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EXPERIMENTAL STUDY OF THE TRANSITION FROM FORCED TO NATURAL CIRCULATION IN EBR-II AT LOW POWER AND FLOW†

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ABSTRACT

A series of tests have been conducted in EBR-II which studied the dynamics of the transition from forced to natural circulation flow in a liquid-metal-cooled fast breeder reactor (LMFBR). Each test was initiated by abruptly tripping an electromagnetic pump which supplies 5-6% of the normal full operational primary flow rate. The ensuing flow coast-down reached a minimum value after which the flow increased as natural circulation was established. The effects of secondary system flow through the intermediate heat exchanger and reactor decay power level on the minimum in-core flow rates and maximum in-core temperatures were examined.

INTRODUCTION

During the normal operation of a liquid-metal-cooled fast breeder reactor (LMFBR), fuel temperatures are controlled within prescribed limits by the forced circulation of liquid sodium. However, during certain events postulated to occur during the operating life-time of a reactor plant, power to the primary pumps (which maintain the forced flow) can be lost. As the forced flow starts to decrease, the plant protective system will, with extremely high reliability, shut down the plant. However, due to the generation of radioactive material within the fuel and structural components of the reactor during power operation, a certain amount of residual decay heat will be released, even though the primary fission reactions have ceased. Therefore, there must be some continued circulation of coolant, albeit at a much reduced rate, following this event in order to prevent an undesirable overheating of the reactor.

Prudent engineering design necessitates the inclusion of at least one back-up system to provide sufficient coolant flow to maintain reactor temperatures within prescribed limits. Particular systems which have been designed include redundant electrical power supplies, "pony-motors" on the primary pumps which may be powered by diesel generators, auxiliary pumps driven by storage batteries, or other related concepts, all of which provide some form of emergency electrical or mechanical energy supplies. However, in order to assure continued reactor coolability even in the extremely remote event of a complete failure of these types of engineered safe-

guards, the heat transport circuits of the plant are designed to promote natural circulation of the coolant driven by buoyancy forces. Thus, the natural circulation capability of a plant is the ultimate, "fall-back" heat transport mechanism of reactor cooling.

In this context, it then becomes either desirable or necessary (depending upon the licensing process) to demonstrate by analysis, simulation experiments, and/or actual plant tests that a particular plant design is capable of benignly surviving a transition from forced flow, power operation to natural convective decay heat removal. The use of analytical methods presupposes the existence of experimental data against which the modeling and assumptions used in the codes can be tested. Unfortunately, there are very little data available for liquid metals undergoing forced to natural convective transitions, especially in geometries relevant to reactors. One source of information is from the work of Kerian, *et al.* [1] in which loss of forced flow transients with constant heat flux were studied utilizing NaK in a vertical, rectangular channel. In these tests, the channel had a cross section of 6.4 by 76.2 mm and was heated over a 254 mm height of the broad faces. The experiments examined the transition between forced and natural or mixed convective flow caused by an elimination or reduction in the externally applied axial pressure gradient. During these transients, it was found that the flow rate decreased to a level less than the ultimate steady-state natural convective value, resulting in transient temperatures exceeding the final steady-state natural convective values. The flow under-shoot effect was apparently caused by a delay in the establishment of the increased temperature of the riser above the test section as the flow rate decreased, thus resulting in a delay in the development of the buoyancy forces. The magnitude of the resultant overheating in the test section during the transient relative to the final steady convective condition was found to increase with increasing Grashof number (or increasing heat flux) and increasing rate of flow coastdown. Additionally, it was observed that the effects of buoyancy remained negligible during the transient until the ratio Gr/Re^2 reached a value of about 250 (where Gr and Re are the Grashof and Reynolds numbers, respectively).

In a similar test [2] which was conducted in the Experimental Breeder Reactor No. II (EBR-II) and involved a sudden transition from operation at an initial steady-state of 5.3% of full flow (auxiliary primary pump on) and 1.6% of full power to a final purely natural convective state at 1.7% of full flow, a significant undershoot in flow rate and corresponding overshoot in temperature occurred when the pump was

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abruptly de-energized. During the transient period, the total reactor coolant flow rate dropped to 1.1%, while that in an instrumented fueled subassembly (XX07) dropped to 0.7% of their full values.

In general, the results obtained from this EBR-II plant test agreed quite well in a qualitative sense with the out-of-pile coastdown tests with NaK [1]. In both situations, a significant delay in the establishment of natural convective flow occurred following the loss of forced flow which could be attributed to the gradual establishment of the hot leg (upper reactor internals and/or outlet piping) temperatures corresponding to the final steady natural convective conditions.

Natural circulation experiments have also been conducted in other sodium-cooled reactors, e.g., SEFOR [3], HALLAM [4], Rapsodie [5], Phenix [6], KNK [7], and PRF [8]. However, because of a lack of in-core instrumentation and a lack of documentation of sufficient detail of the specific test results, these experiments are of limited value to the general technical community. The purpose of this paper is to present the results of an extensive series of transient natural circulation tests conducted in EBR-II which greatly extend the data presented in [2]. This series of tests, all of which were conducted at low power and flow conditions, investigated the effects of decay power level, initial temperature (or buoyancy) distribution in the primary heat transport circuit, secondary heat transport system dynamics, and intersubassembly phenomena upon the transition from forced to natural convective flow.

DESCRIPTION OF EXPERIMENTS

Reactor and Instrumentation Description

The Experimental Breeder Reactor No. II (EBR-II) is a sodium-cooled fast breeder reactor with a complete steam-electric system. The plant is located in the Idaho National Engineering Laboratory (INEL) and is operated by Argonne National Laboratory for the U. S. Department of Energy. The reactor consists of 16 rows of subassemblies, the inner 7 constituting the fueled core and the outer nine containing reflector and blanket subassemblies. The active fuel length in the core is 0.343 m, with 91 elements in a metal fuel driver subassembly. These driver fuel elements are 4.42 mm in diameter and are contained within a hexagonal can which has a flat-to-flat dimension of 56.1 mm. The spacing of the individual fuel elements is maintained by wire wraps with a diameter of 1.24 mm and an axial pitch of 152 mm. The normal operating power of the reactor is 60.0 MWt at a total coolant flow rate of 0.516 m³/s. Additional descriptive information on EBR-II is available in [9] and a sketch of the primary heat transport circuit is shown in Fig. 1.

The instrumentation utilized in the EBR-II tests includes both the normal plant sensors and special in-core sensors. The in-core instrumentation was located in a modified driver subassembly placed in a converted control-rod position in the fifth row of the core. This experimental subassembly, designated XX08, consists of 61 elements, 58 of which are fueled, and are contained within a hexagonal can which measures 46.4 mm across its inside flats resulting in a pin bundle hydraulic diameter of 2.76 mm. Within this subassembly, there are two inlet, permanent magnet flowmeters, six fuel centerline thermocouples, referred to as "FTCs," located 21.6 mm below the top of the fuel (at beginning of fuel life), and 16

Fig. 1 Schematic Diagram of the Primary Heat Transport Circuit of EBR-II

coolant thermocouples mounted as wire wrap spacers. Two of these coolant thermocouples, called "BTCs," are located at the core bottom, two, called "4TCs," are at 0.4 of the core height (0.137 m above core bottom), two, called "7TCs," are at 0.7 of the core height (0.240 m above core bottom), nine, called "TTCs," are near the core top (same height as the fuel thermocouples), and one, called "15TC," is at 1.5 of the core height (0.514 m above core bottom or 0.171 m above core top). In addition, two coolant thermocouples, called "OTCs," are located above a flow mixer near the subassembly exit, but within the hex can and 0.933 m above core bottom, and measure the mixed-mean coolant outlet temperature. A schematic diagram indicating the axial location of the XX08 instrumentation is shown in Fig. 2. Additional details on the instrumented subassembly XX08 are available in [10].

The general plant instrumentation is calibrated on a periodic basis, however, special calibration efforts were undertaken for the tests reported in this paper. The temperature sensors (including both thermocouples and resistance thermometers) and their associated signal conditioning equipment were checked for off-sets by operating the plant at full (100%) flow and extremely low decay power. This resulted in an isothermal primary system (except for the slight temperature rise across the core which was accounted for) and all of the signals from the temperature sensors could thereby be corrected. With only a few exceptions, the corrections obtained from this calibration were less than 1 C, and therefore, it is felt that all of the temperature measurements herein reported have uncertainties of less than about ± 0.5 C. Although the flowmeters were not calibrated in-place, extensive out-of-pile calibration tests were performed for those units located within the XX08 subassembly. Thus, the sensitivity of these flowmeters is known explicitly over the range of about 0.3×10^{-4} to 25×10^{-4} m³/s (about 1 to 100% of full flow), and was obtained for lower flow rates by extrapolation between 0 and 0.3×10^{-4} m³/s. The calibration uncertainty was estimated to be about $\pm 0.2\%$ for flow rates

Fig. 2 Schematic of the Instrumented Fueled Subassembly, XX08, Including Sensor Locations

greater than about $6 \times 10^{-4} \text{ m}^3/\text{s}$, about $\pm 1\%$ for flow rates between 0.3×10^{-4} and $0.6 \times 10^{-4} \text{ m}^3/\text{s}$, and about $\pm 5\%$ for lower flow rates. Because of the additional uncertainties involved with the signal conditioning equipment and the actual reactor environment, it is estimated that the final reported flow rates have inherent uncertainties of about twice these calibration values.

General Experimental Procedures

The series of natural circulation transients described in this paper were limited to relatively low initial reactor powers and flow rates. The rationale for conducting tests under these conditions was primarily one of conservatism, i.e., since these transients were the first such events experienced by EBR-II, they were specifically planned to result in relatively low peak temperatures. Thus, the potential risk to the reactor and the in-core irradiation samples was essentially negligible. Following the successful completion of these transients, subsequent natural circulation experiments could then be conducted at higher power levels with justifiable confidence.

The basic purpose of the present tests was to experimentally ascertain the effects of reactor decay power level, initial buoyancy distribution in the primary heat transport circuit, primary-secondary system thermal-hydraulic interactions, and intersubassembly phenomena upon the transition from forced to natural circulation. In order to accomplish this objective while still satisfying the necessary conservatism as described in the preceding paragraph, the following general experimental procedure was followed: 1) at

the end of a normal full power operating run, the reactor was shut down, 2) a wait period ranging from about one hour to seven days was used to permit the decay heat load to drop to the desired level, 3) the primary pumps were shut down and the secondary pump adjusted to the specified value (primary forced flow now being provided only by the primary auxiliary pump), and 4) the actual transient initiated by disconnecting the electrical power supply to the auxiliary pump. Electrical power to the secondary pump was either continued or also disconnected according to the specific test requirements. Thus, all of these tests were initiated from an initial condition of approximately 5-6% of the rated primary forced flow (100% primary flow equals $0.516 \text{ m}^3/\text{s}$), secondary forced flow of 2.6% to 10.4% of rated flow (100% secondary flow equals $0.379 \text{ m}^3/\text{s}$), and a reactor decay power level of 0.16% to 1.6% of rated (100% power equals 60.0 MWt). The specific test conditions and resulting temperature and flow rate measurements are discussed in the next section.

EXPERIMENTAL RESULTS

Five separate transient natural circulation tests were conducted in the current program with most of them consisting of a sequence of pump trips and coast-downs (referred to as "phases"). As discussed previously, these tests were conducted over a range of reactor decay power levels and secondary system flow rates. A summary of the initial conditions of several of the most important parameters are shown in Table 1. The test described as "F" was conducted as part of an earlier EBR-II experiment and utilized a similar instrumented subassembly (XX07) to obtain local in-core temperature and flow rate data. The following sections of this paper will discuss the details of these tests in terms of the major parametric effects of plant operation upon the extremums of core temperatures and flow rates.

General Characteristics of Transient Behavior

Although this test series was conducted over a range of reactor decay power levels and secondary system flow rates, the general characteristics of the in-core transient behavior were similar from test to test. A typical relationship is that in Fig. 3 where the XX08 in-core flow rate vs. time curve is shown for transient 1E-1. The in-core flow decreases very rapidly when the power to the auxiliary pump is turned off. The minimum value (the flow "undershoot") is reached as buoyancy forces sufficiently develop in order to establish natural circulation flow. Later sections of this paper will show the dependence of the minimum flow rate on the decay power level and on the secondary system flow rate.

Figure 4 shows the in-core temperatures at various axial locations in XX08 for the same transient. These data clearly show the peaking of the coolant temperatures at each axial level within the subassembly and that the time required to reach the maximum temperatures increases with increasing axial distance. The magnitude of the peak coolant temperatures as functions of decay power level and secondary system flow is discussed in further detail in subsequent sections of this paper.

The redistribution of subchannel coolant flow within a subassembly and the resultant effect on the radial temperature profile under natural circulation conditions has been discussed in [11]. This effect was again observed during the XX08 test program.

TABLE 1. Initial Conditions for Natural Circulation Transients

Test-Phase	Reactor Decay Power (%) ¹	Reactor Primary Flow (%) ²	IHX Primary Inlet Temp (C)	Sec. System Flow (%) ³	IHX Sec. Inlet Temp (C)	IHX Sec. Outlet Temp (C)
1A-1	0.16	5.40	374	3.3	261	375
1A-2	0.16	5.60	374	5.1	267	375
1A-3	0.16	5.77	370	7.0	274	371
1A-4	0.16	5.99	364	10.4	292	356
1B-1	0.34	5.47	374	3.3	267	375
1B-2	0.34	5.68	375	7.8	278	377
1C	0.34	5.91	370	7.8	297	370
1D-1	0.68	5.87	379	9.8	294	381
1D-2	0.66	6.04	380	7.3	300	380
1D-3	0.64	5.67	378	3.1	299	380
1E-1	0.19	5.48	374	2.6	264	374
1E-2	0.19	5.58	370	5.8	278	370
1E-3	0.19	5.71	366	6.6	289	366
1E-4	0.19	5.95	361	9.0	298	357
F	1.60	5.30	370	2.0	-	-

¹The values are expressed in percent of full reactor power of 60.0 MWt.

²The values are expressed in percent of full reactor flow of 0.516 m³/s.

³The values are expressed in percent of full secondary flow of 0.379 m³/s.

Fig. 3 Transient Coolant Flow Rate in XX08 During Test 1E-1

Fig. 4 Transient Coolant Temperatures in XX08 During Test 1E-1

Coolant temperatures measured at various times during the loss-of-flow transient 1E-1 for several of the top of core (TTC) locations are shown in Fig. 5. The data shown here also include information obtained under steady-state full power and flow conditions during the reactor run preceding the natural circulation test. (Actual temperatures can be compared here because the coolant inlet temperature for transient 1E-1 was essentially the normal operating value of 371 C.) These data show that the radial temperature profile which exists under full power conditions becomes essentially flat during these transients. The hot channel peaking factors therefore are significantly reduced and in fact, approach a value of unity under these conditions.

Effect of Intersubassembly Heat Transfer

The importance of intersubassembly heat transfer in reducing radial temperature gradients during natural circulation events has been previously investigated and reported in [11]. The data presented in [11] showed that coolant temperature differences from subassembly-to-subassembly are significantly reduced when an LMFBF changes from a condition of high power and high forced flow to low power and natural circulation flow. This section will discuss new experimental data which have been obtained under different irradiation conditions and further substantiate and quantify the importance of intersubassembly heat transfer during

Fig. 5 Transverse Coolant Temperature Profiles in XX08 at Full Power and Flow and During a Natural Circulation Transient (Test 1E-1)

natural circulation transients. These data were obtained by conducting two separate, but nearly identical, transient natural circulation tests with the fueled, instrumented subassembly XX08 in the EBR-II core. The significant difference in these two tests was that for the first test (1A), the XX08 subassembly had not been previously irradiated whereas it had been irradiated prior to the second test (1E).

Test 1A was initiated approximately 6-3/4 days after the shutdown of the preceding reactor run. At this time the overall fission product decay power was calculated to be 0.16% of the full operational power. Although the XX08 subassembly had not been irradiated prior to Test 1A and therefore did not constitute a fission product decay source, it was a susceptor of gamma and beta activity emitted by the surrounding subassemblies which had been irradiated. Calculations done for these conditions yielded a value of 145 watts for the XX08 power. This power corresponds to approximately 21% of the decay power which would have been present if XX08 had been irradiated during the preceding reactor run. During the initial forced flow part of Test 1A therefore, XX08 was much cooler than the average of the surrounding subassemblies. Test 1E was begun five days after a reactor shutdown when the estimated decay power was 0.19%. For XX08, this value corresponds to a power of approximately 750 watts. Because XX08 and the surrounding fueled subassemblies all had the same percentage power and flow during the forced flow parts of Test 1E, their coolant temperature rises were in the same ratio as for full power and flow conditions. As XX08 was designed to operate with a higher coolant temperature than the surrounding subassemblies under full power conditions, it also had a higher coolant temperature during the forced flow portions of Test 1E. It was believed that by conducting two tests in which XX08 was cooler than its surroundings in one and warmer in the other, the effects of intersubassembly heat transfer and flow redistribution during natural circulation transients could be observed and separated from other factors.

Some typical results are shown in Fig. 6 where the measured transient flow rate through the XX08 instrumented subassembly is shown for the first phase of Tests 1A and 1E. Each of the other three phases (where each phase consists of a coastdown from forced

to natural circulation flow at a different value of secondary system sodium flow through the intermediate heat exchanger) yielded similar results. It is apparent from an examination of this figure that the two curves are nearly identical in shape and value, including the "undershoot" at the transition from forced to natural circulation flow. An implication of this result is that whatever dynamic flow redistribution occurred during these tests, it was not significantly affected by the local power level in XX08.

The transient coolant temperatures which correspond to these coolant flow rates are shown in Fig. 7. These particular temperatures were obtained from thermocouples located at the top-of-core (TTC) and at the outlet of the XX08 subassembly (OTC). A distance of approximately 0.6 m separates these two thermocouple locations. The different effects of intersubassembly heat transfer in the two tests can be seen in the relationship between the top-of-core-thermocouples (TTC) and the outlet thermocouples (OTC). For Test 1A where XX08 was cooler than its surroundings, the initial OTC reading is higher than the TTC value. This indicates that heat is transferred into XX08 in the 0.6 m long, nonfueled region separating these thermocouples. For Test 1E on the other hand, the TTC reading is higher than the OTC, thus indicating heat is transferred out of XX08 to the cooler surroundings.

The overall shape of each set of curves is similar to the other with the peak top-of-core temperatures reached approximately 2-1/2 minutes after the pump trip and with the peak outlet temperatures delayed until approximately 5 minutes after pump trip. In each test a near quasi-steady-state was attained after about 12 minutes.

Fig. 6 Comparison of XX08 Coolant Flow Rates During Tests 1A-1 and 1E-1

The importance of intersubassembly heat transfer can be seen by noting the relative temperature rises prior to the pump trip and comparing them to those obtained during natural circulation. For example, the top-of-core temperature rises differ by approximately a factor of 5 prior to the pump trip but the peak values of 43 C and 22 C for Tests 1E and 1A respectively differ by only about 95%. A similar result is seen for the outlet coolant temperatures where the peak values of 32 C and 22 C (a 45% difference) were reached even though the initial temperature rises differed by a factor of 4. These data show that under drastically different conditions (XX08 powers differing by approxi-

Fig. 7 Comparison of Top-of-Core and Subassembly Outlet Temperature Transients in XX08 During Tests 1A-1 and 1E-1

mately a factor of 5 and the initial coolant temperature rises by approximately a factor of 4-5) the peak coolant temperature rises across the subassembly during a natural circulation transient differed by only about 50-100%. However, it should be noted that although the local temperature differences at each axial location were markedly reduced under natural circulation conditions, the absolute temperature rises were increased. As the hydraulic characteristics of XX08 or the reactor as a whole had not changed from one test to the other, the only mechanism which can cause such a reduction in temperature gradients is intersubassembly heat transfer.

The data presented above further substantiate the importance of intersubassembly heat transfer in reducing local temperature gradients in an LMFBR under natural circulation conditions. Data such as these can be used to reduce calculated local hot channel factors used in the safety analyses of loss-of-forced-flow transients. Any such reductions in these hot channel factors could result in more efficient and economical design and operation of future LMFBRs while still maintaining a reasonable margin of safety.

Effect of Secondary/Primary System Thermal Interactions

The secondary and primary heat transport circuits in EBR-II are thermally interconnected by an intermediate heat exchanger (IHX). In the IHX, the primary sodium flows downward, so that the net buoyancy of the primary heat transport circuit is aided by the cooling of the sodium in this component. Since the average density of the primary sodium in the IHX is determined by its axial temperature gradient, the dynamics of the heat transfer between the counterflowing primary and secondary sodium will have a direct effect upon the contribution of the IHX to the circuit buoyancy pressure forces. At steady-state full power operating conditions, the primary and secondary flow rates are sufficiently balanced so that the axial temperature gradient within the IHX is approximately linear. However, during situations when the ratio of the secondary-to-primary flow rates is quite large,

most of the heat transfer occurs at the top of the IHX near the primary sodium entrance. This results in most of the primary side temperature drop occurring near the IHX top, and thus in a higher average density of the primary sodium in the IHX than under balanced flows.

Conversely, when the secondary-to-primary flow rate ratio is quite small, most of the heat transfer occurs near the bottom of the IHX, resulting in most of the primary side temperature drop existing near the bottom. This yields a lower average density of the primary sodium in the IHX than under balanced flow. The effect of increased secondary flow rate relative to primary flow rate, then, is to cause an increase in the net buoyancy of the primary heat transport circuit, while a decrease in secondary flow rate causes an opposite effect.

During loss of flow transients the ratio of secondary-to-primary flow rates will change considerably and will affect the transient core temperatures through its influence on the net buoyancy. In this section, the experimentally measured effect of secondary flow rate upon the transient core temperatures and flow rates will be discussed. It should be noted here that the secondary sodium IHX inlet temperature did vary somewhat from test to test, and this variation will affect the primary sodium density. However, in most all cases studied here, this effect was of secondary importance and the data will be represented relative to only the secondary flow rate.

In all but one of the following tests, the secondary flow rate was held essentially constant during the primary system flow coastdown. Thus, the thermal transient in the IHX was driven only by the primary system sodium flow coastdown. Since the initial primary forced flow rate was always about 5 to 6% of rated, and the secondary flow rate varied from about 2 to 10%, ratios of secondary-to-primary flow rates of less than and greater than one were investigated.

A summary of the effect of secondary system flow on the minimum primary system flow rate is shown in Fig. 8. The primary system flow rates shown are the minimum values measured in the XX08 instrumented subassembly and thus represent the flow rate at the onset of natural circulation flow. These data indicate a strong effect of secondary system flow on the minimum in-core flows. This effect is particularly evident in the approximate 5-8% range with asymptotic behavior on either side of this range. The 5-8% range represents a flow rate of 0.019-0.030 m³/s in the secondary system which is roughly comparable to the approximately 0.028 m³/s primary system flow prior to the initiation of the transient.

The effect of secondary system flow on the maximum coolant temperature rises at the outlet of the XX08 instrumented subassembly is shown in Fig. 9. Although no sharp break points are noted such as was seen with the minimum XX08 flow rates, the strong effect of secondary system flow can be seen. Figure 10 shows the same effect with the temperature rises being the average values of the peak measured temperature rises at the outlet of several different in-core subassemblies. All of these experimental observations indicate the important role played by the buoyancy of the primary sodium in the IHX and the strong influence of the secondary flow rate thereupon.

As was noted earlier, one of these transients was conducted with a changing secondary system flow rate. For Test 1C, the primary auxiliary pump and the secondary electromagnetic pump were simultaneously tripped. The secondary system flow therefore coasted down to its natural circulation level of approximately 5% in about

Fig. 8 Effect of Secondary Coolant Flow Rate Upon the Minimum Natural Circulation Flow Rate in XX08

Fig. 9 Effect of Secondary Coolant Flow Rate Upon the Peak Subassembly Outlet Coolant Temperature in XX08

Fig. 10 Effect of Secondary Coolant Flow Rate Upon the Peak Subassembly Outlet Coolant Temperature (Average of 17 Subassemblies)

1 minute. Comparisons of the results from Test 1C with those from Test 1B-2 which had the same initial conditions except for the IHX secondary inlet temperature, show that the minimum in-core flow rates and coolant temperature rises are very nearly equal. These

results indicate that for these particular tests, the dynamics of the secondary system coastdown were too slow to affect in-core conditions during the first minutes of the transient. The quasi-steady-state in-core flow rates obtained after several minutes did, however, show the effect of decreased secondary system flow with values of approximately 2.0 and 1.5% for Test 1B-2 and Test 1C respectively. This resultant relative independence of primary and secondary system convective flow transients is caused by a combination of a very rapid primary flow coast-down rate and a long fluid transport time from the reactor core to the IHX. For the much longer primary flow coastdowns resulting from trips of the primary centrifugal pumps, a much stronger interaction between the primary and secondary systems exists.

Effect of Reactor Power Level

For the types of transients described in this paper, the reactor power level affects the reactor core temperatures and the natural circulation flow in a similar, but somewhat more complex manner than the IHX thermal conditions. When the reactor power is increased at initially steady natural convective flow conditions, the coolant axial temperature gradient in the core will correspondingly increase. However, this causes the coolant density to decrease, i.e., the buoyancy forces to increase. This causes an increased convective flow rate and a lowered coolant axial temperature profile. The net effect, however, is for the coolant temperatures to rise with increasing reactor power. During a loss of forced flow transient at constant reactor decay power, a similar interaction occurs. For larger initial power levels, the initial buoyancy forces are larger due to the increased axial temperature gradient, so that the transition from forced to natural convective flow will occur at a greater minimum flow rate. However, due to higher power level, the rate of heat input to the coolant is also larger. Thus, the net effect of power level upon peak transient coolant temperatures is dependent upon the relative dynamics of the flow de- and acceleration and energy storage and transfer.

The experimental tests were designed so that a variety of decay power levels at given secondary flow rates could be attained and their affect upon the minimum coolant flow rate and peak coolant temperature during the loss-of-forced-flow transients studied. In order to illustrate the effects of reactor decay power on the minimum flow rate at the transition from forced to natural circulation flow, the data of Fig. 8 are cross plotted in Fig. 11 with secondary system flow rate as a parameter. These data indicate a near linear relationship between decay power and minimum in-core flow rate. Each of the curves shown appears to have a different intercept for zero decay power. These different intercepts reflect the contribution of the buoyancy forces in the IHX to the minimum in-core flow. The data point at 1.6% decay power and 2% secondary system flow was obtained in 1974 as part of the XX07 instrumented subassembly natural circulation test program [2]. The measured minimum flow rate of 0.68% is seen to be consistent with the values obtained in the current XX08 test program.

The maximum coolant temperature rises measured at the outlet of XX08 are shown in Fig. 12 as a function of reactor decay power. As before, these data were obtained by cross plotting the corresponding temperature vs. secondary system flow curves. The data for each curve in Fig. 12 appear to indicate a zero temperature rise at zero decay power which is the expected

Fig. 11 Effect of Power Level Upon the Minimum Natural Convection Flow Rate in XX08

Fig. 12 Effect of Power Level Upon Peak Subassembly Outlet Coolant Temperature in XX08

Fig. 13 Effect of Power Level Upon Peak Subassembly Outlet Coolant Temperature (Average of 17 Subassemblies)

result. As with the minimum flow curve, the measured temperature rise obtained during the earlier XX07 test appears to be reasonably consistent with the XX08 test data.

Figure 13 shows the average values of the peak measured temperature rises at the outlet of several in-core subassemblies plotted vs. reactor decay power. These data show the same relationship as was observed for the XX08 outlet temperature data. The point at 90 C and 1.6% decay power represents the XX07 Test F result. Although this test was terminated before all of the outlet temperatures had reached their peak

values, the 90 C value shown here is believed to be approximately the average peak value which would have been reached if the test had not been terminated.

All these data indicate that the slope of the peak coolant temperature vs. decay power curves is positive, but decreases with increasing power. Thus, although increased power levels increase the peak transient coolant temperature, the increased buoyancy caused by the resulting larger axial temperature gradients somewhat reduces the magnitude of this effect by increased convective flow rates. This same phenomena is responsible for the redistribution of coolant flow rates across a reactor core and the resultant reduced transverse temperature gradients observed during natural circulations events in LMFBR's [11].

An interesting observation can also be gleaned from these and related studies in that Le Chatelier's principle appears to be valid for natural circulation. In other words, a system in natural circulation will tend to change in such a manner to reduce the effects of imposed perturbations.

CONCLUSIONS

Each of the tests conducted as part of the natural circulation test program in EBR-II has resulted in a smooth transition from forced to natural circulation flow following an abrupt loss of pumping power. The minimum in-core flow rate has been shown to be a nearly linear function of reactor decay power level. The effect of secondary system coolant flow through the IHX (a parameter representative of the buoyant driving forces in the IHX) was observed to have a strong effect, particularly over the range where initial primary flow and secondary flow were reasonably closely matched. The test data on flows and temperature rises obtained in the current test program appear to be consistent with the data obtained in an earlier test done as part of the XX07 instrumented subassembly program.

The data obtained from these tests is being used in the development of computer models for analytical studies of flow coastdown/natural circulation events. It has also been used as a basis for planning and conducting additional tests which will further enhance our understanding of natural circulation phenomena. The results of these measurements and the analytical predictions will be presented in a forthcoming paper.

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REFERENCES

- 1 Kerian, J. L., Singer, R. M. and Parkinson, T. F., "An Experimental Study of Combined Liquid Metal Heat Transfer During Flow Transients," *ASME Paper No. 72-WA/HT-45* (1972). Presented at the 1972 Winter Annual Meeting of the ASME, New York.

2 Gillette, J. L., et al., "A Flow Cooldown Transient to Natural Convective Conditions in EBR-II," Trans. Am. Nucl. Soc., 22, pp. 594-596 (1975).

3 Unpublished document, Natural Circulation Testing of the SEFOR Reactor Coolant System.

4 Sunderman, I. O., "Hallam Nuclear Power Facility Post Critical Test Completion Report, Scram, and Loss-of-Load," Atomics Intl. Report AI-PC-660 (Feb. 1964).

5 Chetail, J. C., Pointer, R., and Delisk, J. P., "Emergency Cooling of the Rapsodie-Fortissimo Core," ASME Paper No. 74-WA/HT-51 (1974). Presented at the 1974 Winter Annual Meeting of the ASME, New York.

6 Guillemard, B., "Start-up of Phenix," Nucl. Engrg. Intl., 19, No. 216, pp. 411-414 (May 1974).

7 Hopfenfeld, J., U. S. ERDA/RRD, personal communication.

8 Jones, P., Central Electricity Generating Board, Berkeley Nuclear Laboratories, UK, personal communication.

9 Koch, L. J., et al., "Hazard Summary Report: Experimental Breeder Reactor II (EBR-II), Argonne National Laboratory Report No. ANL-5719 (May 1957) and ANL-5719 (Addendum) (June 1962).

10 A. Smaardyk, et al., "EBR-II Environmental Instrumented Subassembly XX08: Engineering and Assembly," Argonne National Laboratory Report ANL-78-9 (May 1978).

11 Singer, R. M. and Gillette, J. L., "Measurements of Subassembly and Core Temperature Distributions in an LMFB," AIChE Symp. Series, 73, No. 164, p. 97 (1977).