THERMAL-HYDRAULIC ASPECTS OF FLOW INVERSION IN A RESEARCH REACTOR

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ABSTRACT

PARET, a neutronics and thermal-hydraulics computer code, has been modified to account for natural convection in a reactor core. The code was then used to analyze the flow inversion that occurs in a reactor with heat removal by forced convection in the downward direction after a pump failure. Typical results are shown for a number of parameters. Research reactors normally operating much above ten MW are predicted to experience nucleate boiling in the event of a flow inversion. Comparison with experimental results from the Belgian BR2 reactor indicated general agreement although nucleate boiling that was analytically predicted was not noted in the BR2 data.

INTRODUCTION

Coolant flow in forced convection cooled research reactors is normally in the downward direction. This may be done to minimize nuclear radiation at the surface of the reactor pool or to lessen the resistance during shutdown rod insertion.

If pumping power is lost during operation of such a reactor, decay heat is ultimately removed by natural convection in the upward direction. In the process of transition from forced to natural convection, however, a period of very low flow and flow inversion will occur. It is of interest to be able to analyze the thermal-hydraulic aspects of this situation to predict whether excessive fuel and/or clad temperatures will occur.

The correlations for heat transfer, onset of nucleate boiling (ONB), and departure from nucleate boiling (DNB) for steady operation both in upflow and downflow were described by Sudo, et al. Critical heat flux (CHF) correlations were developed for both upward and downward flow in a research reactor by Mishima, et al.

Chen, et al. considered the thermal-hydraulics of loss of flow resulting in a flow coastdown to a minimum value maintained by the emergency pump. In this way flow inversion was avoided. Using a modified version of the COBRA computer code, it was predicted that CHF would not occur for sufficiently slow coastdown rates or for sufficiently low radial peaking factors. Loss of flow transients were studied using the PARET code by Matos, et al. Cloak peak
temperatures reached in cases reported were below the saturation temperature. The THSP code was used by Klein, et al.\textsuperscript{5} to predict the thermal response to a loss of coolant flow. Coolant flowrate and clad temperature histories were plotted.

PARET CODE

The basis for the PARET code was documented in a report by Obenchain.\textsuperscript{6} It is a coupled neutronics-hydrodynamics-heat transfer code designed for use in predicting the course and consequences of nondestructive reactivity accidents in small reactor cores. The core can be divided in as many as four regions, each having different power generation, coolant mass flow rate, and hydraulic parameters, and each represented by a single fuel pin or plate plus its associated coolant channel.

A transient problem is forced through specification of externally inserted reactivity versus time or average core power versus time. For specified reactivity the power of the reactor is determined through a numerical solution of the point reactor kinetics equations. Solution of these equations is effected subsequent to an estimation of the reactivity fed back up to the current time. The feedback mechanisms include fuel rod expansion, moderator density effects, and temperature effects in the fuel.

The hydrodynamics are based on a one-dimensional modified Momentum Integral Method which involves the solution of equations for conservation of mass, momentum, and energy. The solution yields core pressure drop along with fluid enthalpy, pressure, and mass flow rate along each channel. Coolant flow rate is determined by specifying either the mass flow rate or the core pressure drop as a function of time. Coolant flow reversal is allowed for in the model.

Some revisions to the PARET code have been made by Woodruff.\textsuperscript{7,8} These include optional use of SI units on input and output and a selection of heat transfer, flow instability, and DNB correlations. A tabulation of decay heat power based on the ANS curve for fission product decay heat has also been added. The revised PARET code gives good agreement with the SPERT I transient tests.\textsuperscript{8}

MODIFICATIONS TO THE PARET CODE

In order to add the buoyant forces associated with natural convection, the core pressure drop mode of operation was modified. The buoyant force was calculated from the difference between the weight of coolant in the channel and the weight of a similar column of fluid at the initial inlet temperature of the coolant. The failed pump is assumed to slow down at a prescribed exponential rate. The pressure drop due to the decaying pumping head is superimposed on the buoyant component. Initially the latter is overwhelmed but in 5-30 sec. depending on pump decay rate, the two are about the same magnitude. As the downflow decreases, the coolant temperature increases and buoyancy effects are enhanced. The net force turns upward prior to the moment
of flow inversion because of inertial effects. As the flow reverses the coolant that fails to escape the bottom of the reactor is heated further as it flows upward causing the buoyant forces and the mass flow to peak just after flow inversion.

Coolant reentering the bottom of the core has just passed through the reactor and will therefore be hotter than the normal entering coolant. This effect has been allowed for by increasing the reentering coolant enthalpy by 15–25 KJ/KG. By the time the pump head has fallen at an exponential rate about three orders of magnitude, its force is considered to be dissipated. It would be difficult to show exactly when and how the pump head goes to zero but a continued exponential decrease does not seem reasonable. Therefore, after this time, its magnitude is decreased linearly to zero in two sec. This has an effect on the result since the linear interval occurs at the time that the competing forces (pump and buoyancy) are of the same order. The rate of linear decrease does not have a large effect.

The axial lengths of the plenums above and below the reactor core will contribute to the buoyant force. The temperature of the coolant in these legs is assumed to be at the corresponding inlet or outlet temperature. This is an approximation because the temperature will vary in the plenum region.

TYPICAL RESULTS

Cases were run for a typical MTR plate-type fueled test reactor at three power levels and for three pump coast down rates. Characteristics of the reactor are given in Table 1. The model included two channels: an average power channel and a hot channel which had roughly 90% more power than the average channel. The reactor is assumed to trip when the flow drops to 85% of its normal value. The total power drops rapidly to about 13% and then more slowly (to 3-4% after one min.). After reactor trip, a little more than 70% of the heat generated was assumed to be deposited in the fuel with the balance going to the water. Previous operating history was not an important factor in determining decay heat level. Power levels were chosen to retain single phase flow throughout the transient. Data presented are for the hot channel.

Table 2 lists the cases run and certain results. At powers above ten MW, for this core, nucleate boiling would be predicted. Figures 1 and 2 are plots of mass flow through the core with time. Because the flowrate prior to trip is far greater than that achievable in natural convection, the upward velocity scale has been expanded in this figure. The downward velocity is nearly identical for the six and ten MW cases in Fig. 1. However in natural convection the higher power case causes the flowrate to peak sooner and at a considerably higher value. The effect of the pump coastdown rate is seen in Fig. 2. The pump time constant is the time for the flow to fall to 1/e of its current value. Thus flow inversion is delayed for the larger time constant but after about 25 sec, the two flows are nearly identical. The time constants used in these cases are not necessarily typical for research reactor installations.
Table 1. Characteristics of Typical and BR2 Reactors

<table>
<thead>
<tr>
<th>Item</th>
<th>Typical</th>
<th>BR2</th>
</tr>
</thead>
<tbody>
<tr>
<td>Element Type</td>
<td>Plate</td>
<td>Circular (See Fig. 7)</td>
</tr>
<tr>
<td>No. of Elements</td>
<td>17</td>
<td>14</td>
</tr>
<tr>
<td>Plates per Element</td>
<td>16</td>
<td>6 (Circular)</td>
</tr>
<tr>
<td>Fuel Height (m)</td>
<td>0.6096</td>
<td>0.762</td>
</tr>
<tr>
<td>Plenum Height (m)</td>
<td>0.3048</td>
<td>2.4</td>
</tr>
<tr>
<td>Thicknesses (mm)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel</td>
<td>0.508</td>
<td>0.508</td>
</tr>
<tr>
<td>Clad</td>
<td>0.508</td>
<td>0.381</td>
</tr>
<tr>
<td>Coolant Channel</td>
<td>2.39</td>
<td>6-7</td>
</tr>
<tr>
<td>Pressure (atm)</td>
<td>1.5</td>
<td>2.0 (After Trip)</td>
</tr>
<tr>
<td>Inlet Water Temperature (°C)</td>
<td>20</td>
<td>35</td>
</tr>
<tr>
<td>Trip Set Point on Low Flow</td>
<td>85%</td>
<td>85%</td>
</tr>
<tr>
<td>Peak to Average Power</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Axial</td>
<td>1.31 (Symmetric)</td>
<td>1.85 (Skewed)</td>
</tr>
<tr>
<td>Radial</td>
<td>1.905</td>
<td>1.55</td>
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Table 2. Cases Run on Typical Reactor

<table>
<thead>
<tr>
<th>Case No.</th>
<th>Full Power (MW)</th>
<th>Pump Time Constant (sec)</th>
<th>Time of Flow Inversion (sec)</th>
<th>Maximum Upflow Kg/(sec m²)</th>
<th>Peak Temperature Clad °C</th>
<th>Peak Temperature Coolant °C</th>
</tr>
</thead>
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<tr>
<td>1</td>
<td>6</td>
<td>2</td>
<td>7.12</td>
<td>99</td>
<td>91</td>
<td>78</td>
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<td>2</td>
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<td>3</td>
<td>10</td>
<td>2</td>
<td>6.90</td>
<td>153</td>
<td>114</td>
<td>96</td>
</tr>
<tr>
<td>4</td>
<td>8</td>
<td>1</td>
<td>5.25</td>
<td>134</td>
<td>107</td>
<td>89</td>
</tr>
<tr>
<td>5</td>
<td>8</td>
<td>4</td>
<td>10.25</td>
<td>117</td>
<td>99</td>
<td>84</td>
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</table>
Fig. 1. Mass Flow Histories for Typical Reactor, Pump Time Constant: 2s

Fig. 2. Mass Flow Histories for Typical Reactor, 8 MW Core
Temperature plots for the ten MW case are shown in Figs. 3 and 4. The peak clad and coolant temperature histories, Fig. 3, drop rapidly after trip and then rise rapidly as the flow decreases to zero. The peak temperatures are reached several seconds after the moment of flow inversion. As mentioned the high coolant temperature enhances the buoyant force causing the coolant flow peaks seen in the first two figures. After the hot slug of fluid is ejected from the top of the reactor, buoyancy forces and flowrate quickly fall off. The slight overshoot is due to the inertia of the coolant in the core. Axial profiles in the coolant are seen in Fig. 4 before and after the flow inversion. The hot slug of coolant can be seen to be moving upward at 9 sec. and reach the top at 13 sec.

Peak clad and coolant temperatures are plotted in Fig. 5 versus peak local heat flux at full power. A 3.5 MW case was also run to extend these curves. Total core power is also plotted in this figure.

![Graph showing temperature history of clad and coolant, 10 MW Core](image-url)
Fig. 4. Axial Temperature Distributions in Coolant, 10 MW Core

Fig. 5. Variation of Clad and Coolant Peak Temperature and Core Power with Fuel Plate Heat Flux
BR2 is a pressurized water-cooled thermal test reactor. During its initial operation tests were run to show that the reactor could operate safely at its design power of 400 W/cm$^2$. Among these tests was the loss of primary coolant while operating the reactor at a series of constant power levels.

The startup core and the primary coolant loop shown in Fig. 6 were taken from Ref. 9. The sketch of the fuel element cross section in Fig. 7 was taken from Ref. 10. The normal operating pressure is 13.6 atmospheres but during these tests a depressurization dropped the pressure at the core midplane to two atmospheres prior to flow inversion. Thermocouples recorded temperatures on the outer plate of three elements during the tests (see Fig. 7). They were placed at the core midplane as well as 150 and 300 mm above and below that elevation. Coolant temperatures in the upper and lower plenum were also recorded. The coolant velocity prior to pump loss was 11 m/sec. After pump loss the velocity decayed to half its value every 5 sec. This corresponds to a time constant of 7.2 sec. Other details of the BR2 reactor are given in Table 1.

A PARET model of the BR2 tests was prepared with the intent of verifying the natural-convection/flow-inversion enhancements made to PARET. Results are reported here for the case of a maximum nominal heat flux of 370 W/cm$^2$ as given in Table 3 of Ref. 9. Comparison is made to data presented in Fig. 2(b) of Ref. 11.

The calculated and experimental clad temperature is shown in Fig. 8. Good agreement is seen until shortly before flow inversion which occurs at about 25 sec. At 23 sec. the calculated clad temperature rises rapidly to 125°C where subcooled nucleate boiling occurs. The heat transfer in this mode of operation is adequate to handle the heat generated. Boiling ceases when sufficient upflow, due to natural convection, is attained. The experimental results in the five sec. before and after the flow inversion do not show the constant temperature associated with boiling. The reason for this difference has not been determined. Even for cases at considerably lower power nucleate boiling is predicted in the PARET calculations. The laminar heat transfer present for a time before and after flow inversion do not seem to be high enough to maintain a low clad temperature.

Coolant temperatures above and below the core are plotted in Fig. 9. The calculated values correspond to the top and bottom of the active core while the experimental values were recorded some distance past the end of the active core. In particular the lower thermocouple was located about 0.9 m below the fueled section and therefore temperature peaks such as seen in the calculated lower plenum temperature at flow inversion are not observed in the experimental values. A rapid rise in the upper plenum temperature is predicted and observed at 28 sec. but the calculated temperature rapidly drops off while the experimental value peaks higher and cools much more slowly.

The higher clad and lower coolant temperatures calculated after 30 sec. could be partially due to a lower heat transfer coefficient used by PARET than actually existed. Another possibility investigated was the thermal moderating effect of the relatively massive beryllium matrix that was adjacent to the
Fig. 6. Core and Circuit for BR2 Reactor (From Ref. 9)

Fig. 7. BR2 Fuel Element Cross Section (From Ref. 10)
Fig. 8. Clad Temperature History for BR2 Test

Fig. 9. Coolant Temperature History for BR2 Test
fuel plate in which the thermocouple was embedded. A model of this was prepared with the Argonne thermal analysis computer code, THTB. From these calculations it was concluded that, while the beryllium matrix did tend to smooth out coolant temperature, the effect was small. Subcooled boiling would still occur and the difference in upper plenum coolant temperature after 30 sec. (Fig. 9) is not due to this effect.

CONCLUSIONS

The PARET computer code was well suited to this application. Its capability to handle both core physics and thermal hydraulics in a reasonably complete manner was necessary in dealing with the complex phenomena of flow inversion.

For a given research reactor with known geometry and operating characteristics, the modified PARET code can model the flow inversion process in a fairly detailed manner. The pump coast down rate may not be known accurately but because peak temperatures are not highly sensitive to it, a reasonable estimate should be adequate.

The typical reactor described in this paper with full power ratings below ten MW, should not experience nucleate boiling during a pump loss - flow inversion incident. For reactors operating at somewhat higher powers, boiling may occur. However, as long as the bulk coolant remains subcooled, excessive clad temperatures should not result.

Reports of experimental investigations of the flow inversion phenomena in the literature are sparse. As a part of its initial operation, the BR2 reactor was put through such a test. While some aspects of an attempt to model these results with the PARET code were at variance, general agreement was obtained. If additional experimental data becomes available, it would be beneficial to make further comparisons.

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REFERENCES


