THE 4TH INTERNATIONAL TOPICAL MEETING ON
NUCLEAR THERMAL HYDRAULICS, OPERATIONS AND SAFETY
April 6-8, 1994, Taipei, Taiwan

THE STATUS OF THERMAL-HYDRAULIC STUDIES ON THE DECAY HEAT REMOVAL BY NATURAL CONVECTION USING RAMONA AND NEPTUN MODELS

H. Hoffmann, K. Hain, K. Marten, H. Ohira*, K. Rust, D. Weinberg
Kernforschungszentrum Karlsruhe GmbH (KfK)
Institut für Angewandte Thermo- und Fluiddynamik (IATF)
Postfach 3640, 76021 Karlsruhe, Federal Republic of Germany
*Power Reactor and Nuclear Fuel Development Corporation, Tokyo, 107, Japan

ABSTRACT

Thermal-hydraulic experiments were performed with water in order to simulate the decay heat removal by natural convection in a pool-type sodium-cooled reactor. Two test rigs of different scales were used, namely RAMONA (1:20) and NEPTUN (1:5). RAMONA served to study the transition from nominal operation by forced convection to decay heat removal operation by natural convection. Steady-state similarity tests were carried out in both facilities. The investigations cover nominal and non-nominal operation conditions. These data provide a broad basis for the verification of computer programs. Numerical analyses performed with the three-dimensional FLUTAN code indicated that the thermal-hydraulic processes can be quantitatively simulated even for the very complex geometry of the NEPTUN test rig.

I. INTRODUCTION

The European Fast Reactor (EFR) is an advanced sodium-cooled plant with a total thermal power of about 3,600 MW; the compact primary system is designed in pool-type configuration; passive measures are the guiding principle in the safety area.

The decay heat removal (DHR) concept comprises two diverse systems as shown in Fig. 1.

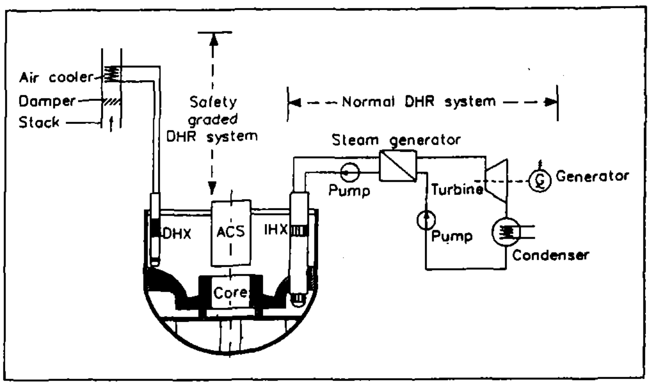

Fig. 1. Normal and safety graded heat removal systems of the European Fast Reactor.

- Under nominal operating conditions, the decay heat is removed via intermediate heat exchangers (IHX) to the water/steam circuits and the final heat sink.
- In the case of an unavailability of the steam plant heat sink or of a total loss of station service power, the decay heat will be removed by the safety graded system, which operates entirely by natural convection. This passive DHR concept is based on six direct reactor cooling (DRC) systems with a total capacity of 6×15 MW operating independently from each other. Each of the systems consists of a sodium/sodium decay heat exchanger (DHX) immersed in the primary vessel and connected via an intermediate sodium loop to a heat sink formed by a sodium/air heat exchanger (AHX) installed in a stack with air inlet and outlet dampers. The decay heat is removed by natural convection on the sodium side and natural draft on the air side.

To demonstrate the capability of the DHR system, many experiments have been carried out in variably scaled test facilities using water and sodium. The group of water tests includes the three-dimensional (3D) vessel experiments RAMONA and NEPTUN used for reactor typical steady-state and transient tests. To make sure that the transfer of such experimental results to practical situations is possible, it is required that the most important phenomena can be taken into account.

In addition, non-nominal conditions of the decay heat removal have been investigated utilizing both models in order to assess the main parameters influencing the thermal-hydraulics in the upper plenum. These are:

- Only half of the available DHXs are in operation for decay heat removal, i.e., the decay heat is removed from the primary tank in a highly asymmetrical mode.
- The above core structure (ACS) is impermeable and permeable, respectively. In case of a permeable ACS, the fluid of the upper plenum penetrates this component.

The test facilities RAMONA and NEPTUN were actually initiated during the former German SNR-2 project. So there are geometrical differences to the EFR. Despite of these, the experiments give an insight into the relevant EFR phenomena and provide a broad basis for the development and assessment of computer programs.

II. TEST FACILITIES

The test facilities are schematically represented in Fig. 2 together with the main dimensions in millimeters. Figure 2a shows a cross-section of the RAMONA apparatus (360°, scale 1:20) and the major components of the primary vessel. The core with a maximum power of 75 kW is composed of nine individually heatable rings forming eight annular flow channels with a width of 2 mm. Downstream of the heated portion having a height of 130 mm, a 70 mm high unheated section is arranged. Straight-tube heat exchangers operating on the counterflow principle serve to simulate eight IHXs and four DHXs. Each of the four speed controlled primary pumps is equipped with two feed lines to the core. About 250 thermocouples (TCs) are installed in the plena and components. Up to now, nearly 200 test runs have been performed in the RAMONA set-up. Forced and natural convection experiments operated under steady-state conditions as well as studies on the transition from forced to natural convection have been carried out. The influence of different design geometries (DHXs [1] and ACS [2]) and operating parameters (core power [3], primary pump flow coastdown, IHX secondary side flow coastdown, DHX startup delay time [4], DHX standby operation [5], etc.) on the thermal-hydraulics have been taken into account. The investigations simulate the conditions after a reactor scram from 40 percent nominal load. This corresponds to a model power of 30 kW. These tests have been supplemented recently by simulating a scram from 100 percent nominal load equivalent to a model power of 75 kW. In all cases the decay heat is removed by the DHXs exclusively.

The NEPTUN test facility (360°, scale 1:5) is schematically represented in Fig. 2b. The core is designed for a maximum power of 1,600 kW. It is simulated in radial direction by 253 heatable fuel rod bundles, 60 reflector elements, 84 heatable storage elements and 252 shielding elements. In this study only the 253 fuel elements on a core diameter of 986 mm are heated. All additional elements are assembled without hydraulic connection to the core inlet flow. Each heatable subassembly consists of 19 rods which are hexagonally arranged in circular tubes with an inner diameter of 50 mm. The heater rods (rod diameter of 8.5 mm and rod pitch of 9.52 mm) have a heated portion of 220 mm and unheated sections of 300 (upstream) and 360 mm (downstream) simulating the fertile and reflector zones, respectively. About 220 mm below the upper edge of the core, a pad plane with a porosity of 6.8 % of the free interwrapper flow area supports the core elements at a prescribed wrapper-to-wrapper pitch. The very detailed modeling of the core geometry allows the investigation of the thermal-hydraulic interaction between the core and the hot upper plenum including

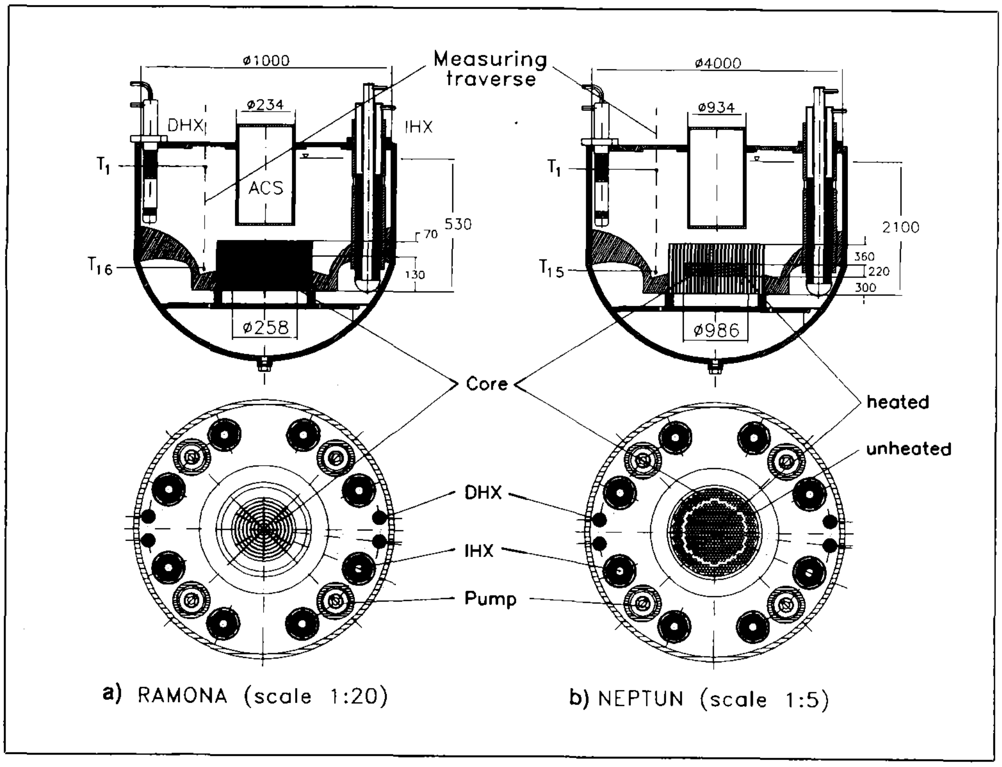

Fig. 2. Geometry of the RAMONA and NEPTUN test facilities with the main dimensions in millimeters.

the interwrapper phenomena. The four DHXs are built as straight-tube heat exchangers operating on the counterflow principle. At the present state, the eight IHXs and four primary pumps serve as dummies to simulate the hydraulic behavior of these components. With other words, only natural convection experiments operated under steady-state conditions are being conducted. About 1,200 TCs are installed in the flow areas of interest.

III. NUMERICAL SIMULATION METHODS

The numerical simulation of both the uniform and the non-uniform cooling modes associated with the core-to-plenum interaction presents a formidable challenge to the computer codes. A wide spectrum of potential methods is being assessed ranging from lumped-parameter systems codes, offering advantageous computing-to-real-time ratios, to 3D codes which offer a high degree of spatial resolution. The number of mesh cells, however, depends on the degree of detail required to resolve the fluid field, the phenomena being modeled, and practical restrictions such as computing time and computer storage limitations.

The highly vectorized thermal-hydraulic computer code FLUTAN [6] has been applied for the numerical simulation of the RAMONA and NEPTUN experiments. The computer code is a tool to analyze combined fluid dynamics and heat transport for three-dimensional, laminar and turbulent, steady-state and transient problems. An essential feature of the computer program is the self-optimizing algorithm CRESOR for solving the Poisson equation. RAMONA and NEPTUN are modeled by a 3D noding scheme representing 90° sectors with about 15,000 and 19,000 volume cells, respectively. Laminar flow conditions are supposed. In simulating the NEPTUN experiments, locally turbulent flows are considered.

The cores of RAMONA and NEPTUN are described by porosities and permeabilities taking into account the heat capacity of the heating elements. The DHXs are modeled by permeabilities and real heat transfer surfaces. The thermal-hydraulic characteristics of the components (core, subassemblies, DHX) are specified for the data input by making use of pretest measurements with the original model components [7].

Figure 3 shows typical velocity fields computed for RAMONA and NEPTUN with a decay heat power of 8 and 133 kW, respectively. The graphs indicate the results for the core/upper plenum DHX cross-section as well as the results for the interwrapper space/upper plenum IHX cross-section. The interwrapper flow can only be shown for NEPTUN as the RAMONA core geometry is composed of annular rings suppressing this flow pattern.

The velocity vectors plotted in Figs. 3a and 3b indicate clockwise flow paths. The vertically upward flowing water is heated up in the core channels. At the top end of the core, the warm flow is being deflected and mixed with the radial cold stream coming from the DHXs. Near the water surface, the warm fluid flows to the DHXs again, where it is cooled down. Cold fluid leaves the DHXs, sinks, and impinges on the intermediate bottom (redan), which separates the upper from the lower plenum. A second part of the warm fluid circulates via the IHX and the pump to the high pressure plenum and enters the core again as illustrated by Fig. 3c. This plot shows also the interwrapper flow. A clockwise circulating flow can be detected within the interwrapper space. The flow is initiated by the heated central part of the core, as the core periphery remains unheated. A vortex can be identified below the pad plane. Above the plane, the cold DHX flow penetrates into the interwrapper space in direction to the center part of the core. The cold stream cools the outlet region of

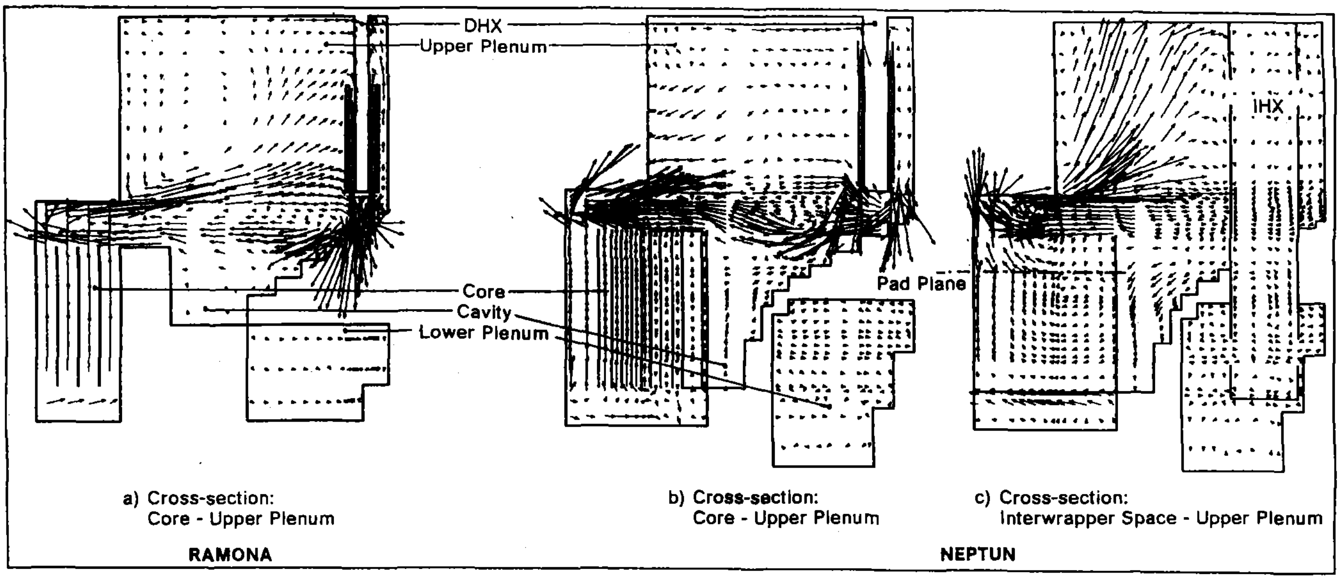

Fig. 3. Computed velocity vectors of the RAMONA and NEPTUN test facilities.

the core and mixes with the outlet flow leaving the core at its upper end.

IV. RESULTS OF THE INVESTIGATIONS

A. Transition from forced to natural convection

Various investigations have been carried out in the RAMONA test facility to study the transition from forced to natural circulation [2,8]. Two typical results are shown in Fig. 4 for a scram simulating a 40% (30 kW) and 100% (75 kW) nominal load case, respectively. The main test conditions are:

- Reduction of core power from 30 to 1 and from 75 to 2 kW, respectively, within 1.5 s after scram.
- Reduction of core flow mass rate from 840 and 2100 g/s, respectively, which corresponds to the half-time of a primary pump coastdown of 10 s. The pumps are stopped at 130 s.
- Reduction of mass flow rate per IHX secondary loop from 90 and 225 g/s, respectively, to 0 g/s within 15 s. At the secondary side of the IHX, the inlet temperature is kept constant at 23 °C.
- DHX startup, delayed by 240 s, of each of the four secondary loops of the DHXs within 160 s by a linear increase in the mass flow rate from 0.5 to 6.4 g/s. At the secondary-side, the mean fluid temperature amounts to 25°C.

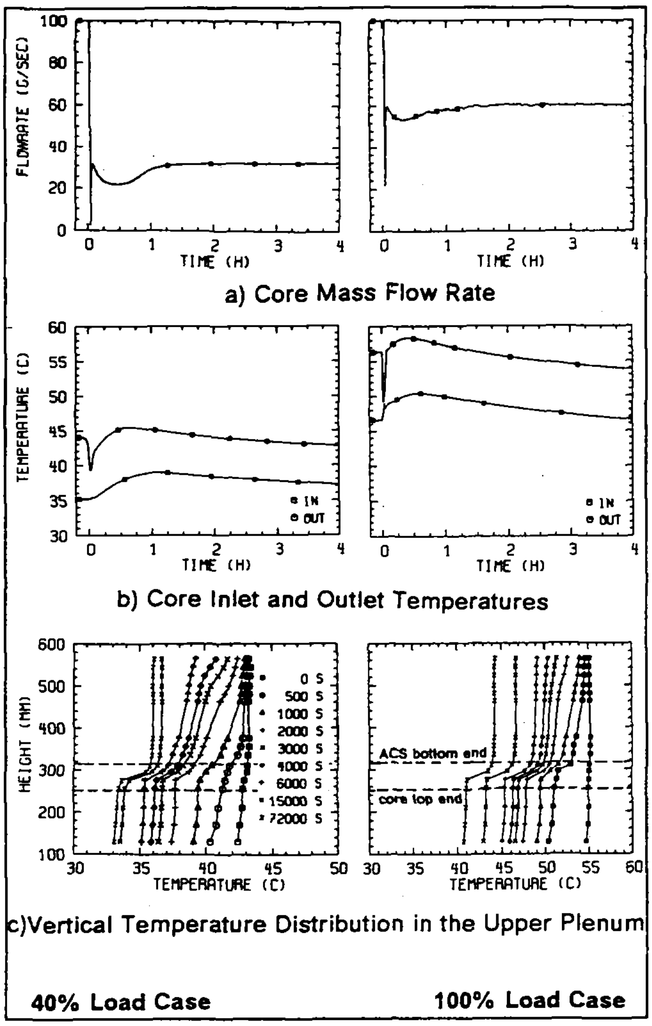

Fig. 4. Influence of the load case on the thermal-hydraulics of RAMONA during the transition from forced to natural convection.

For a given core power, the primary pumps control the core flow and, hence, the core temperatures. The flow rate measurements are shown in Fig. 4a. At the moment of the pump stop (t=130 s after scram), the flow rates drop considerably and begin to rise again with the onset of natural convection. This is caused by the increasing temperature differences in the core (Fig. 4b), which depend on the core power level after scram. The temperatures at the inlet and outlet sides of the core are plotted as the mean values of all thermocouples. The outlet temperatures exhibit a decrease just after scram and a rise with increasing time. The buoyancy forces in the cooling channels increase with increasing core power. Therefore, the measured short- and long-term core flow rates are proportional to the core power after scram. Consequently, the core flow rate is higher in the case of a power reduction from 75 to 2 kW compared to a reduction from 30 to 1 kW. A second small minimum of the core flow rates can be observed 30 min after scram. This flow reduction is caused by the fluid of high temperature in the hot plenum entering the IHX and reducing its natural convection pressure head.

Figure 4c shows the fluid temperatures which have been measured in the upper plenum along the vertical traverse indicated in Fig. 2. Starting with forced convection at 0 s, a uniform temperature profile is registered. This is the scram point where the core power is reduced within 1.5 s from 30 to 1 kW or from 75 to 2 kW. With the time the temperature profiles show an increasing difference between the upper and the lower part of the hot plenum. This temperature difference is strongly influenced by the start of the DHXs operation. 400 s after scram they are in full service and deliver cold fluid into the cavity surrounding the core. The cavity gets colder and colder and cools higher-level positions of the upper plenum. It results in the development of a temperature stratification in the upper plenum. The cold fluid reaching the core outlet level counteracts the hot fluid from the core and, hence, the upward buoyancy forces. This effect superimposes the reduced natural convection pressure head of the IHX, and both effects lead to increases in the core temperature difference and core flow reduction within the first hour after scram. In all cases, a minimum core flow is observed just before the cooler fluid reaches the IHX inlet windows at about 2000 s after scram. As a result, the core flow rate is increased followed by a continuous reduction in the gradient of the temperature increase at the outlet side of the core (Fig. 4b). As soon as cold fluid reaches the core inlet, both core inlet and outlet temperatures begin to decrease. The temperature stratification of the upper plenum is fully developed at about 2 h after scram and steady-state conditions are reached within a time interval of 20 h.

B. Core power

Under steady-state operating conditions of the DHX, a temperature stratification is formed indicated by a sharp temperature gradient above the upper edge of the core. This behavior is demonstrated by the measurements represented in Fig. 5a. The temperatures are registered by the TCs arranged along the vertical traverse as shown in Fig. 2. The data are plotted for RAMONA in the power range of 1 to 3.6 kW and for NEPTUN in the power range of 133 to 221 kW keeping the DHX inlet conditions constant. The profile of the upper plenum temperatures measured in RAMONA and NEPTUN reveal identical physical characteristics for both test facilities. Figure 5b shows the results of analytical predictions using the FLUTAN computer code. The temperature profiles are obtained for the same positions as for the measurements. These computed temperature distributions represent as well temperature gradients in the region between core outlet and the bottom end of the ACS. In general, these gradients are a little smaller than the measured ones.

Detailed temperature field measurements available from steady-state tests provide the basis for isotherm plots of the upper plenum behavior. Fig-

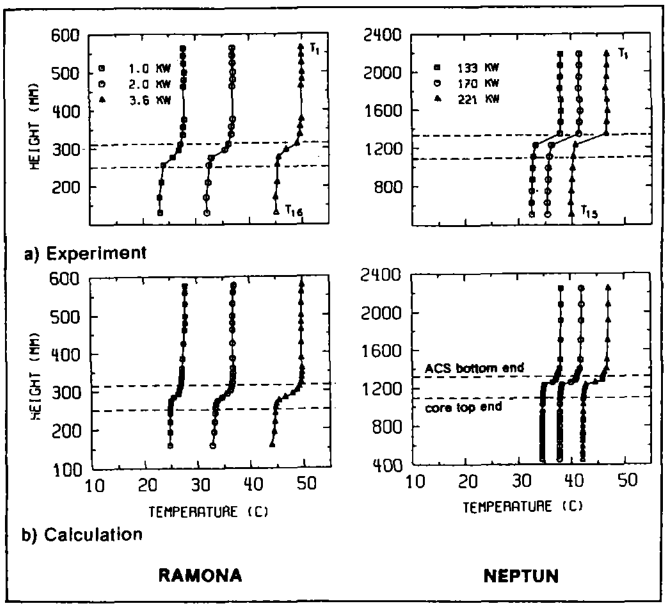

Fig. 5. Influence of the core power on the measured and computed temperatures in the hot plena of the RAMONA and NEPTUN test facilities.

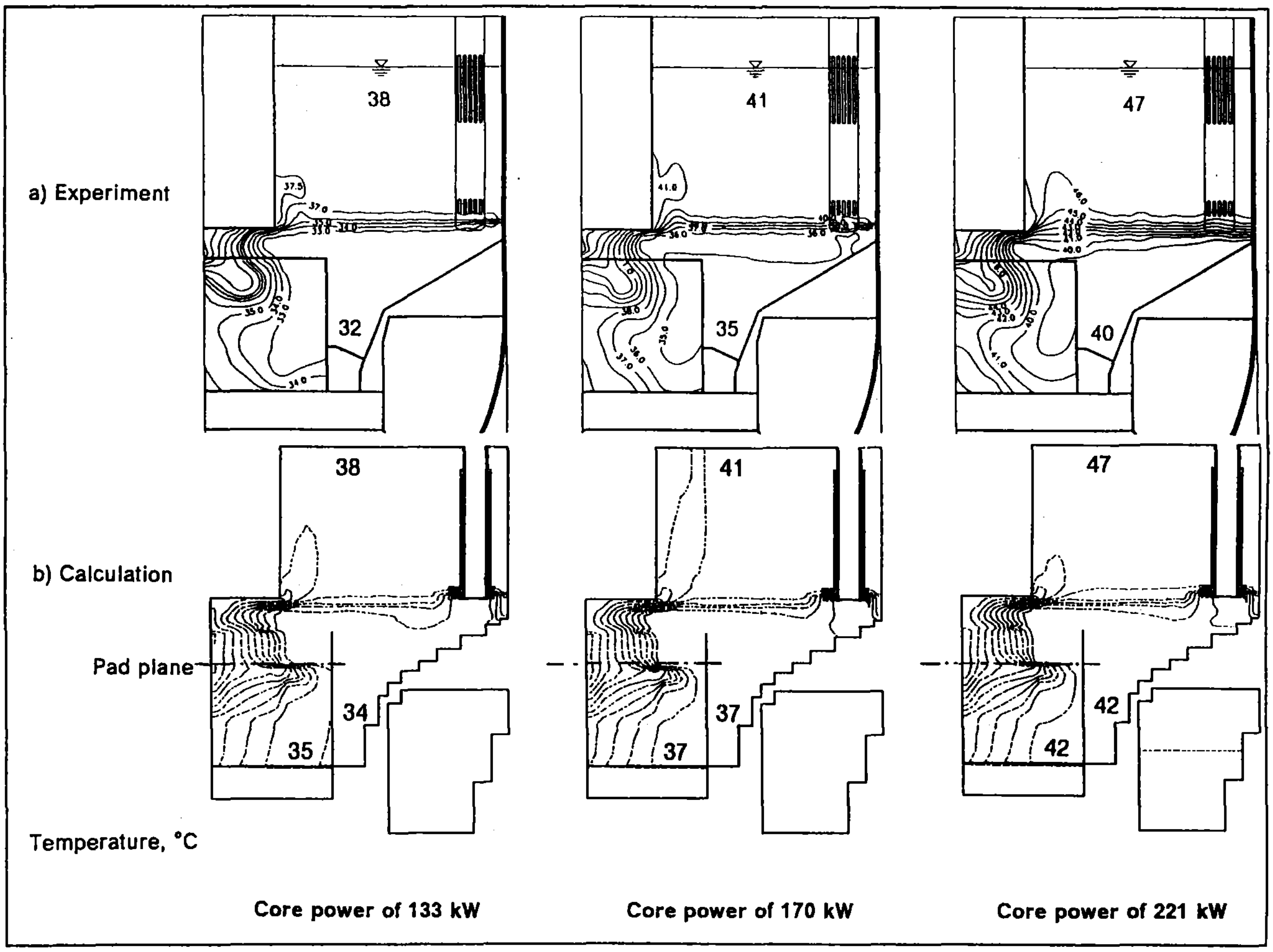

Fig. 6. Influence of the core power on the measured and computed isotherms of the NEPTUN test facility.

ure 6 shows such graphs for the NEPTUN investigations only. The experimental results and the FLUTAN calculations reflect the horizontal character of the isotherms indicating the formation of a stratified flow in the upper plenum. The close spacing of the isotherms (intervals of 1 K) also points out that the entire temperature rise occurs in the region between top end of the core and outlet windows of the DHXs. The highest temperatures occur above the core, and they drop with increasing distance from the core center-line. The results of NEPTUN differ from those registered for RAMONA [7] in the following two points:

- At the DHX outlet windows, RAMONA shows a very inhomogeneous field of isotherms resulting in large temperature gradients at the bottom of the redan. This could be demonstrated by very detailed local measurements. In contrast to this, the isotherm field measured in NEPTUN show a horizontal extension over the outlet region of the DHX. The redan is always found to be at identical temperatures.
- In NEPTUN, interstitial flow is possible. This means that cold fluid delivered during the operation time of the DHX passes the interwrapper space. The cold flow results in the temperature distributions typical for the experiments. The cold interstitial flow influences the uniform distribution of the core temperature and contributes remarkably to the decay heat removal. In RAMONA, interstitial flow is excluded due to the annular geometry of the cooling channels.

Comparing the computed and measured NEPTUN results, it can be seen that the temperature distributions in the upper plenum are well represented by the calculations. In the interstitial flow region of NEPTUN, the agreement between calculation and experiments is satisfactory for these first calculations.

C. Asymmetric decay heat removal

Fundamental pretests have been performed in a slab geometry to investigate the in-vessel cooling modes caused by the position and number of DHXs being in operation [9]. A highly asymmetrical decay heat removal occurs under the condition that only two out of four pairwise installed DHXs serve as heat sink. To assess the capability of the passive DHR system, reactor typical experiments have been carried out in the NEPTUN test facility for a core power of 133 kW. Figure 7 shows a comparison of local temperatures measured during both uniform and non-uniform heat removal. In Fig. 7a, core inlet and outlet temperatures are plotted versus the core radius. In comparison to the symmetrical cooling mode, the temperature level of the asymmetrical case is roughly 12 K higher. The reasons are the core power and the inlet conditions at the secondary side of the DHXs which have been kept constant for both cases. In the upper part of Fig. 7b, temperature measurements are represented along the vertical thermocouple positions indicated in Fig. 2. The lowest temperatures have been recorded in the cavity. Between the top end of the core and the bottom end of the ACS, the measurements show a nearly continuous increase of the temperatures due to the mixing process of the cold and warm fluid streams. Almost uniform temperatures are prevailing in the upper plenum. The lower diagram of Fig. 7b illustrates that even under non-uniform cooling conditions the tempera-

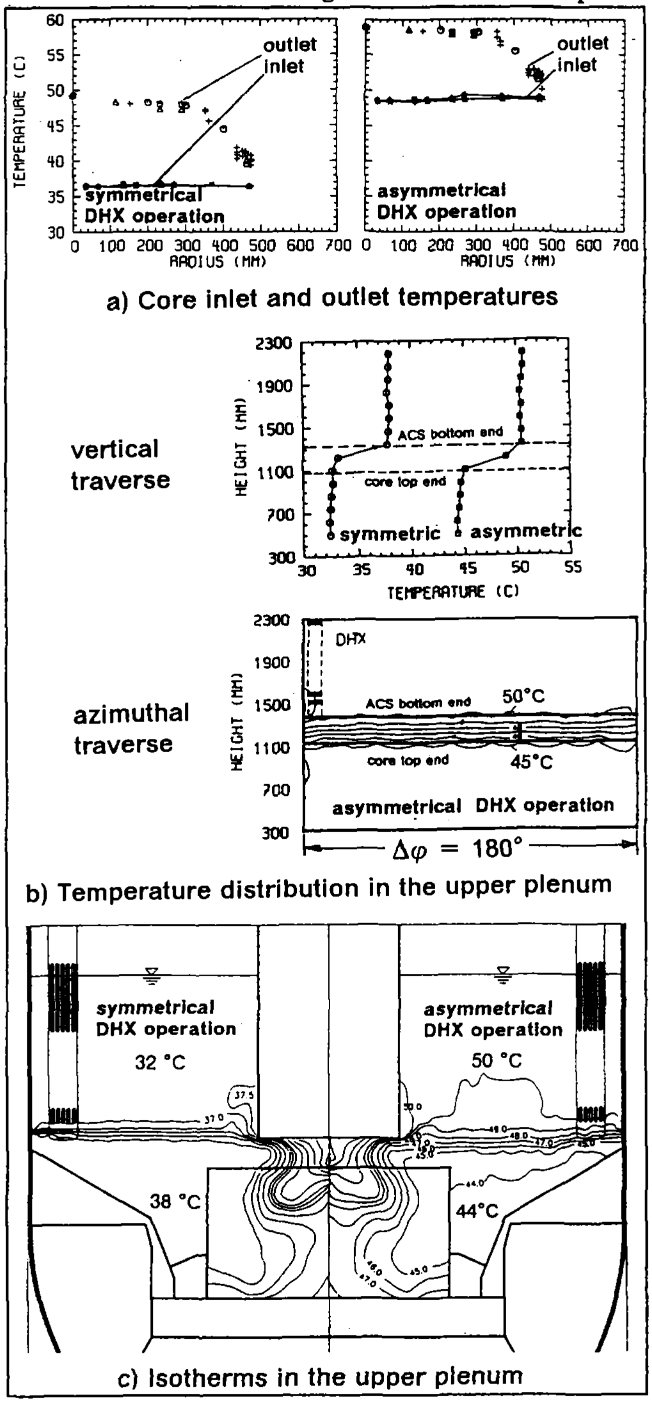

Fig. 7. Influence of the symmetrical and asymmetrical operation of the decay heat exchangers on the measured temperatures in the hot plenum of the NEPTUN test facility for a core power of 133 kW.

ture behavior remains nearly unaffected in azimuthal direction. These temperature profiles have been gained by rotating the measuring device clockwise through 180 degrees. Such detailed measurements of the temperature field serve to plot the isotherm field with an isotherm interval of 1K. Figure 7c shows the establishment of a stratified flow on the way from the DHX outlet region to the top end of the core. The close spacing of isotherms reflects in additon that the whole temperature rise mainly takes place in the region between the core and the ACS. The highest temperatures occur above the core at its center-line and decrease with increasing radial distance from this area. The field of the isotherms doesn't vary with the azimuthal measuring position except near the outlet region of the DHXs.

D. Design of the Above Core Structure

The results discussed so far refer to a non-permeable surface of the ACS. In the following, the corresponding results using a permeable ACS are described and compared to the previous findings. In this case, a part of the upper plenum fluid penetrates the ACS. The permeability of the ACS amounts to $\varepsilon_1 = 15\%$ at the cylindrical surface and $\varepsilon_2 = 12.5\%$ at the bottom.

For the permeable and non-permeable ACS, Fig. 8a shows the temperature distribution for the vertical thermocouple positions indicated in Fig. 2. It can be seen that the temperature in the cavity are about 2 K lower in case of a permeable ACS than in tests with impermeable ACS surfaces. The temperature gradient between the head of the subassemblies up to the bottom end of the permeable ACS is reduced. With increasing height, however, the temperatures of the upper plenum rise. That is in contrast to the findings of a non-permeable ACS. The reason for this behavior is attributed to the fact that the radial cold flow leaving the DHX deflects the upward directed hot flow under mixing and penetrates into the ACS at its bottom end. This can be readily seen from the isotherm fields represented in Fig. 8b.

As a consequence of this flow behavior, the highest temperatures occur at the center-line of the model. A stratified flow is present throughout the upper plenum. The highest temperature gradient is still in the region between the top end of the core and the bottom end of the ACS. In contrast to the impermeable ACS, however, the gradient is diminished. These findings are confirmed by the numerical simulations. Figure 8b shows the calculated isotherms. A comparison of the analytical predictions against the experimental data reflects a good agreement of the findings.

V. SUMMARY

In order to assess the capability of the passive decay heat removal system design for an advanced pool-type liquid-metal-cooled reactor, water experiments were conducted in the RAMONA and NEPTUN test facilities. Studies on the transition from forced to natural convection as well as natural ciruclation tests performed under quasi steady-state

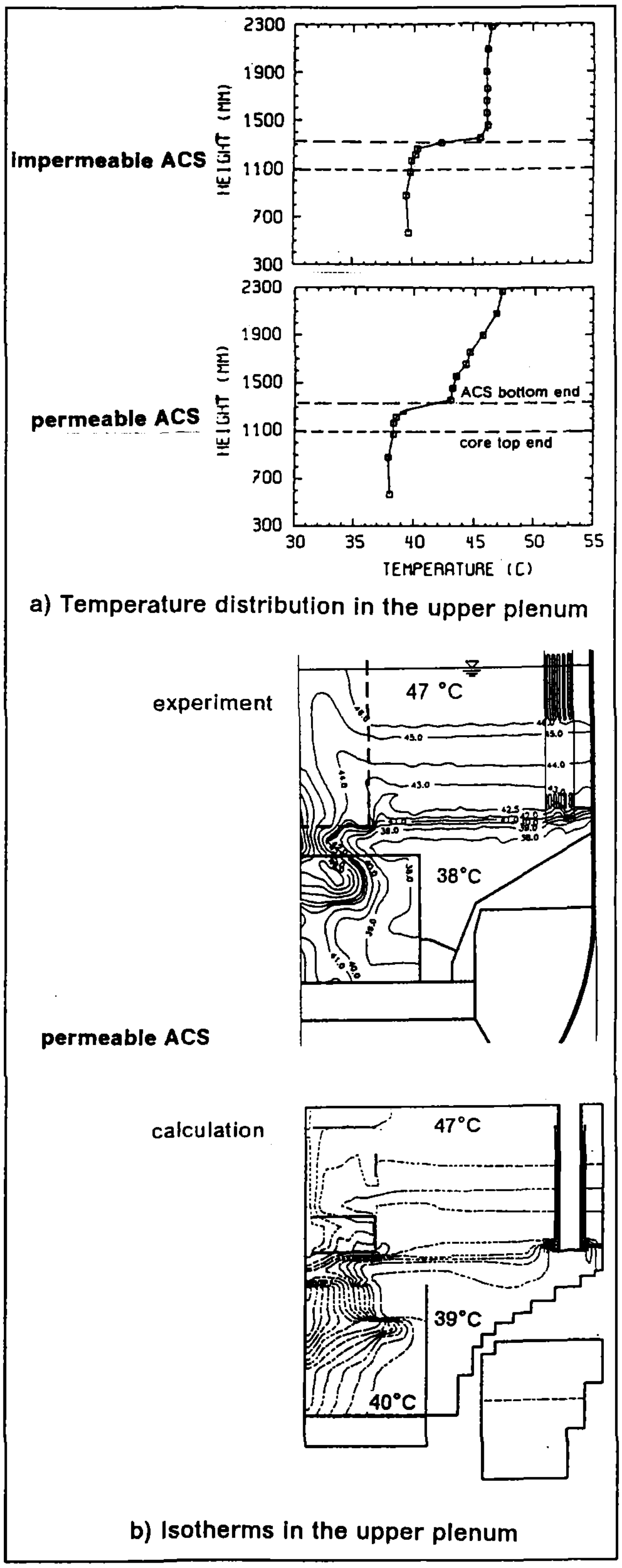

Fig. 8. Influence of the design of the above core structure on the measured temperatures in the hot plenum of the NEPTUN test facility for a core power of 221 kW.

conditions were carried out. The influence of different operating and design parameters on the thermal-hydraulic behavior were taken into account. Analytical simulations utilizing the computer program FLUTAN were compared against the experimental data. The investigations lead to the following statements:

- The transient DHR tests performed in RAMONA were completed with tests using a prescram power of up to 75 kW. Generally, the anticipated primary system behavior is confirmed.
- The NEPTUN model forms the link between the small RAMONA rig and the full-size reactor plant. The thermal-hydraulics of the upper plenum are found to be identical in both models. The interwrapper thermal-hydraulics simulated in NEPTUN contributes to the decay heat removal.
- Asymmetric heat removal (only two DHXs being in operation) results in a temperature increase in the system, but, doesn't lead to changes in the general temperature distribution.
- The permeable ACS shell influences the flow paths and temperature distributions in the upper plenum. Such a design causes an altered temperature stratification.
- Analytical predictions utilizing the FLUTAN computer code lead to a remarkably good agreement of the data if the geometrical and operational conditions are modeled accurately. Deviations mainly result from the lack of nodalization which has to be accepted for reasons of the required computing time.
- So far, all water model tests show that decay heat can be safely removed by natural convection. Additional sodium and water tests under identical geometrical and flow conditions are running to guarantee the transferability of the results.

REFERENCES

1. H. Hoffmann, K. Marten, H. Ohshima, D. Weinberg, "Investigations on Natural Convection with the 3D Model RAMONA. Influence of the Configuration of the Immersion Coolers on the Upper Plenum Thermohydraulics and Core Mass Flows", 7th Proc. of Nuclear Thermal-Hydraulics: 1991 Winter Meeting of the American Nuclear Society, San Francisco, CA, U.S.A., November 10-14, 1991, pp- 212-220.

2. H. Hoffmann et al. Investigations on Decay Heat Removal by Natural Circulation in LMFBR's, Fourth Internat. Top. Meeting on Nuclear Reactor Thermal-hydraulics, NURETH-4, Karlsruhe, Germany, Vol. 1, pp. 384-391, October 1989.

3. H. Hoffmann, K.Marten, D. Weinberg, Y. Ieda, "Investigations on Natural Circulation in Reactor Models and Shut Down Heat Removal Systems for LMFBR's", 5th Proc. of Nuclear Thermal-Hydraulics, 1989, Winter Meeting of the ANS, San Francisco, CA, U.S.A., Nov. 26-30, 1989, pp. 448-454.

4. H. Hoffmann, K. Marten, D. Weinberg, H. Kamide, "Thermal-hydraulic Model-Experiments and Calculations on the Transition from Forced to Natural Circulation for Pool-type Fast Reactors", Nuclear Technology, 99 (1992) pp. 374-85.

5. K. Marten, D. Weinberg, H. Hoffmann, B. Kohtes, "Die Thermohydraulik im Primärsystem des EFR bei mitlaufenden Tauchkühlern im Leistungsbetrieb", (in German) Jahrestagung Kerntechnik '93, Köln, 25.-27. März 1993, pp. 105-108.

6. H. Borgwaldt, W. Baumann, G. Willerding, "FLUTAN, Input Specifications", KfK 5010, Kernforschungszentrum Karlsruhe, Germany (1992).

7. H. Hoffmann et al., "Scaled Model Studies of Decay Heat Removal by Natural Convection for the European Fast Reactor". 6th Internat. Topical Meeting on Nuclear Reactor Thermal-hydraulics, (NURETH-6), Grenoble, France, Oct. 5-8, 1993.

8. D. Weinberg, H.H. Frey, H. Tschöke, K. Rust, "Comparison of Transient 3D Calculations with RAMONA Model Experiments". Proc. of the 5th Internat. Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-5), Salt Lake City, UT, U.S.A., September 21-24, 1992, Vol. II, pp. 571-79.

9. K. Rust, H. Tschöke, D. Weinberg, K. Becker, "Influence of Immersion Cooler Positions on the Thermal-hydraulics in a Slab Geometry, - A Comparison of Analytical and Experimental Data -", 6th Internat. Topical Meeting on Nuclear Reactor Thermal-hydraulics (NURETH-6), Grenoble, France, Oct. 5-8, 1993.