DHR system based on natural circulation alone is independent of any power source. The DHR system consists of immersion coolers (ICs) installed in the hot plenum and connected to air coolers, each via intermediate circuits. During the postscram phase, the decay heat is to be removed by natural circulation from the core into the hot plenum and via the ICs and intermediate loops to the air coolers. The function of this DHR system is investigated and demonstrated in model tests with a geometry similar to the reactor, though on a different scale. RAMONA is such a three-dimensional model set up on a 1:20 scale. It is operated with water.

The steady-state tests for natural-circulation DHR operations have been conducted over a wide range of operational and geometric parameters.[1] To study the transition from nominal to DHR conditions, experiments were defined to investigate the onset of natural circulation in the postscram phase (transient tests). For these investigations RAMONA was provided with active components (see Fig. 1):

1. The core consists of nine controllable annular heaters representing fuel and blanket regimes.

2. Four speed-controlled primary pumps, each with two feed lines to the core; each feed line contains a mass flow meter.

3. Eight intermediate heat exchangers (IHXs): The inlet temperatures of the secondary loops are adjusted by coolers and heaters, and the secondary mass flow is controlled.

4. Four ICs installed pairwise in a 180-deg angular position in the hot plenum with one secondary loop each: The mass flow and the inlet temperature of each loop are controlled.

5. Approximately 250 stationary thermocouples in all components of interest and on the most important measurement traverses in the plena.

The reactor scram can be followed by various operational modes of the system and its time-dependent thermohydraulic behavior must be known. These are

1. decay heat level[2]
2. coastdown curve of the primary pump flow rate[2]
3. coastdown curve of the secondary flow rate of the IHX
4. delay time up to the start of the ICs after a scram.

This study concentrates on the investigations of the effects of IHX secondary-side conditions; pump stops at 15, 120, 240, and 1200 s after scram; and on the delay time of 0, 240, and 3000 s up to the start of the ICs after scram. The investigations simulate a scram from 40% load operation of a pool-type fast reactor. The time-dependent mass flows and temperatures were measured, and special attention was given to the onset of natural convection.

The experiments were analyzed using the one-dimensional LEDHER code.[3] LEDHER is a network analysis code for the long-term DHR of a fast reactor developed at Power Reactor and Nuclear Fuel Development Corporation in Japan. The results of the experiments and calculations allow the following conclusions:

1. The IHX secondary-side flow coastdown shows that all parameters that reduce the IHX primary-side temperatures increase the natural circulation flow rate within the reactor tank. If natural circulation in the secondary loops can be established, the core temperatures are significantly reduced.

2. The formation of the thermal stratification in the hot plenum is the key parameter for the onset of a very effective natural circulation. Longer time delay between shutdown and onset of IC operation provides higher core flow rates. The higher flow rate during the down time of the IC together with the large heat capacity of the system influences the core temperature only to a minor extent.

3. The calculations agree with the experimental results as long as the control volumes are well represented. The analysis of local effects in the hot plenum due to the interaction of hot fluid from the core and cold fluid from the immersion coolers needs a multidimensional computational method. These calculations are in progress.

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3. Steam Condensation and Liquid Hold-Up in Steam Generator U-Tubes During Oscillatory Natural Circulation, *G. F. De Santi (CEC-Italy), F. Mayinger (Tech Univ of Munich-FRG)*

In many accident scenarios, natural circulation is an important heat transport mechanism for long-term cooling of light water reactors. In the event of a small pipe break, with subsequent loss of primary cooling fluid loss-of-coolant accident (LOCA), or under abnormal operating conditions, early tripping of the main coolant pumps can be actuated. Primary fluid flow will then progress from forced to natural convection. Understanding of the flow regimes and heat-removal mechanisms in the steam generators during the entire transient is of primary importance to safety analysis.

Flow oscillations during two-phase natural circulation experiments for pressurized water reactors (PWRs) with inverted U-tube steam generators occur at high pressure and at a primary inventory range between two-phase circulation and reflux heat removal.

This paper deals with the oscillatory flow behavior that was observed in the LOBI-MOD2 facility during the transition period between two-phase natural circulation and reflux condensation. Flow oscillations during natural circulation have been observed in the integral system facilities at the high-pressure range, Semiscale,[1] Large-Scale Test Facility,[2] and in ad hoc test apparatus in the low-pressure range.[3,4] De Santi[5,6] identified these oscillations to occur in the transition region between two-phase circulation and reflux condensation during the LOBI experiments. These oscillations, originating in the steam generator U-tubes, consist of alternating phases of stalled flow and rapid two-phase circulation. When the flow in parallel U-tubes oscillates in phase, the oscillations can eventually extend to the entire cooling circuit and have implications for coolant distribution inside the primary system and for core coolability. Using integral system data from LOBI-MOD2 experiments, it is demonstrated that the predominant mechanism that governs the duration of the interrupted flow phase is steam condensation at the top of the U-tubes. The two-phase natural circulation oscillations are caused by synchronized periodic interruptions of steam generator U-tube flow, due to vapor trapped at the top. As steam condenses, liquid columns develop in the ascending and descending legs of the U-tubes;

as they reach the top, flow starts again. The duration of the "stalled" phase of the flow oscillations in the LOBI natural-circulation experiments is controlled by the speed of steam condensation at the top of the U-tubes. By increasing the core power, the oscillation cycle becomes shorter (higher frequency) with smaller amplitude. The frequency is not governed by mechanisms related to countercurrent flow limiting (CCFL). Similar flow instabilities can occur during small-break LOCA depressurization transients.

There are several possible mechanisms of liquid hold-up in the vertical U-tube region which are necessary for a closed condensation space. This may be CCFL but could also be simply related to the water inventory in the circuit. Based on results from low-pressure experiments, the onset of flow oscillation during reflux condensation was related to the occurrence of flooding in the vertical pipes.[7,8] The CCFL would allow only part of the condensate to flow backward from the U-tubes to the core and the remainder to be dragged upward and held up by the steam. A liquid column developed in the U-tubes above the two-phase condensation region, establishing an increasing gravitational pressure drop, which reduced the water inventory in the core. This paper demonstrates that the mechanism to form the liquid column must not necessarily be CCFL; it may also be simply a "loop-full" condition. During the stalled phase, liquid accumulated in the vertical steam generator tubes by condensation of primary vapor at the top of U-tubes accumulates. As the liquid column reaches the top, it can flow over the U-bend and a rapid two-phase circulation is reestablished (carryover phase). This cycle does not require CCFL. The liquid column in the rising leg, necessary for formation of a closed steam condensation volume, may be formed simply at a relatively high primary inventory.

Furthermore, the influence of pressure shows that the phenomenon can only be truly observed in a high-pressure facility. Hypothetical experiments at near atmospheric pressure could only reproduce the observed oscillations at temperatures higher than for the high-pressure case. It may be difficult to perform low-pressure experiments without the presence of noncondensibles, due to the reduced solubility of air in water.

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4. Testing and Analysis of Passive Decay Heat Removal in Liquid-Metal Systems, *Richard C. Soucy, Ravnesh Amar (ETEC)*

Natural circulation tests were conducted in sodium to demonstrate passive removal of decay heat and to validate the computer codes that predict decay heat removal.

Two types of heat removal systems were tested:

1. direct reactor auxiliary cooling system (DRACS) utilizing an in-reactor-vessel sodium-to-sodium heat exchanger to transfer heat to an external natural draft sodium-to-air heat exchanger

2. reactor air-cooling system [(RACS)/radiant vessel auxiliary cooling system (RVACS)] that relies on natural-circulation air cooling on the outside of the vessel.

The purpose of testing is to produce generic data for the validation of three-dimensional multipurpose codes. Design and licensing of liquid-metal reactors (LMRs) requires validation of these codes by comparison with experimental thermal-hydraulic data.

TEST SYSTEM

The system designed to test the RVACS and DRACS concepts is shown schematically in Fig. 1. The DRACS test system consists of a sodium-filled vessel, internal heater assembly, an internal sodium-to-sodium heat exchanger with connecting piping, pump, air cooler, and instrumentation to control and monitor test conditions. The heat rejection loop connected to vessel T-5 includes the test system piping, electromagnetic (EM) pump, flowmeter, thermocouples, and heat exchangers.

Typically, the test tank and external loop were heated to the initial test temperature 340°C (650°F). To simulate decay heat from a reactor core, the central heaters were turned on, generating up to 100 kW power and a thermal up-transient in the T-5 test vessel. Heat rejection by the DRACS system was simulated by sodium flow through the external loop and heat removal through the air cooler as temperature increased in the sodium pool.

RESULTS

In general, core tank cooling was shown to occur as predicted as heat transferred through the immersed heat exchanger causing natural circulation of primary and secondary sodium through the DRACS loop and heat rejection through the air cooler.

A typical test was started with 100-kW power input to the central heater. Approximately 90 kW was transported from the test tank and was removed by the DRACS heat rejection loop. Test data compared closely with the DASHR computer code prediction by Amar and Estes.[11] COMMIX code analysis recently performed by Tzanos and Pedersen[2] produced temperature profile predictions in good agreement with test data.

RVACS/RACS SYSTEM

The program was extended to include a demonstration of the cooling characteristics of the RACS, also referred to as the RVACS included in Fig. 1. The RVACS/RACS test system consists of a sodium-filled vessel with an internal test assembly that simulates the internal configuration of a modular reactor and two external shrouds to simulate the reactor guard vessel and the external air shroud. The air-cooling shroud en-