a steady configuration with no flow and level water surface.

The vessel model used was obtained from the USPWR test problem. It has 4 radial zones with boundaries at 1.0919, 1.6855, 1.9376, and 2.1971 meters and 8 evenly spaced azimuth zones. There are 17 axial zones with upper faces at the following meters:

1.3672	1.9389	2.4469	3.0882	3.6400	4.2800
5.0100	5.9200	6.7458	7.1395	7.5523	7.9650
8.7015	9.2655	1.0137	10.926	12.575	

The void fraction in the vessel is set to 0 for levels 1 through 10; to 0.5 for level 11; and to 1.0 for levels 12 through 17. The air partial pressure is initially set equal to the total pressure (0.1 MPa) in all cells.

Connections to the vessel are through simple, single-cell PIPE components representing the nozzle sections of the hot and cold legs. The hot legs have a cell length of 0.825 m, cell volume of 0.408 m^3, cell face adjacent to the vessel with 0.8-m^2 area, and cell face adjacent to the BREAK of 0.427 m^2. The cold legs have a cell length of 1.163 m, a cell volume of 0.445 m^3, a cell face adjacent to the vessel of 0.6297 m^2, and a cell face adjacent to the FILL of 0.383 m^2.

The type 3 FILLs connected to the cold-leg PIPEs have geometries (DX and VOL) matching the adjacent PIPE cell. Both inlet velocities are set to 0. The type 0 BREAKs are also given geometries matching the adjacent PIPE cell.

E.2. Bubble Rise Test Problems

Purpose: The primary purpose of these problems is to check the logic for an initial increase above zero void fraction when the gas entering a cell is primarily non-condensable. As the run approaches steady state and timestep size increases, the problem also provides a valuable test of stability associated with interfacial drag.

Additionally, these tests provide a simple assessment of the low-void interfacial drag and can be used as a check for changes in the low-void evaporation model or in numerical diffusion.

Success Metrics: The output of Control block -100 at the end of the second timestep should be compared with the total mass of air that has flowed in from the FILL during the first two timesteps. This is a good flag for problems in the transition from single-phase liquid to bubbly flow.

Time history plots should be compared for timestep size, as should those of the void fraction, air partial, and vapor velocity midway up the tank. This information will permit detection of instabilities. When stable results are obtained, the information can be compared with data on bubble rise velocities and can be used to indicate changes in the evaporation model at low void.

Problem Description: These problems follow bubbles injected into the bottom of a tank of water. The tank is 2 m high and 2 m in diameter and is initially full of water. At time zero, air bubbles are injected at the bottom at velocity of 0.132 m/s from a source that has a void fraction of 0.03. Only the air enters from this source. The air-bubble velocity has been set to match the bubble rise velocity obtained from the present 1D interfacial drag correlation in TRAC for a void fraction of 0.03. It should be changed if the interfacial drag correlation is changed. The liquid velocity at this boundary is set to zero. The top of the tank is bounded by a pressure boundary condition of 0.1001 MPa. All temperatures in the tank and boundary conditions are 300 K. After ~15 s, the bubbles have spread uniformly through the tank, and a steady state should follow.

In all decks, a type 3 FILL supplies the air. The FILL's total pressure and air partial pressure are set to 0.1001 MPa. Its liquid and vapor temperatures are set to 300 K, and its void fraction set to 0.03. The volume of the fill is 0.3145926 m^3, and the length is 0.1 m. The liquid fill velocity is zero and the vapor velocity is 0.132 m/s. All decks also share the same upper-boundary pressure condition. This is provided with a type 0 BREAK, which has the same pressure, temperatures, void fraction, and geometry as the FILL.

The 1D versions of the test problems model the tank with a 20-cell PIPE. Each cell is 0.1 m long, with cell volume and cell edge flow areas calculated automatically from the 2-m, hydraulic diameter and the assumption of a uniform, circular cross section (FA and VOL set to -1.0 in the input). Two 1D problems have been created that differ only in the final maximum timestep size. One input deck runs to steady state, while the second develops a bounded instability due to a higher requested maximum timestep. As the code is improved, the final maximum timestep should be increased to maintain one stable problem and one with instability.

The 3D versions of the problem replace the central 18 cells of the pipe with an equivalent vessel configured with 1 radial ring, 1 theta zone, and 18 axial levels, each 0.1 m high. One pair of problems results in stable and unstable runs analogous to those for the pure 1D. A final problem was created with the new reflood model activated (the rod temperatures are all 300 K).

E.3. Falling Droplet Test Problems

Purpose: The two primary purposes of this test series are to check logic for initial decrease from a void fraction of one- to two-phase dispersed flow, and to test for stability problems associated with interfacial drag. Additionally, this test provides a simple assessment of the low-void interfacial drag and can be used as a check for changes in the low-void evaporation model or in numerical diffusion.

Success Metrics: Computed total system mass should be compared for all large edits in the calculation, with particular attention paid to the first three edits. This is used as a flag for problems in the transition from single-phase gas to dispersed flow.

Time history plots should be compared for timestep size, as should those of the void fraction, air partial and vapor velocity midway up the standpipe. This information will permit detection of instabilities. When stable results are obtained, the information can be compared with data droplet velocities and can be used to indicate changes in the evaporation model at high void.

Problem Description. This problem follows drops injected at the top of an air-filled standpipe. The standpipe is 2 m high and 2 m in diameter and initially contains only air. At time zero, water is injected into the top of the pipe at velocity of 0.2287 m/s from a source that has a void fraction of 0.99. Only the liquid enters from this source. This velocity has been set to match the droplet terminal velocity obtained from the current 1D interfacial drag correlation in TRAC for a void fraction of 0.99. This injection velocity should be changed if the interfacial drag correlation is changed. The gas velocity at this upper boundary is set to zero. The bottom of the standpipe is connected to a pressure boundary condition of 0.100 MPa. All temperatures in the standpipe and boundary conditions are 300 K. After ~10 s, the droplets have spread uniformly through the system and a steady state should follow.

In all decks a type 3 FILL supplies the liquid. The FILL's total pressure and air partial pressure are set to 0.100 MPa. Its liquid and vapor temperatures are set to 300 K, and its void fraction set to 0.03. The volume of the fill is 0.3145926 m^3, and the length 0.1 m. The gas fill velocity is 0, and the liquid velocity is 0.2287 m/s. All decks also share the same lower-boundary pressure condition. This is provided with a type 0 BREAK, which has the same pressure, temperatures, void fraction, and geometry as the FILL.

The 1D version of these test problems models the standpipe with a 20-cell PIPE. Each cell is 0.1 m long, with cell volume and cell edge flow areas calculated automatically from the 2-m, hydraulic diameter and the assumption of a uniform circular cross section (FA and VOL set to -1.0 in the input). Only a single 1D problem has been created because the code is stable for all timestep sizes currently permitted. As the code timestep control is improved, a second 1D test may be needed to mark a threshold of instability.

The 3D versions of the problem replace the central 18 cells of the pipe with an equivalent vessel configured with 1 radial ring, 1 theta zone, and 18 axial levels, each 0.1 m high. One pair of problems results in stable and unstable runs analogous to those for the pure 1D. A final problem pair was created with the new reflood model activated (although the rod temperature is 300 K).

E.4. Boron Transport

Purpose: The primary purpose of this test set is to provide a quantitative measure of the numerical diffusion associated with the code's boron transport equations. It has as a secondary purpose the introduction of a method by which the numerical diffusion of any of the mass or energy equations may be measured.

Success Metrics: The key output variable is the value of control block -120. The numerical value should be that predicted by the C-curve method for the conditions used in the calculation. The final value is of prime interest; however, a time history plot of this variable should be examined to be certain that it has ceased to change.

Problem Description: This problem models the propagation of a 1-s-long square pulse of boron with a peak concentration of 0.002 and a base concentration of 0. Flow is through a pipe 10 m in length and 1 m in diameter. Velocity of the pure liquid flow is maintained at 2.0 m/s. Temperature of the liquid is 577.6 K, and pressure at the outlet is fixed at 15.51 MPa.

A type 10 FILL drives flow. Input is set to only take boron concentration from a control block, other variables are taken as constants. The FILL void fraction is fixed at 0, the liquid velocity is at 2.0 m/s, the liquid temperature is at 577.6 K, and pressure is at 15.51 MPa. The volume associated with the FILL is 0.785398 m^3, and the length is 1.0 m. The control block supplying boron concentration (CB -5) is simply a table with entries of 0.002 at 0 and 1 s and 0 at 1.001 and 10000.0 s.

The PIPE component has 20 cells each 0.5 m long with cell volumes and face areas computed internally from the 1.0-m hydraulic diameter. Initial conditions in the pipe are set to give velocity of 2 m/s at all faces, and temperature of 577.6 K, pressure of 15.51 MPa, and void fraction of 0 in all cells.

Conditions at the PIPE outlet are provided by a type 0 BREAK component. Fluid conditions and geometry of the BREAK match those of the FILL, except that boron concentration is fixed at 0.

A key feature of the test problem is a set of control blocks (-1, -2, and -10 through -120) that implement the C-curve method to provide a quantitative measure of the numerical diffusion associated with the propagation of the boron. The method was originally developed for analysis of experimental data on turbulent mixing (Levenspiel, "Chemical Reaction Engineering," Second Edition, Wiley, 1972) and has been adapted for quantifying numerical diffusion.

APPENDIX F

CANDIDATE TESTS FOR THE TRAC-M COMMON LBLOCA VALIDATION TEST MATRIX

In this appendix, we present the candidate experimental facilities for the TRAC-M common LBLOCA validation test matrix. For each PIRT local-level (LL) process/phenomena identified in Section 4 (Table 4-1), we provide a table. Each table lists the experimental facilities that have produced data, which are candidates for inclusion in the validation test matrix. Where possible, specific tests have been identified, but we acknowledge that more effort is required in this regard. Local-level PIRT phenomena are covered in Tables F-1 through F-15. Component- and system-level PIRT phenomena are covered in Tables F-16 through F-22.

TABLE F-1
CANDIDATE COMMON EXPERIMENTAL FACILITIES: BOILING–FILM

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill, Reflood				
PIRT Parameter	Boiling–Film				
	Plant Range	Test Facility			
Plant Parameter		Stewart	Laperriere	Winfrith	THEF/INEL
P (MPa)	0.2–15.4	0.009 - 2.03	9.6 - 10	0.2 - 7	0.2 - 7
q (W/cm^2)	1–46	0.16 - 0.19	0.17 - 0.4	1 - 30	0.8 - 22.5
v (m/s)	0–4				
G (kg/m^2-s)	10–2455	360 - 2783	2815 - 4406	50 - 2000	12 - 70
Comments		Ref. 2: Fundamental tube data	Ref. 3: Fundamental tube data	Fundamental tube data. Ref. 4 facility #10.4	Fundamental tube data. Ref. 4 facility #11.3

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill, Reflood				
PIRT Parameter	Boiling–Film				
	Plant Range	Test Facility			
Plant Parameter		Fung/U. of Ottawa	Lehigh	TPTF/JAERI	Blowdown HT/RS37
P (MPa)	0.2–15.4	0.1	0.1 - 1.0	0.5 - 12	1.3 - 15
q (W/cm^2)	1–46		< 10	3 - 25	74 - 163
v (m/s)	0–4				
G (kg/m^2-s)	10–2455		< 300	20 - 410	3300 - 3828
Comments		Ref. 5: Fundamental tube data, includes void fraction.	Fundamental rod-bundle data. Ref. 4 facility #11.42.	BWR and PWR core geometries; Ref. 4 facility #6.1.	25-rod bundle; Ref. 4 facility # 4.5.

Nomenclature
P, pressure

q, heat flux
v, velocity
G, mass flux

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1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
2. J. C. Stewart, "Low Quality Film Boiling at Intermediate and Elevated Pressures," M.Sc. thesis, University of Ottawa, Ottawa, Canada (1981).
3. A. Laperriere, "An Analytical and Experimental Investigation of Forced Convective Film Boiling," M.Sc thesis, University of Ottawa, Ottawa (1983).
4. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
5. K. K. Fung, "Subcooled and Low Quality Film Boiling of Water in Vertical Flow at Atmospheric Pressure," Ph.D. Thesis, University of Ottawa (1981).

TABLE F-2
CANDIDATE COMMON EXPERIMENTAL FACILITIES: BOILING–TRANSITION

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill, Reflood				
PIRT Parameter	Boiling–Transition (Note 1)				
	Plant Range	Test Facility			
Plant Parameter		U. of Cincinnati	Argonne SGTF	U. of Ottawa	Johannsen
P (MPa)	0.2–15.4	0.1 - 0.4	7 - 15.3	0.1	0.1 - 1.2
q (W/cm^2)	1–46	2 - 75		40 - 250	20 - 800
v (m/s)	0–4				
G (kg/m^2-s)	10–2455	7.3 - 144	70 - 320	68 - 203	25 - 200
Comments		Refs. 2-3: Fundamental tube and annulus data (Note 3)	Ref. 4: Fundamental tube data (Note 3)	Ref. 5: Fundamental tube data	Ref. 7: Fundamental tube data

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill, Reflood				
PIRT Parameter	Boiling–Transition (Note 1)				
	Plant Range	Test Facility			
Plant Parameter		Bennett	FZK Single Rod	NEPTUN	
P (MPa)	0.2–15.4	6.9	0.1	0.41	
q (W/cm^2)	1–46	7 - 100	0–56		
v (m/s)	0–4				
G (kg/m^2-s)	10–2455	< 5500	150	15–150	
Comments		Ref. 8: fundamental tube data	Refs. 9-10 Note 4	Refs 9, 11: rod bundle tests 5036 and 5050, Note 4	

Nomenclature
P, pressure
q, heat flux
v, velocity
G, mass flux

References

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2. S. Wang, "A Study of Transition Boiling Phenomnena with Saturated Water at 1-4 Bar," Ph.D Thesis, College of Engineering, University of Cincinnati, Ohio (1981).
3. S. Wang, Y. K. Kao, J. Weisman, "Studies of Transition Boiling Heat Transfer in a Vertical Round Tube," Nuclear Engineering Design, Vol. 70, pp. 223-243 (1982).
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5. S. C. Cheng, W. W. L. Ng, K. T. Heng, and D. C. Groeneveld, "Measurements of Transition Boiling Data for Water Under Forced Convective Conditions," Transactions of the ASME, Journal of Heat Transfer, Vol. 100, pp. 382-384 (May 1978).
6. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
7. K. Johannsen, P. Weber, and Q. Feng, "Experimental Investigation of Heat Transfer in the Transition Boiling Region," Technische Universitat Berlin document EUR-13135 (October 1990).
8. A. W. Bennett, G. F. Hewitt, H. A. Kearsey, and R. K. F. Keeys, "Heat Transfer to Steam-Water Mixtures Flow in Uniformly Heated Tubes in Which the Critical Heat Flux Has Been Exceeded," Atomic Energy Research Establishment document AERE-R-5373 (March 1968).
9. E. Elias, V. Sanchez, and W. Hering, "Development and Validation of a Transition Boiling Model for RELAP5/MOD3 Reflood Simulation," Nuclear Engineering and Design, Vol. 183, pp. 269-286 (1998).
10. P. Hoffman and V. Noack, "Experiment on the Quench Behavior of the Fuel Rod Segments," Second International Quench Workshop, Karlsruhe (September 1996).
11. M. Richner, G. Th. Analytis, and S. N. Aksan, "Assessment of RELAP5/MOD2, cycle 36.02, Using NEPTUN Reflooding Experimental Data," Paul Scherrer Institut document PSI104, UREG/IA-00103 (October 1991).

Notes

1. In his review (Ref. 1), Johannsen states "The main conclusions of Refs. 1-5: There is a lack of a reliable empirical database for heat transfer in the transition and inverted annular flow film boiling region, especially at low flows and pressures; the available correlations and analytical models are not very accurate; and problems still exist in understanding the physical mechanisms."

2. The OECD/CSNI separate effect test matrix report (Ref. 6) identifies tests for "Heat Transfer: Post-CHF in the Core . . " but does not subdivide the post-CHF area further to identify tests that may have usable data for validating the transition boiling model.
3. Per Ref. 4: "It is important to differentiate between transition boiling phenomena in internal and external flows where the hydrodynamics are significantly different."
4. Used for validation of RELAP5/MOD3 transition boiling model (See Ref. 9). Data for NEPTUN Test 5050 is in the NEA data bank.

TABLE F-3
CANDIDATE COMMON EXPERIMENTAL FACILITIES: CONDENSATION–INTERFACIAL

Plant	Westinghouse 4-Loop PWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Condensation–Interfacial heat and mass transfer				
	Plant Range	Test Facility			
Plant Parameter		Lee, et al.	Kim, et al.	Akimoto, et al.	Celata, et al.
P (MPa)	0.1	1.0		0.05–0.2	0.1–1.0
G_g, (kg/m^2-s)			Re_v =2500–30000	0–74	to 20 kg/hr
G_l, (kg/m^2-s)			Re_l =800–15000	0–1000	To 120 kg/hr
Superheat (K)		10–40			40
Comments		Ref. 1: Cocurrent stratified horizontal condensing flows (See Note 1)	Ref. 2: counter-current steam-water stratified flow (See Note 2)	Ref. 3-4: water injected into flowing steam at 90° angle.	Ref. 5-6: near stagnant superheated steam condensing on a slowly-moving subcooled water surface

Nomenclature
P, pressure
G_g, gas mass flux
G_l, liquid mass flux

References

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2. H. J. Kim, S. C. Lee, and S. G. Bankoff, "Heat Transfer and Interfacial Drag in Countercurrent Steam-Water Stratified Flow," International Journal of Multiphase Flow, Vol. 11, pp. 593-606 (1985).
3. H. Akimoto, Y. Tanaka, Y. Kozawa, A. Inoue, and S. Aoki, "Oscillatory Flows Induced by Direct Contact Condensation of Flow Steam with Injected Water," Journal of Nuclear Science and Technology, Vol. 22, No. 4, pp. 269-283 (April 1985).
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6. G. P. Celata, M. Cumo, G. E. Farello an G. Focardi, "A Theoretical Model of Direct Contact Condensation on a Horizontal Surface," International Journal of Heat and Mass Transfer, Vol. 30, No. 3, pp. 459-467 (1987).

Notes

1. Inlet liquid temperatures were between 30 and 62°C.
2. Conducted at aspect ratios between 4 and 87 degrees. Vapor and liquid Reynolds numbers reported as between 2,500–30,000 and 800–15,000, respectively.

TABLE F-4
CANDIDATE COMMON EXPERIMENTAL FACILITIES: DRAINING

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Long-Term Cooling
PIRT Parameter	Draining

	Plant Range	Test Facility			
Plant Parameter		Foster	Lubin and Springer	Georgia Institute of Technology	
P (MPa)	0.2	0.1	0.1		
q (W/cm^2)					
G (kg/m^2-s)	0.0–4150 (note 1)	0.0–4150 (specify)	1580		
Comments		Ref. 1: formula provides the time to empty a vertical cylinder, the top of which is open to atmosphere	Ref. 2: SET experiment-data on draining water from a 5-1/2 in cylinder, the top of which is open to atmosphere	Refs. 3-4: SET experiment for draining of a sealed vertical cylinder induces 2-phase countercurrent flow	

	Plant Range	Test Facility	
Plant Parameter		ROSA-AP600	PACTEL
P (MPa)	0.2	0.2–7	
q (W/cm^2)			
G (kg/m^2-s)	0.0–4150 (note 1)		
Comments		Ref. 5: IET experiments (note 2). Need to acquire actual data reports.	Ref. 6: IET experiments in PACTEL, a scaled IET of a 6-loop VVER-440 type PWR. Assessment would also demonstrate adequacy of TRAC for this plant application

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. T. C. Foster, "Time Required to Empgy a Vessel," Chemical Engineering, Vol. 95, No. 5, pp. 171-172 (1990).
2. B. T. Lubin and G. S. Springer, "The Formation of a Dip on the Surface of a Liquid Draining From a Tank," Journal of Fluid Mechanics, Vol. 29, Part 2, pp. 385-390 (1967).
3. K. H. Lillibridge, S. M. Ghiaasiaan, and S. I. Abdel-Khalik, "An Experimental Study of Gravity-Driven Countercurrent Two-Phase Flow in Horizontal and Inclined Channels," Nuclear Technology, Vol. 105, pp. 123 (1994).
4. S. M. Ghiaasiaan, B. K. Kamboj, and S. I. Abdel-Khalik, "Modeling of Gravity-Driven Oscillatory Countercurrent Two-Phase Flows," Nuclear Science and Technology, Vol. 117, pp. 22-32 (1994).
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6. J. Tuunanen, V. Riikonen, J. Kouhia, and J. Vihavainen, "Analysis of PACTEL Passive Safety Injection Experiments GDE-21 through GDE-25," Nuclear Engineering and Design, Vol. 180, pp. 67-91 (1998).

Notes
1. Based upon TRAC-PF1/MOD2 intermediate break loss-of-coolant accident (LA-UR-95-1785). Maximum IRWST flows are 100 kg/s and 30 kg/s for the broken and intact loops, respectively. Maximum broken loop CMT flow is 50 kg/s. IRWST delivery line is 0.15405-m diameter. CMT delivery line is 0.17305-m diameter.
2. Reference 5 lists the following experiments as demonstrating a variety of Core Makeup Tank processes (SB1, CL4, CL3, CL6, CL7, CL5, PB2, SG1, DV1, CL8, PB1, AD1, and SG2).

TABLE F-5
CANDIDATE COMMON EXPERIMENTAL FACILITIES: ENTRAINMENT/DEENTRAINMENT

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, Reflood				
PIRT Parameter	Entrainment/deentrainment				
	Plant Range	Test Facility			
Plant Parameter		Cousins and Hewitt	Steen and Wallis	Lopez de Bertodano et al.	Paras and Karabelas
P (MPa)	0.2	0.22		0.14–0.66	
j_f (m/s)		0.06–0.39	0.08–0.319	0.074–0.54	0.02–0.2
j_g (m/s)		24-47		24.5–126	31–66
Comments		Ref. 1, 3: upward flow air-water in vertical round tube	Ref. 2, 3: downward air-water flow in 1.07 to 1.59-cm tubes	Ref. 4-5: adiabatic upward flow air-water loop.	Ref. 6: adiabatic horizontal air-water flow

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, Reflood				
PIRT Parameter	Entrainment/deentrainment				
	Plant Range	Test Facility			
Plant Parameter		Williams			
P (MPa)	0.2				
j_f (m/s)					
j_g (m/s)					
Comments		Ref. 7: adiabatic horizontal air-water flow in pipe			

Nomenclature

P, pressure

j_f, liquid phase volumetric flux (superficial velocity)

j_g, gas phase volumetric flux (superficial velocity)

References

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2. D. A. Steen and G. B. Wallis, "The Transition from Annular to Annular-Mist Concurrent Two-Phase Down Flow," Atomic Energy Commission Report NYO-3114-2 (1964).
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4. M. A. Lopez de Bertodano, C.-S. Jan, and S. G. Beus, "Annular Flow Entrainment Rate Experiment in a Small Vertical Pipe," Nuclear Engineering and Design, Vol. 178, pp. 61-70 (1997).
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6. S. V. Paras and A. J. Karabelas, "Droplet Entrainment and Deposition in Horizontal Annular Flow," International Journal of Multiphase Flow, Vol. 17, No. 4, pp. 455-468 (1991).
7. L. R. Williams, "Entrainment Measurements in a 4-Inch Horizontal Pipe," University of Illinois M.Sc. Thesis (1986).

TABLE F-6
CANDIDATE COMMON EXPERIMENTAL FACILITIES: EVAPORATION

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Blowdown, Refill, Reflood
PIRT Parameter	Evaporation

	Plant Range	Test Facility			
Plant Parameter			Allesandrini, et al.	Wurtz	Hewitt
P (MPa)	0.2–15.4		5.0	7.0	Low pressure
q (W/cm^2)	1- 46		Adiabatic	Adiabatic	61-65
G (kg/m^2-s)	0–2455		1500	500-1000	297
Subcooling (K)					
Comments			Ref. 2: See Note 1 Steam-water data	Ref. 3: See Note 1 Steam-water data	Ref. 4: See Note 1 non-equilibrium entrainment data

	Plant Range	Test Facility			
Plant Parameter		Becker	Lehigh Tube	INEL	Winfrith
P (MPa)	0.2–15.4	1-16		0.48 – 7.07	0.199 – 1.009
q (W/cm^2)	1- 46	10-300		0.8 – 22.5	1 - 30
G (kg/m^2-s)	0–2455	500-3000		12.1 - 70.7	51 - 2014
Subcooling (K)		10			
Comments		Ref. 5: See Note 2 Single tube-diameter and length 0.015 and 7 m, respectively; 5 different heat flux profiles.	Refs. 6-7: Internal flow in heated tube using hot-patch technique.	Refs. 8-9: Internal flow in heated tube using hot-patch technique Also entered for film boiling.	Refs. 10-11: Internal flow in heated tubes. Also entered for film boiling.

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	Evaporation				
	Plant Range	Test Facility			
Plant Parameter		Lehigh Bundle	Flecht-Seaset		
P (MPa)	0.2–15.4	0.105 – 0.120			
q (W/cm^2)	1- 46	< 10			
G (kg/m^2-s)	0–2455	< 300			
Subcooling (K)		0.4 - 40			
Comments		Ref. 12: 3x3 rod bundle with 98 fixed-CHF points and 278 slow-moving CHF data points. Wall temperatures and heat fluxes vs distance above the quench front. Vapor superheats at two axial locations. Used hot-patch technique. Also entered for film boiling.	Ref. 13: Use forced-reflood bundle experiment 31504. Flecht-Seaset used a core simulator consisting of 161 electrically heated rods within a 17x17 square matrix.		

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

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10. D. Swinnerton, R.A. Savage, and K. G. Pearson, "Heat Transfer Measurements in Steady-State Post-Dryout at Low Quality and Medium Pressure," AEA Thermal Reactor Services, Physics and thermal Hydraulic Division Report AEA-TRS-1045, Winfrith, United Kingdom Atomic Energy Report AEEW-R 2503 (1990).
11. D. Swinnerton, M. L. Hood, and K. G. Pearson, "Steady State Post-Dryout at Low Quality and Medium Pressure Data Report," Winfrith, United Kingdom Atomic Energy Report AEEW-R 2267 (1988).
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Notes

1. As cited in S. Gao, D. C. Leslie, and G. F. Hewitt, "An Improved TRAC Code for Two-Phase Annular Flow Modeling," submitted for publication in Nuclear Engineering and Design (1998).
2. As cited in B. J. Azzopardi, "Prediction of Dryout and Post-Dryout Heat Transfer with Axially Non-Uniform Heat Input by Means of an Annular Flow Model," Nuclear Engineering and Design, Vol. 163, pp. 51-57 (1996).

TABLE F-7
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLASHING–INTERFACIAL

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Blowdown
PIRT Parameter	Flashing–Interfacial heat and mass transfer

	Plant Range	Test Facility			
Plant Parameter		Sozzi and Sutherland	Edwards & O'Brian	CANON SUPER CANON VERTICAL CANON	BNL Nozzle
P (MPa)	5.1–15.4		0.1–7	3.2; 15.0; 13.0	0.7
G (kg/m^2-s)	0–2455				3130–7010
Subcooling (K)					T_{inlet} = 300 K
Comments		Ref. 1: Flashing discharge through a pipe with various entrance character-istics.	Ref. 2: Pipe blowdown	Ref. 3: Pipe blow-down. OECD/SET Facility Numbers 3.3 and 3.4 (See Note 1)	Refs. 4-5: Converging diverging nozzle

	Plant Range	Test Facility			
Plant Parameter		MOBY DICK SUPER MOBY DICK	OMEGA		
P (MPa)	5.1–15.4	0.15–12	0.1–15		
G (kg/m^2-s)	0–2455	4200–10300	W=10-19 kg/s		
Subcooling (K)		Subcooled Saturated	T_{inlet} = 558 K		
Comments		Ref. 3: steady-state critical flow in tubes and nozzles over a spectrum of pres-sures. OECD/SET Facility Number 3.1, 3.2	Ref. 3: SET test for blowdown of rod bundle. OECD/SET Facility Number 3.15		

Nomenclature
P, pressure
G, mass flux
W, mass flow

References

1. G. L. Sozzi and W. A. Sutherland, "Critical Flow of Saturated and Subcooled Water at High Pressure," General Electric Co. document NEDO-13418 (1975).
2. A. R. Edwards and T. P. O'Brian, "Studies of Phenomena Connected with the Depressurization of Reactors," Journal of the British Nuclear Energy Society, V. 9, pp. 125-135 (1970).
3. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
4. N. Abuaf, B. J. C. Wu, G. A. Zimmer, and P. Saha, "A Study of Nonequilibrium Flashing of Water in a Converging Diverging Nozzle," Vol. 1: Experimental, Vol. 2: Modeling, Brookhaven National Laboratory document NUREG/CR-1864 and BNL-NUREG-51317 (June 1981).
5. P. Saha, N. Abuaf, and B. J. C. Wu, "A Nonequilibrium Vapor Generation Model for Flashing Flows," Transactions of the ASME, Journal of Heat Transfer, Vol. 106, pp. 198-203 (February 1984).

Notes

1. Some of the CANON series of data have been used for TRAC-PF1/MOD1 assessment, and the results are reported in NUREG/IA reports 0001 and 0023.

TABLE F-8
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLOW–CRITICAL

Plant	Westinghouse AP600				
Transient	Large-, Intermediate, and Small-Break Loss-of-Coolant Accident				
Transient Phase	Dependent upon break size				
PIRT Parameter	Critical Flow in Break				
	Plant Range	OECD Test Facility (Ref. 1)			
Plant Parameter	Break	Super Moby Dick (CEA - France)	Rebeca (CEA - France)	Marviken (Sweden)	Piper (University of PISA, DCMN / Italy)
Break Diam. (m)	0.0254 - 0.5588	0.020	0.030	0.2 - 0.509	0.01 - 0.05
Break L/D	1 - >10	0 - 20	0	0.3 to 3.7	
P (MPa)	15.78 - 0.102	2 - 12	0.2 - 0.8	0.1 - 5.2	1 - 9
G (kg/m^2-s)	1.2e06 - 10	8140-62000			
Void fraction	0.0 - 1.0	0 - 0.94	0.981 - 0.999	0 - 1.0	0.0 - 0.9
Subcooling (°C)	71.2 - 0.0	63.8 - 0.0	0	50 - 0	0 - 150
T_{liq} (K)	548.1 - 373.2	421.7 - 597.8			
T_{vap} (K)	619.3 - 400.0	485.5 - 597.8			
	OECD Facility ID	3.2	3.25	8.2	5.17
Comments	Plant parameter ranges are from TRAC AP600 LBLOCA, IBLOCA, and SBLOCA analyses (Refs. 2-4).	Vertical upflow, steady-state facility. Three nozzle configurations tested. Super Moby Dick was one of the critical flow tests used to assess TRAC-PF1/MOD1 Version 14.3 (Ref. 5).	Vertical downflow steady-state facility. Two convergent-divergent nozzle geometries tested. Steam and steam-air mixtures.	Large scale critical flow facility (Ref. 6). A number of nozzle geometries were tested ranging from 0.2 m to 0.509 m in diameter with length-to-diameter ratios from 0.3 to 3.7. TRAC-PF1/MOD2 has been assessed against six tests (Ref. 7).	The Piper facility is primarily for BWR blowdown experiments. The test section is a vertical cylindrical tube, 0.19 m ID, 3m length.

Plant	Westinghouse AP600				
Transient	Large-, Intermediate, and Small-Break Loss-of-Coolant Accident				
Transient Phase	Dependent upon break size				
PIRT Parameter	Critical Flow in Break				
	Plant Range	OECD Test Facility (Ref. 1)			
Plant Parameter	Break	TPFL (Two-Phase Flow Loop, USA)	Critical Flow Rig (GE - USA)	Edwards Blowdown Experiment (UK)	Additional Test Facilities (See Notes).
Break Diam. (m)	0.0254 - 0.5588		0.0127 - 0.0762	0.073	
Break L/D	1 - >10		0.0 - 140.0	56.1	
P (MPa)	15.78 - 0.102	2.0 - 6.0	4.1 - 6.9	6.9 - 0.1	
G (kg/m^2-s)	1.2e06 - 10			17500 - 200	
Void fraction	0.0 - 1.0		0.0 - 0.13	0.0 - 1.0	
Subcooling (°C)	71.2 - 0.0			55.0 - 0.0	
T_{liq} (K)	548.1 - 373.2				
T_{vap} (K)	619.3 - 400.0				
	OECD Facility ID	11.35	11.54		
Comments	Plant parameter ranges are from TRAC AP600 LBLOCA, IBLOCA, and SBLOCA analyses (Refs. 2-4).	Multipurpose support facility to LOFT LOCA experiments. Tee/critical flow experiments performed. The facility has been used for different kinds of experiments but no relevant information is available.	These tests investigated low-quality critical flow, including effects of geometry, length, and L/D. The tests covered 7 different types of nozzles with different nozzle test section lengths. (See Ref. 8)	The Edwards blowdown experiment is not one of the CSNI facilities but is included in the matrix because it simulates a double-ended break of a primary loop pipe. (See Ref. 9)	

Plant	Westinghouse AP600				
Transient	Intermediate, and Small-Break Loss-of-Coolant Accident				
Transient Phase	Dependent upon break size				
PIRT Parameter	Critical Flow in Valves				
	Plant Range	OECD Test Facility (Ref. 1)			
Plant Parameter	ADS Valves	Safety Valve (CISE-SIET, Italy)	Valve Blowdown Facility (CEGB-MEL / UK)	Additional Test Facilities (See Notes).	
Valve Diam. (m)	0.0615 - 0.1767	0.0203, 0.0045			
Valve L/D	>10	0			
P (MPa)	5.5 - 0.102	6.0 - 19.0	28.2		
G (kg/m^2-s)					
Void fraction	0+ - 1.0	0 - 1.0	0 to 1.0		
Subcooling (°C)	0.0				
T_{liq} (K)			513.0		
T_{vap} (K)					
	OECD Facility ID	5.5	10.21		
Comments	Plant parameter ranges are from TRAC AP600 IBLOCA and SBLOCA analyses (Refs. 3-4).	Tested PWR primary loop safety valve behavior in LOCA and operational transients and two-phase flow conditions. Two scaled safety valves tested: (1) 1:7.4 Crosby Type HB valve, 6 M6 orifice and (2) 1:133 SPES pressurizer safety valve.	High flowrate, high pressure test facility for research, development, and testing on primary circuit overpressure protection system valves for the Sizewell B PWR.		

Nomenclature

P	Pressure
G	Mass Flux
T_{liq}	Liquid Temperature
T_{vap}	Vapor Temperature

References

1. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
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5. B. Spindler and M. Pellissier, "Assessment of TRAC-PF1/MOD1 Version 14.3 Using Separate Effects Critical Flow and Blowdown Experiments, Volumes 1 and 2," USNRC Report NUREG/IA-0023 (SETh/LEML/88-138) (April 1990).
6. R. R. Schultz and L. Ericson, "The Marviken Critical Flow Test Program," Nuclear Safety, Vol. 22, No. 6, (1981) pp. 712-724.
7. J. L. Steiner and J. F. Lime, "Comparison of TRAC-PF1/MOD2 Calculated Results with Critical-Flow Test Data," Los Alamos National Laboratory report LA-UR-98-2565 (May 1998).
8. G. L. Sozzi and W. A. Sutherland, "Critical Flow of Saturated and Subcooled Water at High Pressure," 1975 ASME Winter Annual Meeting Symposium on "Non-Equilibrium Two-Phase Flows" held in Houston, Texas.
9. A. R. Edwards and T. P. O'Brien, "Studies of Phenomena Connected with the Depressurization of Water Reactors, *J. Br. Nucl. Energy Soc.* **9**, 125-135 (1970).
10. E. D. Hughes and B. E. Boyack, "TRAC-P Validiation Test Matrix," Los Alamos National Laboratory report LA-UR-97-3990 (September 1997).

Notes

There are a number of other facilities selected in the TRAC-P Validation Test Matrix report (Ref. 10) for critical-flow assessment. The following is from the Validation Test Matrix. Not included in the list are those facilities already cited (Super Moby Dick, Marviken, and Critical Flow Rig).

Test Facility	Description
CISE Blowdown	A vertical-pipe blowdown experiment studied depressurization and heat-transfer phenomena of initially flowing subcooled water.
LOFT Valve/Wyle	Studied small-break blowdown from a horizontal round pipe through a 16.0 mm diameter nozzle (may be OECD/CSNI facility 11.5 or 11.34, but no data sheet for either).
ROSA APCL - 03	ROSA 1-inch Cold Leg Break Test.
Carofano-McManus	Studied critical flow of two-phase water at about 0.16 Mpa.
Cumulus Critical Flow	Critical flow of superheated vapor and subcooled liquid through the pressurizer relief valves of a French PWR.
Deich Critical Flow	Studied two-phase critical flow at 0.12 Mpa.
Fincke-Collins Critical Flow	Studied critical flow of subcooled water at pressure from 0.09 to 0.30 MPa.
Neussen Critical Flow	Studied critical flow of two-phase water at pressure from 0.84 to 6.5 Mpa.
VAPORE	Two-phase critical flow through the full-scale automatic depressurization system (ADS) valve trains for the AP600.

TABLE F-9
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLOW–DISCHARGE

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, Reflood				
PIRT Parameter	Discharge				
	Plant Range	Test Facility			
Plant Parameter		LOFT L3-1 (Note 1)	SRL Gas Pressurizer (Note 2)	KMR-2	
P (MPa)	0.2–5.0	1.5–4.5			
q (W/cm^2)					
G (kg/m^2-s)	0–16100				
Comments		Ref. 1	Ref. 3	Ref. 5	

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. P. D. Bayless, J. B. Marlow, and R. H. Averill, "Experimental Data Report for LOFT Nuclear Small-Break Experiment L3-1," EG&G Idaho, Inc. document NUREG/CR-1145, also EGG-2007 (January 1980).
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3. W. L. Howarth and R. A. Dimenna, "SRS Supplemental Safety System Injection (Gas Pressurizer) Test," Westinghouse Savannah River Company report WSRC-MS-92-519 (May 3, 1993).
4. W. L. Howarth and R. A. Dimenna, "RELAP5 MOD3 Analysis of SRS Supplemental Safety System Injection (Gas Pressurizer) Test," Westinghouse Savannah River Company report WSRC-MS-92-519X (December 29, 1992).
5. A. S. Devkin and B. F. Balunov, "RELAP5/MOD3 Assessment for the Depressurization Processes at the Test Facility KMR-2 with Gas-Steam Pressurizer," Proceedings of the International Conference on New Trends in Nuclear System Thermohydraulics, Pisa, Italy, Volume 1, pp. 429-33 (May 30 - June 2, 1994).

Notes

- This test was used to validate the accumulator model in RELAP5/MOD3 as described in Ref. 2, Section 2.2.7.
- This test was used to validate the accumulator model in RELAP5/MOD3 as described in Ref. 4.

TABLE F-10
CANDIDATE COMMON EXPERIMENTAL FACILITIES: HEAT CONDUCTANCE–FUEL-CLAD GAP

Plant	Westinghouse AP600		
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)		
Transient Phase	Blowdown		
PIRT Parameter	Gap conductance		
	Plant Range	Test Facility	
Plant Parameter		Modified Pulse Design (low pressure)	Modified Pulse Design (high pressure)
Gas pressure (MPa)	2.5	0.1	
Temperature (K)	294	293 - 873	
Gas composition (Note 1)	Helium (94.7%) Air (4.4%) Argon (0.5%) Xenon (0.34%) Krypton (0.06%)	helium (100), argon (100), xenon, (100) helium/argon (51.8/48.2), and helium/xenon (89/11)	
Interfacial surface morphology or ISM (μm)		Depleted UO_2: ISM-I =14.4±2.8; ISM-II = 1.6±0.7; and ISM-III = x±0.05 Zircaloy-4: ISM-I =4.5±0.4; ISM-II = 0.4±0.2; and ISM-III = x±0.05	
Gap width (μm)	10	2.7 - 33.0	
Comments	Above as-built conditions	Source of data is Ref. 3. Reference 4 reports use of the data to validate a modified model.	Source of data is Ref. 5.

Plant	Westinghouse AP600		
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)		
Transient Phase	Blowdown		
PIRT Parameter	Gap conductance		
	Plant Range	Test Facility	
Plant Parameter		Power Burst Facility	Halden assembly IFA-226
Gas pressure (MPa)	2.5		
Temperature (K)	294		
Gas composition (Note 1)	Helium (94.7%) Air (4.4%) Argon (0.5%) Xenon (0.34%) Krypton (0.06%)		helium, argon, xenon, krypton, nitrogen, hydrogen
Interfacial surface morphology or ISM (μm)			
Gap width (μm)	10		210 - 250
Comments	Above as-built conditions	Source of data is Ref. 6.	Source of data is Ref. 7, as reported in Ref. 8

Nomenclature
See Plant Parameters

References

1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
2. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
3. J. E. Garnier and S. Begej, "Ex-Reactor Determination of Thermal Gap and Contact Conductance Between Uranium Dioxide: Zircaloy-4 Interfaces - Stage I - Low Gas Pressure," Pacific Northwest Laboratories document PNL-2697, NUREG/CR-0330 (January 1979).
4. V. K. Chandola and S. K. Loyalka, "Gap Conductance and Temperature Transients in Modified Pulse Design Experiments," Nuclear Technology, Vol. 56, pp. 434-446 (March 1982).

5. J. E. Garnier and S. Begej, "Ex-Reactor Determination of Thermal Gap and Contact Conductance Between Uranium dioxide: Zircaloy-4 Interfaces - Stage II: High Gas Pressure," Pacific Northwest Laboratories document PNL-2232, NUREG/CR-0330, Vol. 2 (July 1980).
6. G. A. Berna, et al., "Gap Conductance Test Series-2 test Results Report for Tests GC 2-1, GC 2-2, and GC 2-3," NUREG/CR-0300, TREE-1268 (November 1978).
7. E. T. Laats, P. E. MacDonald, and W. J. Quapp, "USNRC-OECD Halden Project Fuel Behavior Test Program - Experiment Data Report for Test Assemblies IFA-226 and IFA-239," Idaho Nuclear Engineering Laboratory (December 1975).
8. P. E. MacDonald and J. Weisman, "Effect of Pellet Cracking on Light Water Reactor Fuel Temperatures," Nuclear Technology, Vol. 31, pp. 357-366 (December 1976).

Notes

1. Gas composition used in B. E. Boyack, et al., "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Methodology to a Large-Break Loss-of-Coolant Accident," EG&G Idaho, Inc. document NUREG/CR-5249, also EGG-2552 (October 1989).
2. See Ref. 1 for a brief description of the current TRAC model, section 3.4.5, pg. 3-85 to 3-86.
3. Gap conductance is not identified as an experimental parameter in Ref. 2.
4. Experimental results show that fuel pellets crack, relocate, and are eccentrically positioned within the sheath. As a result, the heat transfer across the fuel-sheath gap is significantly greater than that which is calculated with fuel pellet modeling as solid concentric cylinder (See Ref. 8).

TABLE F-11
CANDIDATE COMMON EXPERIMENTAL FACILITIES: HEAT TRANSFER–FORCED CONVECTION TO VAPOR

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Forced Convection to Vapor (Note 1)				
	Plant Range	Test Facility			
Plant Parameter		Babus'Haq	Davies &Al-Arabi		
P (MPa)	0.1				
q (W/cm^2)	1				
v (m/s)	0-4				
G (kg/m^2-s)	10-20				
Re (core)	$1.4\text{-}2.8 \times 10^4$	$1.2\text{–}5.5 \times 10^4$			
Comments		Ref. 1: Tests performed with air rather than steam	Ref. 2: Tests performed with water		

<u>Nomenclature</u>
P, pressure
q, heat flux
v, velocity
G, mass flux
Re, Reynolds Number

<u>References</u>

1. R. F. Babus'Haq, "Forced-Convective Heat Transfer from a Pipe to Air Flowing Turbulently Inside It," Experimental Heat Transfer, Vol. 5, pp. 161-173 (1992).
2. V. C. Davies and M. Al-Arabi, "Heat Transfer Between Tubes and a Fluid Flowing Through Them with Varying Degrees of Turbulence Due to Entrance Conditions," Proc. Inst. Mech. Eng, Vol. 169, pp. 993-1006 (1955).

TABLE F-12
CANDIDATE COMMON EXPERIMENTAL FACILITIES: HEAT TRANSFER–STORED ENERGY RELEASE

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Blowdown
PIRT Parameter	Stored Energy Release

	Plant Range	Test Facility	
Plant Parameter		Power Burst Facility Test PCM-2 (Ref. 1-2)	Power Burst Facility Test LOC-11C (Ref. 3-4)
P (MPa)	5.1–15.4	13.53	
q (W/cm²)	7–46	136	
G (kg/m²-s)	0–2455	750–1361	
Comments	Above as-built conditions	Unirradiated fuel used.	

	Plant Range	Test Facility	
Plant Parameter		PHEBUS LBLOCA Test 212 (Ref. 5)	LOFT L6-8B-1 and L6-8B-2 (Ref. 6-7)
P (MPa)	5.1–15.4		14.6 rising to 15.7 decreasing to 14.2
q (W/cm²)	7–46		
G (kg/m²-s)	0–2455		
Comments	Above as-built conditions	Nuclear fuel rods used.	Fuel centerline temperature available during slow transient with controlled core conditions.

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. Z. R. Martinson, "Power-Cooling-Mismatch test serest test PCM-2 Test Results Report," Idaho National Engineering Laboratory document NUREG/CR-1038 (1977).
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3. J. R. Larson, et al., "PBF-LOCA Test Series, Test LOC-11 Test Results Report," NUREG/CR-0618, TREE-1329 (March 1979).
4. P. E. MacDonald, J. M. Broughton, and J. W. Spore, "An Evaluation of the Thermal-Hydraulic and Fuel Rod Thermal and Mechnical Behavior During the First Power Burst Facility Nuclear Tests," Nuclear Technology, Vol. 44, pp. 401-410 (August 1979).
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6. D. B. Jarrell, J. M. Divine, and K. J. McKenna, "Experimental Data Report for LOFT Anticipated Transient Slow and Fast Rod Withdrawal Experiment L6-8," NUREG/CR0-2930 (July 1982).
7. C. L. Nalezny, "Summary of Nuclear Regulatory Commission's LOFT Program Experiments," NUREG/CR-3214 (July 1983).

TABLE F-13
CANDIDATE COMMON EXPERIMENTAL FACILITIES: INTERFACIAL DRAG (CORE AND DOWNCOMER)

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Reflood				
PIRT Parameter	Core Interfacial Drag				
	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		Dadine	Pericles Rectangular	Pericles Cylindrical	Erset Rod Bundle
P (MPa)	0.333-0.396	0.1-0.6	0.2-0.4	0.2-0.4	0.1-0.6
q (W/cm^2)		1-3	2.27-4.36	1.5-4.2	1-7
Wall Temp. (K)	860-1197	300-600	385-700	355-600	300-900
Flooding Rate (cm/s)	0-14		0-5	1-19	1-12
G (kg/m^2-s)	45.5-98.4	20-150	25-50	2-190	10-120
Subcooling (°C)		20-50	30-90	60	20-80
Void fraction	0-1.0				
Comments		Heated tube	Rect. 357-rod core	Cylind. 368-rod core	36-rod bundle

	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		Rebeca	TPTF Jaeri	SCTF Jaeri	CCTF Jaeri
P (MPa)	0.333-0.396	0.2-0.8	3.1-12	≤0.6	≤0.6
q (W/cm^2)					
Wall Temp. (K)	860-1197		≤920K		
Flooding Rate (cm/s)	0-14		≤120		
G (kg/m^2-s)	45.5-98.4		17-94		
Subcooling (°C)			≤20		
Void fraction	0-1.0				
Comments		Critical flow	Horizontal two-phase flow and core heat transfer facility (25-, 24-, and 39-rod core geometries)	2D 8 fuel-rod bundle core	3D 32 fuel-rod bundle core

Plant	Westinghouse AP600
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Reflood
PIRT Parameter	Core Interfacial Drag

	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		Frigg/Froja	Neptun-1 & Neptun-2 Reflood	Achilles Reflood Loop	Thetis Bundle
P (MPa)	0.333-0.396	3-8.7	0.1-0.41		0.13-0.4
q (W/cm^2)		21-89			
Wall Temp. (K)	860-1197		757-867		
Flooding Rate (cm/s)	0-14		1.5-15	4-30	1-6
G (kg/m^2-s)	45.5-98.4	470-2160			
Subcooling (°C)		2-30	11-78		
Void fraction	0-1.0				
Comments		6-rod (Froga) and 36-rod (Frigg) test sections	33 rod test section	68 rod test section ballooned and unballooned tests	7x7 rod test section

	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		Flecht-Seaset/W	THTF/ORNL	G2/W	BCL
P (MPa)	0.333-0.396	0.14-0.41			
q (W/cm^2)					
Wall Temp. (K)	860-1197				
Flooding Rate (cm/s)	0-14	1.5-15			
G (kg/m^2-s)	45.5-98.4				
Subcooling (°C)		3-78			
Void fraction	0-1.0				
Comments		17x17 rod bundle	Ref. 5, Note 4	No info sheet	No info sheet

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).

2. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).

3. C. Unal and R. A. Nelson, "A Phenomenological Model of the Thermal-Hydraulics of Convective Boiling During the Quenching of Hot Rod Bundles Part II: Assessment of the Model with Steady-State and Transient Post-CHF Data," *Nuclear Engineering and Design* **136**, 298-318 (1992).

4. C. Unal, E. Haytcher, and R. A. Nelson, "Thermal-Hydraulics of Convective Boiling During the Quenching of Hot Rod Bundles Part III: Model Assessment Using Winfrith Steady-State Post-CHF Void Fraction and Heat Transfer Measurements and Berkeley Transient Reflood Test Data," *Nuclear Engineering and Design* **140**, 211-227 (1993).

5. D. G. Morris, G. L. Yoder, and C. B. Mullins, "An Experimental Study of Rod Bundle Dispersed-Flow Film Boiling with High-Pressure Water," *Nuclear Technology*, **69**, 82-93 (April 1985).

Notes

1. The CCTF-Run 14 and the Lehigh rod-bundle reflood test 02/24/85-20 were used in Ref. 3 to assess the interfacial drag during reflood.
2. A series of steady-state Winfrith heated tube tests were used in Ref. 4 to assess the axial void-fraction profile.
   - The core interfacial drag has also been indirectly assessed with Flecht-Seaset Tests 31504 and 33436, CCTF Core-II Run 54, and STCF Run 719.
   - Reference 5 is just one of many ORNL documents that must be examined to determine the appropriate tests to be used.

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Downcomer Interfacial Drag				
	Plant Range	Test Facility (Ref. 2)			
Plant Parameter		UPTF	CCTF JAERI	1/15 +1/30 Vessel/Creare	BCL
P (MPa)	0.333-5.06	1-2	≤0.6	0.1-0.45	
Rod Temp. (K)	765-1140				
G (kg/m^2-s)	-357 - 243				
Subcooling (°C)				0-110	
Void fraction	0-1.0				
Comments		1:1 German (KWU) PWR core simultor	3-D 32 fuel-rod bundle core	1/15 and 1/30 vessel downcomer flow tests	No info sheet

Nomenclature
P, pressure
G, mass flux

References

1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).

2. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).

TABLE F-14
CANDIDATE COMMON EXPERIMENTAL FACILITIES: LEVEL

Plant	Westinghouse AP600				
Transient	Small-, Intermediate, and Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Any phase of transient where there is two-phase flow in the vessel or vertical sections of the RCS				
PIRT Parameter	Liquid Level in Pipes				
	Plant Range (Note 1)	OECD Test Facility (Ref. 2)			
Plant Parameter		Vertical Canon	Tapioca	UPTF	Battelle BWR
P (MPa)	0.333-0.396	13	15	1-2	54, 70, 88 bar
q (W/cm^2)		1-3		1.5-4.2	
Wall Temp. (K)	860-1197	300-600		355-600	
Flooding Rate (cm/s)	0-14				
G (kg/m^2-s)	45.5-98.4	20-150		2-190	
Subcooling (°C)		20-50		60	
Void fraction	0-1.0				
Temperature		500-590K	280°C		256-302°C
OECD Facility ID		3.4	3.6	4.1	4.4
Facility Description		Vertical Blowdown, 4.5 m, 0.1 m diam. tube, break at top, 3 to 15 mm diam. Used for TRAC-PF1/MOD1 critical flow assessment	Blowdown facility, 0.324 m ID, 2.6 m length, 0.2144 m^3 volume, break locations at side, top, bottom, middle; break size 2, 5, 10, 20, 35 mm ID	1:1 German (KWU) PWR core simulator	1:80 volume scale of BWR Vessel, 0.6 m ID, to evaluate steam line and feedwater LOCAs, electrical heater, 600kW, 42 heater tube bundle. Discharge nozzle at 6.4, 10.0, 11.2 m height, break diam.: 33, 45, 64, and 76 mm

Plant	Westinghouse AP600				
Transient	Small-, Intermediate, and Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Any phase of transient where there is two-phase flow in the vessel or vertical sections of the RCS				
PIRT Parameter	Liquid Level in Pipes				
	Plant Range (Note 1)	OECD Test Facility (Ref. 2)			Ref. 4
Plant Parameter		Marviken	Lotus	Single Tube Level Swell	Shoukri Subcooled Boiling
P (MPa)	0.333-0.396	1-5.2	1.7-3.77 bar	0.1	0.15 - 0.17
q (W/cm^2)					
Wall Temp. (K)	860-1197				
Flooding Rate (cm/s)	0-14				
G (kg/m^2-s)	45.5-98.4		4-290 air 5-1000 water		
Subcooling (°C)		0 - 50			
Void fraction	0-1.0	0 - 1.0			
OECD Facility ID		8.2	10.13	10.14	
Facility Description		Large scale critical flow facility. Test T-11 is a level swell experiment with the break located at the top of the vessel (See Ref. 3)	Vertical air-water annular flow tube section, 31.8 mm ID, 20 m length, upflow	Vertical electrically heated tube, steady state level swell tests, 3 m length, 12.5 mm ID, stainless steel	Vertical stainless steel tube, 12.7 mm ID and 30.6 cm length. (See Ref. 4)

Plant	Westinghouse AP600				
Transient	Small-, Intermediate, and Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Any phase of transient where there is two-phase flow in the vessel or vertical sections of the RCS				
PIRT Parameter	Liquid Level in Core				
	Plant Range (Note 1)	OECD Test Facility (Ref. 2)			
Plant Parameter		Pericles Cylindrical	TPTF Jaeri ROSA IV Program	ECN Reflood and Boildown	FRIGG
P (MPa)	0.333-0.396	0.2-0.4 1.0 - 6.0	3-12 MPa 0.5-12 MPa	2-6 bar	5 MPa
q (W/cm^2)		1.5-4.2 1-2	3-18	1.7-5	
Wall Temp. (K)	860-1197	355-600°C 600°C	≤ 920 K		
Flooding Rate (cm/s)	0-14		≤ 1.2 m/s	1.4-9	
G (kg/m^2-s)	45.5-98.4	1-19 g/cm2 s 1.7 - 3 g/cm2 s	13-98 kg/m2s		
Subcooling (°C)		60°C < 10°C	≤ 20°C	20-80°C	
Void fraction	0-1.0				
OECD Facility ID		3.9	6.1	7.1/7.2	8.3
Facility Description		Cylindrical 368-rod core, 17 x 17 array, for low pressure and high pressure reflooding, also boil-off steady-state and transient tests, 0.95 cm OD, 3.656 m length	Horizontal two-phase flow and core heat transfer facility (25-, 24-, and 39-rod core geometries); Low flow heat transfer tests, boil-off tests, and reflood tests	36 rod test section, 10.7 mm diam, 3 m length, boiloff and reflood tests	6-rod (FROGA) and 36-rod (FRIGG) test sections, Marviken BHWR fuel element design. Extensive number of tests

Plant	Westinghouse AP600				
Transient	Small-, Intermediate, and Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Any phase of transient where there is two-phase flow in the vessel or vertical sections of the RCS				
PIRT Parameter	Liquid Level in Core				
	Plant Range (Note 1)	OECD Test Facility (Ref. 2)			
Plant Parameter		Neptun-1 Boiloff	Achilles Reflood Loop	Thetis	GE Level Swell
P (MPa)	0.333-0.396	1-5 bar 1-4.1 bar			
q (W/cm^2)		24.6 - 75.1 kW 2.45-4.19 kW/rod			
Wall Temp. (K)	860-1197	757, 867°C			
Flooding Rate (cm/s)	0-14	1.5 - 15			
G (kg/m^2-s)	45.5-98.4				
Subcooling (°C)		0-39°C 11-78°C			
Void fraction	0-1.0				
OECD Facility ID		9.1	10.1	10.2	11.44
Facility Description		33 rod test section, emergency core cooling heat transfer tests in PWR core geometry, boil-off and reflood tests	68 rod test section ballooned and unballooned tests	7x7 test section, PWR core heat transfer during LOCA, reflood tests with clad ballooning blockage, single phase heat transfer tests, level swell tests	Blowdown facility, 14 ft pressure vessel with different size orifice plates to control depressurization

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Liquid Level in Downcomer				
	Plant Range (Note 2)	OECD Test Facility (Ref. 2)			
Plant Parameter		UPTF	CCTF Jaeri	1/15 +1/30 Vessel/Creare	BCL
P (MPa)	0.333-5.06	1-2	≤0.6	0.1-0.45	
q (W/cm^2)					
Wall Temp. (K)	765-1140				
G (kg/m^2-s)	-357 - 243				
Subcooling (°C)				0-110	
Void fraction	0-1.0				
OECD Facility ID		4.1	6.15	11.13	
Facility Description		1:1 German (KWU) PWR core simultor	Full height 3-D 32 fuel-rod bundle core. Each bundle has 57 heater rods, 10.7-mm OD, 3.66-m heated length, 7 nonheated rods 13.8-mm OD, 8x8 square lattice with 14.3-mm pitch, 4 loops with 2 steam generators, 4 pump simulators, ECCS injection in cold legs only	1/15 and 1/30 vessel downcomer flow tests	No info sheet

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. B. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
2. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
3. M. A. Grolmes, A. Sharon, C. S. Kim, and R. E. Paul, "Level Swell Analysis of the Marviken Test 11," *Nuclear Science and Engineering*, **93** (3), 229-239 (1986).
4. M. Shoukri, B. Donevski, R. L. Judd, and G. R. Dimmick, "Experiments on Subcooled Flow Boiling and Condensation in Annular Channels," in Proceedings of the International Seminar on Phase Interface Phenomena in Multiphase Systems (Hemisphere Publishing, 1991), pp. 413–422.

Notes
1. Plant range shown is for reflood phase of AP600 LBLOCA in core.
2. Plant range shown is for refill phase of AP600 LBLOCA in downcomer.

TABLE F-15
CANDIDATE COMMON EXPERIMENTAL FACILITIES: NONCONDENSIBLE EFFECTS

Plant	Westinghouse 4-Loop PWR
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)
Transient Phase	Reflood
PIRT Parameter	Noncondensable Effects

	Plant Range	Test Facility (Note 1)			
Plant Parameter		MIT Steam Condensation	MIT Single-Tube Experiment	UCB Steam Condensation	
P (MPa)	0.1	1.5–4.5			
Re_i			5000–11400		
F (%)		Air:35–85	Air:10–35 Helium: 2–10		
Comments		Ref. 1	Refs. 2-3	Refs. 4-5	

Nomenclature
P, pressure
Re_i, inlet mixture Reynolds number
F, noncondensable fraction

References
1. Dehbi, M. W. Golay, and M. S. Kazimi, "The Effects of Non-Condensable Gases on the Steam Condensation under Turbulent Natural Convection Conditions," Massachusetts Institute of Technology document MIT-ANP-TR-004 (June 1990).
2. M. Siddique, "The Effects of Noncondensable Gases on Steam Condensation under Forced Convection Conditions," Ph.D. Thesis, Massachusetts Institute of Technology (January 1992).
3. M. Siddique, M. W. Golay, and M. S. Kazimi, "Local Heat Transfer Coefficients for Forced-Convection Condensation of Steam in a Vertical Tube in the Presence of a Noncondensable Gas," Nuclear Technology, Vol. 102, pp. 386-402 (1993).
4. M. Vierow and V. E. Schrock, "Condensation Heat Transfer in Natural Circulation with Noncondensable Gas," Department of Nuclear Engineering, University of California at Berkeley document UCB-NE 4170 (May 1990).
5. S. Z. Kuhn, V. E.Schrock, and P. F. Peterson, "Final Report on U. C. Berkeley Single Tube Condensation Studies," University of California Berkeley document UCB-NE-4201, Rev. 2 (1994).

Notes

1. The MIT steam condensation, MIT single-tube experiment, andUCB steam condensation experiments were previously used for assessing the noncondensable model in RELAP5/MOD3 (Y. A. Hassin and S. Banerjee, "Implementation of a Non-Condensible Model in RELAP5/MOD3," Nuclear Engineering and Design, Vol. 162, pp. 281-300 (1996).

TABLE F-16
CANDIDATE COMMON EXPERIMENTAL FACILITIES: ASYMMETRIES

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, Refill				
PIRT Parameter	Asymmetric Flow				
	Plant Range	Test Facility (note 2)			
Plant Parameter		LOFT L2-5			
P (MPa)	0.1 - 15.4	0.1 - 14.95			
q (W/cm^2)	0.1 - 46	0.72 – 36.0 MW			
G (kg/m^2-s)	1 - 2455	192.4 kg/s			
Comments		Refs. 1-2			

Nomenclature
P, pressure
q, heat flux
G, mass flux

References
1. C. L. Nalezny, "Summary of Nuclear Regulatory Commission's LOFT Program Experiments," Idaho National Engineering Laboratory document EGG-2248, also NUREG/CR-3214 (July 1983).
2. P. D. Bayless and J. M. Divine, "Experiment Data Report for LOFT Large Break Loss-of-Coolant Experiment L2-5," Idaho National Engineering Laboratory document EGG-2210 also NUREG/CR-2826 (August 1982).

TABLE F-17
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLOW–COUNTERCURRENT

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Countercurrent Flow–Downcomer				
	Plant Range	Test Facility			
Plant Parameter		Dartmouth	Bankoff	BCL	Creare
P (MPa)	0.1			0.1–0.4	
$T_{ECC\ inj}$ (K)				277-366	288–366
G_v (kg/m^2-s)				8.3 lb/s	0–5.5 lb/s
G_l (kg/m^2-s)				575 gpm	0–1500 gpm
Comments		Ref. 1:	Refs.5-6	Ref. 2: Note 1	Ref. 3: Note 2

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Countercurrent Flow–Downcomer				
	Plant Range	Test Facility			
Plant Parameter		UPTF	UPTF		
P (MPa)	0.1				
$T_{ECC\ inj}$ (K)		50 subcooled			
G_v (kg/m^2-s)		100 kg/s			
G_l (kg/m^2-s)		735–1465 kg/s			
Comments		Ref. 4: Test 6 Downcomer	Ref. 4: Test 10C Upper tie plate		

Nomenclature

P, pressure
G, mass flux

References

1. G. B. Wallis, P. C. DeSicyes, P. J. Roselli and J. Lacombe, "Countercurrent Annular Flow Regimes for Steam and Subcooled Water in a Vertical Tube," Electric Power Research Institute document NP-1336 (January 1980).
2. R. P. Collier, L. J. Flanigan, and J. A.Dworak, "Data Report on ECC Bypass Tests for TRAC Assessment," Battelle Columbus Laboratories document (July 1980).
3. C. J. Crowley, P. H. Rothe, and R. G. Sam, "1/5 Scale Countercurrent Flow Data Presenation and Discussion," Creare, Inc. document NUREG/CR-2106 (November 1981).
4. "Test No. 6 Downcomer Countercurrent Flow Test," 2D/3D Program Upper Plenum test Facility Experimental Data Report, Siemens/KWU document U9 316/89/14 (1989).
5. S. G. Bankoff, R. S. Tankin, M. C. Yuen, and C.L. Hsieh, "Countercurrent Flow of Air/Water and Steam/Water through a Horizontal Performated Plate," International Journal of Heat and Mass Transfer, Vol. 24, No. 8, pp. 1381-1395 (1981).
6. I. Dilber and S. G. Bankoff, "Countercurrent Flow Limits for Steam and Cold Water through a Horizontal Perforated Plate with Vertical Jet Injection," International Journal of Heat and Mass Transfer, Vol. 28, No. 12, pp. 2382-2385 (1985).

Notes

1. BCL operated a $1/15^{th}$-scale model at 60 psi and a $2/15^{th}$-scale facility at low pressures.
2. Creare operated several facilities in scales ranging from 1/30 to 1/5.

TABLE F-18
CANDIDATE COMMON EXPERIMENTAL FACILITIES: FLOW–MULTIDIMENSIONAL

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	Multidimensional flow (note 1)				
	Plant Range	Test Facility (note 2)			
Plant Parameter		OST (note 3)	Rectangular clarifier (note 4)	Slab Core Test Facility (note 5)	Pericles (note 6)
P (MPa)	5.1 - 15.4	0.1–5.0	0.1	0.2	0.2–0.55
q (W/cm^2)	7 - 46		Isothermal		1.35–5.0
G (kg/m^2-s)	2455 - 0		4 - 11 (estimated)		
Comments		Problem has been calculated as reported in Ref. 3.	Data reported in Ref. 4; analysis using data reported in Ref. 5.		

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	Multidimensional flow (note 1)				
	Plant Range	Test Facility (note 2)			
Plant Parameter		Cylindrical Core Test Facility (note 7)			
P (MPa)	5.1 - 15.4	0.2			
q (W/cm^2)	7 - 46				
G (kg/m^2-s)	2455 - 0				
Comments					

Nomenclature
P, pressure
q, heat flux
G, mass flux

References
1. "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation," Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency document NEA/CSNI/R(93)14/Part 1/Rev (September 1993).
2. E. Boyack, "TRAC-PF1/MOD2 Adequacy Assessment Closure and Special Models," Los Alamos National Laboratory document LA-UR-97-232 (February 21, 1997).
3. H. Stadtke, G. Franchello, and B. Worth, "Numerical Simulation of Multi-Dimensional Two-Phase Flow Based upon Flux Vector Splitting," Proceedings of the 7th International Meeting on Nuclear Reactor Thermal-Hydraulics NURETH-7, NUREG/CP-0142, Vol. 1, pp. 648-672 (September 10-15, 1995).
4. Imam, "Numerical Modeling of Rectangular Clarifiers," Ph.D. Thesis, University of Windsor (1981). Will request data after it is determined that this is a valid element of the validation test matrix.
5. H. Gerges and J. McCorquodale, "Modelling of Flow in Rectangular Sedimentation Tanks by an Explicit Third-Order Upwinding Technique," International Jouranl for Numerical Methods in Fluids, Vol. 24, pp. 537-561 (1997).
6. B. E. Boyack, P. R. Shire, and S. C. Harmony, "TRAC-PF1/MOD1 Code Assessment Summary Report For SCTF Core-III," Los Alamos National Laboratory restricted distribution document LA-CP-90-71 (February 8, 1990).
7. H. J. Stumpf, "CCTF Run 76 TRAC-PF1/MOD1 Analysis," Los Alamos National Laboratory document LA-2D/3D-TN-86-6 (April 1986).
8. H. J. Stumpf, "CCTF Run 77 TRAC-PF1/MOD1 Analysis," Los Alamos National Laboratory document LA-2D/3D-TN-86-5 (May 1986).

Notes
1. For the blowdown phase, multidimensional phenomena in the core was highly ranked. This phenomenon appears in the OECD/CSNI test matrix (Ref. 1) as Category 10, Global Multidimensional Fluid Temperature, Void and Flow Distribution with the following plant components identified: upper plenum, core, downcomer, and steam-generator secondary side.
2. We have attempted to list the experimental facilities moving from most fundamental separate effect tests to integral tests.
3. Should be considered as "Other Standard Test" or OST in the "concept category," as described in Ref. 2. Problem models the blowdown of a partially filled pressure vessel through a horizontal discharge line.
4. Parameters do not correspond to AP600 blowdown parameters. Should consider this test as basic proof of principle, i.e., used to evaluate the degree to which basic two-dimensional phenomena are calculated in an isothermal condition.
5. Use SCTF Runs 718, 719, 720 which characterize multidimensional core flows with the multidimensionality induced by the radial core power profile. Run 718 has a uniform radial core power profile; Run 719 has 1.36, 1.20, 1.10, 1.00, 0.91, 0.86, 0.81, and 0.76 peak-to-average power ratios across the 8 test assemblies; Run 720 has 0.81, 0.91, 1.1, 1.36, 1.20, 1.00, 0.86, and 0.76 across the 8 test assemblies. All three tests have previously been used for TRAC assessment (See Ref. 6). These tests most directly apply to the refill and reflood phases. SCTF is OECD/CSNI SET facility 6.14 (Ref. 1).

6. Multidimensionality induced by the radial core power profile with the radial peaking factor between 1 and 1.85. These tests most directly apply to reflooding and boiloff. Pericles is OECD/CSNI SET facility 3.8 (Ref. 1).
7. Use CCTF Runs C2-16/76, the base case for the CCTF upper plenum injection tests or C2-18/78, the UPI best estimate case. Both tests have previously been used for TRAC assessment (See Refs. 7-8).

TABLE F-19
CANDIDATE COMMON EXPERIMENTAL FACILITIES: OSCILLATIONS

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill				
PIRT Parameter	Oscillations				
	Plant Range	Test Facility			
Plant Parameter		U-Tube Manometer (Ref. 1)	Frigg Dynamic tests (Refs. 2-4)	Flecht-Seaset (Refs. 5-7)	Slab Core Test Facility (Refs. 8-9)
Plant Parameter					
P (MPa)	0.2	0.1			
q (W/cm^2)					
G (kg/m^2-s)	0.0 - 4150				
Comments	Check core and downcomer flows during refill and enter in plant parameter section	Single phase liquid – analytical solution exists	Tests 662101, 662105, 662107, 662113, 462053, and 462101, See Note 1.	Test 33437 - See Note 2.	Test S2-08 (Run 613). See Note 3

Nomenclature
P, pressure
q, heat flux
G, mass flux

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10.

Notes
1. Rohatgi, et al., used the FRIGG data of Refs. 2-3 to assess the RAMONA-3B code. The oscillations are externally induced by core power variations. The geometry includes a downcomer and core connected by a horizontal pipe. FRIGG is closer to a SET than IET and it appears that the FRIGG data is a good candidate for assessment of the code's capability to predict oscillatory phenomena measured in a facility with two-phase flow that is simpler than IET facilities.
2. Current TRAC input deck exists and was used in the assessment reported in Ref. 7.
3. Test has previously been assessed for TRAC-PF1 as reported in Ref. 9.

TABLE F-20
CANDIDATE COMMON EXPERIMENTAL FACILITIES: POWER–DECAY HEAT

Plant	Westinghouse AP600				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, Reflood, Long-Term Cooling				
PIRT Parameter	Decay Heat				
	Plant Range	Test Standard			
Plant Parameter		ANS-5.1-1994 (Ref. 1)	AESJ (Ref. 2-3)	ISO (Ref. 4)	
T (s)	$0.0 - 10^{10}$	$0.0 - 10^{10}$	$0.0 - 10^{10}$	$0.0 - 10^{10}$	
Comments		American National Standard	Proposed Japanese Standard	Proposed International Standards Organization Standard	

Nomenclature
T, Time

References

12. "American National Standard: For Decay heat Power in Light Water Reactors," American Nuclear Society standard ANSI/ANS-5.1-1979(R1985) (1985).
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TABLE F-21
CANDIDATE COMMON EXPERIMENTAL FACILITIES: PUMP PERFORMANCE, INCLUDING DEGRADATION

Plant	Westinghouse 4-Loop PWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	Pump Degradation				
	Plant Range	Test Facility			
		Semiscale	EPRI	KWU	
P (MPa)	5.1–15.4				
Head (m) (Note 1)			~95	92	
Specific Speed		18	82	130	
Comments		Ref. 1: Pump is of the radial-flow type.	Ref. 2: Pump is of the mixed-flow type.	Ref. 3: Pump is of the axial-flow type used in KWU reactors.	

Nomenclature
P, pressure
G, mass flux

References

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2. "Pump Two-Phase Performance Program," Electric Power Research Institute document EPRI NP-1556, Volumes 1-8 (September 1980).
3. W. Kastner and G. J. Seeberger, "Pump Behavior and Its Impact on a Loss-of-Coolant Accident in a Pressurized Water Reactor," Nuclear Technology, Vol. 60, pp. 268-277 (February 1983).

Head
1. Steady-state design point single-phase head.

TABLE F-22
CANDIDATE COMMON EXPERIMENTAL FACILITIES: REACTIVITY–VOID

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No tests identified.

APPENDIX G

EXPANDED LISTING OF TRAC-M INPUT DECKS FOR COMMON AND PWR-SPECIFIC SETS, IETS AND PLANTS

Table G-1 lists the available common and PWR-specific TRAC-M SETs input decks. For each facility input deck, a brief description of the facility, test type, test number, and report reference in addition to the latest code version on which the input deck was exercised are provided. Table G-2 lists the available common and PWR TRAC-M IET input decks in the same format. Table G-3 lists the available PWR TRAC-M plant input decks in the same format.

TABLE G-1
TRAC-M INPUT DECKS FOR SEPARATE EFFECT TESTS

Facility	Type of Test	Test ID	References	Input Deck	Comments
Akimoto	Condensation		G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
Bankoff	CCFL		G-1	TRAC-M/F77, Version 5.5	Air-water and steam-water. Equivalent to MOD2 input deck
BCL	Downcomer counter-current flow	26204, 26502, 29111	G-2	PD2	Deck stored in LANL TRAC Input Deck Archive (TIDA)
Bennett	Heated-tube CHF	5336, 5431, and 5442	G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
Berkeley	Reflood heat transfer		1991 Dev. Assessment	Early MOD2	Deck stored in LANL TIDA
CISE	Critical Flow	4	G-3	PD2	Deck stored in LANL TIDA
CREARE	Downcomer counter-current flow		G-3	Listings in Appendix F	No deck found
Dartmouth	Air-water counter-current flow	2-in pipe and 6-in. pipe	G-3	MOD1	Deck stored in LANL TIDA
Edwards	Critical Flow		G-3	MOD2	Deck stored in LANL TIDA
FLECHT	Reflood heat transfer	4831 17201	G-3	PD2	No deck found
FLECHT-SEASET	Reflood heat transfer	31504 33436	G-1	TRAC-M/F77, Version 5.5	Forced and gravity reflood tests. Equivalent to MOD2 input deck

TABLE G-1 (cont)
TRAC-M INPUT DECKS FOR SEPARATE EFFECT TESTS

Facility	Type of Test	Test ID	References	Input Deck	Comments
Lehigh	Reflood heat transfer		G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
Marviken	Critical Flow	4, 13, 20, 22, and 24	G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
THETIS	Boildown level-swell test		1991 Dev. Assessment	Current MOD2	Deck stored in LANL TIDA
THTF	Rod-bundle blowdown heat transfer	177	G-3	PD2	Deck stored in LANL TIDA
Winfrith	Heated tube CHF		1991 Dev. Assessment	MOD2	Deck stored in LANL TIDA
Moby-Dick	Critical flow	403, 408, 455, 79, 172	G-4	MOD1	Typical input data deck in report
Super-Moby-Dick	Critical flow	1-15	G-4	MOD1	Typical input deck in report
Cannon	Blowdown	D, L, I	G-4	MOD1	Typical input deck in report
Super-Canon	Blowdown	P, X, Q	G-4	MOD1	Typical input deck in report
Vertical-Canon	Blowdown	9, 22, 24	G-4	MOD1	Typical input deck in report
Omega-Tube	Blowdown	3, 6, 8, 9, 29, 30	G-4	MOD1	Typical input deck in report
Omega-Bundle	Blowdown	2, 3, 9, 11, 13, 18, 19	G-4	MOD1	Typical input deck in report
Strathclyde	Refill phase LB LOCA	B/B2; C/C2; D/D2	G-5	MOD1	1/10th scale model of a PWR downcomer
Achilles	Forced/gravity reflood	23, 28	G-6	MOD1	Typical input deck in report
UPTF	LOCA loop flow pattern LOCA downcomer flow pattern	8b 6	G-1	TRAC-M/F77, Version 5.5	Cold-leg flow and downcomer tests. Equivalent to MOD2 input deck Deck stored in LANL TIDA
CCTF	LOCA refill and reflood	14	G-1	TRAC-M/F77, Version 5.5	Direct ECC water injection into lower plenum. Equivalent to MOD2 input deck

TABLE G-2
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
PKL	Natural circulation	ID1-4	G-7	MOD1	Deck stored in LANL TIDA
	Natural circulation	ID1-9	G-7	MOD1	Deck stored in LANL TIDA
	Reflux cooling	ID1-14	G-7	MOD1	Deck stored in LANL TIDA
	Gravity reflood	K9	G-2	MOD1	Deck stored in LANL TIDA
	Gravity reflood	K5.4A	G-2	MOD1	Deck stored in LANL TIDA
Semiscale Mod-1	200% cold-leg break without ECCS	S-02-8	G-3	MOD1	Deck stored in LANL TIDA
	200% cold-leg break with ECCS	S-06-3	G-3	MOD2	Deck stored in LANL TIDA
Semiscale Mod-3	2.5% cold-leg break, early pump trip	S-SB-P1	G-7	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break, delayed pump trip	S-SB-P2	G-7	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break, late pump trip	S-SB-P7	G-7	MOD1	Deck stored in LANL TIDA
	10% cold-leg break with delayed ECCS and secondary blowdown	S-07-10D	G-7	MOD1	Deck stored in LANL TIDA
	2.5% hot-leg break, pumps off	S-SB-P3	G-2	MOD1	Deck stored in LANL TIDA
	2.5% hot-leg break, pumps on	S-SB-P4	G-2	MOD1	Deck stored in LANL TIDA
	Natural circulation	S-NC-2B	G-8	MOD1	Deck stored in LANL TIDA
	Natural circulation	S-NC-5	G-8	MOD1	Deck stored in LANL TIDA
	Natural circulation	S-NC-6	G-8, G-9	MOD1	Input listing in Reference G-10 Deck stored in LANL TIDA
	Natural circulation	S-NC-7C		MOD1	Deck stored in LANL TIDA

TABLE G-2 (cont)
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
Semiscale Mod-2a	10% cold-leg break with upper-head injection (UHI)	S-UT-2	G-8	MOD1	Deck stored in LANL TIDA
	5% cold-leg break without UHI	S-UT-6	G-8	MOD1	Input listing in Reference G-10 Deck stored in LANL TIDA
	5% cold-leg break with UHI	S-UT-7	G-8	MOD1	Input listing in Reference G-10 Deck stored in LANL TIDA
LOFT	Isothermal DEGB blowdown	L1-4	G-3	MOD1	Deck stored in LANL TIDA
	50% power, DEGB cold-leg break	L2-2	G-7	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break in broken cold leg	L3-1	G-2	MOD1	Deck stored in LANL TIDA
	15% cold-leg break in broken cold leg	L3-7	G-7	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break in intact cold leg, early pump trip	L3-5	G-2	MOD1	Deck stored in LANL TIDA
	2.5% cold-leg break in intact cold leg, late pump trip	L3-6	G-2	MOD1	Deck stored in LANL TIDA
	20.7% cold-leg break in broken cold leg	L5-1	G-2	MOD1	Deck stored in LANL TIDA
	20.7% cold-leg break in broken cold leg with delayed ECCS	L8-2	G-2	MOD1	Deck stored in LANL TIDA
	200% cold-leg break, pumps on	L2-3	G-2	MOD1	Deck stored in LANL TIDA

TABLE G-2 (cont)
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
	200% cold-leg break, early pump trip	L2-5	G-2	MOD1	Deck stored in LANL TIDA
	200% cold-leg break, higher power	L2-6 (LP-02-6)	G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
	Loss of feedwater transient	L9-1/L3-3	G-8	MOD1	Deck stored in LANL TIDA
	Cooldown transient	L6-7/L9-2	G-8	MOD1	Deck stored in LANL TIDA
	Loss of steam load	L6-1	G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
	Pump trip	L6-2	G-9	MOD1	Input listing in Reference
	Excessive-load increase	L6-3	G-9	MOD1	Input listing in Reference
Crystal River Transient	Anticipated transients—non-nuclear instrumentation failure		G-8	MOD1	Deck stored in LANL TIDA
CCTF	Core-I reflood base case	14		MOD1	Deck stored in LANL TIDA
	Core-II reflood low power	54	G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
	Core-II upper plenum injection	57	G-10	MOD1	Deck stored in LANL TIDA
	Core-II upper plenum injection	59	G-10	MOD1	Deck stored in LANL TIDA
	Core-II upper plenum injection	72	G-10	MOD1	Deck stored in LANL TIDA
	Refill/reflood with asymmetric injection	76	G-10	MOD1	Deck stored in LANL TIDA
	Refill/reflood with UPI	78	G-10	MOD1	Deck stored in LANL TIDA

TABLE G-2 (cont)
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
	Downcomer injection/vent valves closed	58	G-11	MOD1	Deck stored in LANL TIDA
	Cold- and hot-leg injection	79	G-12	MOD1	Deck stored in LANL TIDA
	Best-estimate	71	G-13	MOD1	Deck stored in LANL TIDA
MIST	Delayed HPI/PORV feed-and-bleed cooling	330302	G-14	MOD1	Deck stored in LANL TIDA
	50-cm^2 SBLOCA	320201	G-15	MOD1	Deck stored in LANL TIDA
	10-cm^2 SBLOCA	3109AA	G-16	MOD1	Deck stored in LANL TIDA
	STGR	3404AA	G-17	MOD1	Deck stored in LANL TIDA
ROSA-IV LSTF	Single- and two-phase natural circulation	ST-NC-02	G-18	MOD1	Deck stored in LANL TIDA
SCTF	refill/reflood	S2-SH2 (Run 605)	G-19	MOD1	Deck stored in LANL TIDA
		OS1	G-20	MOD1	Deck stored in LANL TIDA
		S3-9 (Run 713)	G-21	MOD1	Deck stored in LANL TIDA
		(Run 704)	G-22	MOD1	Deck stored in LANL TIDA
		(Run 714)	G-23	MOD1	Deck stored in LANL TIDA
		S2-03 (Run 608)	G-24	MOD1	Deck stored in LANL TIDA
		S2-08 (Run 613)	G-25	MOD1	Deck stored in LANL TIDA
		S2-09 (Run 614)	G-26	MOD1	Deck stored in LANL TIDA

TABLE G-2 (cont)
TRAC-M INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
		S2-SH1 (Run 604)	G-27	MOD1	Deck stored in LANL TIDA
		S2-12 (Run 617)	G-28	MOD1	Deck stored in LANL TIDA
		Run 605	G-29	MOD1	Deck stored in LANL TIDA
		S2-06 (Run 611)	G-30	MOD1	Deck stored in LANL TIDA
		S3-15 (Run 719)	G-1	TRAC-M/F77, Version 5.5	Equivalent to MOD2 input deck
UPTF		17	G-31	MOD2	Deck stored in LANL TIDA
		21	G-32	MOD2	Deck stored in LANL TIDA
		27	G-33	MOD2	Deck stored in LANL TIDA
LOBI	1% cold-leg SB LOCA	A2-81	G-34	MOD1	ISP-18 exercise Deck stored in LANL TIDA
	3% cold-leg SB LOCA	BL-02	G-35	MOD1	

TABLE G-3
TRAC-M INPUT DECKS FOR NUCLEAR POWER PLANTS

Vendor Plant	Transient or Accident	References	Decks	Comments
Westinghouse				
AP600	LB LOCA	G-36	TRAC-M/F77	Deck stored in LANL TIDA
CSAU Plant	LB LOCA for code scaling, applicability, uncertainty (CSAU)	G-37	MOD1	Deck stored in LANL TIDA
R. E. Ginna	Steam generator tube rupture	G-38	MOD2	Deck stored in LANL TIDA
H. B. Robinson	SB LOCA, steam generator tube rupture	G-39	MOD2	Deck stored in LANL TIDA
South Texas Project	SB LOCA	G-40	MOD2	Deck stored in LANL TIDA
USPWR 15x15 fuel	LB LOCA	G-41	MOD1	Deck stored in LANL TIDA
USPWR 17x17 fuel	LB LOCA	G-42	MOD1	Deck stored in LANL TIDA
Zion-1	Main feed-line break/loss of feedwater	G-43	MOD2	Deck stored in LANL TIDA
CE				
Arkansas Nuclear One-2	Turbine trip transient	None	MOD1	Deck stored in LANL TIDA. Converted from a RETRAN deck.
Calvert Cliffs-1	Loss of offsite power	G-44	MOD2	Deck stored in LANL TIDA
B&W				
Bellefonte	Steady state only	G-45	MOD2	Deck stored in LANL TIDA
Crystal River	Plant transient of February 26, 1980	G-8	MOD1	Deck stored in LANL TIDA
Davis-Besse	Loss of feedwater	G-46	MOD2	Deck stored in LANL TIDA
Oconee-1	SB LOCA	G-47	MOD2	Deck stored in LANL TIDA
Three Mile Island-2	SB LOCA (TMI-1 accident)	G-48	MOD2	Deck stored in LANL TIDA

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G-40. J. F. Lime, "Deck Conversion of the South Texas Project Plant Model from TRAC-PF1/MOD1 to TRAC-PF1/MOD2," Los Alamos National Laboratory document LA-CP-96-0026 (January 1996).

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G-43. S. B. Rodriguez, "Conversion and Validation of a Zion-1 Input Deck to TRAC-PF1/MOD2, Version 5.4.05," Los Alamos National Laboratory document LA-CP-95-214 (1995).

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G-48. S. B. Rodriguez, "Conversion and Validation of a Three Mile Island Input Deck to TRAC-PF1/MOD2, Version 5.4.05," Los Alamos National Laboratory document LA-CP-95-183 (1995).

APPENDIX H
RECOMMENDED TESTS FOR THE BWR LBLOCA VALIDATION TEST MATRIX

In this appendix, we present the experimental facilities recommended for the TRAC-M BWR LBLOCA validation test matrix. For each PIRT local-level (LL), component-level (CL), and system-level (SL) LBLOCA phenomenon identified in Section 4 (Table 4-5), but not addressed in the recommended tests for the TRAC-M common LBLOCA validation test matrix (Appendix F), we provide a table identifying recommended tests.

Additional tables are provided for several phenomena covered in Appendix F for which additional BWR-specific tests are recommended.

Each table lists the experimental facilities that have produced data that are recommended for inclusion in the validation test matrix.

Local-level PIRT phenomena are covered in Tables H-1 through H-12. Component- and system-level PIRT phenomena are covered in Tables H-13 through H-26.

TABLE H-1
PROPOSED BWR EXPERIMENTAL FACILITIES: BOILING–FILM

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood				
PIRT Parameter	**Boiling–Film**				
	Plant Range	Test Facility (See Also Table F-1 for additional tests)			
Plant Parameter		THTF Film Boiling Tests 3.03.6AR 3.06.6B & 3.08.6C			
P (MPa)	0.3 – 5.0	5.17 - 12.4			
Heat Flux (kw/m2)		160 – 1100			
Equil.Quality (%)	0.1-90	0.15 – 100			
Clad Temps (K)	500 – 1400	600 – 1000			
Mass Flux kg/s-m2		129- 1090			
Comments					

References

1. D. G. Morris, et. al., "An Analysis of Transient Film Boiling of High Pressure Water in a Rod Bundle," NUREG/CR-2469, ORNL, March 1982.

TABLE H-2
PROPOSED BWR EXPERIMENTAL FACILITIES: BOILING–NUCLEATE

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Long-term cooling				
PIRT Parameter	**Boiling–Nucleate**				
	Plant Range	Test Facility			
Plant Parameter		ORNL Test 3.07.9N			
P (MPa)	0.1 – 7.0	12.7			
Wall Superheat (K)	0 – 10	14 - 17			
Void Fraction	0 - 0.4	0.17 - 0.89			
Mass Flux (kg/m2-s)	0 – 1500	806			
Heat Flux (MW/m2)	0 - 0.555	0.94			
Subcooling (K)	10 – 60	14.29			
Comments					

Refererences

1. G. L. Yoder et al., Dispersed Flow Film Boiling in Rod Bundle Geometry Steady State Heat Transfer Data and Correlation Comparisons, NUREG/CR-2456, ORNL-5848, March 1982.

TABLE H-3
PROPOSED BWR EXPERIMENTAL FACILITIES: CONDENSATION–INTERFACIAL

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, reflood				
PIRT Parameter	**Condensation–interfacial: ECC Water**				
	Plant Range	Test Facility (See Table F-3)			
Plant Parameter					
Pressure (MPa)	0.1- 5.0				
Void fraction	0.0 -1.0				
ECC Temp (F)	80 – 180				

TABLE H-4
PROPOSED BWR EXPERIMENTAL FACILITIES: DRYOUT–CHF

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood				
PIRT Parameter	**Dryout–critical heat flux (CHF)**				
	Plant Range	Correlations			
Plant Parameter		Biasi	CISE	Zuber	
Pressure (MPa)	0.1 – 5.0	0.1-14.2	7.0	0.1 – 5.0	
Mass Flux(kg/m^2-s)	0 – 6000	100- 6000	300 -1400	< 100	
Quality	0.1 – 1.0	0.2 – 1.0			
Void	0.7 – 1.0				
Comments				Zuber is applied if flow is countercurrent	

References

1. L. Biasi, et al, " Studies on Burnout: Part 3," Energ. Nucl. 14, 1967, pp.530-536.
2. CISE: Heat Transfer Crisis in Steam-Water Mixtures, Energ. Nucl. 12, 1965
3. N. Zuber et al," The Hydrodynamic Crisis in Pool Boiling of Saturated and Subcooled Liquids," Int. Developments in Heat Transfer, 2, 1961, pp. 230-236.

TABLE H-5
PROPOSED BWR EXPERIMENTAL FACILITIES: FLASHING–INTERFACIAL

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood				
PIRT Parameter	**Flashing–interfacial: lower plenum, core, and downcomer**				
	Plant Range	Test Facility			
Plant Parameter		ROSA-III Tests 901, 902, 924, 926, 905	FIST Test 6DBA1B		
Pressure (MPa)	0.1 – 5.0	0.1 – 7.0	0.1 – 7.0		

References

1. Tasaka, et al., ROSA-III Double-Ended Break Test series for a Loss-of-Coolant Accident in a BWR," Nucl. Tech. Vol. 68, Jan 1985, pp. 77-93.

2. H. Kumamaru, et al., "Similarity Study of ROSA-III and FIST Large Break Counterpart Tests to BWR Large Break LOCA," Nucl. Engr. and Design 103, pp. 223-238, June 1986.

TABLE H-6
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–CRITICAL

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	**Flow–Critical**				
	Plant Range	Test Facility (See Table F-8)			
Plant Parameter					
Pressure (MPa)	0.7 – 7.0				
L/D	1 – >10				
Subcooling (K)	0 – 20				

TABLE H-7
PROPOSED BWR EXPERIMENTAL FACILITIES: HEAT CONDUCTANCE–FUEL-CLAD GAP

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Heat conductance–fuel-clad gap**	Governs temperature distribution and removal of stored heat			
	Plant Range	Test Facility (See Table F-10)			
Key Physical Parameter					
Pellet:					
k (W/m-K)	(7.5-18.5)10E+3				
T (K)	>530				
Gap:					
h (W/sq.m-K)	(3.3-13.1)10E+3				
Burnup (MWD/T)	0-40,000				
Comments					

TABLE H-8
PROPOSED BWR EXPERIMENTAL FACILITIES: HEAT TRANSFER–FORCED CONVECTION TO VAPOR

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, reflood				
PIRT Parameter	**Heat transfer–forced convection to vapor**				
	Plant Range	Test Facility (See also Table F-11)			
Plant Parameter		THTF Bundle Uncovery Tests 3.09.10 I,J, K, L, M, & N	G-2 336 Rod bundle Uncovery Tests 718, 722, 727, &731		
Pressure (MPa)	0.1 – 5.0	3.9 – 7.0	0.1 – 5.5		
Void fraction	1.0	1.0	1.0		
Clad Temp (F)	500- 2200	500 – 1500	500 – 1600		
Vapor Temp (F)	500 –1800	500 – 1200	500 -1300		
Vapor Re	100-2000	1100- 18,000	1000- 7000		
Comments		Tests contain level swell and thermal radiation to steam data also	Tests contain level swell data also		

References

1. Anklam, et al., "Experimental Investigations of Uncovered-Bundle Heat Transfer and Two-Phase Mixture Level Swell Under High Pressure Low Heat Flux Conditions," NUREG/CR-2456, ORNL, March 1982.

2. H. Yeh, et. al., "Heat Transfer Above the Two-Phase Mixture Level Under Core Uncovery Conditions in a 336 Rod Bundle," EPRI NP-2161, December 1981.

TABLE H-9
PROPOSED BWR EXPERIMENTAL FACILITIES: HEAT TRANSFER–RADIATION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill/reflood				
Phenomenon/Justification	**Heat transfer–radiation**			Affects the peak clad temperature (in BWR/2)	
	Plant Range	Facility range			
		Separate effect tests		Integral tests	
Key Parameter/Facility		GOETA (Ref. 1)	THTF (Ref. 2)	TLTA-5A (Ref. 3)	
$T_w - T_v$ (K)	~400	550 –850	<400	400 –600	
Emissivity (-)	0.6 – 1.0	0.7	0.4 – 0.6	~0.6	
Geometry	rod-to-rod and wall	rod-to-rod and wall	rod-to-rod and wall	rod-to-rod and wall	
Comments		stagnant steam	steady-state boiloff	LBLOCA/no ECC	

References

1. Test 27: Experimental investigations of cooling by top spray and bottom flooding for a BWR, Studsvik/RL-78/59, June 1978.
2. Test 3.09.10K: Experimental investigations of uncovered bundle heat transfer…, NUREG/CR-2456.
3. Test 6426/Run 1: BWR BD/ECC program, NUREG/CR-2229.

T_w=wall temperature
T_v=steam temperature

TABLE H-10
PROPOSED BWR EXPERIMENTAL FACILITIES: HEAT–STORED

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, reflood				
PIRT Parameter	**Heat–stored (fuel and metal structures)**				
	Plant Range	Test Facility (See Tables F-10)			
Plant Parameter					
Temp (K)	570 - 1000				
Comments	**Metal to volume ratio is an important parameter**				

TABLE H-11
PROPOSED BWR EXPERIMENTAL FACILITIES: INTERFACIAL SHEAR

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown/refill/reflood				
Phenomenon/Justification	**Interfacial shear**	Affects two-phase separation (level), entrainment, and pressure drop			
	Plant Range	Facility range			
		Separate effect tests			
Key Parameter/Facility		CISE (Ref. 1)	GE Level Swell (Ref. 2)	TLTA-5A (Ref. 3)	Pericles (Ref. 4)
P (MPa)	0.1 – 7.2	5.0	0.1 – 7.0		0.2 – 0 .4
G_l (kg/m²-s)		80 – 380		~0 – 360	
G_v (kg/m²-s)		4 – 310		2.4 – 360	
Void (-)	~0 – 1.0	0.2 – 0.9	0 – 1.0	0.1 – 1.0	0.2 – 0.9
Geometry	bundle, plenum, pipe	round tube	vessel 1ft & 4ft OD	full-scale bundle	bundle
Comments		steady-state flow	flashing/blowdown Test 1004-3 Test 5801-13	steady-state boiloff	steady-state boiloff

References

1. Density measurements of steam/water mixture flowing in tube, CISE-R-291, December 1969.
2. J. A. Findlay and G. L. Sozzi, "BWR Refill-Reflood Program - Model Qualification Task Plan," General Electric Company document NUREG/CR-1899 (October 1981).
3. Test 6441: BWR BD/ECC program, NUREG/CR-2229.
4. Study of two-dimensional effects in core of LWR during the reflood phase, CEC, Final Report Contract No. SR) 2F, 1984.

Note: Refs. 3 and 4 are applicable for assessment of interphase drag in bundles.

G_l=mass flux of liquid phase
G_v=mass flux of steam

TABLE H-12
PROPOSED BWR EXPERIMENTAL FACILITIES: REWET

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Rewet**	Determines the transition from film to nucleate boiling			
	Plant Range	Facility range			
		Integral tests			
Key Parameter/Facility		TLTA-5A (Ref. 1)			
T_{wall} (K)	650 – 850	620 – 850			
T_{sat} (K)	~550	~550			
P (MPa)	7.0 – 6.0	~6.5			
x (-)	~0.5	~0.5			
Geometry	channeled 8x8 bundle	full-scale bundle			
Comments		LBLOCA/no ECC			

References

1. Test 6426/Run 1: BWR BD/ECC program, NUREG/CR-2229.

TABLE H-12 (cont)
PROPOSED BWR EXPERIMENTAL FACILITIES: REWET

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Reflood				
Phenomenon/Justification	**Rewet**		Determines the transition from film to nucleate boiling		
	Plant Range	Facility range			
		Separate effect tests			
Key Parameter/Facility		GOETA (Ref. 2)	NEPTUN (Ref. 3)	BWR-FLECHT (Ref. 4)	PWR-FLECHT (Ref. 5)
P (MPa)	0.1 – 1.0	0.7	0.1 – 0.4	0.15 – 0.45	0.15 – 0.30
T_{wall} (K)	600 – 800	850 – 1100	1030 – 1140	1030 – 1220	530 – 1140
$T_{sat} - T_{ECC}$ (K)	25 – 150	75	22 – 134	0 – 90	10 – 80
V_{flood} (cm/s)	2.5 – 10.0	–	1.5 – 15.0	8 – 14.0	1.6 – 3.8
W_{spray} (kg/s) *)	0.5 – 0.75	0.44	–	–	–
Geometry	channeled 8x8 bundle	channeled 8x8 bundle	half-length bundle	7x7 bundle	10x10 bundle
Comments	*) on channel basis	top reflood	bottom reflood	bottom reflood	bottom reflood

References

2. Test 42: Experimental investigations of cooling by top spray and bottom flooding for a BWR, Studsvik/RL-78/59 (June 1978).
3. NEPTUN bundle reflooding experiments, EIR Report No. 386 (1981).
4. Effect of geometry and other parameters on bottom flooding heat transf. associated with nucl. fuel bundle simulators, ANCR-1049 (April 1972).
5. FLECHT – low flooding rate cosine test series, WCAP-8651 (December 1975).

TABLE H-12 (cont)
PROPOSED BWR EXPERIMENTAL FACILITIES: REWET

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Reflood				
Phenomenon/Justification	**Rewet**	Determines the transition from film to nucleate boiling			
	Plant Range	Facility range			
		Integral tests			
Key Parameter/Facility		TLTA-5A (Ref. 6)	FIST (Ref. 7)		
P (MPa)	0.1 – 1.0	0.1 – 1.0	0.3 – 0.5		
T_{wall} (K)	600 – 800	500 – 800	550 – 800		
$T_{sat} - T_{ECC}$ (K)	25 – 150	132	84 – 104		
V_{flod} (cm/s)	2.5 – 10.0	5.1			
W_{spray} (kg/s) *)	0.5 – 0.75	0.67	0.5		
Geometry	channeled 8x8 bundle	full scale bundle	full scale bundle		
Comments	*) on channel basis	LBLOCA	LBLOCA		

References

6. Test 6424/Run 1: BWR BD/ECC program, NUREG/CR-2229.
7. Test 4DBA1: BWR FIST Phase 2, NUREG/CR-4128 (March 1986).

TABLE H-13
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–CHANNEL BYPASS LEAKAGE

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood				
PIRT Parameter	**Flow–Channel Bypass Leakage**				
	Plant Range	Test Facility			
Plant Parameter		ROSA-III Tests 901, 902, 924, 926, 905 (Ref. 1)	FIST Test 6DBA1B (Ref. 2)		
Pressure (MPa)	0.1 – 7.0	0.1 – 7.0	0.1 – 7.0		
Leakage Flow (kg/s)	0 – 1.5		0 – 1.2		
Geometry	Channel bundle	Simulated leakage paths with drilled holes	Prototypical		
Comments		4 channels	one channel		

References

1. Tasaka, et al., ROSA-III Double-Ended Break Test series for a Loss-of-Coolant Accident in a BWR," Nuclear Technology, Vol. 68, pp. 77-93 (January 1985).
2. H. Kumamaru, et. al., "Similarity Study of ROSA-III and FIST Large Break Counterpart Tests to BWR Large Break LOCA," Nuclear Engineering and Design, 103, pp. 223-238 (June 1986).

TABLE H-14
PROPOSED BWR EXPERIMENTAL FACILITIES: COUNTERCURRENT

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill/reflood				
PIRT Parameter	**Flow–countercurrent: upper tie plate**				
	Plant Range	Test Facility			
Plant Parameter		Tobin BD/ECC (Ref. 2)	Jones BD/ECC(GE 8x8 bundle data) (Refs. 1, 3)	Naitoh et. al. (Ref. 4)	GOTA BWR ECC Tests (Ref. 5)
P (MPa)	0.1 – 5.0	Near atmospheric	Near atmospheric	Near atmospheric	0.1- 2.0
Steam Flow (gm/s)	90 – 200	36 – 99	0 – 126	43 –83	
Liquid Flow (cm^3/s)	0 – 1000	549 –972	315 – 916	117 – 1033	0.045 – 2.20 Kg/s
Kf-	0 – 2.1		0.0. – 0.8	0.0 – 0.7	
Kg-	0 - 2.1		1.0 – 2.1	1.0 – 2.1	
Water Temp (°C)	40 – 80	Saturated	38 – 96	27 – 97	37 – 97
Comments	Note that the range for Kf and Kg include the range where CCFL exists. Data on a channel basis	Sat. steam/water	Sat. steam	Steam inlet from bundle bottom	Top spray, 64 rods(CCF in bundle pacers, not in tie plate)

References

1. D. D. Jones, "Test Report TLTA Components CCFL Tests," GE Nuclear Systems Products Division, BD/ECC Program, NEDG-NUREG-23732, (1977).
2. R. Tobin, CCFL Test Results, Phase 1 – TLTA 7x7 Bundle," GE Nuclear System Products Division, BD/ECC Program, GEAP-21304-5 (1977).
3. D. D. Jones, "Subcooled Countercurrent Flow Limiting characteristics of the Upper Region of a PWR Fuel Bundle," GE Nuclear Systems Products Division, BD/ECC Program, NEDG-NUREG-23549, (1977).
4. M. Natitoh, et. al., "Restrictive Effect of Ascending Steam on Falling Water during Top spray Emergency core Cooling," J. of Nuclear Science and Technology, Vol. 15, 11, pp. 806, (1978).
5. "Separate Effects Test Matrix for Thermal Hydraulic Code Validation – Volume I," Organization for Economic Co-operation and Development Nuclear Energy Agency document OECD/GD(94)82 (September 1993).

TABLE H-14 (cont)
PROPOSED BWR EXPERIMENTAL FACILITIES: COUNTERCURRENT

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill/reflood				
PIRT Parameter	**Flow–countercurrent: upper tie plate**				
	Plant Range	Test Facility			
Plant Parameter		UPTF (Ref. 6)			
P (MPa)	0.1 – 5.0	0.3 – 1.5			
Steam flow (Kg/s)	61 – 153	35 – 300			
Liquid flow (Kg/s)	300 – 460	30 – 1200			
Kf^-	0 – 2.1				
Kg^-	0 – 2.1				
Water Temp. (°C)	40 – 80	Sat – 30.0			
Flow cross section, (m^2)	2.6	3.755 (1:1 scale)			
Comments	Flow cross section is for BWR/4, hole dia. is also important. Data on a core basis.	Steady-state, hot-leg water injection			

<u>Nomenclature</u>
P, pressure
Kf^- , Kutateladze No. for liquid
Kg^- , Kutateladze No. for steam

<u>References</u>

6. U. Simon, et al, "UPTF Calibration Tests, Final Report on Research Project BMFT 1500664, Kraftwerk Union, Technischer Bericht
R 54/85/14, December 1985.

TABLE H-15
PROPOSED BWR EXPERIMENTAL FACILITIES: COUNTERCURRENT

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill/reflood				
PIRT Parameter	**Flow–countercurrent: side entry orifice**				
	Plant Range	Test Facility			
Plant Parameter		Jones BD/ECC (GE 8x8 bundle data)			
P (MPa)	0.1 – 2.0	Near atmospheric			
Steam flow (gm/s)	7 – 30	0 – 38			
Liquid flow (cm3/s)	0 – 500	0 – 505			
$Kf^{1/2}$	0 – 3.0	0.0 – 1.2			
$Kg^{1/2}$	0 – 1.8	0.9 – 2.0			
Water Temp. (°C)	40 – 80	Saturated steam/water			
Comments	Flow rates are for channel	Bundle bottom inlet, side entry orifices; five orifices sizes			

Nomenclature
P, pressure
q, heat flux
G, mass flux

References

1. D. D. Jones, "Test Report TLTA Components CCFL Tests," GE Nuclear Systems Products Division, BD/ECC Program, NEDG-NUREG-23732, (1977).

2. K. H. Sun and R. T. Ferdandez, "Countercurrent Flow Limitation Correlation for BWR Bundles during a LOCA," ANS Transactions, Vol. 27, pp. 605 (1977).

3. K. H. Sun, "Flooding Correlations for BWR Bundle Upper tie Plate and side Entry Orifices," Second Multi-Phase Flow and Heat Transfer Symposium Workshop, Miami Beach, Florida, April 16-19, 1979.

TABLE H-16
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–DISTRIBUTION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
PIRT Parameter	**Flow–Distribution: lower plenum**				
	Plant Range	Test Facility			
Plant Parameter		ROSA-III Tests 901, 902, 924, 926, 905 (Ref. 1)	FIST Test 6DBA1B (Ref. 2)	TLTA Tests 6422 Run 3, 6424 Run 1, 6423 Run 3, & 6426 Run 1 (Ref. 3)	SSTF Test EA2-2 (Ref. 3)
Pressure (MPa)	0.1 – 5.0	0.1 – 7.0	0.1 – 7.0	7.1	0.507
					Low plen inj rate = 3.024 kg/s Core steam inj rate= 4.98 kg/s LPCI = 49.21 l/s Subcooling of inj water= 105 K

References

1. Tasaka, et al., ROSA-III Double-Ended Break Test Series for a Loss-of-Coolant Accident in a BWR," Nuclear Technology, Vol. 68, pp. 77-93 (January 1985).
2. H. Kumamaru, et al., "Similarity Study of ROSA-III and FIST Large Break Counterpart Tests to BWR Large Break LOCA," Nuclear Engineering And Design, 103, pp. 223-238 (June 1986).
3. NUREG/CR-2571, "BWR Refill-Reflood Program Task 4.8 – TRAC-BWR Model Qualification for BWR Safety Analysis Final Report," October 1983.

TABLE H-17
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–FORWARD

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Jet pump: forward flow**			Affects coastdown of the core flow	
	Plant Range	Facility range			
		Integral tests		Separate Effects Tests	
Key Parameter/Facility		TLTA-5A (Ref. 1)	FIST (Ref. 2)	LSTF 1/6 Scale Jet Pump (Ref. 3)	Full Scale Jet Pump Data (Ref. 4)
N – Ratio (-)	0.15 – 0.22			2 to -2	0.125 – 0.325
M – Ratio (-)	1.5 – 2.5	2 – 2.25		2 to –2	0.35 – 2.25
Forward flow loss (-)		~4.0	~8.0		
P (MPa)				0.4 –8.16	7.05
Fluid Temp (K)				302 – 562	
Suction Flow (Kg/s)				0 – 13.0	Discharge flow = 300 L/s
Drive Flow (Kg/s)				0 – 4.0	200 L/s
Comments		LBLOCA	LBLOCA		

1. Test 6426/Run 1: BWR BD/ECC program, NUREG/CR-2229.
2. Test 6DBA1B: BWR FIST: Phase 1 results, NURG/CR-3711, March 1985.
3. G. E. Wilson, "INEL One-Sixth Scale Jet Pump Data Analysis," EG& G Idaho, Inc. document EGG-CAAD-5357 (February 1981).
4. A. A. Kurdirka and D. M. Gluntz, "Development of Jet Pumps for Boiling Water Reactor Recirculation System," Journal of Engineering Power, pp. 7 –12, January 1974.

TABLE H-18
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–MULTIDIMENSIONAL

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill/reflood				
Phenomenon/Justification	**Flow–Multidimensional: upper plenum**	Affects CCFL in the upper plenum and top reflood			
	Plant Range	Facility range			
		Component tests			
Key Parameter/Facility		SSTF/UP (Ref. 1)			
P (MPa)	0.1– 1.0	0.2 – 1.0			
W_{spray} (kg/s) *)	~0.5	0.4 – 0.54			
$T_{sat} - T_{ECC}$ (K)	25 – 150	54 – 145			
W_{steam} (kg/s) *)	0.05 – 0.2	0.09 – 0.16			
Geometry	upper plenum	full scale upper plenum			
Comments	*) on channel basis	spray into 2-phase mix			

References

1. BWR refill-reflood program Task 4.4, NUREG/CR-2786, May 1983.

TABLE H-19
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–REVERSE

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Flow–reverse: jet pump**				Affects break flow
	Plant Range	Facility range			
		Component tests		Integral tests	
Key Parameter/Facility		INEL 1/6 jet pump (Ref. 1)		TLTA-5A (Ref. 2)	FIST (Ref. 3)
Reverse flow loss (-)	~0.9			~1.2	~1.3
Comments		Covers wide range of BWR jet pump conditions		LBLOCA	LBLOCA

References

1. G. E. Wilson, "INEL One Sixth Scale Jet Pump Data Analysis," EG&G Idaho, Inc. document EGG-CAAD-5357 (February 1981).
2. Test 6426/Run 1: BWR BD/ECC program, NUREG/CR-2229.
3. Test 6DBA1B: BWR FIST: Phase 1 results, NURG/CR-3711, March 1985.

TABLE H-20
PROPOSED BWR EXPERIMENTAL FACILITIES: POWER–3D DISTRIBUTION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill/reflood				
Phenomenon/Justification	**Power–3D distribution**	Affects the peak clad temp. location and channel grouping			
	Plant Range	Facility range			
		Integral tests			
Key Parameter/Facility		ROSA-III (Ref. 1)			
P_{rad} (-)	0.5 – 1.2	1 – 1.4			
Geometry	channeled bundles	4 half-length bundles			
Comments		LBLOCA			

References

1. Test 926: ROSA-III experimental program, JAERI-1307, November 1987.

TABLE H-21
PROPOSED BWR EXPERIMENTAL FACILITIES: POWER–DECAY HEAT

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown, refill, reflood, long-term cooling				
PIRT Parameter	**Power: Decay Heat**				
	Plant Range	Test Facility (See Table F-20)			
Plant Parameter					
Time (sec)	$0 - 10^{10}$				
Comments					

TABLE H-22
PROPOSED BWR EXPERIMENTAL FACILITIES: PRESSURE DROP

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Pressure drop**	Affects the flow distribution between the shroud and downcomer			
	Plant Range	Facility range			
		Separate effect tests			Integral tests
Key Parameter/Facility		Sher and Greer (Ref. 1)	Muscettola (Ref. 2)	EPRI (Ref. 3)	ROSA-III (Ref. 4)
P (MPa)	0.7 – 7.2	7.6 and 14	6.9	< 0.2	0.7 – 7.2
G (kg/m^2-s)	~30 – 2020	950 – 6780	1145 – 4370	1500 – 2100	~10 – 1100
x (-)		0 – 0.4	0.01– 0.7		
Geometry	bundle, plenum, pipe	rectangular tube	round tube	square tube	4 half-length bundles
Comments		steam-water	steam-water	air-water	LBLOCA

References

1. Boiling pressure drop in thin rectangular chennels, Chem. Symp. Series 23, 61-73, 1959.
2. Two-phase pressure drop – comparison with measurements, AEEW-R-284, 1963.
3. Experimental study of the diversion cross-flow, EPRI NP-3459, Vol. 1, April 1984.
4. Test 926: ROSA-III Experimental Program, JAERI-1307, November 1987.

TABLE H-23
PROPOSED BWR EXPERIMENTAL FACILITIES: PUMP–PERFORMANCE

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Blowdown				
Phenomenon/Justification	**Pump–performance: recirculation pump coastdown**				Determines the core flow
	Plant Range	Facility range			
		Integral tests			
Key Parameter/Facility		ROSA-III (Ref. 1)	FIST (Ref. 2)		
Torque/Inertia (s^{-2})	38 – 58	~100			
Time (s)	5 – 8		5 – 8		
Geometry	centrifugal pump	centrifugal pump	centrifugal pump		
Comments		LBLOCA	LBLOCA		

References

1. Test 926: ROSA-III Experimental Program, JAERI-1307, November 1987.
2. Test 4DBA1: BWR FIST Phase 2, NUREG/CR-4128, March 1986.

TABLE H-24
PROPOSED BWR EXPERIMENTAL FACILITIES: SPRAY DISTRIBUTION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill/reflood				
Phenomenon/Justification	**Spray distribution**	Affects CCFL and its breakdown at the upper tie-plate			
	Plant Range	Facility range			
		Component tests			
Key Parameter/Facility		SSTF (Ref. 1)			
Sparger height (m)	0.15 – 0.7	0.15 – 0.4			
2-phase level (m)	0 – 1.0	0 –0.4			
Geometry	upper plenum	full scale upper plenum			
Comments		different BWR sprays			

References

1. BWR refill-reflood program Task 4.4, NUREG/CR-2133, May 1982.

TABLE H-25
PROPOSED BWR EXPERIMENTAL FACILITIES: VOID DISTRIBUTION

Plant	BWR					
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)					
Transient Phase	Blowdown/reflood					
Phenomenon/Justification	**Void distribution/2-phase level**			Determines heat transfer in the core below and above the 2-phase level, timing of jet pump, inlet orifice, and recirc. suction uncovery		
	Plant Range	Facility range				
		Separate effect tests				
Key Parameter/Facility		Frigg (Ref. 1)	TLTA-5A (Ref. 2)	GE Level Swell (Ref. 3)	SSTF/LP (Ref. 4)
P (MPa)	0.1 – 7 .0	~5.0		0.1 – 7.0	0.2 – 1.0
G (kg/m^2-s)		690 – 1500	2.4 – 360		
Void (-)	0 – 1.0	0 – 0.8	0.1 – 1.0	0 – 1.0	0 – 1.0
Geometry	bundle, plenum, annulus	37-rod bundle	full-scale bundle	vessel 1-ft & 4-ft OD	full-scale lower plenum
Comments		steady-state boiling	steady-state boiloff	flashing/blowdown Test 1004-3 Test 5801-13	flashing experiment

References

1. Frigg-2, Hydrodynamic and heat transfer measurements on a full scale 36-rod Marviken fuel element, ASEA and ABB, 1968.
2. Test 6441: BWR BD/ECC program, NUREG/CR-2229.
3. J. A. Findlay and G. L. Sozzi, "BWR Refill-Reflood Program - Model Qualification Task Plan," General Electric Company document NUREG/CR-1899 (October 1981).
4. BWR refill-reflood program Task 4.4, NUREG/CR-2786, May 1983.

TABLE H-25 (cont)
PROPOSED BWR EXPERIMENTAL FACILITIES: VOID DISTRIBUTION

Plant	BWR					
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)					
Transient Phase	Blowdown/reflood					
Phenomenon/Justification	**Void distribution/2-phase level**			Determines heat transfer in the core below and above the 2-phase level, timing of jet pump, inlet orifice, and recirc. suction uncovery		
	Plant Range	Facility range				
		Separate effect tests		Integral tests		
Key Parameter/Facility		ANL (Ref. 5)		TLTA-A (Ref. 6)	FIST (Ref. 7)	
P (MPa)	0.1 – 7 .0	1.03 – 4.13		0.1 – 7.0	0.1 – 7.0	
G (kg/m²-s)						
Void (-)	0 – 1.0			0 – 1.0	0 – 1.0	
Geometry	bundle, plenum, annulus			full scale bundle	full scale bundle	
Comments		Subcooled and saturated void; heat flux is 17-100 kW/liter; subcooling is 2-19K, inlet velocity is 1-6 m/s.		LBLOCA	LBLOCA	

5. J. F. Marchaterre, "Natural and Forced-Circulation Boiling Studies," Argonne National Laboratory document ANL-5735 (May 1960).
6. Test 6424/Run 1: BWR BD/ECC program, NUREG/CR-2229.
7. Test 4DBA1: BWR FIST Phase 2, NUREG/CR-4128, March 1986.

TABLE H-26
PROPOSED BWR EXPERIMENTAL FACILITIES: FLOW–NATURAL CIRCULATION

Plant	BWR				
Transient	Large-Break Loss-of-Coolant Accident (LBLOCA)				
Transient Phase	Refill, reflood, long term coling				
PIRT Parameter	**Flow–natural circulation**				
	Plant Range	Test Facility			
Plant Parameter		ROSA-III Test NC-1 NC-5 (Refs. 2-3)	FRIGG Test FT 36a 36b, & 36c (Ref. 1)	FIST 6PNCI-4 (Refs. 4-6)	
Pressure (MPa)	0.1- 7.0	7.35, 2.06	1 –7.0	7.0	
Inlet Subcooling (K)	0- 60	0	3 – 58	0.0	
Exit Qual %	10- 80		3 - 73	0-7	
Mass Flux (kg/m2-s)	0.0-1500	100 - 400	195 – 2160	0-1022	
Heat Flux (MW/m2)	0.0-0.555	Core power: 7-20%	0.21-0.89	0.222	
Downcomer Level(m)	1.6	0.6 – 1.7		1 – 1.6	
Comments		The ROSA Nat Circ tests were conducted by changing pressure, core power, and downcomer liquid level (below the scram level) as test parameters			

References

1. O. Nyland, et. al., "Hydrodynamic and Heat Transfer Measurements on a Full Scale Simulated 36-Rod Marviken Fuel Element with Uniform Heat Flux Distribution, FRIGG Loop Project, FRIGG-2g, 1968.
2. K. Tasaka et. al., "Steam Line Break, Jet Pump Drive Line Break and Natural Circulation Tests in ROSA-III Program for BWR LOCA/ECCS Integral Tests," Eleventh water Reactor Safety Research information Meeting, Gaithersburg, MD, October 24-28, 1983.
3. K. Tasaka, et al., "ROSA-III Double-Ended Break Test Series for a Loss-of-Coolant Accident in a BWR," Nuclear Technology, Vol. 68, pp. 77-93 (January 1985).
4. "BWR FIST Phase I Results," NUREG/CR-3711 (March 1985).

5. "BWR FIST Phase II Results," NUREG/CR-4128 (March 1986).
6. "TRAC-BD1/MOD1–An Advanced Best Estimate Computer Program for BWR Transient Analysis, Volume 4, Developmental Assessment," EG&G Idaho, Inc., document NUREG/CR-3633 (April 1984).

APPENDIX I

EXPANDED LISTING OF TRAC-B INPUT DECKS FOR COMMON AND BWR-SPECIFIC SETS, IETS, AND PLANTS

Table I-1 lists the available common and BWR-specific TRAC-B SETs input decks. For each facility input deck, a brief description of the facility, test type, test number, and report reference in addition to the latest code version on which the input deck was exercised are provided. Table I-2 lists the available common and BWR TRAC-B IET input decks in the same format. Table I-3 lists the available BWR TRAC-B plant input decks in the same format. Please note that the TRAC-M input processing can also read TRAC-B format input decks.

Table I-1
TRAC-B INPUT DECKS FOR SEPARATE EFFECT TESTS

Facility	Type of Test	Test ID	References	Decks[a]	Comments
Marviken	Critical Flow	Test 15,24	I-1	Mv7c	10 second blowdown. Can model Tests 15 or 24
CISE	Two-phase flow in an adiabatic vertical pipe	CISE-R-291	I-2	Cistbf1	
THTF	Rod-bundle blowdown heat transfer	306.6B, 308.6C	I-3	Thtf366 Thtf386	
Bennett	Dispersed flow film boiling	Test 5358	I-4	Ben5358	
FRIGG	Natural circulation flow test(36-rod bundle)	Run 301016	I-5	Frgns1,frgnt1	
GE Small Vessel	Level swell	Test 1004-3	I-6	Swl8a	
Edwards Pipe	Critical flow	Blowdown test	I-7	Edpga	
Jet Pump	INEL1/6 scale jet pump		I-8	Jp2cbf1	

[a] TRAC-B Version 014 Input Deck

TABLE I-2
TRAC-B INPUT DECKS FOR INTEGRAL EFFECT TESTS

Facility	Type of test	Test ID	References	Decks	Comments
TLTA	LB LOCA	Test 6423	I-9	TRAC-B Version 014, Tlta	
FIST	LB LOCA	Test 6DBA1B	I-10	Fist6dba1b	
FIST	SB LOCA	Test 6SB2C	I-10	Fist6sb2c	
FIST	ATWS type event	Test 6pmc2	I-10	Pmc1bc1	

TABLE I-3
TRAC-B INPUT DECKS FOR NUCLEAR POWER PLANTS

Plant	Transient or Accident	References	Decks	Comments
Browns Ferry	LB LOCA	I-11	BFLBLOCA-TRAC-B	A TRAC-M deck also exists
Browns Ferry	SB LOCA	I-12	BFSBLOCA-TRAC-B	A TRAC-M deck also exists
Browns Ferry	1-pump trip transient	I-13	BF1PUMP-TRAC-B	
Browns Ferry	2-pump trip transient	I-14	BF2PUMP-TRAC-B	
Browns Ferry	Feedwater pump trip transient	I-15	BFFWTRAC-B	
Browns Ferry	Generator load rejection transient	I-16	BFGLRTRAC-B	
Peach Bottom	Feedwater pump trip transient	None	PBFWTRAC-B	A TRAC-M deck also exists
Generic BWR/6	Small break LOCA	None	Bwrstra	
Generic BWR/6	Large break LOCA	None	Bwrltra	
Generic BWR/6	Recirculation pump trip	None	Blackfox	
Dresden	Recirculation pump trip	None	Dresden	
Generic BWR/4	Recirculation pump trip	None	Gbwr4lds-1dkin	
Grand Gulf	Steady-state deck	None	Grandg	
LaSalle	85% power with recirculation pump trip and reactivity transient	None	Lasalle	

REFERENCES

I-1. "Marviken Full-Scale Critical Flow Tests-Interim Report Description of the Test Facility," MXC-101, Marviken Power Station, Sweden.

I-2. G. Agnostini, et al., Density Measurements of Steam Water mixtures Flowing in a Tubular Channel Under Adiabatic and Heater Conditions, CISE-R-291, December 1969.

I-3. D. G. Morris, et al., "A Preliminary Evaluation of Rod Bundle Post CHF Heat Transfer to High Pressure Water in Transient Upflow," Interim Report for THTF Test 3.06.6B, ORNL, PWR-DBHT Separate Effects Program, November 1980.

I-4. W. Bennett, Heat Transfer to Steam Water Mixtures Flowing in Uniformly Heated Tubes in which the Critical Heat Flux Has Been Exceeded, AERE-R-5373, 1967.

I-5. O. Nyland, et al., "Hydrodynamic and Heat Transfer Measurements on Full-Scale simulated 36 Rod Marviken Fuel Element with Uniform Heat Flux Distribution," FRIGG Loop Project, FRIGG-2g, 1968. See also O. Nyland, et al., FRIGG Loop Project, FRIGG, R4-494/RL.

I-6. J A Findlay, "BWR Reflood-Reflood Program Model Qualification Task Plan," NUREG/CR-1899 October 1981.

I-7. R. Edwards and F. P. O'Brien, "Studies of Phenomena Connected with the Depressurization of Water Reactors," Journal of the British Nuclear Energy Society, **9**, 1987, pp. 125-135.

I-8. G. E. Wilson, INEL, One-Sixth Scale Jet Pump Data Analysis, EGG-CAAD-5357, February 1981.

I-9. L S Lee et al., "BWR Large Break Simulation Tests-BWR Blowdown Emergency Core Cooling Program," 1982.

I-10. G. Stevens, "Full Integral Simulation Test (FIST) Facility Description Report," NUREG/CR-2576, December 1982.

I-11. UMCP-TRAC-012, "Large Break LOCA TRAC-B Input Model for Browns Ferry Unit 3," January 1998.

I-12. UMCP-TRAC-014, "TRAC-B Analysis of the 0.14 Ft^2 Recirculation Line Break for Browns Ferry Unit 3," January 1998.

I-13. UMCP-TRAC-02, "TRAC-B Input Model for the Browns Ferry Nuclear Plant Unit 3 One Pump Trip Transient," September 1997.

I-14. UMCP-TRAC-03, "TRAC-B Input Model for the Browns Ferry Nuclear Plant Unit 3 Two Pump Trip Transient," September 1997.

I-15. UMCP-TRAC-04, "TRAC-B Input Model for the Browns Ferry Nuclear Plant Unit 3 Feedwater Pump Trip Transient," September 1997.

I-16. UMCP-TRAC-05, "TRAC-B Input Model for the Browns Ferry Nuclear Plant Unit 3 Generator Load Rejection Transient," October 1997.

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10. SUPPLEMENTARY NOTES
F. Odar, NRC Project Manager

11. ABSTRACT *(200 words or less)*

This document briefly describes the elements of the Nuclear Regulatory Commission's (NRC's) software quality assurance program leading to software (code) qualification identifies a test matrix for qualifying the modernized Transient Reactor Analysis Code (TRAC-M) to the NRC's software quality assurance requirements. Code qualification is the outcome of several software life-cycle activities, specifically, (1) Requirements Definition, (2) Design, (3) Implementation, and (4) Validation Testing. The major objective of this document is to define the TRAC-M Validation Testing effort. Validation is the process of demonstrating that the as-built software meets its requirements. Testing is the primary method of software validation. We have subdivided the TRAC-M validation test matrix into four elements. The first set of validation activities compares code-calculated results with data from tests other than those employing experimental data, designated Other Standard Tests. The second set of validation activities compares code-calculated results with data from Separate Effect Tests. The third and fourth sets of activities compare code-calculated results with data from Component Effect Tests and Integral Effect Tests, respectively. The four elements identified above constitute the TRAC-M Validation Test Matrix.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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