The Abdus Salam
International Centre for Theoretical Physics

United Nations
Educational, Scientific and
Cultural Organization

IAEA
International Atomic Energy Agency

2152-25

Joint ICTP-IAEA Course on Natural Circulation Phenomena and Passive Safety Systems in Advanced Water Cooled Reactors

17 - 21 May 2010

EXAMPLES OF NATURAL CIRCULATION IN PHWR

P.K. Vijayan and D. Saha

Reactor Engineering Division
Bhabha Atomic Research Centre
Trombay, Mumbai, India

IAEA Training Course on Natural Circulation Phenomena and Passive Safety Systems in Advanced Water-Cooled Reactors, ICTP, Trieste, Italy, 17-21 May 2010

Lecture Note T21a

on

EXAMPLES OF NATURAL CIRCULATION IN PHWR

by

Dilip Saha and P.K. Vijayan

Reactor Engineering Division, Bhabha Atomic Research Centre
Trombay, Mumbai-400085, INDIA

May 2010

EXAMPLES OF NATURAL CIRCULATION IN PHWR

Dilip Saha and P.K.Vijayan

Reactor Engineering Division
Bhabha Atomic Research Centre
Trombay, Mumbai-400 085, India
E-mail : dsaha@apsara.barc.ernet.in

KEYWORDS

PHWR, NAPS-1, Natural Circulation, Flow Reversal

OBJECTIVE

The main objective of this lecture is to provide deep insight into the complex natural circulation phenomena in the core of a Pressurised Heavy Water Reactor. A detailed account of natural circulation tests conducted in an Indian PHWR, Unit-1of Narora Atomic Power Station is given in this lecture. This will enable the participants to appreciate the importance of natural circulation in a nuclear reactor to a greater extent.

1. INTRODUCTION

Natural circulation can take place in a nuclear reactor under following circumstances:

a) As design intent
b) Under accidental condition
c) Under planned experimental programme

Experiments have been conducted in very few nuclear power reactors; Narora Atomic Power Station Unit-1(NAPS-1), the Indian Pressurised Heavy Water Reactor (PHWR) and Dodewaard, the Boiling Water Reactor of Holland, now decommissioned, are among them. Rated power of NAPS-1 is 235 MWe and for Dodewaard it was 50 MWe.

In the following sections experiments on natural circulation conducted in NAPS-1 are described. Natural circulation phenomenon is quite complex in a PHWR because of following features.

a) In PHWR, core is horizontal leading to horizontal flow path in core. In this case, it is difficult to predict direction of flow when started from stationary condition.
b) Different rows of channels are at different elevations leading to difference in driving force for different channels.
c) Hydraulic resistance of flow path is large because of the presence of long inlet and outlet feeder pipes and long fuel channels.

2. DESCRIPTION OF THE REACTOR

The Indian Pressurised Heavy Water Reactors (PHWRs) are dependent on thermosyphon in the primary coolant loop for decay heat removal from core following power failure and reactor trip. Figure 1 shows a simplified flow diagram of the Primary Heat Transport System (PHTS) of a PHWR. It comprises of two passes of the coolant in the core. Flow direction in one pass is just opposite the flow direction in the other pass. The coolant passes through coolant channels in core pass-1, steam generator-1, pump-1, coolant channel in core pass-2, steam generator-2, pump-2, and back to coolant channels in core pass-1. This path of the coolant shown by arrows in Fig. 1 is known as a figure-of-eight loop. In a 235 MWe PHWR, in each core pass there are 153 horizontal fuel channels located at different elevations inside the calandria vessel as shown in Fig.2. Each channel is connected by an inlet and an outlet feeder pipes to large diameter inlet and outlet headers. Large diameter piping connects the headers to the steam generators (SGs) and pumps. Two boiler branches are connected in parallel between a Reactor Outlet Header (ROH) and a Reactor Inlet header (RIH) on each side. Each boiler branch comprises of a SG and a pump. The SGs comprise of large number of vertical inverted U-tubes through which the hot heavy water flows. The top of the SG U-tubes are located at about 16 m above the headers and the headers are located about 7 m above the core centre line. The total difference between the top of the U tube and the centre of the core is 23 m. The density difference between hot and cold legs in this height provides the buoyancy force for natural circulation. Figure 2 shows core configuration.

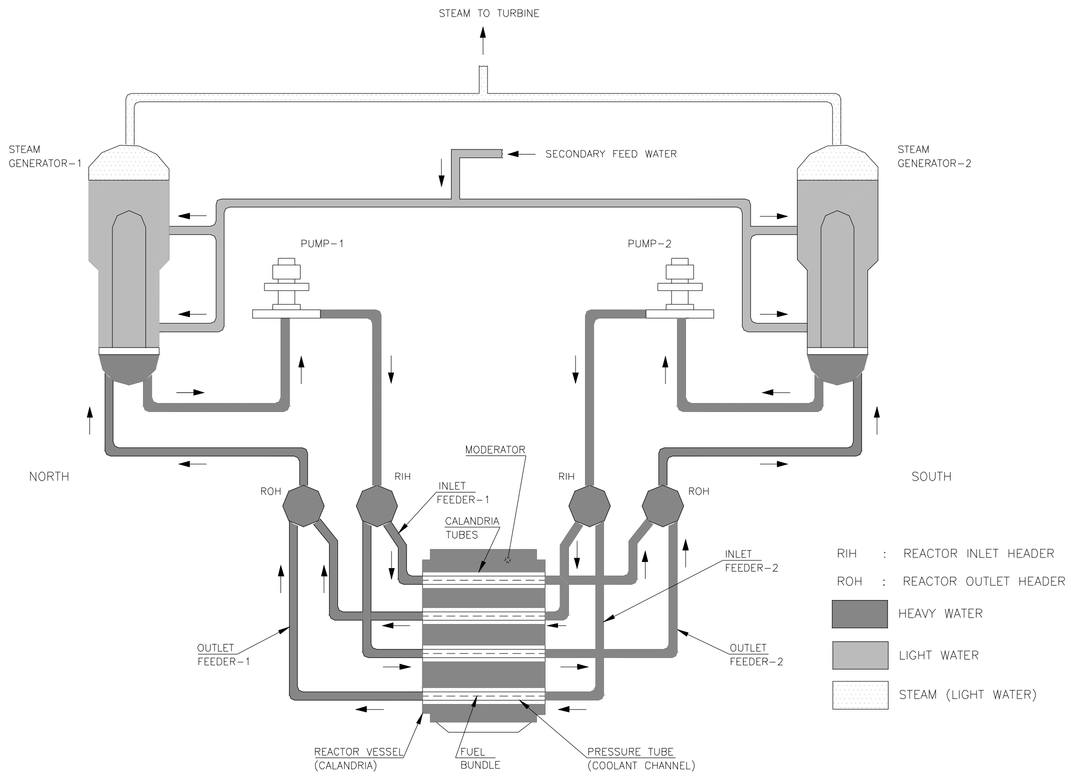

FIG. 1. Schematic of primary heat transport system of a PHWR.

3. TEST OBJECTIVES

To understand the natural circulation behaviour of the primary coolant loop with horizontal fuel channels, phenomenological studies were earlier carried out in a simple figure-of-eight loop [1]. Based on the results of these experimental studies, a computer code TINFLO –III [2] was developed for the analysis of natural circulation in PHWRs. This computer code was used for predicting natural circulation in 235 and 500 MWe PHWRs. To verify these predictions and to experimentally establish the onset of natural circulation in Indian PHWRs, tests were proposed and carried out in Unit-1 of Narora Atomic Power Station (NAPS).

4. TESTS CONDUCTED

Three tests were conducted in Unit-1 of NAPS, with the reactor critical at power levels of 0.956, 1.991 and 3.249 percent of full power (FP). The initial conditions for the tests are described below:

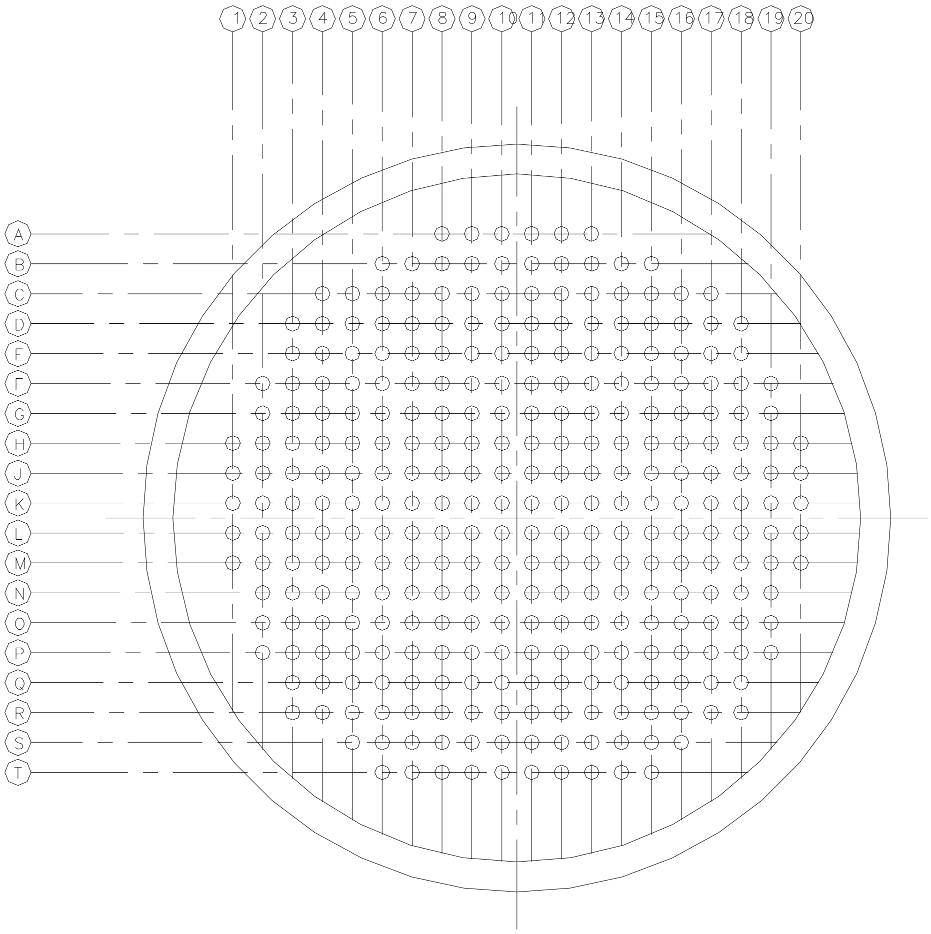

FIG. 2. Core configuration.

4.1. Initial conditions

For all the tests, the secondary side of the SGs was maintained at about 14.6 bar with the help of the Atmospheric Steam Discharge Valves (ASDV). The SG level was maintained at around 12.9 m with the help of the Auxiliary Boiler Feed Pump (ABFP). On the primary side, the pressure was maintained at about 86 bar. The initial conditions of flow and temperature on the primary side were somewhat different for the three tests. For the 0.956% FP and 1.991% FP tests, two primary heat transport system (PHTS) pumps were running (1-1 mode of pump operation). Since the boiler branches with the idle pumps were isolated, the heat removal was accomplished by only two SGs on two sides of the calandria in the 0.956% FP tests. For the 1.991% FP tests, the boiler branches having idle pumps were not isolated and heat removal was by all the four SGs. The flow in the SGs on the idle pump branches was in the reverse direction to start with. For the tests at 3.249% FP, the initial PHTS flow rate, pressure and temperature corresponded to those of steady state thermosyphon at 1.991% FP.

5. TEST PROCEDURES

Tests at 0.956% and 1.991% FP were initiated by switching off the PHTS pumps. The test at 3.249% FP was initiated by increasing the power from 1.991% FP to 3.249% FP when the reactor was thermosyphoning at 1.991% FP. During the tests, the PHTS pressure was maintained with the help of the fuelling machine supply pump.

5.1. Measurements made

The important parameters recorded during the tests are:

1. Channel flows and differential temperatures for 16 channels
2. SG differential temperatures, pressure and level for all the four SGs
3. Differential pressure between RIH and the corresponding ROH on each side
4. PHT system pressure
5. Neutron power and
6. Channel outlet temperature for all the channels

During the tests, the transient variation of channel flows, outlet temperatures and SG were monitored till the steady state was achieved.

6. RESULTS AND DISCUSSIONS

A complete description of the test results is given in Ref. [3]. The channel ΔT transients are plotted in Fig. 3. Typical transient behaviour, representative of most of the channels is shown by channel S-8 (Fig. 3). The ΔT reaches a peak value before a steady state with minor oscillations is reached. The peak and the steady state values of most of the channels are quite close to each other. The time of occurrence of the peak varies from 250 to 380 seconds for the different channels.

The ΔT behaviour of channels N-13 and Q-16, shown in Fig. 3, need special mention. In channel N-13, the ΔT increased as expected upto a time of about 265 seconds and then suddenly started reducing at a very fast rate and went out of the chart range of the visicorder at about 325 seconds. Just before going out of the range of the chart the ΔT indication was about –16.1ºC. i.e., 16.1ºC ΔT with flow reversal. The ΔT variation of channel Q-16, on the other hand, shows a peak (approx. 54.7 ºC) at about 340 seconds and then suddenly reduces to about –25.7 ºC with flow reversal at about 480 seconds. Subsequently, the ΔT decreases and flow direction changes again to the original and the ΔT stabilizes at about 21.4 ºC. Thus flow reversal has taken place twice in this channel and hence the initial and final steady directions in the channel are the same leading to positive ΔT before and after the flow reversals. Channel flow reversals, during transient thermosyphoning, have been reported for CANDU systems [4]. However, instances of a majority of the channels in one core pass flowing in the forward direction with a few flowing in the reverse direction under steady state conditions are not reported for these systems. None of the monitored channel ΔTs showed negative values for the 1.991% and 3.249% FP tests indicating the absence of flow reversal.

From the measurements obtained from the channel temperature monitoring system, the individual channel outlet temperatures of rows having the same flow direction are plotted in Fig. 4. Since the channels at the bottom of the calandria have a larger elevation difference compared to the top channels, the flows in lower channels are expected to be more than those in the top channels. This would indicate lower outlet temperatures for the bottom channels as compared to the corresponding top channels generating the same power. This trend is generally observed for the tests at 1.991% and 3.249% FP (see Fig. 4). This is all the more significant since the bottom channels generate more power (approx. 10% for fresh core) than the corresponding top channels in the same column.

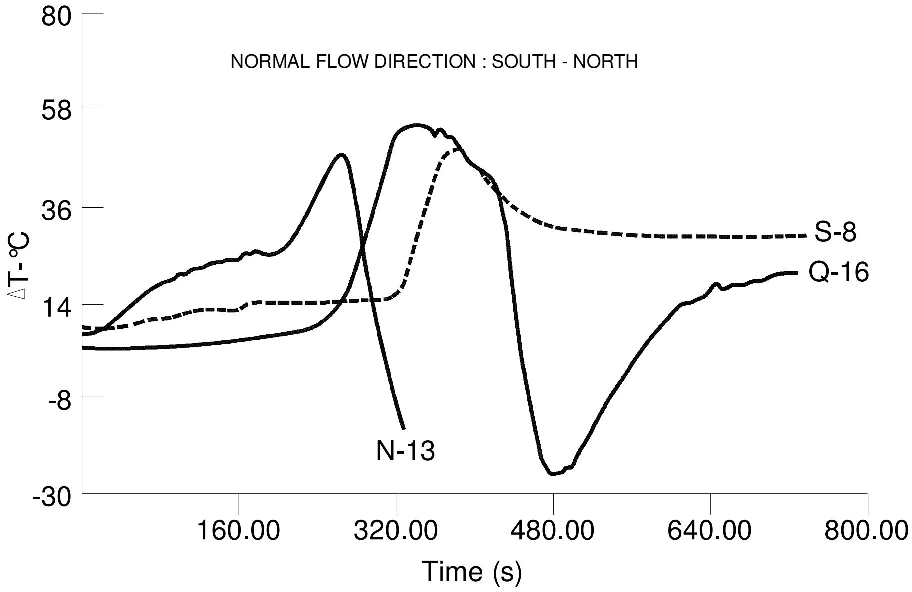

FIG. 3. Channel ΔT variation at 0.956%FP.

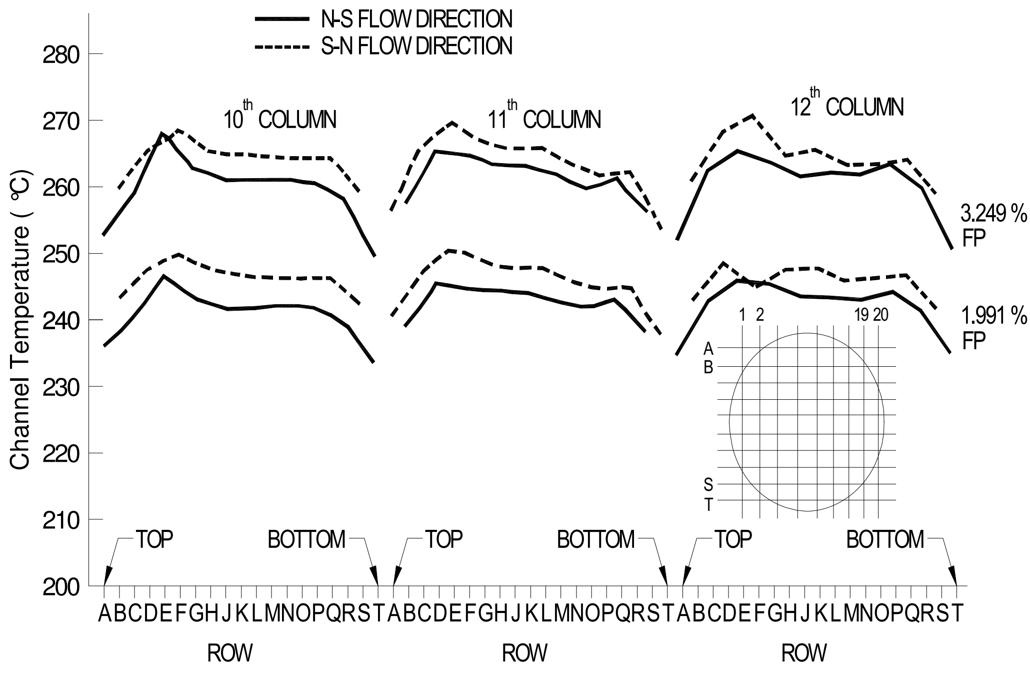

FIG. 4. Channel outlet temperature variation across different vertical columns for tests at 1.991 and 3.249% FP.

7. METHOD OF ANALYSIS

Analysis for test conditions was carried out using the computer code TINFLO-III, which solves the one-dimensional momentum and energy equations using the finite difference method. Flow coast down phase is modeled based on the test data on pump speed vs. time. Steam Generator secondary side is modeled by a single volume. Initial temperature of the SG secondary is specified as input and the primary temperatures are calculated by the code.

The code lumps all parallel flow paths in one core pass into one equivalent flow path. Therefore the code can predict only the average core behaviour and not the individual channel behaviour. Further details of the code and the solution procedure are given in Vijayan et al (1988) [2].

8. COMPARISON OF EXPERIMENTAL AND ANALYTICAL RESULTS

The observed and predicted transient and steady state behaviour are described and discussed below.

8.1. Channel flows

Figures 5 and 6 show the predicted and measured variation of normalized flow, with time, for the tests at 0.956% FP and 1.991%FP respectively. The normalization of flow is with respect to the flow corresponding to 2-2 operation mode. In both the tests, a very good match is obtained between the predicted and measured flows during the initial coastdown phase (upto 60 seconds). Beyond this, the prediction and the test value, though showing somewhat similar trends, differ widely in magnitude. This difference is attributed to the large measurement inaccuracy of the normal plant instrumentation at these low flows (less than10% of the full flow). The steady state core flow rates estimated based on heat balance from the measured SG ΔT and reactor power are also plotted in these figures. It may be noted that these are very close to those predicted by the code.

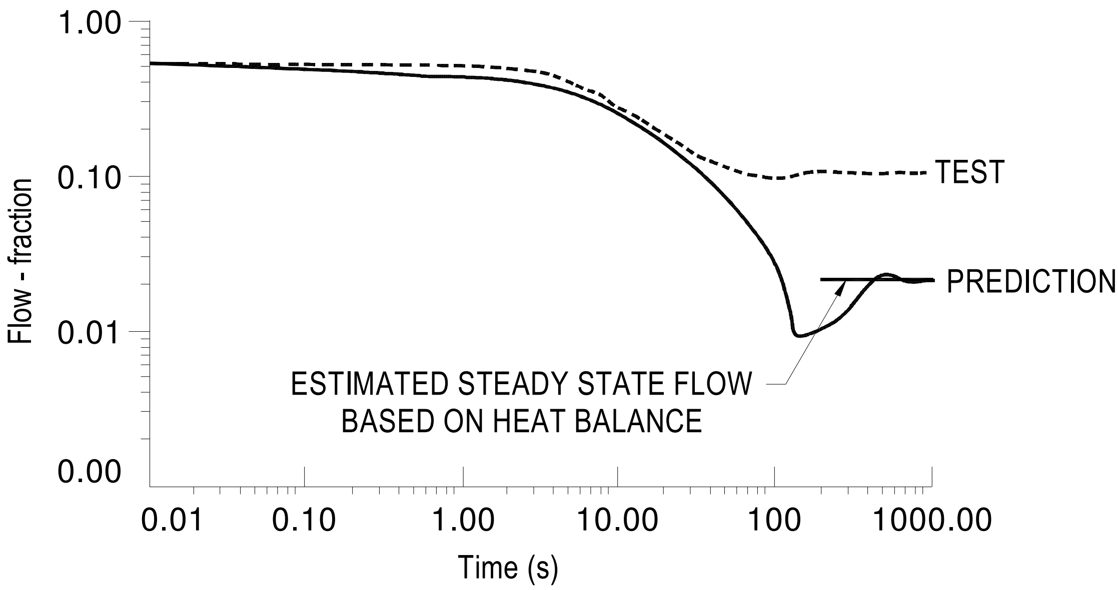

FIG. 5. Variation of flow with time for test at 0.956%FP.

8.2. Header to header pressure drop

Figure 7 shows the measured and predicted variation of header-to-header pressure drop from tests at 1.991% FP. The prediction and the measurements show the same trend. However, there is a time lag between the predicted and measured minimum and peak pressure drops.

At time t=o the ΔP indications are off-scale since these ΔP indicators have a range of only –50 cm to + 250 cm and the ΔP is much higher than this (13000 cm) when the primary pumps are operating. The first indication of header ΔP is obtained at about 30 to 40 seconds after the pumps are switched off.

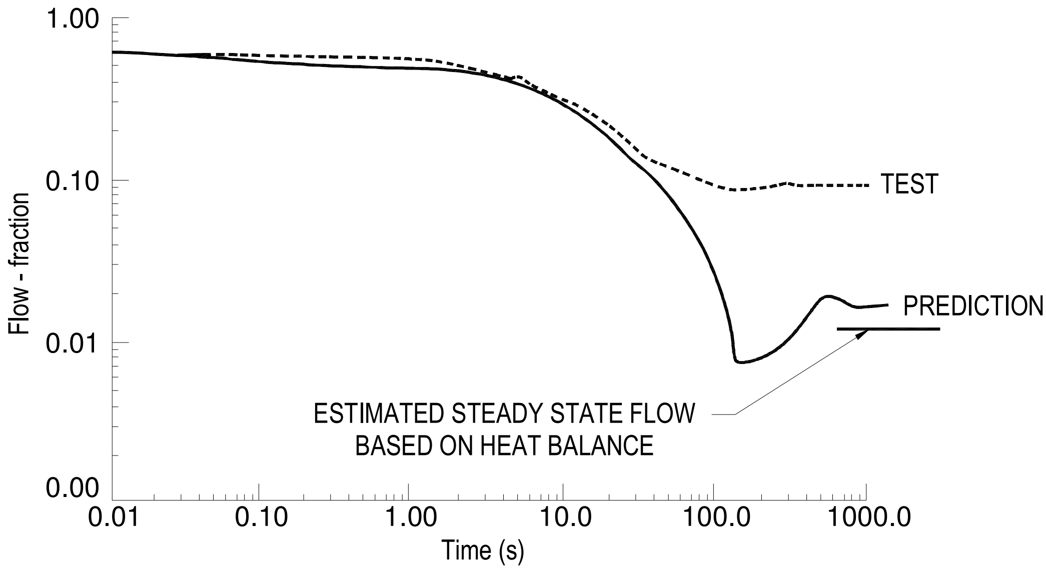

FIG. 6. Variation of flow rate for test at 1.991%FP.

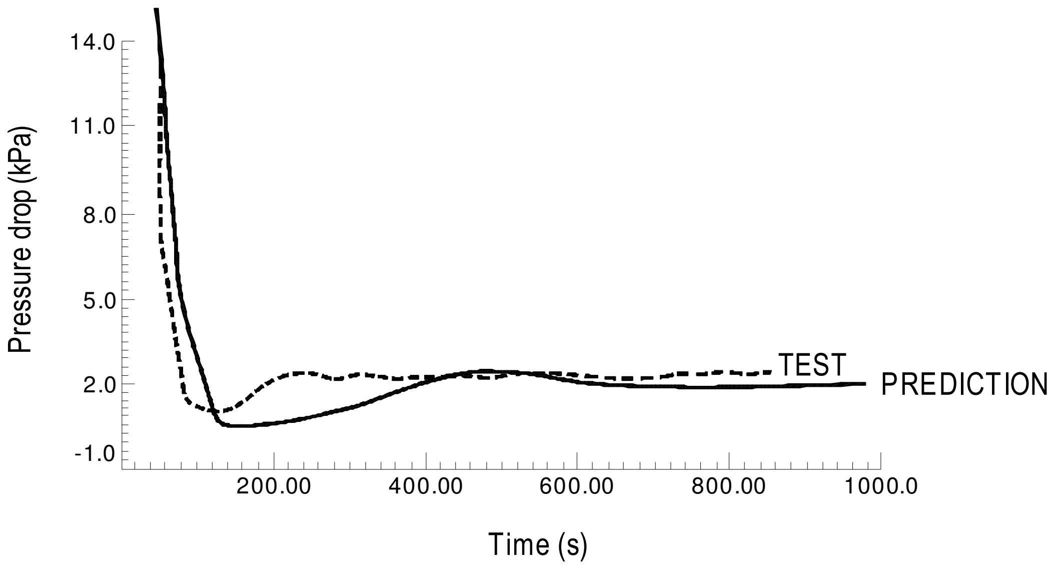

FIG. 7. Variation of header to header ΔP for test at 1.991%FP.

8.3. Channel ΔT variation

Figures 8 and 9 show the measured and predicted channel ΔT variations for the tests at 0.956% and 1.991% FP respectively. The experimental curve is the mean of the seven channel ΔTs. The measured and predicted ΔTs show the same trend. Although the agreement between the measured and predicted steady state values is quite good, the predicted peak value is much larger (about 30%) than that measured. The experimental peak value is observed to be lower than the predicted peak due to following reasons. The different channels have different feeder lengths and are located at different elevations and therefore individual peaks having different values occur at different times. On averaging these ΔTs, a smaller peak value is obtained. However, in the analysis all the channels in one core pass are lumped into one equivalent channel having one length and one elevation. There is a time lag between the peak values predicted and obtained during these tests. This is again attributed to the difference in the lengths of various feeder pipes in the reactor, whereas the prediction is based on an equivalent feeder of average length. Improvement of prediction with regard to this aspect can be obtained by avoiding the lumping of all the parallel channels in a core pass. However, this will involve increase in computational effort.

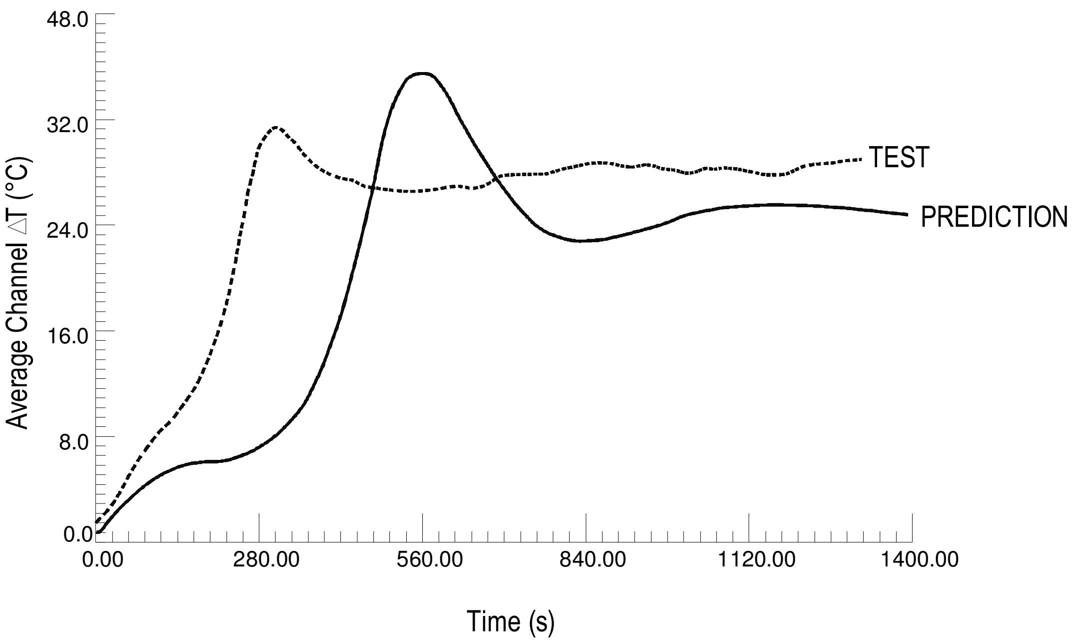

FIG. 8. Variation of average channel ΔT for test at 0.956%FP.

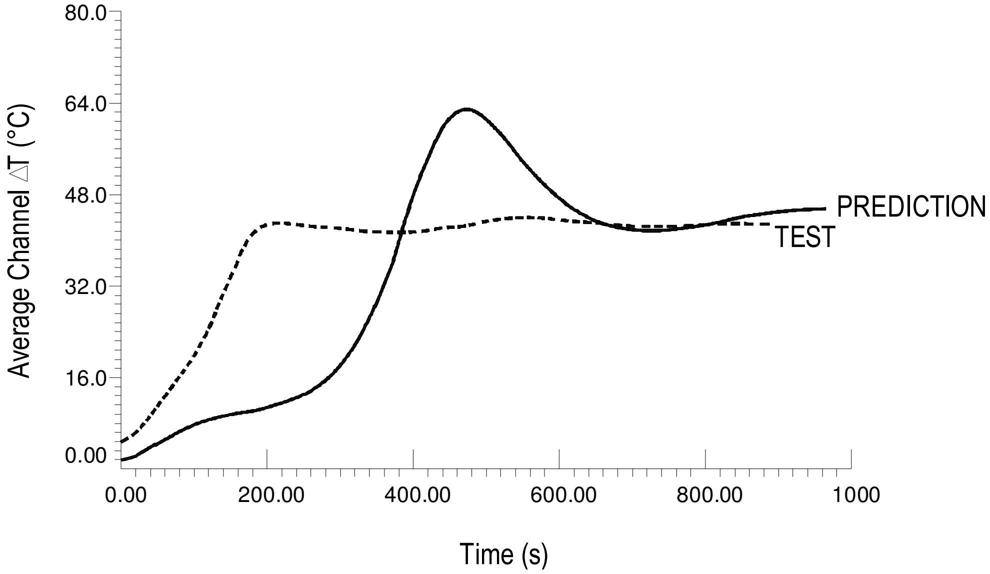

FIG. 9. Variation of average channel ΔT for test at 1.991%FP.

9. CONCLUSIONS

The tests conducted in unit-1 of NAPS have confirmed the efficacy of natural circulation core cooling at decay power levels following failure of forced circulation of the primary coolant.

An interesting observation from the tests is that flow reversal occurred in some channels during the test at 0.956% FP, with only two steam generators valved in. However, flow reversal was not observed at higher power.

The channel elevation plays a significant role in determining the flow rate through the different channels.

The most interesting aspect of the NAPS-1 experimental programme is the fact that the same unit of the same power station could be saved because of natural circulation core cooling in the face of a fire incident. A few years after the experiments, the unit faced a prolonged complete station blackout. Even class-1 power supply was not available since many electrical cables caught fire. Fire-fighting water was injected by diesel pumps into the secondary side of steam generator. Steam pressure was maintained in the secondary side using the Atmospheric Steam Discharge Valves. The main function of removal and transportation of heat from core to SG was achieved by natural circulation very efficiently. No loss of life or injury to plant personnel occurred. There was no release of radioactivity within the plant or to the environment.

ACKNOWLEDGEMENTS

The authors wish to thank IAEA and its consultants for the valuable suggestions provided. Thanks are also due to Mr. D.S.Pilkhwal and Mr. Vikas Jain of RED, BARC for the help rendered in preparing this lecture.

REFERENCES

[1] VIJAYAN, P.K., and DATE, A.W., Experimental and theoretical investigations on the steady state and transient behaviour of a thermosyphon with through flow in a figure-of eight loop, Int. J. Heat Mass Transfer, Vol.33.pp.2479-2489 (1990).

[2] VIJAYAN, P.K., NAGDAUNE, R.K. and VENKAT RAJ, V., Analysis of Flow Coast Down and transition to thermosyphon Cooling in the PHT system of a PHWR following Power Failure, Proceedings of the Indo-Polish Seminar on Development of Advanced Codes and Methods for Transient and Accident Analysis of Nuclear Power plants, pp 10.1-10.26, B.A.R.C. Bombay (1988).

[3] VIJAYAN, P.K., VENKAT RAJ, V. and PILKHWAL, D.S., Thermosyphon Tests conducted on Unit –1 of Narora Atomic Power Station, BARC Internal Report (1991).

[4] CAPLAN, M.Z., GULSHANI, P., HOLMES, R.W., and WRIGHT, A.C.D., Flow behaviour in a CANDU Horizontal Fuel Channel from stagnant subcooled initial conditions, Experimental and Modeling Aspects of Small break LOCA, Proceedings of Specialists' meeting organized by the IAEA, Budapest, Hungary (1983).