ORNL/TM-12677

ornl

OAK RIDGE NATIONAL LABORATORY

MARTIN MARIETTA

Evaluation of the Need for Emergency Heat Exchangers for Long Term Emergency Cooling of the Advanced Neutron Source Reactor

M. Ibn Khayat
J. L. Anderson
R. E. Battle
J. March-Leuba

Advanced Neutron Source

MANAGED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

ORNL/TM-12677

EVALUATION OF THE NEED FOR EMERGENCY HEAT EXCHANGERS FOR LONG TERM EMERGENCY COOLING OF THE ADVANCED NEUTRON SOURCE REACTOR

Mohammed Ibn Khayat
John Anderson
Ron Battle
J. March-Leuba

May 1994

Prepared by
OAK RIDGE NATIONAL LABORATORY
managed by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-84OR21400

CONTENTS

ACRONYMS

ACSL	Advanced Continuous Simulation Language
ANS	Advanced Neutron Source
ANSR	Advanced Neutron Source reactor
CDR	conceptual design report
CFR	critical flux ratio
EHX	emergency heat exchanger
FER	flow excursion ratio
HXP	heat exchanger pool
MHX	main heat exchanger
PCP	pipe chase pool
PRSDYN	Advanced Neutron Source dynamic model
RVP	reactor vessel pool

LIST OF FIGURES

LIST OF TABLES

ABSTRACT

This report summarizes the work performed to evaluate the heat transferred to the light water pools from the primary piping system for the Advanced Neutron Source reactor (ANSR) conceptual design. It has been determined that the ANSR primary piping system will remove sufficient heat from the primary coolant system to the pools for certain design basis event accidents without the emergency heat exchangers if the design parameters, such as pool volumes and pipe sizes (length and surface area), are selected appropriately. Based on this analysis, the emergency heat exchangers might be removed, and their function can be performed by the primary piping passing through the light water pools described in the conceptual design report. This study also shows that connecting the pipe chase pool and the heat exchanger pools improve performance for ANSR emergency heat removal.

1. INTRODUCTION

This study was initiated to estimate the heat rejected to the various pools by the primary coolant system piping and heat exchangers for all operating modes and conditions. This estimation is necessary to define system interfaces for the light water pool systems and to assess the contribution of and the need for the emergency heat exchangers (EHXs) during loss of the reactor secondary cooling system. The objective of this study is to simulate the Advanced Neutron Source reactor (ANSR) cooling circuit, including all relevant components, and to perform analyses to evaluate the heat rejected to the pools for different operating conditions and design basis event scenarios. The configuration of the reactor primary coolant system piping, the volumes of various pools, and the heat transfer correlations for the piping configurations shown in the ANSR conceptual design report (CDR) are included in the ANS dynamic model.

The ANS primary reactor cooling circuit piping is mostly stainless steel, with a 0.35-m diam for each independent loop and a 0.5–0.6-m-diam common leg (hot leg). The other means of heat removal is by the main heat exchanger (MHX). The ANSR conceptual design includes EHXs located in light water pools that remove heat in case of loss of secondary side of the MHXs. The pool light water flowing through the EHXs is driven by natural convection because of the heat transferred from the primary side. The ANS primary coolant piping passes through six separate pools—the pipe chase pool (PCP), which submerges portions of the horizontal common loop hot leg and the horizontal cold legs; the reactor vessel pool (RVP), which submerges the hot leg riser and the vertical cold leg pipe to the reactor inlet; and the four independent heat exchanger pools (HXPs), which submerge each of the main and EHXs and a portion of the piping between them.

2. MODEL DESCRIPTION

This section describes the dynamic model[1] (PRSDYN) used to simulate the heat load and heat transfer by the ANSR cooling system. It is a model developed to simulate accidents and normal operation scenarios for ANSR, including primary and secondary cooling systems. The model is written in modular form as shown in Fig. 1, and it is programmed in the Advanced Continuous Simulation Language (ACSL).* The model includes the following modules: core neutronics, channel thermal hydraulics, pipe thermal hydraulics, vessel bypass and annulus, heat exchangers, main circulation pumps, gas accumulators, makeup system, letdown system, control system, sensor module, pools module, and secondary cooling circuit.

Heat transfer between the light water pools and the primary coolant is a function of the inside and outside heat transfer coefficients and the thickness of the piping system. For this study, it was determined that the best correlation for modeling the heat transfer on the outside of the horizontal piping for natural convection is the Morgan[2] correlation:

$$(h_{nc})_H = \frac{D_e \times c \times (Gr \times Pr)^n}{\kappa} , \qquad (1)$$

where

$(h_{nc})_H$ = natural convection heat transfer coefficient for a horizontal pipe (W/m^2/°C),
D_e = equivalent diameter (m),
c = constant (see Table 1),
Gr = Grashof number,
Pr = Prandtl number,
n = constant (see Table 1),
κ = thermal conductivity (W/m/°C).

Table 1. Constant values for Eq. 1

$Gr^a \times Pr^b$	c	n
10^{-10}–10^{-2}	0.675	0.058
10^{-2}–10^{2}	1.020	0.148
10^{2}–10^{4}	0.850	0.188
10^{4}–10^{7}	0.480	0.250
10^{7}–10^{12}	0.125	0.333

[a]Gr = Grashof number.
[b]Pr = Prandtl number.

*Mitchell and Gauthier Associates, Huntsville, Alabama.

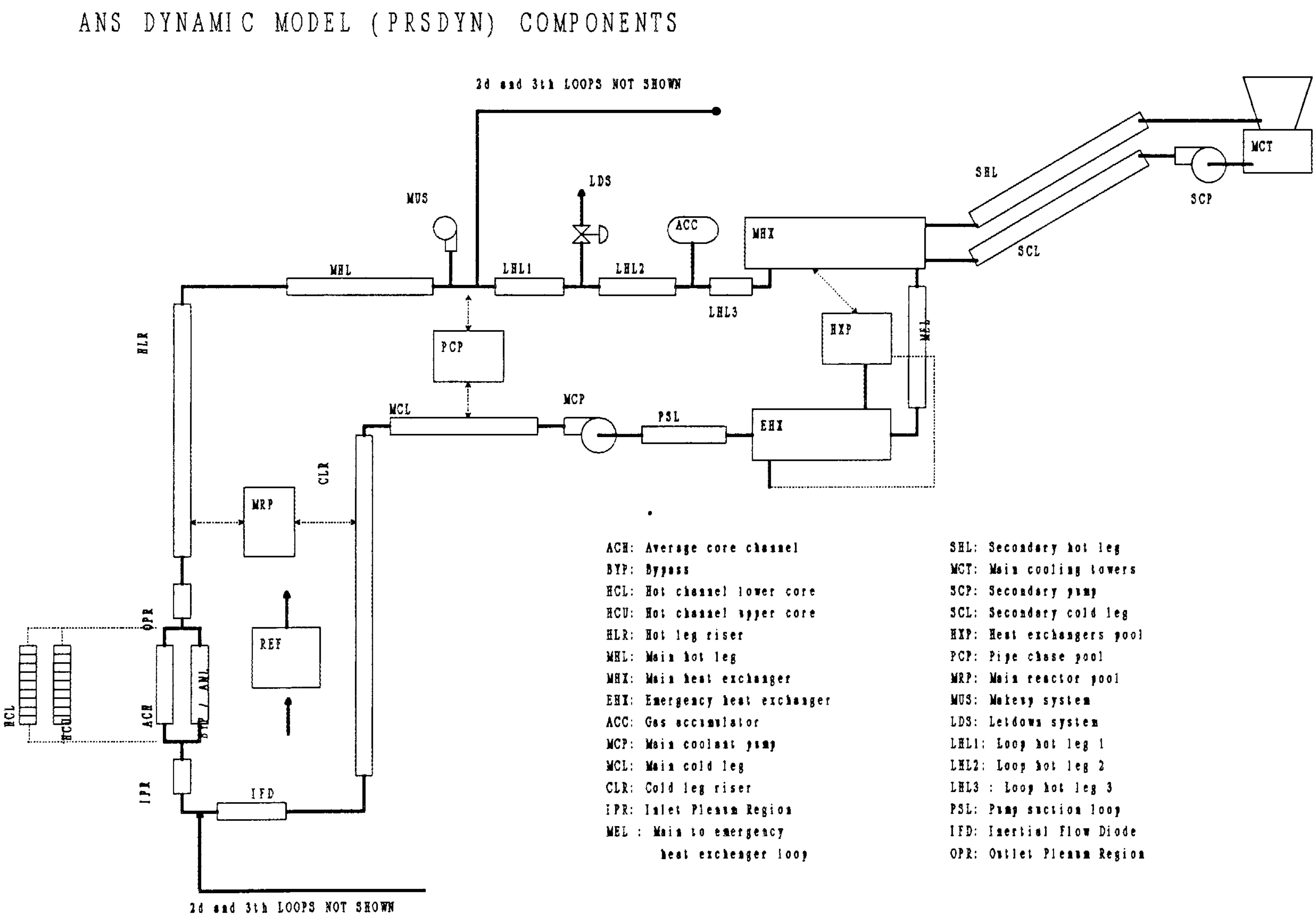

Fig. 1. Advanced Neutron Source reactor dynamic model components.

The correlation used for heat transfer from outside of vertical pipes is the McAdams[3] correlation:

$$(h_{nc})_V = \frac{D_e \times c \times (Gr \times Pr)^n}{\kappa} , \tag{2}$$

where

$(h_{nc})_V$ = natural convection heat transfer coefficient for a vertical pipe (W/m²/°C),
D_e = equivalent diameter (m),
c = constant (see Table 2),
n = constant (see Table 2),
Gr = Grashof number,
Pr = Prandtl number,
κ = thermal conductivity (W/m/s²).

Table 2. Constant values for Eq. 2

Type of flow	$Ga^a = Gr^b \times Pr^c$	c	n	Reference[d]
Laminar	10^4–10^9	0.59	0.25	4
Turbulent	10^9–10^{13}	0.10	0.33	5

[a]Ga = Rayleigh number.
[b]Gr = Grashof number.
[c]Pr = Prandtl number.
[d]Complete citations are in Sect. 7.

The correlation used for flow inside the pipes is the Dittus-Boetler[6] correlation given by Eq. 3.

$$h_{ip} = 0.023 \ Re^{0.8} \times Pr^n , \tag{3}$$

where

h_{ip} = heat transfer coefficient inside circular pipe (W/m²/°C),
Re = Reynolds number,
Pr = Prandtl number.

The conductance coefficient for to heat transfer through the pipe thickness is given by Eq. 4:

$$h_{metal} = \frac{\kappa_{metal}}{\delta d} \tag{4}$$

where

h_{metal} = metal heat transfer conductance coefficient (W/m^2/°C),
κ_{metal} = thermal conductivity of pipe metal (W/m/°C),
δd = pipe thickness (m).

The overall heat transfer coefficient generated by the sum of the inverse of all local heat transfer coefficients and the thermal conductivity of the metal is given by Eqs. 5 and 6.

$$h_{all} = \frac{1}{\frac{1}{(h_{nc})_H} + \frac{1}{h_{ip}} + \frac{1}{h_{metal}}} \quad (\textit{for horizontal pipe}) \tag{5}$$

and

$$h_{all} = \frac{1}{\frac{1}{(h_{nc})_V} + \frac{1}{h_{ip}} + \frac{1}{h_{metal}}} \quad (\textit{for vertical pipe}) \;, \tag{6}$$

where

h_{all} = overall heat transfer coefficient (W/m^2/°C),
h_{ip} = heat transfer coefficient inside circular pipe (W/m^2/°C),
h_{metal} = metal heat transfer conductance coefficient (W/m^2/°C),
$(h_{nc})_V$ = natural convection heat transfer coefficient for a vertical pipe (W/m^2/°C),
$(h_{nc})_H$ = natural convection heat transfer coefficient for a horizontal pipe (W/m^2/°C).

3. SURVIVAL CRITERIA

The analyses in this report are evaluated against the following three criteria. If all criterion are met, the consequences of the transient are acceptable. If at least one of the criterion is violated, the core may not be adequately cooled. These criteria are conservative, and violation of any of them does not necessarily imply core failure because of the low heat fluxes (i.e., decay heat levels) involved in these transients.

1. The temperature of the heavy water (boiling point at 10^5 Pa is ~100°C) at any point in the cooling loop should remain <100°C during the first 72 h after emergency reactor shutdown. This criterion is in place to guarantee natural circulation cooling—even in the case of total depressurization of the ANS system (atmospheric pressure) until power is restored and pool or secondary cooling is resumed. If this criterion were violated, steam voids could accumulate in the upper part of the hot leg, resulting in a vapor lock that would interfere with natural circulation flow.
2. The Costa[7] ratio, also called the flow excursion ratio (FER), must be always >1.0 to prevent parallel channel flow instability in the reactor core.
3. The critical flux ratio (CFR), based on the Gambill-Weatherhead[8] correlation, must always be >1.0 to prevent hot spot burnout.

4. ANALYSES ASSUMPTIONS

All the cases presented in this report assume the following:

- no heat is rejected to the secondary cooling system by the main heat exchangers,
- no emergency heat exchangers are present (removal assumed),
- no heat is removed from the pool by the secondary and essential cooling water systems,
- no heat is rejected to pool walls or building atmosphere,
- no heat is rejected to air filled cells,
- the water inside all the pools is well mixed, and
- the battery backup runs for 4 h (safety analysis assumes 0.5 h).

It is important to note that, when the reactor cooling system is depressurized, primary pump main motors and makeup system pumps are shut down. Assuming no leak to the pools or to the secondary system and using the CDR characteristics, the hot leg riser pressure will be at least 0.16 MPa under these conditions. This minimum pressure exists because the letdown tank gas pressure is at least 0.1 MPa, and the makeup surge line to the hot leg riser provides an extra 6.7 kPa head to the system as a result of the elevation of the makeup pipe surge line (6.7 m).

Three different configurations have been analyzed. Some of the details are summarized in Table 3. The configurations are described below:

- the CDR configuration, which includes models for the reactor vessel light water pool, the pipe chase light water pool, and four independent heat exchanger light water pools with four EHXs;
- the CDR configuration without the EHXs;
- an alternate design, where the containment has only two independent light water pools, the RVP, and a combined pool that includes the pipe chase and the four heat exchanger pools and no EHXs.

Table 3. Analyses parameters

Location	Heat transfer area (m^2)	Heat transfer coefficient (W/°C)	Heat load (kW)
Pump discharge	7.67	2,000	30
Main cold leg	15.50	3,800	60
Cold leg riser	15.50	5,500	80
Flow diode	2.50	1,000	10
Hot leg riser	18.00	5,600	280
Main hot leg	24.00	6,000	300
Loop hot leg	12.00	3,000	150
Loop cold leg	13.50	4,200	150
Pump suction	8.70	2,200	30

5. RESULTS

5.1 CONCEPTUAL DESIGN REPORT CONFIGURATION WITH EMERGENCY HEAT EXCHANGER

All the cases that have been studied with the original CDR design (EHXs available and all pools available including PCP, RVP, and HXPs) were successful. The temperature of the primary cooling circuit at the hot leg location (vessel exit) was always <100°C; therefore the pool temperature never reached boiling. Figure 2 shows a case where the EHX is available, no pony flow is available (natural circulation following the recator shutdown), and the reactor vessel pool is not available. The run was for 4 h only to show the significance of the heat removal by the EHXs.

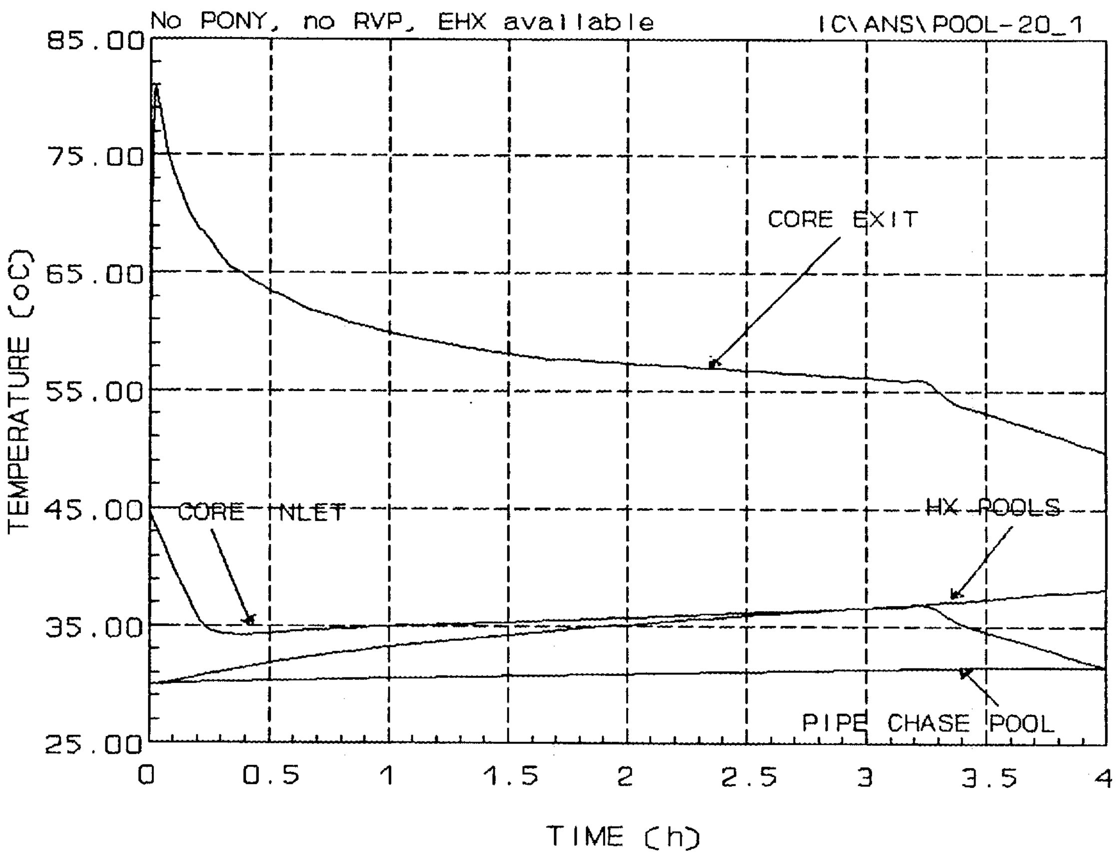

Fig. 2. Three operating loops with emergency heat exchanger available and reactor vessel pool and pony flow not available.

Figure 3 shows the corresponding EHX flow rate driven by natural circulation between the tube side and the pool.

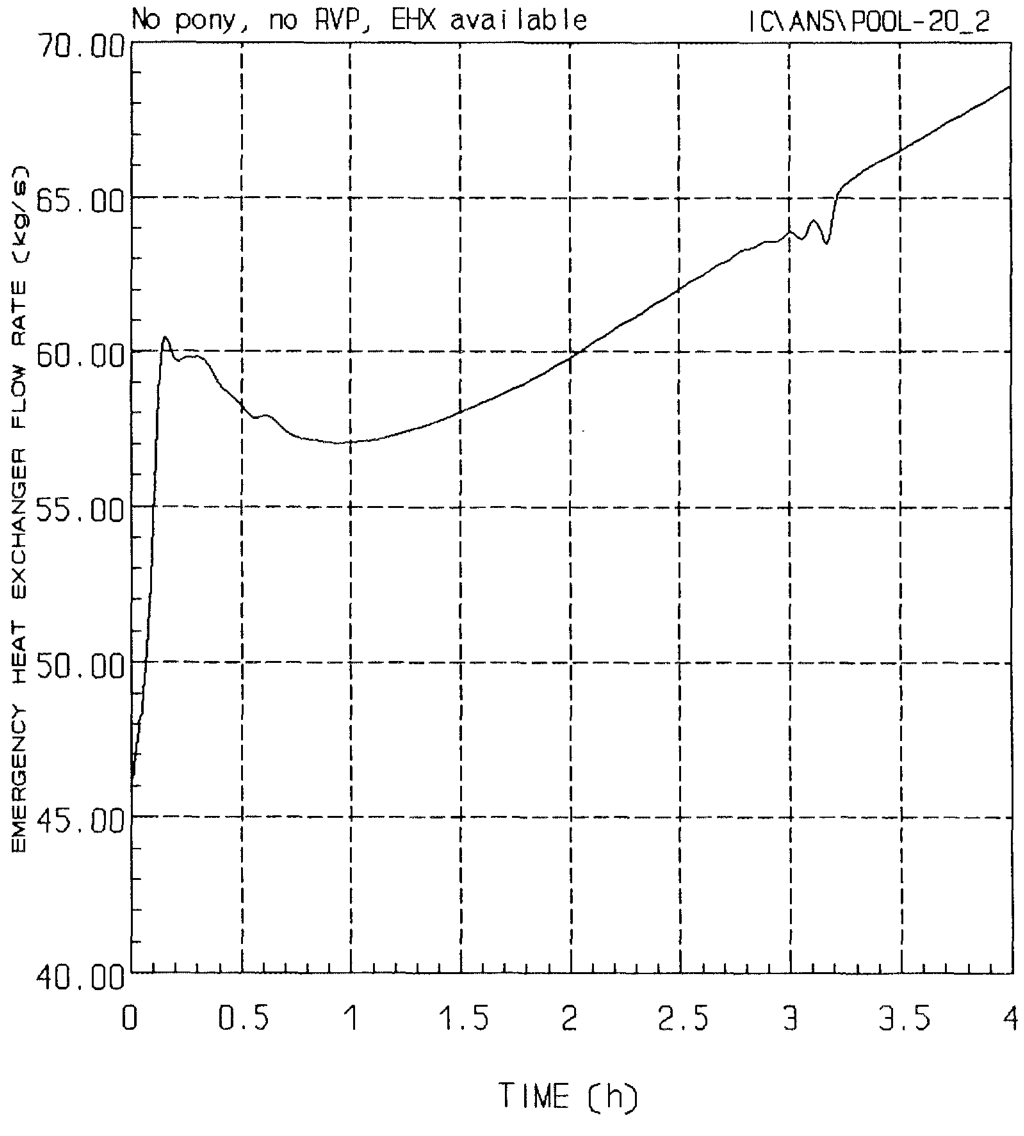

Fig. 3. Three operating loops with the emergency heat exchanger available and the reactor vessel pool and pony flow not available.

5.2 CONCEPTUAL DESIGN REPORT CONFIGURATION WITHOUT EMERGENCY HEAT EXCHANGERS

5.2.1 Three Active Loops with All Pools Active

This test case with three active coolant loops and their pools was performed to simulate the heat load of the pools. The primary pump remained running at 10% of the nominal speed following reactor shutdown for 4 h, which corresponds to the life of the battery backup. Then the pumps were shut down to 0% speed (natural circulation) for 3 d. The makeup pump was also shut down to 0% speed. Most of the heat from the core was generated by decay heat. Figure 4 shows the temperature profiles at the core entrance, core exit, and pools. The hot-leg temperature did not exceed 97°C, nor were the other two survival criteria (Sect. 3) exceeded within 72 h (3 d).

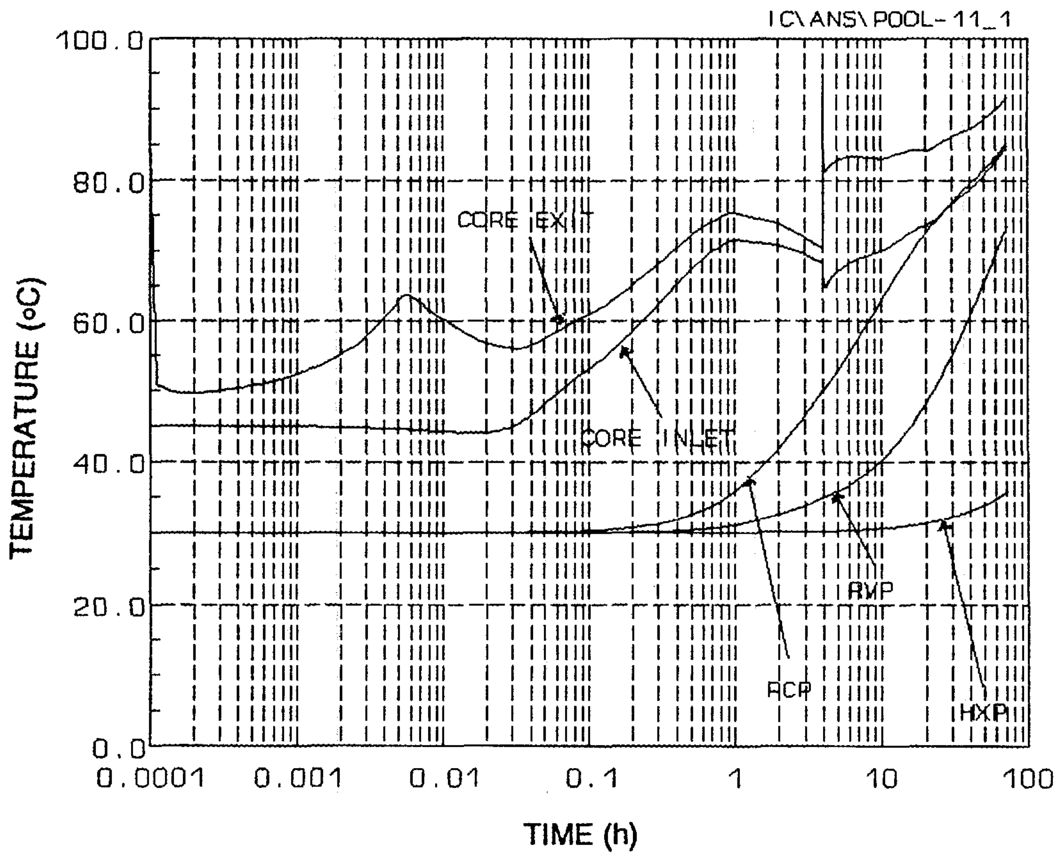

Fig. 4. Three operating loops with all pools active.

5.2.2 Three Active Loops with the Reactor Vessel Pool Dry

This case simulates the event where RVP is not available (i.e., it is assumed dry), the three primary loops are active, and the main pumps run at 10% of nominal speed (pony motor) following the reactor shutdown for 4 h, then shut down to 0% speed (natural circulation) for 68 h. The heat is transferred to the PCP and HXPs only. Figure 5 shows the calculated temperatures at the core inlet, core exit, PCP, and HXP, respectively. The temperature of the core outlet exceeded 100°C in 5h (the maximum temperature is 120°C), which violates one of the safety criteria.

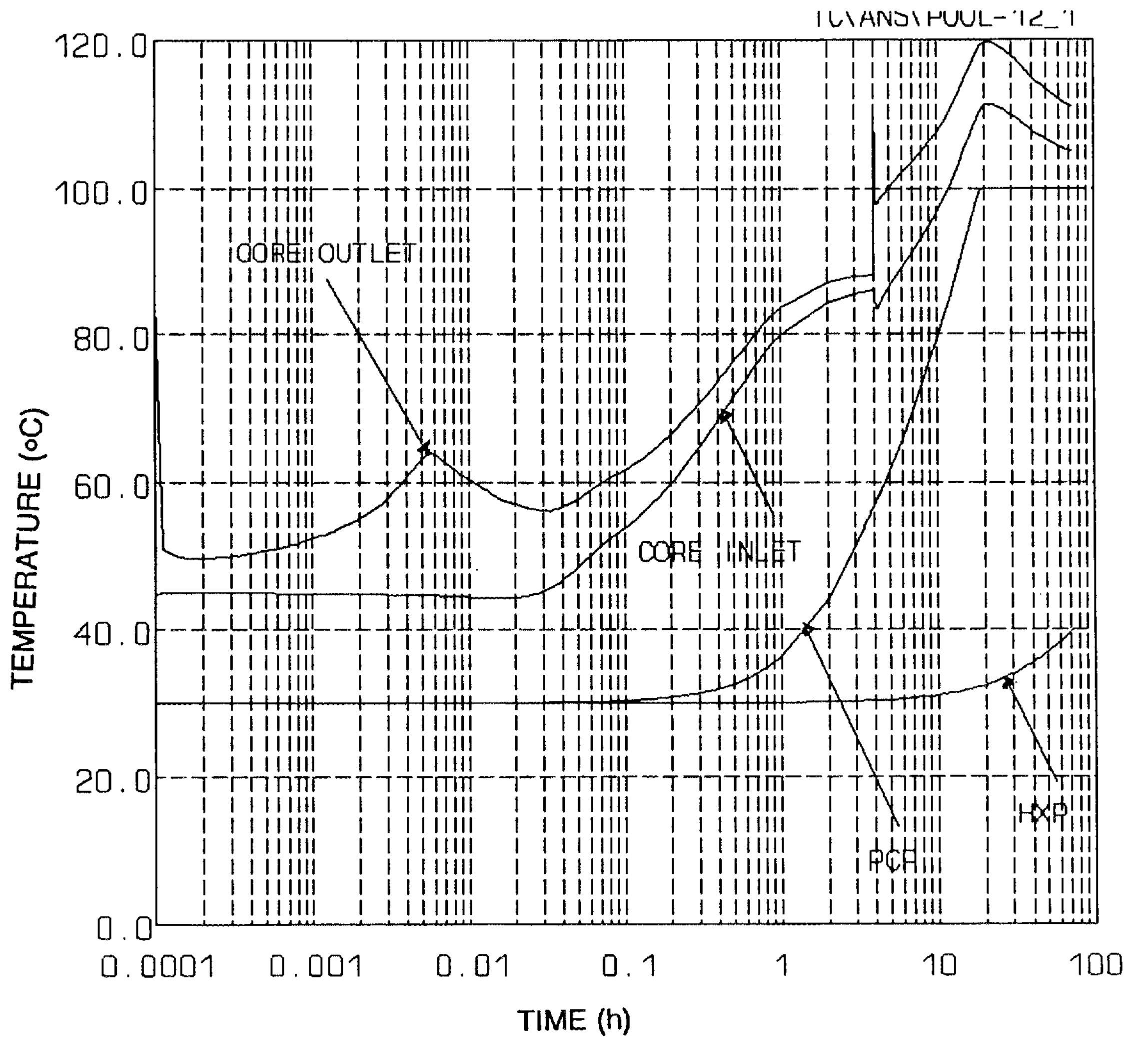

Fig. 5. Three operating loops with reactor vessel pool dry.

5.2.3 Three Active Loops with the Pipe Chase Pool Dry

This test simulates the event where the PCP is not available, the three primary loops are active, the main pumps run for 4 h at 10% of nominal speed (pony motor) following the reactor scram then shut down to 0% speed (natural circulation) for the rest of the transient. The heat is transferred to the RVP and HXPs only. Figure 6 shows the computed temperatures at the core entrance, core exit, RVP and HXPs, respectively. The temperature of the core outlet exceeded 100°C (the maximum temperature is 125°C) for most of the transient.

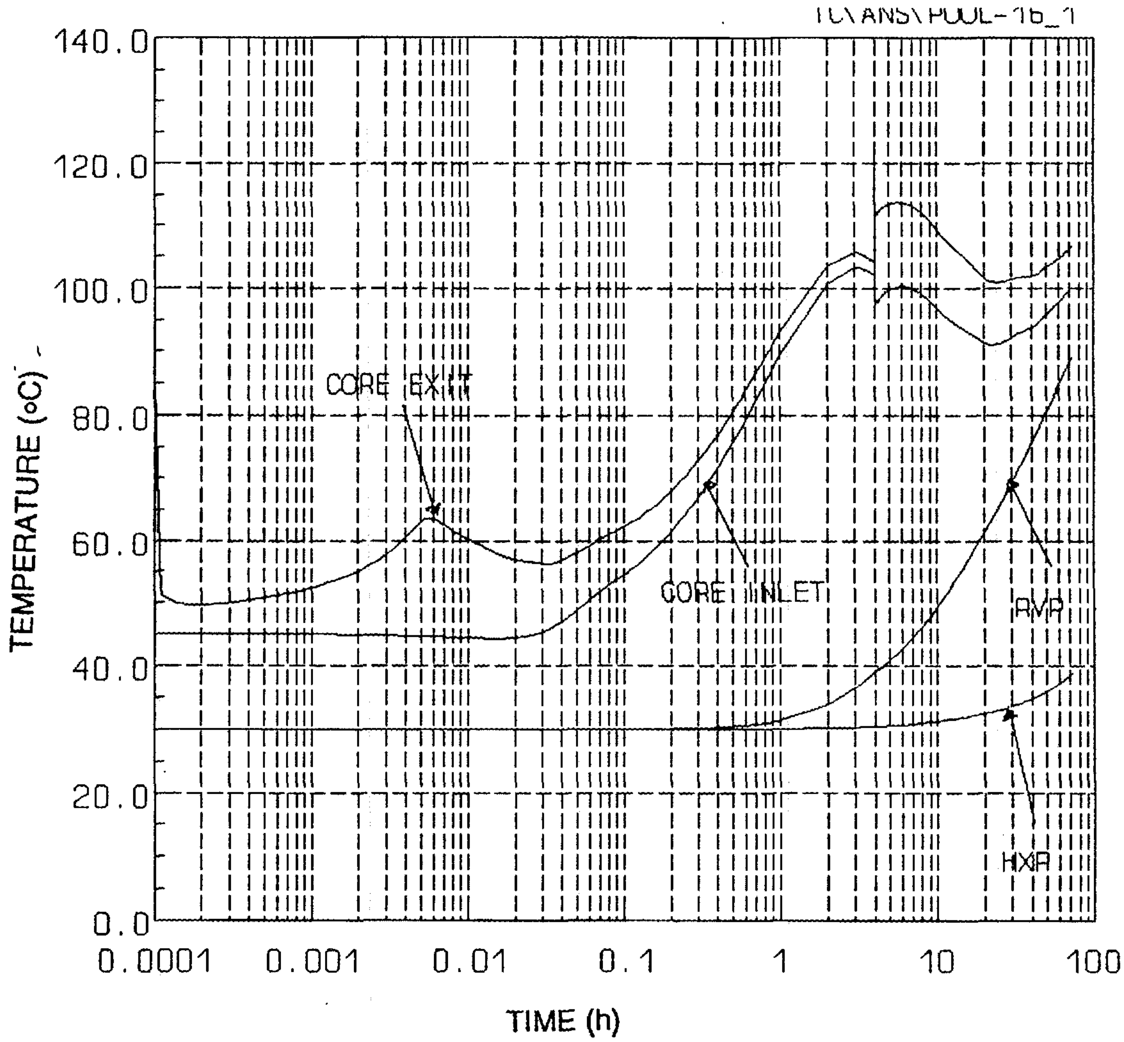

Fig. 6. Three operating loops with the pipe chase pool dry.

5.2.4 Three Active Loops with No Pony Motor and All Pools Available

This case simulates the event where the pony motors are not available and natural circulation starts at time 0. Three primary loops were active, the main pumps were shut down to 0% following the reactor shutdown for 72 h (3 d). All the heat from the primary cooling circuit is transferred to the PCP. The makeup pump was shut down to 0% speed. Thus, the system was depressurized, and the minimum system pressure at the horizontal hot leg is at least 0.16 MPa. Figure 7 shows the computed temperature at the core inlet, core exit, PCP, RVP, and HXP, respectively. The peak temperature of the core outlet was ~90° C within 72 h (3 d), which allows enough margin for the survival criteria.

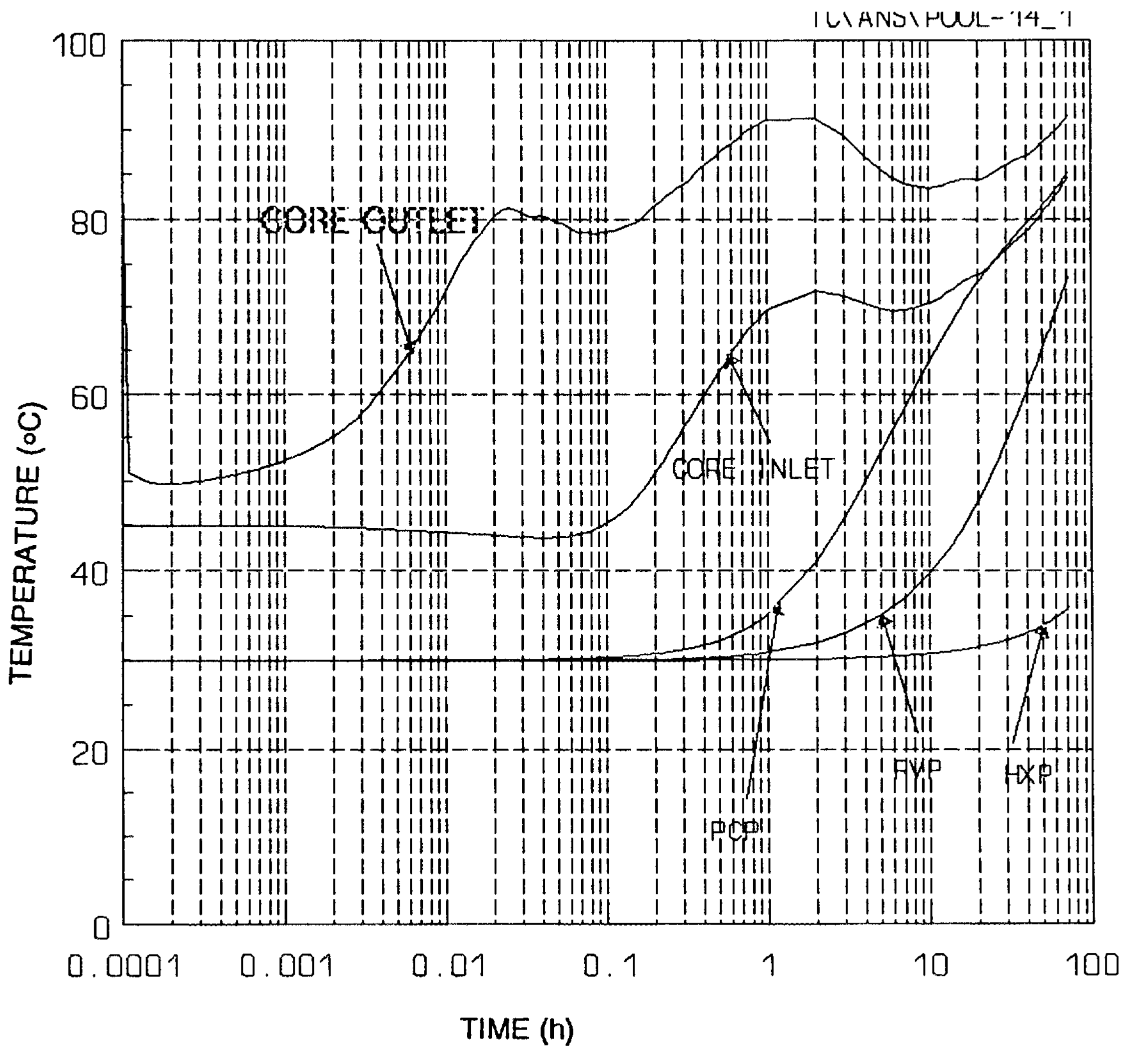

Fig. 7. Three operating loops with all pools active and no pony motor.

5.2.5 Two Active Loops with All Pools Are Available

This test describes the case where two loops were isolated (no flow), and two loops were active with primary pumps running at 10% of nominal speed for 4 h followed by shut down to 0% speed for the rest of the transient (68 h). The duration of this run was three days. Figure 8 shows the temperature shape at the core inlet, core outlet, PCP, HXP, and RVP, respectively. The temperature reached 100°C at the end of the transient (72 h); thus, the core safety limits were not violated in the duration of this event. This transient shows that the submerged primary piping system will eject sufficient heat generated by the reactor core decay heat after shutdown for)72 h.

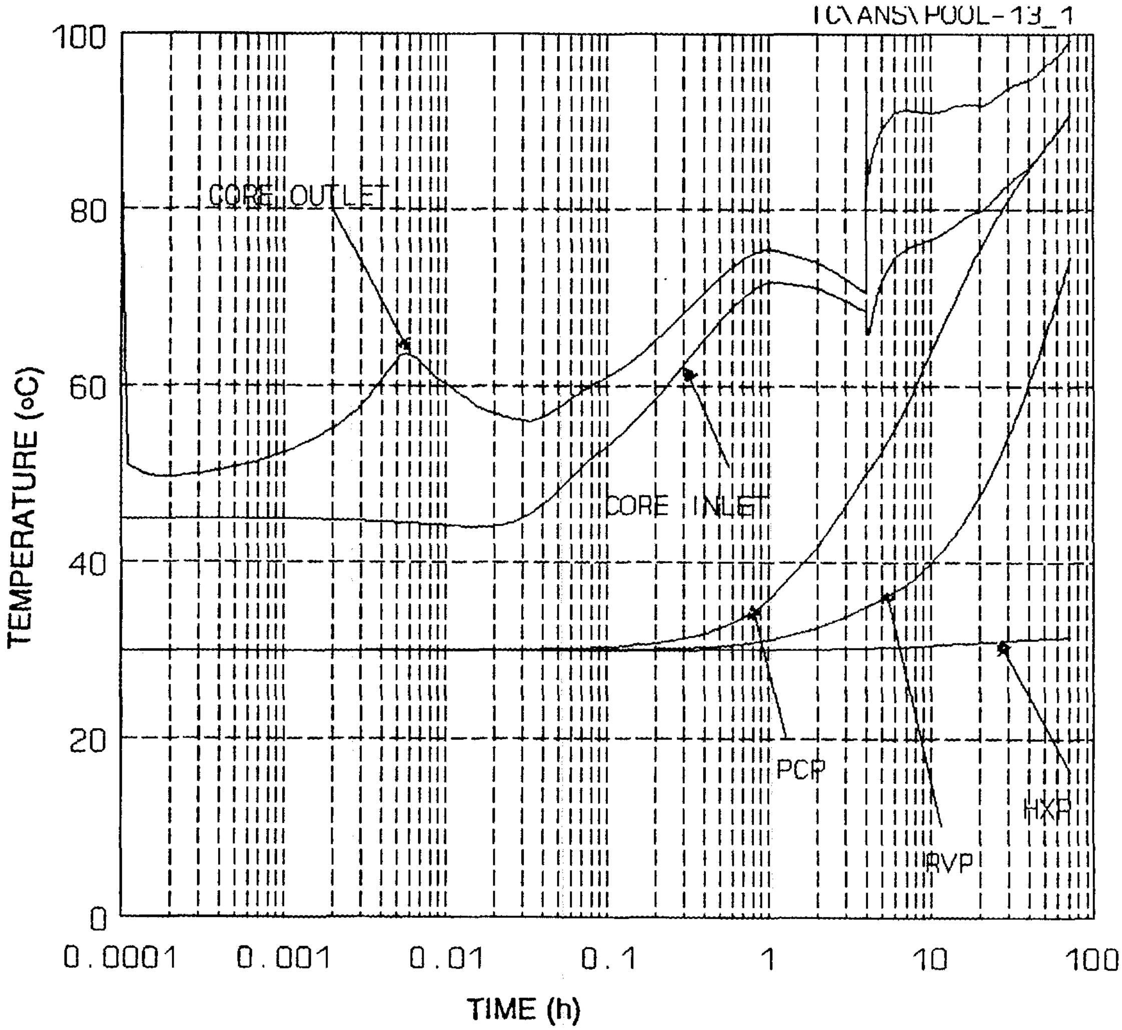

Fig. 8. Two operating loops with all pools active.

5.2.6 Two Active Loops with the Reactor Vessel Pool not Available

This test describes the case where two loops were isolated (no flow), and two loops were active with primary pumps running at 10% of nominal speed for 4 h followed by shut down to 0% speed for the rest of the transient (68 h). The RVP was isolated. The duration of this run was 3 d. Figure 9 shows the temperature profile of the core inlet and outlet, respectively. The core exit temperature exceeded 100°C within 20 h of the transient and reached a maximum of 115°C. Thus, the core safety limits were violated. This transient shows that the actual pool design will not be sufficient to avoid violating safety criteria.

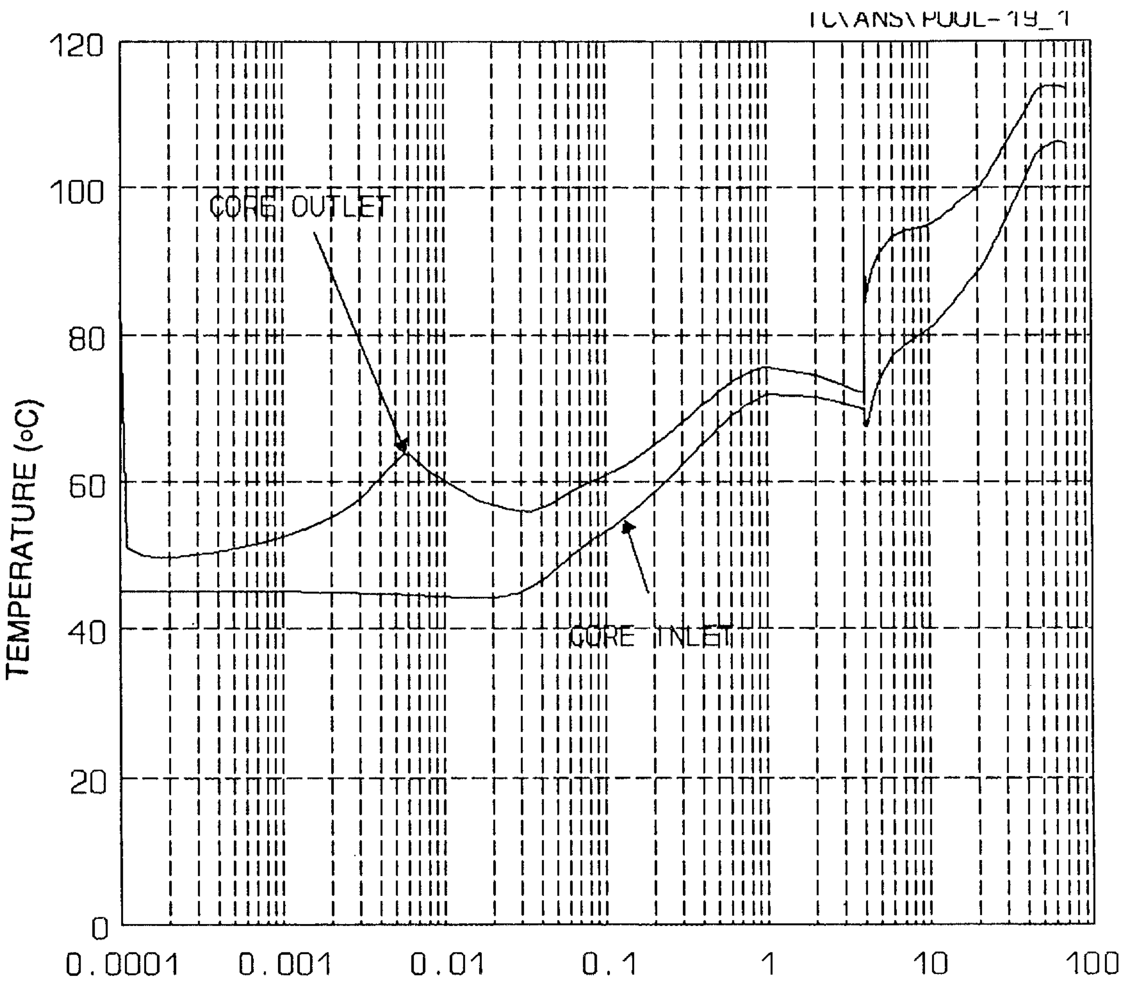

Fig. 9. Three operating loops with all pools active and no pony motor.

5.2.7 Two Active Loops with All Pools Available and No Pony Motor

This event describes the case where only two loops were active and the pumps were shut down to 0% speed following the reactor trip. Figure 10 shows the temperature profile at the core inlet, core outlet, and pools. The core temperature was kept <100°C during the 72-h transient.

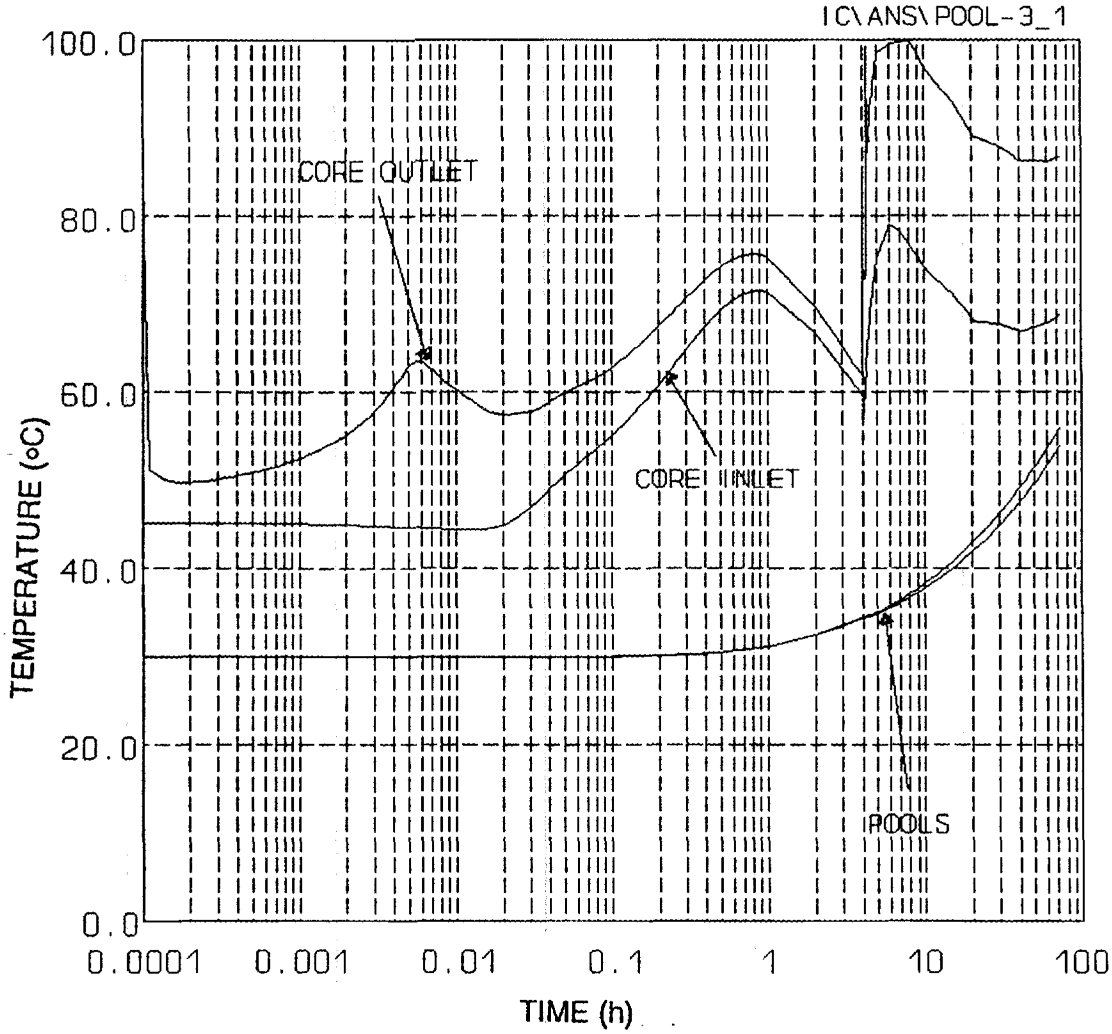

Fig. 10. Two operating loops with all pools active.

5.3 RESULTS FOR THE ALTERNATE CONFIGURATION WITHOUT EMERGENCY HEAT EXCHANGERS

As mentioned before, the difference between this configuration and Option 2 (CDR without EHX), is that the PCP is connected with all four HXPs, so that, for the hot leg (with the same heat transfer area as Option 2), the amount of light water pool available to cool it is larger than the pool available for Option 2 (almost 4 times). This additional water will help the pools to remove the same amount of heat without reaching the 100° C limit, but for the short run (first 2 or 3 h) it will not make a significant difference.

5.3.1 Three Active Loops with All Pools Available

This test case with three active coolant loops and their pools was performed to simulate the heat load of the pools. The primary pump remained running at 10% of the nominal speed, following reactor shut down, for 4 h (which corresponds to the life of the battery backup). Then the pumps were shutdown to 0% speed (natural circulation) for 3 d. The makeup pump was also shut down to 0% speed. Most of the heat from the core was generated by decay heat. Figure 11 shows the temperature at different locations. The hot leg temperature never exceeded 90° C, thus the safety limits were never violated.

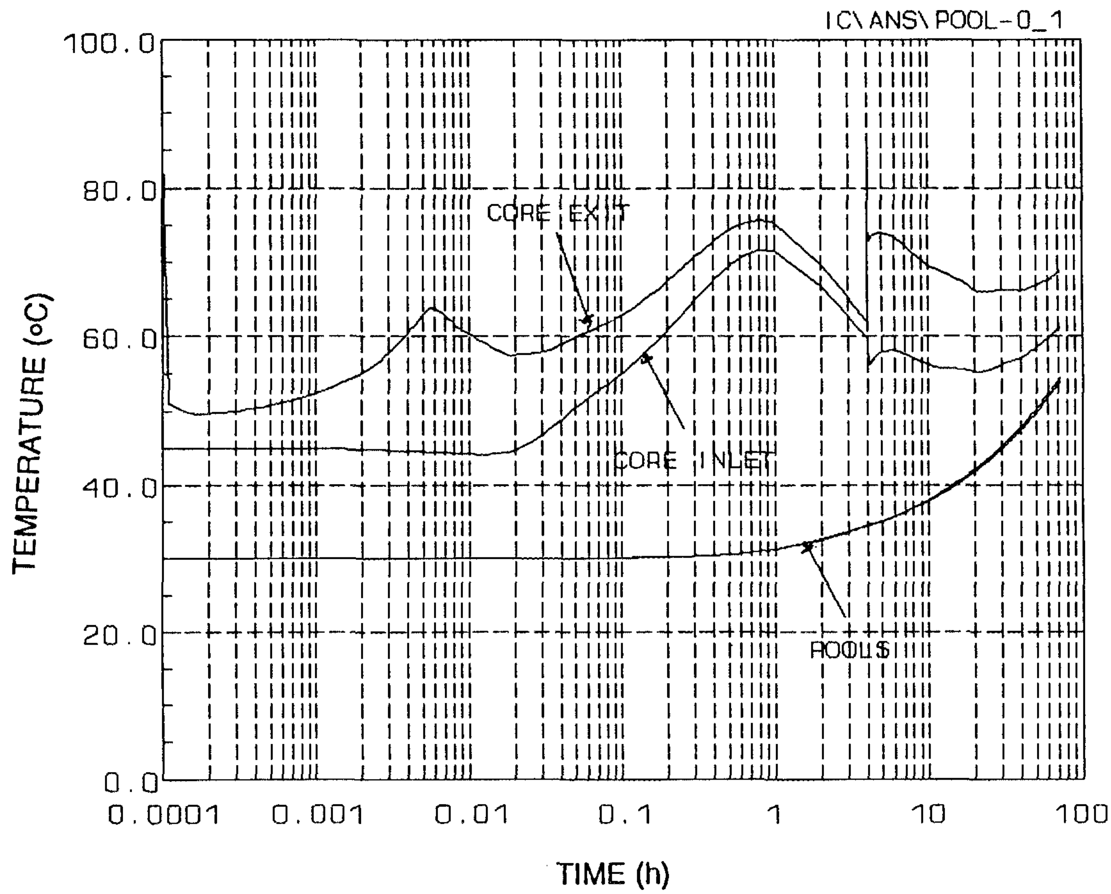

Fig. 11. Three operating loops with available pools are active.

5.3.2 Three Active Loops with the Reactor Vessel Pool Dry

This case simulates the event where the RVP is not available, the three primary loops are active, the main pumps run at 10% of nominal speed (pony motor) following the reactor shut down for 4 h then shut down to 0% speed (natural circulation) for 68 h. The heat is transferred to the PCP only. The system is depressurized, and the minimum system pressure at the horizontal hot leg is at least 0.16 MPa. Figure 12 shows the calculated temperature at the core inlet, core exit, and the PCP, respectively. The temperature of the core outlet never exceeded 90°C, which allows enough margin for the core critical safety criteria. The significant difference between Option 2 and 3 is particularly shown in Fig. 12. In Fig. 5 (same run for Option 2), the temperature reached 100°C and kept increasing 5 h after reactor shutdown. However, in the alternate design (Option 3), the temperature never reached the boiling point.

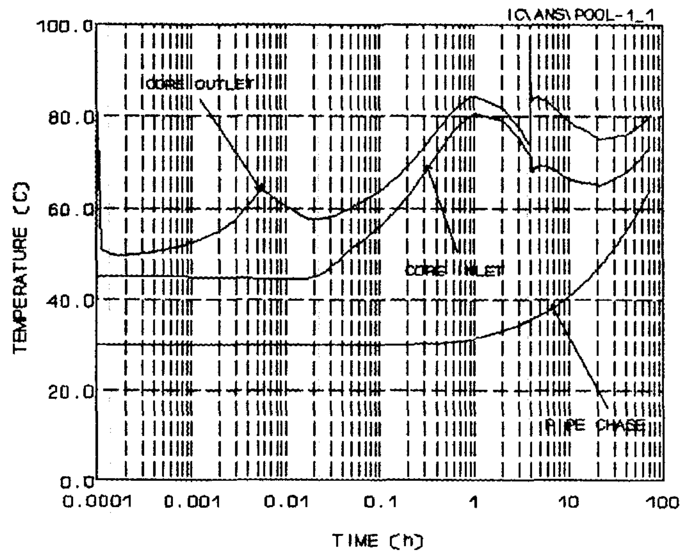

Fig. 12. Three operating loops with the reactor vessel pool dry.

5.3.3 Three Active Loops with All Pools Available and No Pony Motor

This event describes the case where three loops were active and the pumps were shut down to 0% speed following the reactor trip. Figure 13 shows the temperature profile at the core inlet, core exit, and pools. The core temperature was kept <100°C during 72-h transient.

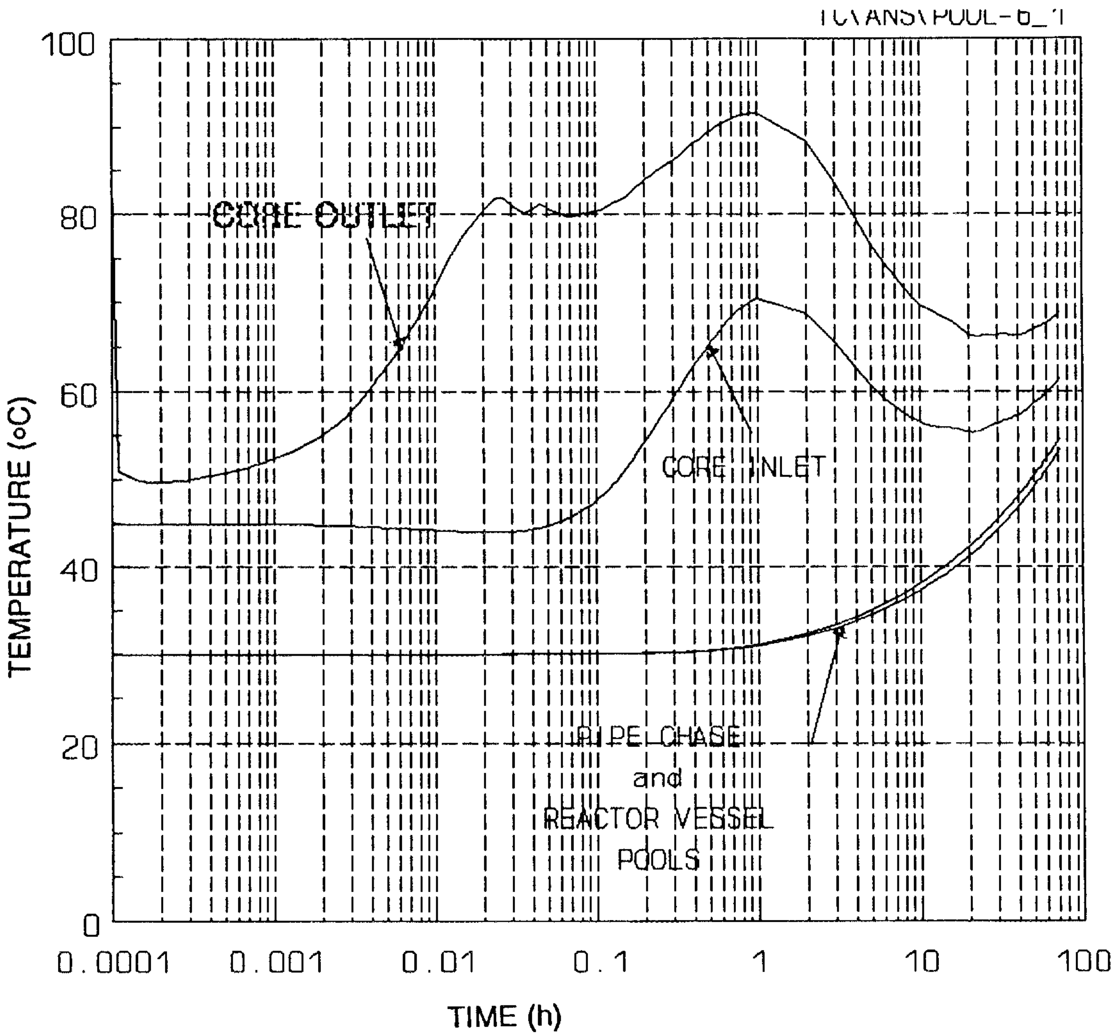

Fig. 13. Three active loops with all pools available and no pony motor.

5.3.4 Two Active Loops with the Reactor Vessel Pool Not Available

This case simulates the event where the RVP is not available, two primary loops are active, and the main pumps run for 4 h at 10% of nominal speed (pony motor) following the reactor scram before being shut down to 0% speed (natural circulation) for the rest of the transient (68 h). All the heat from the primary cooling circuit is transferred to the PCP only. The makeup pump was shut down to 0% speed. Thus, the system was depressurized, and the minimum system pressure at the horizontal hot leg is at least 0.16 MPa. Figure 14 shows the computed temperature at the core inlet, the core exit, and the PCP, respectively. The peak temperature of the core outlet was ~90° C, which allows enough margin to meet the core critical safety criteria.

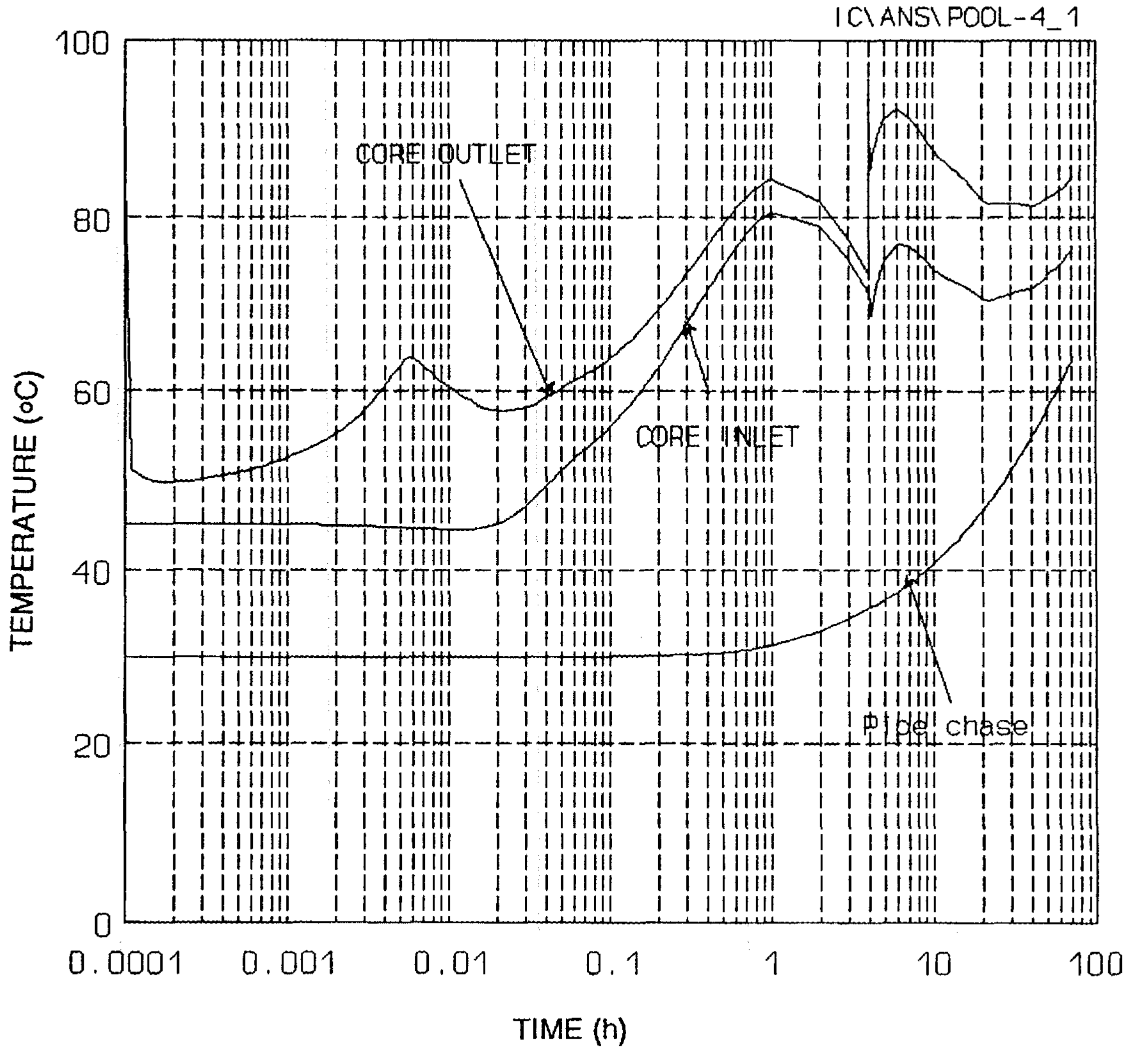

Fig. 14. Two operating loops with the reactor vessel pool dry.

5.3.5 Two Active Loops with All Pools Available and No Pony Motor

This even describes the case where only two loops were active and the pumps were shut down to 0% speed following the reactor trip. Figure 15 shows the temperature profile at the core inlet, the core outlet, and pools. The core temperature was kept ‹100oC during the 72-h transient.

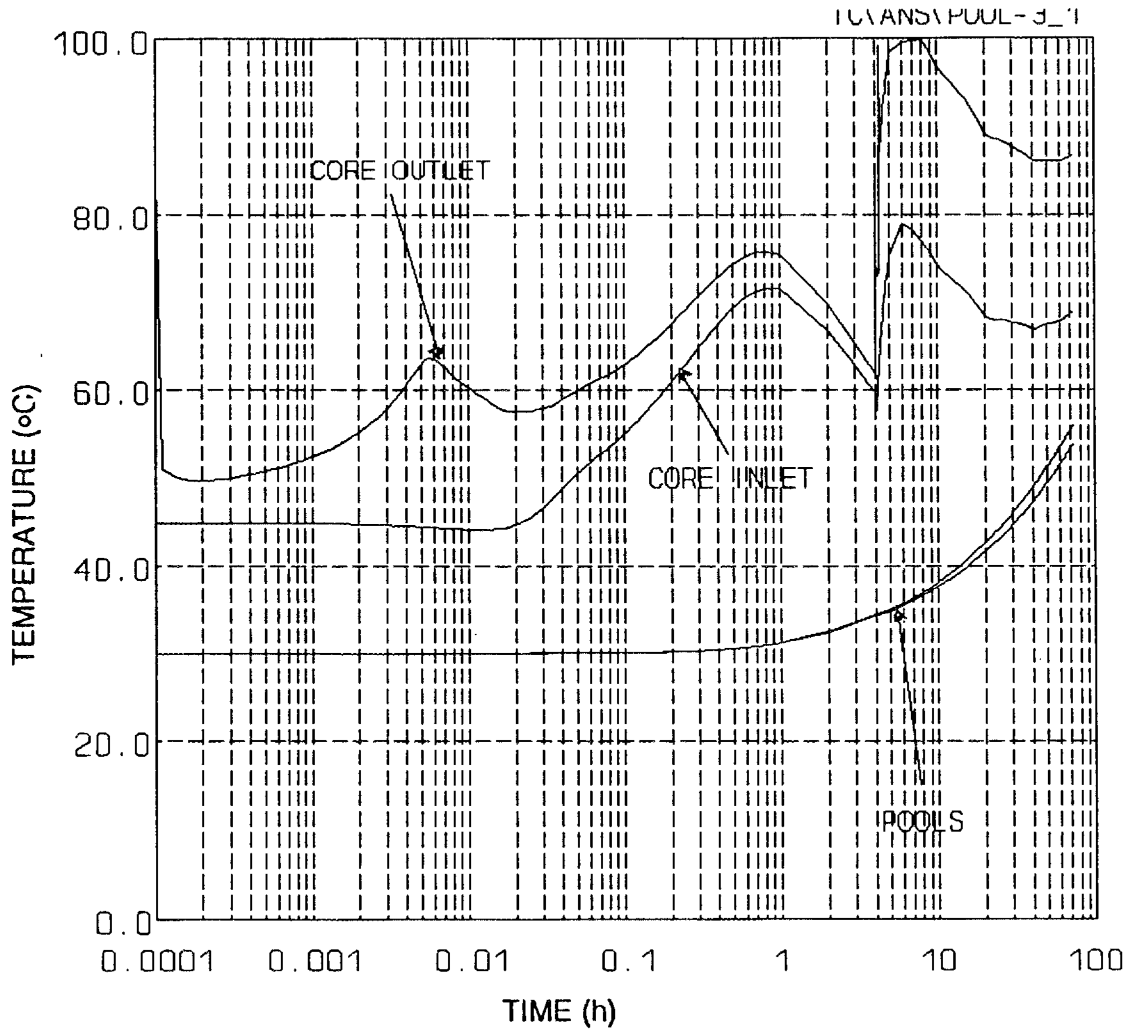

Fig. 15. Two operating loops with all pools active and no pony motor.

6. CONCLUSIONS AND RECOMMENDATIONS

Table 4 summarizes the results of these analyses. From this table, the minimum requirements to survive the class of loss of secondary cooling events are derived for analysis. The requirements are dependent on the number of loops in operation as described below.

1. If the emergency heat exchangers are installed as per the conceptual design, then the Advanced Neutron Source reactor will survive all the cases studied.
2. If the emergency heat exchangers are omitted, then all of the pools must be available for the conceptual design to work .
3. For the alternate design, where the pipe chase pool is connected to all the four heat exchanger pools, all cases were successful.

The main conclusion that comes from these analyses is that the Advanced Neutron Source reactor cooling system does not need the emergency heat exchangers for the event types presented in this report to remove heat generated by the decay heat after reactor shutdown. The primary submerged piping system will remove all the heat from the primary side to the pools. This study also demonstrates that the heat removal response would be more advantageous for the system if the pipe chase pool and all the heat exchangers pools were connected.

Table 4. Summary of results

Case No.	Configuration						Results	
	Type	EHX[a]	# Loops	RVP[b]	PCH[c]	PONY[d]	T_{max}[e] (°C)	$t_{s}\_T_{100}$[f] (h)
1	All the studied cases were successful							
2	CDR[g]	No	3	No	Yes	Yes	120	4
				Yes	No	Yes	125	2
				Yes	Yes	No	92	*[h]
				Yes	Yes	Yes	100	*
			2	No	Yes	Yes	115	20
				Yes	Yes	No	105	1
3	ALTERNATE	No	3	No	Yes	Yes	90	*
				Yes	Yes	No	90	*
			2	No	Yes	Yes	92	*
				Yes	Yes	No	99	*

[a]EHX = emergency heat exchanger.
[b]RVP = reactor vessel pool.
[c]PCH = pipe chase pool.
[d]PONY = 10% of the nominal pump speed.
[e]$t_{s}\_T_{100}$ = time when core outlet temperature reached 100°C.
[f]T_{max} = maximum core outlet temperature.
[g]CDR = conceptual design report.
[h]* = did not exceed safety limits within 72 h (3 d).

7. REFERENCES

1. M. I. Khayat, The Advanced Neutron Source Dynamic Modeling and Inertial Flow Diode Concept, Ph.D. dissertation, The University of Tennessee, Knoxville, 1992.

2. V. T. Morgan, The Overall Convective Heat Transfer from Smooth Circular Cylinders, pp. 199 264, *Advances in Heat Transfer*, Vol. 16. Academic Press, New York, 1975.

3. W. H. Mc Adams, *Heat transmission*, 3d ed., McGraw-Hill, New York, 1954.

4. F. J. Bayley, Analysis of Turbulent Free Convection Heat Transfer. *Proceedings of the Institute of Mechanical Engineers*, London, **169**, 361 (1955).

5. C. Y. Warner and V. S. Arpaci, An Investigation of Turbulent Natural Convection in Air at Low Pressure along a Vertical Heated Flat Plate, *Int. Journal Heat and Mass Transfer*. **11**, 397 406 (1968).

6. F. W. Dittus and L. M. K. Boetler, Title of Article, *University of California, Berkeley, Public Engineering*, **2**, 443 (1930).

7. J. Costa, Measurement of the Momentum Pressure Drop and Study of the Appearance of Vapor and Change in the Void Fraction in Subcooled Boiling at Low Pressure, presented at the meeting of the European Group Double-Phase, Winfrith, France, 1967. Translated from French as ORNL/TR-90/21, Martin Marietta Energy Systems, Inc,. Oak Ridge National Laboratory, 1990.

8. R. J. Weatherhead, *Nucleate Boiling Characteristics and the Critical Heat Flux Occurrence in Subcooled Axial-Flow Water System*, ANL-6675, Argonne National Laboratory, Chicago, March 1963.

Internal Distribution

1. J. L. Anderson
2. B. R. Appleton
3. R. E. Battle
4 8. J. H. Campbell
9. N. C. J. Chen
10. D. Fry
11. R. M. Harrington
12 15. R. L. Johnson
16. J. A. March-Leuba
17. B. S. Maxon
18. G. R. McNutt
19. D. L. Moses
20. F. J. Peretz
21. C. C. Queen
22. D. L. Selby
23. J. P. Schubert
24. M. Siman-Tov
25. M. W. Wendel
26. C. D. West
27. G. L. Yoder
28. ORNL Patent Office
29 30. Central Research Library
31. Y-12 Technical Library, Document Reference Section
32 33. Laboratory Records
34. Laboratory Records (RC)

External Distribution

35. R. Awan, U.S. Department of Energy, NE-473, Washington, DC 20585
36. K. K. Conway, Laboratory Facilities Branch, U.S. Department of Energy, Oak Ridge Operations Office, CE-523, P.O. Box 2001, Oak Ridge, TN 37831-2001
37. R. A. Hunter, Director, Office of Facilities, Fuel Cycle, and Test Programs, Nuclear Energy Division, U.S. Department of Energy, NE-47, Washington, DC 20585
38. T. L. Kerlin, University of Tennessee, College of Engineering 315 Paxqua Engineering Building, Knoxville, TN 37996-2300
39. J. P. Mulkey, U.S. Department of Energy, NE-473, Washington, DC 20585
40. W. T. Oosterhuis, Materials Sciences Division, Office of Energy Research, U.S. Department of Energy, ER-132, Washington, DC 20585
41. I. Thomas, Director, Materials Sciences Division, Office of Energy Research, U.S. Department of Energy, ER-13, Washington, DC 20585
42 43. U. S. Department of Energy, ANS Project Office, Oak Ridge Operations Office, FEDC, MS-8218, P.O. Box 2009, Oak Ridge, TN 37831-8218
44. Office of Assistant Manager for Energy Research and Development, U.S. Department of Energy, Oak Ridge Operations Office, P.O. Box 2001, Oak Ridge, TN 37831-2001
45 46. Office of Scientific and Technical Information, P.O. Box 62, Oak Ridge, TN 37831