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BWR ATWS MITIGATION BY FINE MOTION CONTROL ROD

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ABSTRACT

Two main methods of ATWS mitigation in a SBWR are: Fine Motion Control Rods (FMCRD) and Boron injection via the Standby Liquid Control System (SLCS). The study has demonstrated that the use of FMCRD along with feedwater runback mitigated the conditions due to reactivity insertion and possible ATWS in a BWR which is similar to SBWR.

I. INTRODUCTION

Historically, safety analysis of the Light Water Reactor (LWR) has focused on the Loss Of Coolant Accident (LOCA) and the Anticipated Transient Without Scram (ATWS). The new generation of advanced LWRs such as the Simplified Boiling Water Reactor (SBWR)[1] designed by General Electric (GE) utilizes passive mechanisms to enhance the safety with regard to these accidents. The present work represents a study on the SBWR safety regarding the ATWS using the upgraded version of RAMONA-3B[2] with 3D neutron kinetics. The latest upgraded version is called RAMONA-4B[3].

The strong nuclear-thermal-hydraulic coupling in a SBWR, due to the natural circulation used for the reactor coolant flow, requires an accurate calculation of soluble boron in the multiple parallel channels of the core. To this end, a new detailed boron transport model has been implemented in the upgraded version RAMONA-4B.

The Standby Liquid Control System (SLCS) of a SBWR is a backup shutdown system to be used in case of failure of the normal scram system. The system consists of an accumulator tank, maintained at a high pressure, a piping system with control logic, and a high velocity core injection system.

The SBWR has an additional shutdown system called Fine Motion Control Rod Drive (FMCRD) system. This is a much slower shutdown mechanism than the normal scram system. GE claims that the FMCRD system in conjunction with feedwater runback, can shut down the reactor in the event of an ATWS. The present work assesses the capability of the FMCRD system along with feedwater runback in shutting down the reactor in the events leading to reactivity insertion and possible ATWS. The RAMONA-4B code has been used for the analysis. This study is part of a larger study on ATWS mitigation and instabilities in SBWR.

II. CALCULATIONAL MODEL

RAMONA-4B is a detailed best-estimate thermal-hydraulics computer code with 3D neutron kinetics, capable of modeling a full core with 800 neutronic channels and 200 thermal-hydraulic channels along with 24 axial cells. The hydraulic model is based on non-equilibrium drift-flux formulation for two-phase flow with provision for flow reversal.[4] The neutronic model is based on a well-established 1½-group diffusion theory.[2] The three-dimensional neutron kinetics is an important feature of the calculational model described below.

The RAMONA-4B calculational model used in the present assessment is shown in Figure 1. It includes the reactor pressure vessel with all important internal components (reactor core, upper plenum and riser, steam separator and dryer, steam dome, downcomer, lower plenum) and the steam lines and control systems. The reactor core is modeled with 101 neutronic channels and 25 thermal-hydraulic channels assuming eighth-core symmetry as shown in Figure 2. Twenty four axial cells are used in each of the multiple core channels in order to obtain accurate axial power and void distributions.

The nuclear parameters for the 3D neutron kinetics correspond to Browns Ferry Unit 3, a typical BWR4. The cross sections and their feedback coefficients were generated to represent the end of cycle 5 (8766 MWD/MTU). The three-dimensional exposure and the history-dependent void distributions were taken into account using the auxiliary code BLEND[5] to produce 77 sets of cross sections and the corresponding feedback coefficients. These cross section sets have been used to predict both the radial and axial power distributions in very good agreement with the Browns Ferry-3 cycle-5 measurements.[6]

III. ANALYSIS OF TRANSIENTS

In the present study, a loss of feedwater heating transient [Chpt 15.8, Ref 7] was simulated to investigate the adequacy of the FMCRD system in mitigating reactivity insertion event and possible ATWS in a BWR similar to the SBWR.

1. Transient Description

The transient selected for this analysis is an ATWS event induced by the loss of feedwater heating together with the assumed failure of the normal and alternate scram systems except for the FMCRD. The loss of feedwater heating can be caused by either of the two ways: (1) steam extraction line to a heater is closed, and (2) feedwater is bypassed around the heaters. The total number of unavailable feedwater heaters determines the net loss of heating. Figure 3 shows a schematic of a balance of plant for SBWR[7]. In the SBWR the maximum reduction in feedwater temperature for a single failure case is limited to 55.6 °C. The loss of feedwater heating will result in an increase in core inlet subcooling. This will lead to an increase in the reactor power due to the negative void reactivity feedback in the core. The thermal power increases slightly to a new equilibrium value. This transient does not activate any ATWS logic. In order to investigate the effectiveness of the new SBWR feature, Fine Motion Control Rod Drive (FMCRD) run-in, it is assumed that the FMCRD run-in will be initiated manually after the loss of feedwater heating. The consequences of FMCRD is to move the power peaking towards the exit of the channel as the colder fluid moves up in the channel.

The geometric data and the setpoints are specified in accordance with the conceptual design of the GE SBWR[7]. However, the core represents a regular BWR core with the Browns Ferry Unit 3 cross sections at the end of cycle 5. This is due to the unavailability of the SBWR cross sections at the time of the present study.

2. Results of Calculations

A feedwater temperature reduction of 55.6 °C was initiated at 5 seconds into the transient as shown in Figure 4. The temperature reduction of 55.6 °C is conservative, since a temperature drop of 16.7 °C indicated by the feedwater control system (FWCS) sends a signal to the Selected Control Rod Run-In (SCRRI), to reduce core power and thereby, avoid scram. The loss of feedwater heating will result in an increase of the core inlet subcooling. The inlet subcooling will change from 10.1 to 17.5 °C as shown in Figure 5. The Figure 6 shows that the reactor power increases by 13% within 80 seconds due to the increased inlet subcooling. The reactor condition settles to a new steady state after 75 seconds. Figure 7 shows the axial power profile at the beginning of the transient and at 60 seconds. As the cold water reaches the core inlet, the voiding decreases leading to a bottom peaking power profile.

The loss of feedwater heating transient is a slow transient where the normal and alternate scram systems, except for the FMCRD, were assumed to have failed during this transient. The loss of feedwater heating transient is followed by the FMCRD run-in at 80 seconds, which allows slow insertion of the control rods and by the shutdown of feedwater flow in 10 seconds, as shown in Figure 8.

Figures 7, 9 and 10 indicate that the FMCRD insertion begins at 80 seconds and continues until 180 seconds for full insertion. The slow insertion of control rods results in a quasi-steady power profile, which is skewed to the top of the core. Figure 8 shows that the steam flow rate has decreased to 7.5% of the rated value within 100 seconds. The pressure also reaches a new equilibrium value within 100 seconds as shown in Figure 11. Figure 12 shows the hot channel fuel temperature along with the average channel fuel temperature during the transient. The hottest channel fuel temperature is within the safety limit. The peak cladding temperature was found to be limited to 296 °C, as shown in Figure 13, which is within the saf[illegible] limit.

Figure 14 shows the fuel temperature for the case where feedwater flow was maintained at the nominal value. The peak fuel temperature is about 300°C higher then in the case with the feedwater pump tripped (as shown in Figure 12).

IV. SUMMARY AND CONCLUSIONS

The adequacy of the FMCRD in ATWS mitigation for the SBWR was investigated by using the RAMONA-4B code. The following conclusions can be drawn from the present study:

1. The FMCRD together with the feedwater runback shut down the reactor during the postulated loss of feedwater heater [Chpt 15.8, Ref 7] event without any fuel damage.

2. Because of the very slow rod insertion (100 seconds to move 2.743 m), the FMCRD without the feedwater runback would lead to fuel he.'ıp. It may also result in a hot spot exceeding the fuel melting point. This clearly indicates that the feedwater runback is indeed needed for this type of event as postulated for the SBWR.

ACKNOWLEDGEMENTS

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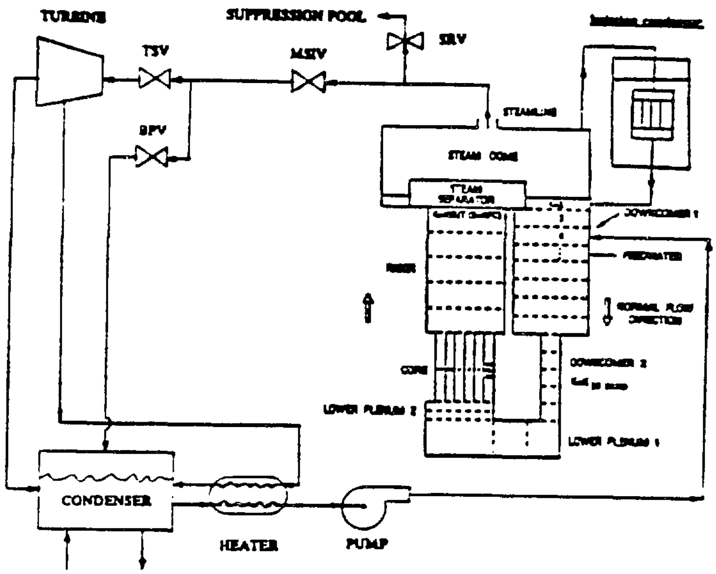

Fig. 1. SBWR schematic diagram for RAMONA-4B calculation model.

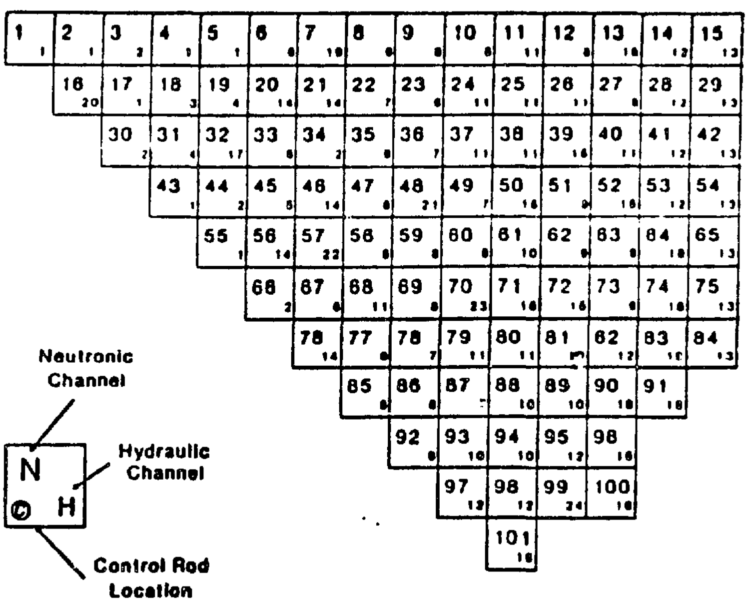

Fig. 2. RAMONA-4B 1/8 core model for the present study.

DISCLAIMER

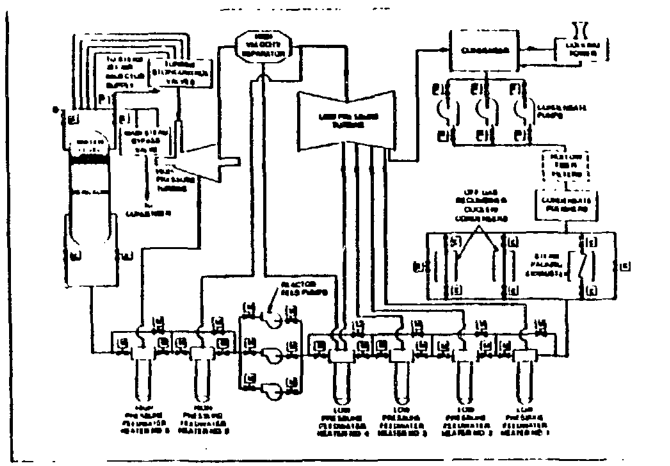

Fig. 3 SBWR balance of plant schematic.

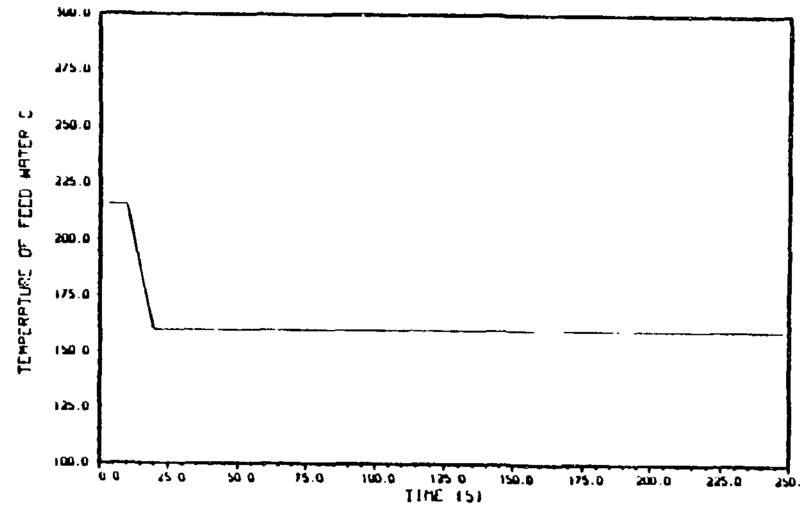

Fig. 4 Feedwater temperature during the transient.

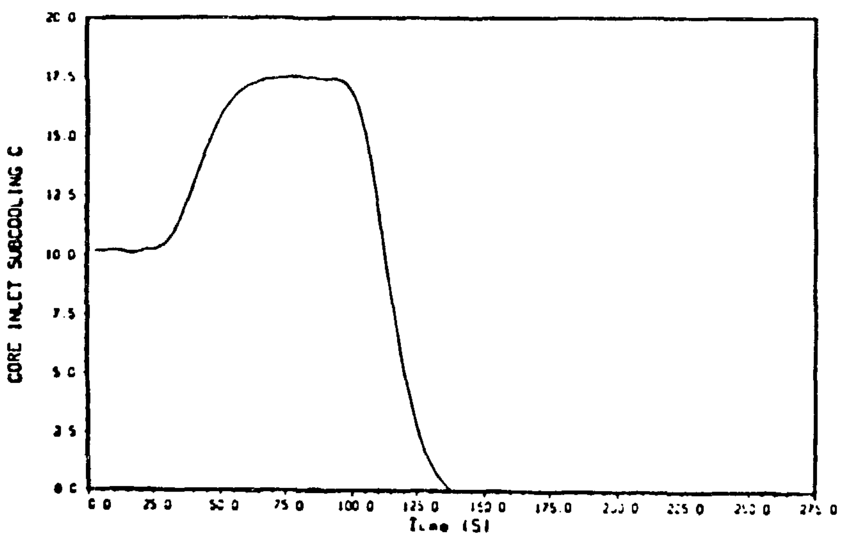

Fig. 5 Core inlet subcooling for loss of feedwater heating ATWS with fine motion control rod insertion.

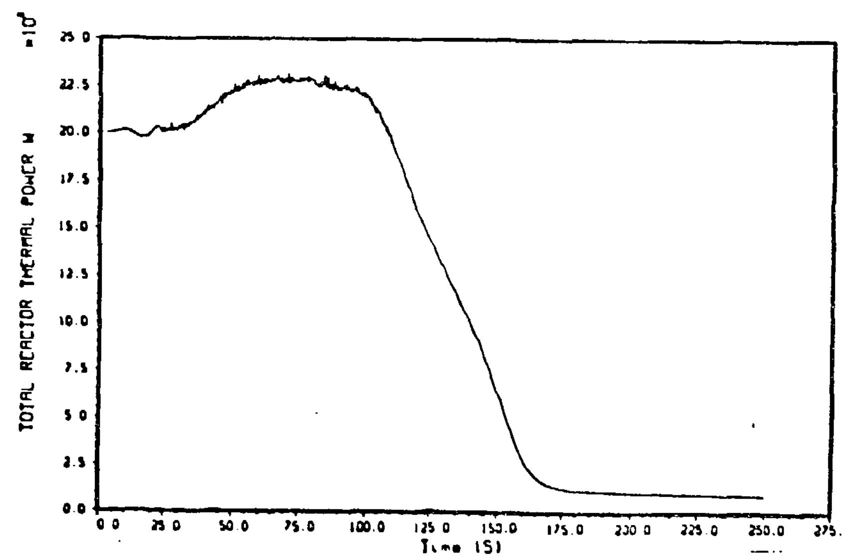

Fig. 6 Thermal power during loss of feedwater heating ATWS with fine motion control rod insertion.

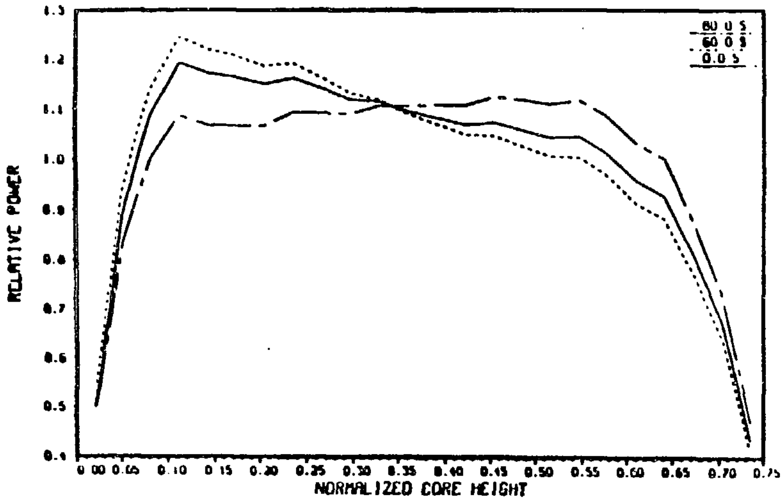

Fig. 7 Axial power profile between 0 s and 80 s.

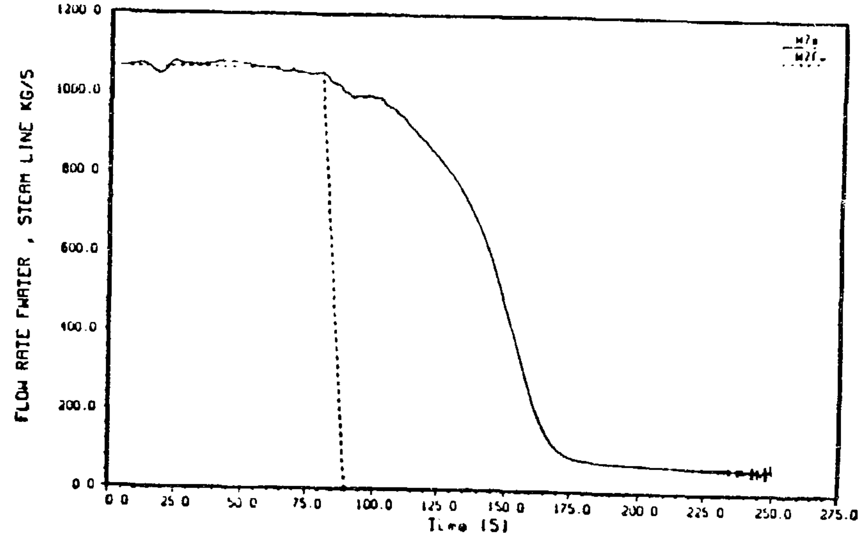

Fig. 8 Feedwater and steam flow during loss of feedwater heating ATWS with fine motion control rod insertion.

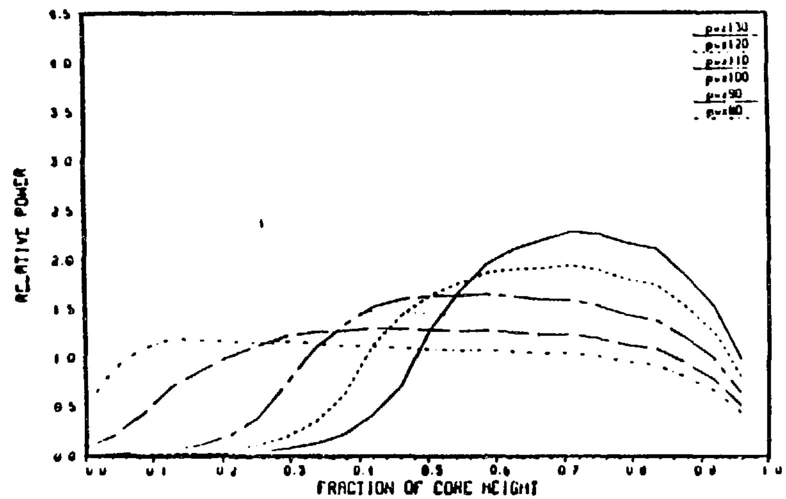

Fig. 9 Axial power profile in the core between 80 s and 130 s.

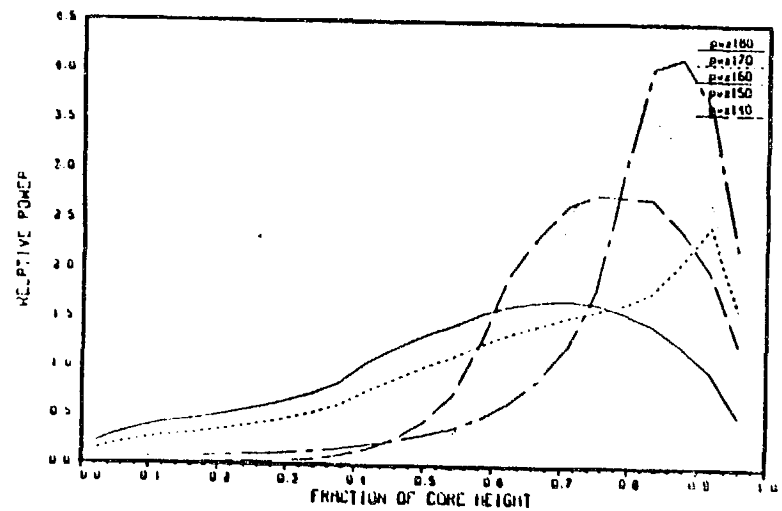

Fig. 10 Axial power profile between 140 s and 180 s.

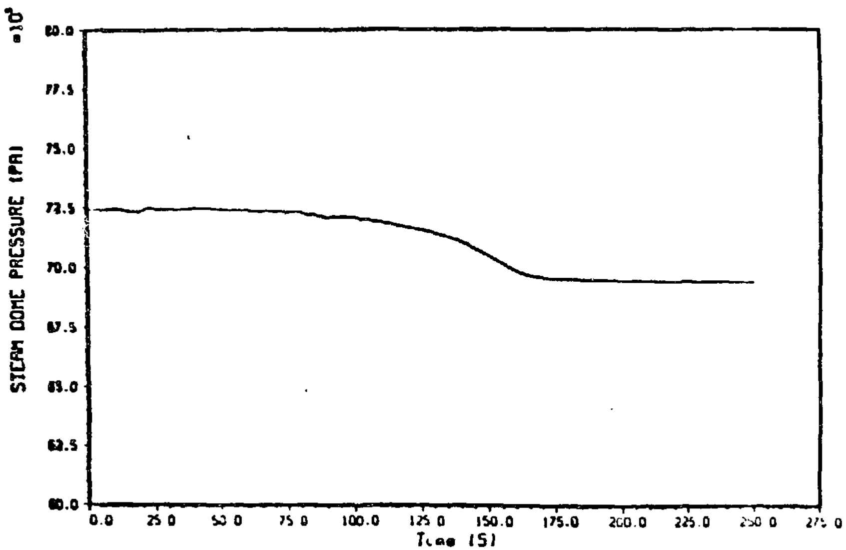

Fig. 11 Pressure profile for loss of feedwater heating ATWS with fine motion control rod insertion

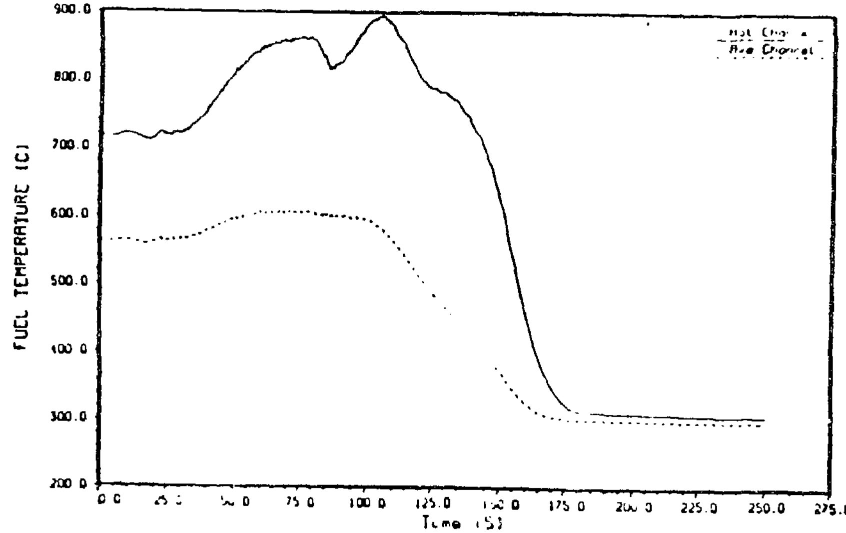

Fig. 12 Fuel temperature profile during loss of feedwater heating ATWS with fine motion control rod insertion.

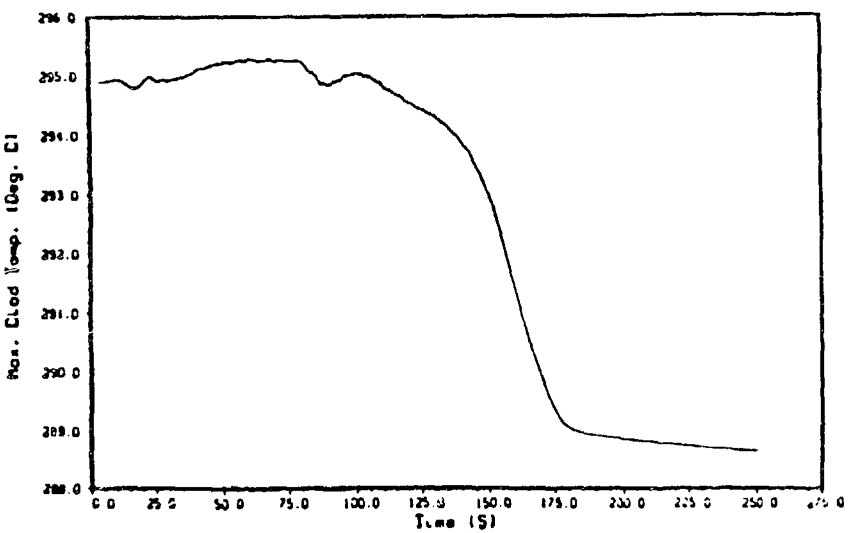

Fig. 13 Maximum cladding temperature during loss of feedwater heating ATWS with fine motion control rod insertion.

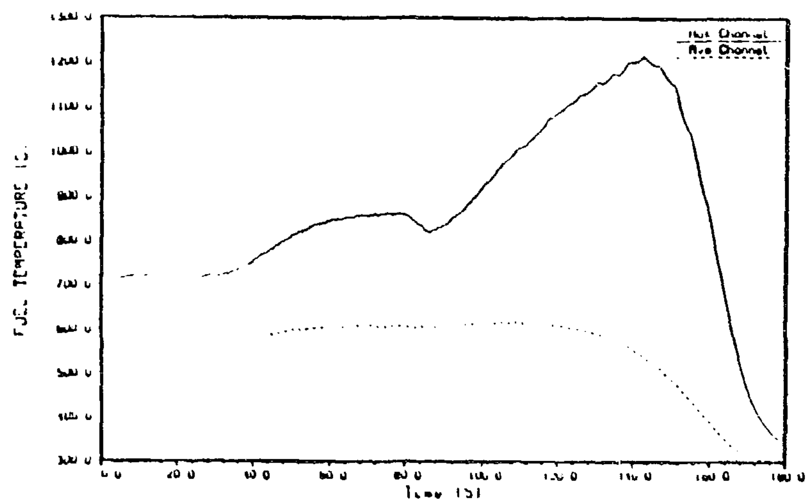

Fig. 14 Fuel temperature for the hot rods for the core with nominal feedwater flow.