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ABSTRACT

The long-term heat dissipation capability of LMFBR plants is investigated by simulating steady-power tests in a number of plants. Two analytical approaches are considered. Advanced thermohydraulic system simulation codes SSC-L and SSC-P are used to study the response of the CRBRP, FFTF and PHENIX plants. Analyses were also applied to actual tests in the PFR and PHENIX reactors. All of these studies have shown that the mixed-mean reactor core temperature rise is proportional to the 0.58 exponent of the steady-power level.

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NOMENCLATURE

a,b	constants in the equation of state for sodium
D	rod diameter, m
De	equivalent diameter, m
f	friction factor
g	acceleration due to gravity, m/s^2
H	wire-wrap lead, m
P	reactor power, MW
Po	normalized reactor power
Re	Reynolds number
T	temperature, C
W	coolant flow rate, kg/s
x	exponent in the overall friction factor correlation
ΔZ_{TC}	relative separation of thermal centre of the core and the IHX
ε	roughness factor, m
ρ	density, kg/m^3
ψ	parameter in the friction factor correlation

INTRODUCTION

The shutdown heat from liquid metal fast breeder reactor (LMFBR) plants is generally removed via forced flow through the main heat transport circuit. Alternately when the main circuit is either completely or partially unavailable, an auxiliary cooling system or path is provided. Such an auxiliary system can be either completely independent (external core cooling system) or semi-independent. In either case availability of pumping power is needed to force sodium through the reactor core. Although this power supply may be made as reliable as dictated by some standard, the consequences of a complete loss of all electric power must be examined to assure the public safety. In this situation the buoyancy head that may exist between the hot and cold legs of the circuit could establish an adequate level of natural circulation to dissipate the shutdown heat. Because of the obvious safety and design implications, the adequacy of this passive mode of heat transfer has been the subject of great interest as shown, for example, in the proceedings of a recent Specialists' Meeting[1].

The scenario for a natural circulation transient may be summarized as follows: complete loss of all electric power to an operating LMFBR plant followed by the reactor scram via the plant protection system. One is now interested in evaluating the state of the plant as a function of time. In the event that adequate natural circulation does get established, a gradual decrease in long-term sodium temperatures should result since the shutdown heat is gradually but steadily decreasing. Prior to this quasi-steady behaviour it is possible to have a hump in the coolant temperatures as a result of the power-to-flow mismatch during the early part of the transient.

It is convenient to break down the analysis of the natural circulation transients into two parts - (a) to show that natural circulation does get established without causing excessive temperatures in the system, and (b) to show that the resulting flow is adequate for long-term heat dissipation. In the first part of the problem, the entire system must be modelled dynamically [2]. The second problem, on the other hand, can be modelled simply by disregarding all time-dependent and inertia terms. A further simplification can be made by ignoring the slow decrease in decay heat generation rate. In this paper we concentrate on the second issue, i.e., to examine the adequacy of long-term heat removal via natural circulation.

Experimentally the adequacy of the natural circulation can be demonstrated in various ways. The most direct one being the simulation of a complete loss of all electric power from an established full power and flow conditions. Such a simulation may also be desirable from reduced power/flow conditions to represent plant operation with one of the heat transport loops under maintenance. In experiments to date, however, tests have been carried out either from a reduced power level and by turning off all electric power followed by the reactor scram, or at low steady-power values without scram after tripping of the pumps. The steady-power levels are taken to be representative of the decay heat values (say, up to 4% of the full power). It should be added that in the steady-power experiments, control rod position is adjusted to counteract the negative feedback reactivity due primarily to the thermal expansion of the core. This category of experiments is helpful in providing data for long-term heat dissipation capability of the plant. The former category of experiments, on the other hand, can provide data for transients as well as long-term natural circulation in the plant.

In this paper we will concentrate on the steady-power tests and their analyses to evaluate the long-term heat dissipation capability of LMFBR plants. After discussing two analytical approaches, applicable to both loop- and pool-type designs, results obtained will be compared with the PFR and PHENIX tests. It will be shown that a simplified correlation can be developed.

MODELLING

In order to ascertain the shutdown heat dissipation capability of an LMFBR plant via natural circulation, one needs to provide for adequate modelling of all processes such as coolant flow through fuel and blanket assemblies, piping and the heat exchanger tubes, and heat transport from the source to the sink. Particular emphasis must be given to the pressure drop calculations at the expected low flow conditions. Different components will experience different flow regimes. For example, sodium flow through blanket assemblies will be laminar while that in fuel assemblies could be in the laminar or transition regime. Fluid flow through the piping and bulk regions may still be characterized as turbulent.

Two different modelling approaches are considered in this paper. In the first one use is made of detailed dynamic computer programs. Test conditions are simulated by turning the electric power off and observing the steady temperature distributions in the system when all time-dependent terms reduce to zero. This method is obviously a more tedious one than that in which all subsystems are modelled for the steady-power case. Yet this approach is taken since detailed dynamic models in the form of computer codes are available [2]. The second approach is based on writing down the continuity, momentum and energy conservation equations in their time-independent form on a global basis, i.e., the details of any component are not considered. With judicious manipulation, expressions relating power, flow and temperatures are obtained.

Detailed Modelling

Schematic drawings of the primary sodium circuit for both the loop-type and pool-type designs are shown in Figures 1 and 2, respectively. These two designs differ from each other in that in the pool-type LMFBRs the intermediate heat exchangers and the primary sodium pumps are located in a large tank along with the reactor core. In the loop system these components are interconnected via large pipes. The intermediate sodium circuit and the steam generating system are not shown since they affect the natural circulation in the primary circuit through the primary sodium temperature distribution in the IHXs. Furthermore, these simplified diagrams allow us to represent other alternate

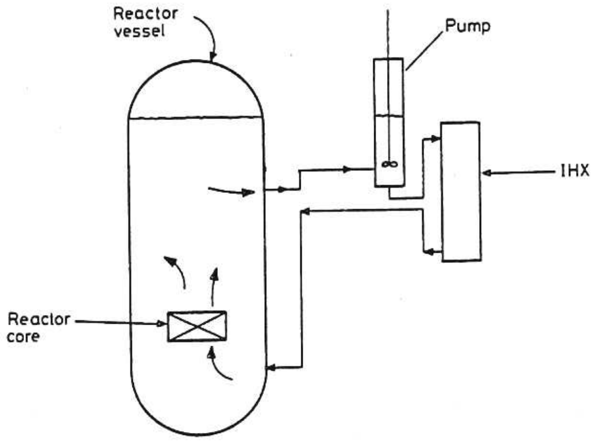

Fig.1 Primary circuit of a loop-type LMFBR.

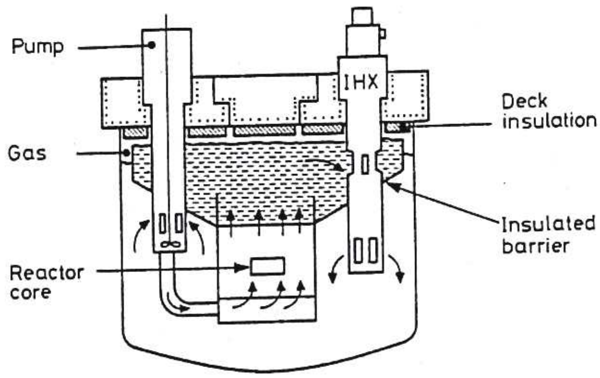

Fig.2 Primary circuit of a pool-type LMFBR.

intermediate circuits that may be incorporated in a specific plant. The U.K. Prototype Fast Reactor, for example, employs an independent decay heat removal system.

In our modelling we have made use of the SSC-L code [3] for simulation of loop-type plants and the SSC-P code [4] for pool-type designs. In either case, all of the processes of interest are modelled in one spatial dimension. For example, fluid flow is modelled in one-spatial dimension in the direction of flow. Heat conduction in the fuel rods is treated in the radial direction only. The reactor core is represented by a number of one-dimensional parallel channels which are coupled hydraulically at the lower and upper plenum regions. Each channel represents at least one or more fuel or blanket assemblies.

Because of the importance of the frictional pressure losses, we note down the expressions for the friction factor for wire-wrapped fuel assemblies. Addito n et al [5] have reported that the fluid flow through wire-wrapped rod bundles does not show the classical transition that is observed for flow in pipings between the turbulent and laminar flow regimes. For pitch-to-diameter ratio of 1.24, they report

$$f = \frac{84}{Re} \qquad Re \leq 2000 \tag{1}$$

which is in close agreement with a more recent correlation by Engel et al [6] for the same value of P/D and wire-wrap lead of 0.30m. In the case of CRBRP and FFTF the Reynolds number encountered in fuel assemblies ranged up to 1800 for steady-power values to 4% of the normal.

Friction factor for blanket assemblies (P/D ∿ 1.08 and H = 0.10m) is correlated by Engel et al [6] as

$$f = \frac{110}{Re} \qquad Re<400 \tag{2}$$

$$= \frac{110}{Re}\sqrt{1-\psi} + \frac{0.48}{Re^{0.25}}\sqrt{\psi}, \quad 400 \leq Re \leq 5000 \tag{3}$$

$$= \frac{0.48}{Re^{0.25}} \qquad Re>5000 \tag{4}$$

where the intermittency factor ψ is defined as

$$\psi = \frac{Re-400}{4600} \tag{5}$$

Note that the expression for the turbulent regime has 0.48 in the numerator as corrected by the authors.

Friction factor for turbulent flow in the pipes is represented [7] as

$$f = 0.0055\left\{1 + \left(20000\frac{\varepsilon}{De} + \frac{10^6}{Re}\right)^{1/3}\right\}, \quad Re>3000 \tag{6}$$

and for laminar flows,

$$f = \frac{64}{Re} \qquad Re<2000 \tag{7}$$

An interpolation of f between the turbulent and laminar correlations is used for the transition region (2000 ≤ Re ≤ 3000).

Since we are interested in the mixed-mean core temperature rise at steady-power conditions, the details of the core representation are not relevant and they can be found elsewhere [8]. Further, the heat exchanger is also modelled as one equivalent heat transfer tube.

Both the SSC-L and SSC-P codes are dynamic simulation codes. The steady-power natural circulation test conditions were simulated as follows:

(1) Reactor was brought to steady 100% power, 100% flow operating conditions.
(2) Total loss of all electric power was assumed to occur at zero time followed by reactor trip as dictated by the plant protection system.
(3) In the CRBRP and FFTF cases a close to steady natural circulation results in the primary sodium circuit within a few minutes of the transient. Since the heat generating rate decreases rather slowly, in the interest of computing time, the reactor power was forced to the desired power levels and then held steady. When the effects of this transient died out, the steady primary flow rate (which is completely due to natural circulation) and temperature changes are noted. The same procedure is used for pool-type plants except that the steady-state sodium flow in the primary circuit is attained at a much later time owing to the large thermal inertia of the system.

Simplified Modelling

Consider the primary sodium circuit of either the loop or pool designs. In a single spatial dimension, the continuity equation for incompressible fluid requires that the flow rate in the system is independent of position. However, in general it is a function of time. Expressions for the momentum conservation equation can be written for each conveniently divided subsection. Since there is no net pressure drop in going through the entire circuit, the following equation is obtained:

$$\frac{1}{2}\frac{W^2}{\rho_0}\sum_i \frac{f_i(W)L_i}{d_i A_i^2} + \Delta\rho . g . \Delta Z_{TC} = 0 \tag{8}$$

where the summation is to be carried over all subsections of the circuit, changes in the sodium density in both the reactor core and the IHX are assumed to vary linearly. Alternately, the friction term in Eq.(8) can be expressed as a single, equivalent term. The energy conservation can be expressed as

$$P = c_p \; W \; \Delta T \tag{9}$$

where ΔT is the mixed-mean sodium temperature rise across the reactor core.

A relation involving two of the three variables (P, W and ΔT) can now be obtained by combining Eqs. (8) and (9) with the equation of state. For incompressible sodium the equation of state shows that the density varies linearly with temperature ($\rho = a - bT$). Thus we get

$$\bar{f}(W) . W^3 = c . \Delta Z_{TC} . P \tag{10}$$

where c is a constant determined from the geometric details of the system. Note that the above form of Eq(10) explicitly shows the role of the distance between the heat generating (reactor core) and heat sink (IHX) thermal centers. The quantity $\bar{f}(W)$ is an effective friction factor. It should be added here that a relationship between the mixed-mean core temperature rise and the reactor power can be obtained by eliminating W from Eqs.(9) and (10).

The functional dependence of $\bar{f}(W)$ on the flow rate is complicated since it contains contributions from flow in the rod bundles, form losses, losses due to stationary pump rotor, check valves, pipes etc and their individual dependence on flow will, in general, be different from each other. In the case that the effective friction factor can be represented as

$$\bar{f}(W) = c_1 \; W^{-x} \tag{11}$$

where c_1 is some constant then the following correlations are obtained for the natural circulation flow rate and the mixed-mean temperature rise across the reactor core:

$$W = C \; (\Delta Z_{TC})^{\frac{1}{3-x}} \; P^{\frac{1}{3-x}} \tag{12}$$

and

$$\Delta T = C' \; (\Delta Z_{TC})^{-\frac{1}{3-x}} \; P^{\frac{2-x}{3-x}} \tag{13}$$

where C and C'are some other constants. Equations (12) and (13) can be used to estimate the natural circulation flow rate and the temperature rise as a function of the power level.

RESULTS

Role of Frictional Losses

To illustrate the delicate balance between the buoyancy head and various frictional losses under natural circulation condition we show some sample dynamic results in Fig.3. These results were obtained

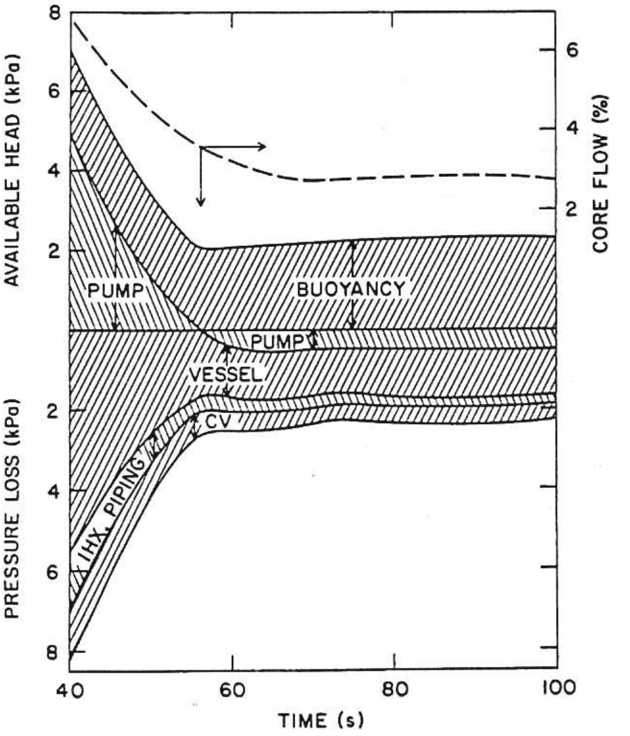

Fig.3 Core flow, pressure heads and losses in CRBRP primary system during coastdown to natural circulation.

[9] for the CRBRP reactor using the SSC-L code. The plant was assumed to be operating at its full power and flow conditions prior to the loss of electric power at time zero. The reactor was scrammed at about 0.5 s later. The SSC-L code then computed pressures, temperatures and flow rates as a function of the transient time throughout the plant. For the present purpose of illustration we show computed time history of pressure head and component pressure losses in the primary sodium circuit from 40 to 100 s into the transient. The time history of the system response for the first forty seconds is not shown since the role of buoyancy is insignificant and the system dynamics is dominated by the stored kinetic energy of the pump rotors. Note that the pump head becomes zero around 55 s and beyond this time it also begins to be a significant contributor to the overall pressure loss. The buoyancy head stays nearly constant at 2.3 kPa(o.33psi). The pressure loss associated with sodium flow through the reactor vessel, much of which is due to flow in the rod bundles, is the dominant contributor.

The pressure loss due to the coolant flow through fuel and rod bundles is roughly half of the total presure losses under natural circulation conditions. It is thus evident that the frictional losses in rod bundles must be evaluated as accurately as feasible. The second most important contributors to the total pressure losses are due to the pump rotors and the check valve. The role of IHX and the piping is not as significant in the computation of the overall losses.

The coolant flow through the reactor primary circuit appears to attain roughly 3% of the nominal full value at about 100 seconds after reactor scram. Since there still is a slight excess of the available head over the pressure losses, the coolant flow will rise. At the same time the decay heat is constantly decreasing and hence the available head will also show a gradual decline. The heat generation at about 100 s after the reactor scram is in the range of 4 to 5% of the nominal full value and the flow is about 3% therefore the mixed-mean core temperature rise at this time is expected to be higher than that at the normal operating levels.

Simplified Model

The functional dependence for the natural circulation flow rate for steady-power condition was shown to be given by Eq.(12). A major input to this equation is the value of the exponent in the effective friction factor. For conditions typical of LMFBR plants, parts of the entire circuit could very well be represented as turbulent (x=0.2) while other sections could exhibit laminar flow regime (x=1.0). Let us assume that the entire circuit could be represented by either of the two limiting behaviours. In that case the flow rates and the temperature increases will behave in the manner indicated in Table I.

TABLE I. FUNCTIONAL DEPENDENCES FOR STEADY-POWER NATURAL CIRCULATION CIRCUITS

Flow Regime	Flow Rate	Temperature Rise
Laminar (x=1.0)	$W \alpha P^{0.50}$	$\Delta T \alpha P^{0.50}$
Turbulent (x=0.2)	$W \alpha P^{0.36}$	$\Delta T \alpha P^{0.64}$

The frictional losses in the pump rotors and the check valves appear to be represented by the laws for the turbulent regime. On the other hand, the Reynolds number for flows in the fuel and blanket assemblies is in the laminar range. The overall pressure loss should therefore be a weighted average of the importance of the pressure losses in the pump and check valve, and that in the rod bundles. From Fig.3 we note that these two losses are about the same. Hence it appears that, to a zeroth order approximation, the overall frictional losses should be characterized by Eq.(11) with the value of x midway between the turbulent and laminar regimes. We have therefore used a value of x=0.6 to represent the overall losses. In this case, Eq.(13) gives

$$\Delta T = C' \ (\Delta Z_{TC})^{-0.42} \ P^{0.58} \tag{14}$$

If the relative separation of the thermal centers of the core and the IHX remain unchanged as a function of steady-power levels then the mixed-mean core temperature rise is proportional to 0.58 exponent of the heat generation rate.

Computer Simulation

The steady-power natural circulation tests were simulated for two loop-type designs (CRBRP and FFTF) and one pool-type (PHENIX) LMFBR plants. These simulations were made using the SSC-L and SSC-P codes respectively, in a manner outlined earlier. Code predicted mixed-mean core temperature rises are displayed in Table II as a function of the shutdown power levels. The shutdown power levels are normalized to the nominal full power levels. Thus, 1% of the power level corresponds to the level of decay heat at about 5000 seconds after the shutdown. The

TABLE II. PREDICTIONS FOR STEADY-POWER NATURAL CIRCULATION IN CRBRP, FFTF AND PHENIX

Normalized Power %	Mixed-mean Core Temperature Rise, C					
	CRBRP		FFTF		PHENIX	
	SSC-L	Correlation	SSC-L	Correlation	SSC-P	Correlation
4	193.0	193.0	198.3	198.3	169.1	169.1
3	162.0	163.3	167.0	167.8	138.6	143.1
2	126.0	129.1	131.1	132.7	102.9	113.1
1	91.0	86.4	81.9	88.7		75.7

actual time at which this level of decay heat occurs will naturally depend upon the fuel cycle. Both the SSC-L and SSC-P predicted temperature rises appear to fit the following correlation

$$\Delta T = K\, P_o^{0.58} \qquad (15)$$

where the constant of proportionality K is equal to 1248.4 for CRBRP, 1282.7 for FFTF and 1093.8 for PHENIX and P_o is the fractional power normalized to unity as full power. These normalization constants were obtained on the basis of the 4% power value. It should be added that the code predicted temperatures for the case of 1% power had not quite reached their steady values.

The normalization constant in the proposed correlation depends upon several factors including (1) design parameters of the entire primary circuit, (2) design parameters of the secondary circuit, and (3) the operating characteristics of the secondary circuit. If, for example, one were to maintain the secondary circuit flow at a rate higher than the natural circulation flow rate then the thermal centre of the IHX will move upwards. This will result in higher buoyancy head and, hence, higher flow rate in the primary sodium circuit. The resulting temperature rises will be smaller. Thus, the constant K will take lower value. This observation is in agreement with the simplified relations noted in Eq.(14). Furthermore this trend was also seen in a transient calculation in which the secondary pump inertia was parametrically varied. The net effect of such a variation was to elevate the effective thermal centre of the IHX when the rotor inertia was lowered and vice versa. This resulted in increased natural circulation in the primary heat transport system[10].

In Eq.(15) the mixed-mean core temperature rise is correlated in terms of the power generation rate. This equation can also be applied to predict the hot channel temperature rise by scaling the constant with an appropriate hot channel factor. The exponent of the power is not expected to change, at least, to the first order of approximation. Any effect associated with inter-assembly flow redistribution and interassembly heat transfer will be reflected in the normalization constant.

Comparison with Test Data

So far we have discussed computer simulations and the resulting correlations. We now turn to applying the correlations to actual plant test data. A series of tests in support of the natural circulation as a reliable mode of decay heat removal from LMFBRs have been performed in PHENIX, EBR-II and PFR. In the case of EBR-II in-core instrumented assemblies (XX07 and XX08) have provided rather detailed temperature measurements [11]. These tests were primarily transient tests and thus provide very valuable data to support the establishment of natural circulation. The power level continued to fall since the reactor was scrammed. Thus, the EBR-II tests were indeed transient tests and not steady-power tests only.

Steady-power natural circulation tests were performed in the British PFR and recently reported by Gregory et al [12]. In these tests, the reactor was brought to a steady power, in steps, to 18-19 MW and steady conditions were maintained for a considerable time. In each step, the power levels were maintained by moving the control rods to counteract the negative feedback owing to the thermal expansion of the core which resulted from the rise in sodium temperature. The measured coolant temperature rise through the core (central assembly) was recorded. This data is shown in Fig 4 as a function of the

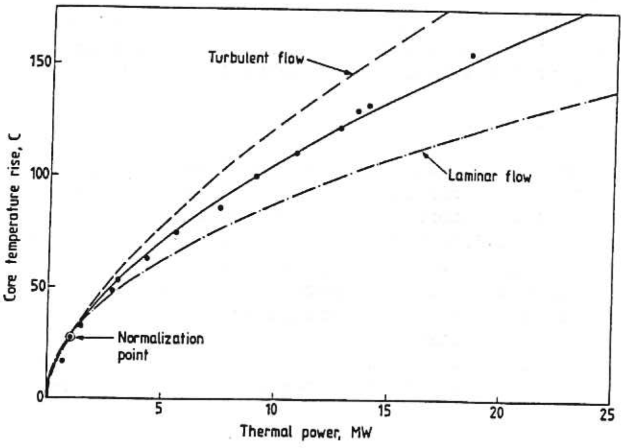

Fig.4 Experimental and correlated core (central assembly) temperature rises in steady-power natural circulation tests in PFR.

steady-power levels. It is observed that all of the results lie on the same smooth curve showing a remarkable consistency despite a range of different initiating conditions.

An attempt was made to analyze these PFR tests. A detailed simulation with the SSC-P code was not attempted since it required detailed design data. Therefore, results of the simplified analysis are discussed here. The constant of proportionality in Eq.(13) was obtained by normalizing the measured ΔT of 27.2 C at 0.95 MW. Results are also shown in Fig.4. The dashed curves are based on strictly laminar or turbulent characterization of the effective friction factor. The solid curve is based on the assumption that the flow dependence of the effective friction factor is exactly half-way between the two flow regimes. It should be noted that similar assumption was found to result in good agreement for the CRBRP and FFTF predicted values. It is seen that almost all of the data points can be predicted adequately.

A correlation for the PFR was also obtained. By normalizing the steady-power levels in terms of the nominal full-power level of 600 MW, the constant of proportionality K in Eq.(15) was found to be 1145.

The above analysis was also applied to the natural circulation experiments that were performed in PHENIX [13]. From the results of the steady 4 MW power test, the observed sodium temperature rises across the central assembly and the average of all assemblies were 65.8C and 38C, respectively. The authors [13] have stated that the mean core outlet temperature is much closer to the value given by only the central assemblies than by an average of all of them because cold sodium surrounding the core induces some conduction cooling at the outlet of peripheral assemblies. We, therefore, use 65.8C as the mixed-mean core temperature rise for this 4MW steady-power test. In this case, the constant of proportionality K in Eq.(15) was found to be 1160. The corresponding value predicted by the SSC-P code is 1093.8 - roughly 5% too low. This difference in the normalization constant could be due to a number of factors such as the characterization of the secondary circuit, the effect of the emergency cooling circuit which was in operation during the transient, and a slightly lower value for the mixed-mean core ΔT than 65.8C. The SSC-P code predicted mixed-mean temperature rise for this 4MW test is 62.1C which appears to be well within the reading error from the published results in Ref.13.

CONCLUSION

An attempt was made to correlate the steady-power natural circulation tests in LMFBR plants. Analytically, correlations were developed by simulating steady-power tests in both loop-type and pool-type designs. In so doing use was made of the SSC-L and SSC-P dynamic thermohydraulic system codes. The form of the resulting correlation was in agreement with the one obtained from a simplified analysis. The effect of the secondary sodium circuit was included in the correlation in terms of the IHX thermal centre.

The functional dependence of the correlation was found to be in agreement with the whole plant tests in both PFR and PHENIX reactors. In the case of PHENIX, the SSC-P code predicted normalization constant also agreed well with the test data.

The analysis, and its comparison, presented in this paper shows that the shutdown heat from a properly designed LMFBR plant can adequately be dissipated once the natural circulation is established. Additional on-going analytical work in conjunction with the anticipated test results of FFTF will further substantiate LMFBR's capability of dissipating shutdown heat via natural circulation.

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