CONF-950904--11

RELAP 5 ANALYSES OF TWO HYPOTHETICAL FLOW REVERSAL EVENTS
FOR THE ADVANCED NEUTRON SOURCE*

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For presentation at the
7th International Meeting on
Nuclear Reactor Thermal-Hydraulics
(NURETH-7)
Saratoga Springs, New York
September 10-15, 1995

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*Research sponsored by the Office of Energy Research, U.S. Department of Energy, under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

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RELAP5 ANALYSES OF TWO HYPOTHETICAL FLOW REVERSAL EVENTS FOR THE ADVANCED NEUTRON SOURCE REACTOR

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ABSTRACT

This paper presents RELAP5 results of two hypothetical, low flow transients analyzed as part of the Advanced Neutron Source Reactor safety program. The reactor design features four independent coolant loops (three active and one in standby), each containing a main circulation pump (with battery powered pony motor), heat exchanger, an accumulator, and a check valve. The first transient assumes one of these pumps fails, and additionally, that the check valve in that loop remains stuck in the open position. This accident is considered extremely unlikely. Flow reverses in this loop, reducing the core flow because much of the coolant is diverted from the intact loops back through the failed loop. The second transient examines a 102-mm-diam instantaneous pipe break near the core inlet (the worst break location). A break is assumed to occur 90 s after a total loss-of-offsite power. Core flow reversal occurs because accumulator injection overpowers the diminishing pump flow.

Safety margins are evaluated against four thermal limits: $T_{\text{wall}} = T_{\text{sat}}$, incipient boiling, onset of significant void, and critical heat flux. For the first transient, the results show that these limits are not exceeded (at a 95% non-exceedance probability level) if the pony motor battery lasts 30 minutes (the present design value). For the second transient, the results show that the closest approach of the fuel surface temperature to the local saturation temperature during core flow reversal is about 39 °C. Therefore the fuel remains cool during this transient. Although this work is done specifically for the ANSR geometry and operating conditions, the general conclusions may be applicable to other highly subcooled reactor systems.

INTRODUCTION

Currently in the advanced conceptual design stage, the Advanced Neutron Source Reactor (ANSR) [1] is a research reactor to be built at the Oak Ridge National Laboratory. It will provide the highest continuous neutron flux levels of any reactor in the world. The reactor is cooled and moderated by highly subcooled heavy water (D_2O). To accommodate the high core power density (4.5 MW/l), coolant flows upward

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through the core at a high velocity (25 m/s). In addition, the ANSR features a submerged primary coolant loop configuration and passive gas-pressurized accumulators to improve the depressurization behavior.

The conceptual safety analysis report (CSAR) [2] was completed and included pipe break [3-9], Loss-of-Offsite Power (LOSP) [10,11], and reactivity insertion accident calculations. Since the CSAR calculations were completed, several updates to the RELAP5 input model have been made. Changes are enumerated here: (1) the power distribution in the cores has been updated to an improved core design with a new U^{235} distribution, (2) a refined nodalization has been used to represent more accurately the propagation of expansion waves that immediately follow an instantaneous pipe break, (3) a total accumulator volume of 7.52 m³ and 0.52-m³ bubble size have been assumed for each accumulator, consistent with the current design, (4) the piping layout is now consistent with the Conceptual Design Report [13]. (The major difference between the assumed piping layout and the present one is the elevation of the main circulation pumps), (5) the main circulation pump was changed from a Byron Jackson [12] design to a Westinghouse design that maintains a higher flow rate at lower developed pump head, and (6) the primary system pressure sensor location (used for initiating reactor scram and primary pump trip) was moved closer to the core to provide more rapid and sensitive low pressure detection. Also, a first-order time lag of 0.03 s was used for the pressure sensor response lag instead of a straight 0.03-s delay.

In the CSAR calculations, the Costa [14] correlation was used to predict the onset of significant void (OSV), while the Gambill/Weatherhead correlation was used to predict the occurrence of critical heat flux (CHF) [15]. Three additional thermal limit criteria have since been included in the code. The first is a modification of the Saha-Zuber [16] OSV correlation. Unlike the Costa correlation, the modified Saha-Zuber correlation has a reasonable low-subcooling limit. The second limit is the $T_{\text{wall}} = T_{\text{sat}}$ criterion, which is exceeded if the fuel surface temperature (as predicted by the Petukhov [17] correlation at high flow or a constant-Nu correlation at low flow) exceeds the bulk saturation temperature in the coolant channel. The third additional limit is the Bergles-Rohsenow [18] correlation for incipient boiling.

The objective of this work is to determine the effects of flow reversal events on the ANSR thermal hydraulic performance. The most recent version of the RELAP5/MOD3/VERSION 1.1.1 input model (including post-CSAR updates) was used to simulate two hypothetical transients that lead to flow reversal. The events are defined and calculated results (with the derived safety margin) are presented and discussed. It is shown that neither of these flow-reversal events present a significant safety threat to the ANSR.

FACILITY DESCRIPTION

The ANSR core design features an axially split core cooled by upward flowing heavy water (see Fig. 1). Each core half is constructed with a series of involute fuel plates arranged in an annular array to provide spanwise uniform coolant gaps. Inner control and shutdown rods and outer shutdown rods are located in the central hole region and the reflector tank, respectively. The core assembly is contained within a double-walled aluminum CPBT that separates the high pressure heavy water core coolant from the low pressure heavy water in the reflector tank that surrounds the entire core region.

A simplified reactor primary cooling system diagram is shown in Fig. 2. The system consists of four identical loops connected in parallel to the reactor vessel and to a common reactor outlet pipe. Each loop contains a gas charged accumulator, a main heat exchanger, an emergency heat exchanger, a reactor coolant pump (each with an AC-powered main motor and a DC-powered pony motor), a strainer, isolation valves, connecting piping, and instrumentation. During normal, full power reactor operation, three of the four loops are in operation with the fourth loop in standby mode.

MODEL DESCRIPTION

An ANSR RELAP5 system model has been developed based on the advanced conceptual design. The model includes three major regions as shown in Fig. 3. The core model (region 1) consists of two core halves, core bypass channels, and a central control rod region. The core is surrounded by the CPBT, which separates the high-pressure primary system and the low-pressure moderator tank.

Core power is calculated using a point kinetics model with reactivity feedback based on coolant density and control rod position. Power is distributed among the various metal and fluid regions and is distributed axially within the fueled region based on a specific fuel loading design.

Each of the two fuel elements is modeled as an average channel (which incorporates all but two of the fuel plates) and two hot channels representing the most limiting axial relative power density profile in each element. For each fuel element, one of these hot channels reflects 95% non-exceedance probability uncertainty levels (for analysis of unlikely events), and one represents 99.9% non-exceedance probability uncertainty levels (for analysis of anticipated events). The purpose of the hot channels is to calculate the most severe axial bulk temperature profile within the core. Within each of these channels, correlations are compared at hot spot conditions. The characteristics of these spots are defined using uncertainties that affect local heat flux conditions, but do not affect bulk coolant conditions because they are so localized.

The loop model (region 2) contains three independent heat exchanger loops. Each loop consists of an isolation valve, a hot/cold leg, an accumulator, horizontal U-tube main and emergency heat exchangers, a centrifugal main circulating pump, and an inertial flow diode (a preferred flow direction device). Both heat exchanger models are calibrated to design specifications. The single-phase homologous curves defining the performance of the main circulation pumps were developed from three-quadrant Westinghouse design curves, and two-phase corrections were based on Semiscale data [19].

An open-loop representation of the letdown and pressurizing system (region 3) is included in the model. Letdown flow is extracted from the inlet plena of the three main heat exchangers. In the design, core outlet pressure is controlled through modulation of the letdown valves. This is modeled by establishing the nominal valve opening size required to allow a nominal flow during normal conditions and controlling about that point.

A main pressurizing pump provides injection flow through a makeup line at the hot leg distribution header. Injection flow is drawn from a constant temperature heavy-water source by the main pressurizing pump. Following letdown isolation, flow through the pressurizing pumps is assumed to continue until the integrated injected flow reaches the makeup tank capacity.

RELAP5 MODIFICATIONS FOR APPLICABILITY TO ANSR

Although the RELAP5 code was designed for pressurized water reactor and boiling water reactor applications, the applicability of the code has been improved for the ANSR-specific fuel plate geometry. Three specific changes were implemented. The first replaced the Dittus-Boelter [20] single-phase forced convection correlation by the Petukhov correlation. The second change incorporated the Gambill/Weatherhead CHF correlation [14] into the existing Groenveld et al. [21] look-up table. Modification three altered the interfacial drag in the slug flow regime to reflect the Griffith [22] drift flux behavior in narrow channels.

In the low flow regions two additional changes were implemented. The friction factor correlation used to determine the flow resistance in the core (about 80% of the total loop pressure drop) was changed from the laminar friction factor correlation for circular tubes ($f = 64/Re$) normally used by RELAP to one for long, thin rectangular channels ($f = 93/Re$). The second change replaced the laminar constant-Nu correlation for circular tubes ($Nu = 4.36$) by one for rectangular channels ($Nu = 7.63$). Although the heat transfer coefficients that are used to determine the fuel surface temperature do not influence OSV, they are important in establishing the $T_{wall} = T_{sat}$ criterion. Experience has shown that RELAP5 will switch from the turbulent Petukhov correlation to the laminar, constant-Nu correlation at a Reynolds number of about 1000. RELAP5 simply takes the maximum of the two heat transfer correlations.

DESCRIPTION OF THE TRANSIENTS

The first transient involved the failure of a single main circulation pump with a stuck-open check valve, leading to flow reversal in the failed loop. The following assumptions were made: (1) the motors (both main AC and pony DC) driving one of the main circulation pumps fail leading to pump coastdown, (2) the check valve in the tripped pump outlet pipe fails to close to prevent reverse flow, and (3) following reactor scram, the core outlet temperature falls sharply, shrinking the coolant and leading to a low-pressure trip of the remaining two functional main circulation pumps. (Here, to force a coolant pump trip, the setpoint was assumed to occur at a pressure 5% higher than the nominal trip setpoint). This is an unlikely or an extremely unlikely event. Therefore, a 95% non-exceedance probability level was used for the thermal limit comparison.

The second transient considered is a 102-mm-diam instantaneous core inlet break into the reactor pool (at about 3 atmospheres). The break is assumed to occur after the reactor and main circulation pumps are shut down, but while the system is still at high pressure. This shutdown-prior-to-break sequence may occur during an earthquake or during a normal shutdown period. For instantaneous breaks [7,9] that occur before shutdown, the pressure expansion wave propagating from the break site into the fuel channels is the limiting phenomenon, because it significantly reduces the subcooling in the fuel channels, causing thermal limits to be challenged (the Costa OSV limiting heat flux is proportional to subcooling). However, here the pipe break occurs after scram so the core exit subcooling is much larger, and is therefore only slightly affected by the oscillations in saturation temperature caused by the pressure expansion wave. Such an accident, however, leads to diminished core flow and possibly even to a flow reversal, if the break is large enough. Flow reversal could occur if the main circulation pumps (running at 10% pony motor flow after shutdown) are unable to maintain upflow against the accumulator (located in the hot leg) discharge. The accumulators act to maintain higher core exit pressure, but the core inlet pressure is decreased due to the break flow. Parametric simulations have shown that the pipe-break time most likely to result in flow reversal is approximately 90 s after reactor and main circulation pump shutdown. This event is also considered an unlikely or extremely unlikely event. Therefore, all limits are evaluated at a 95% non-exceedance probability level.

RESULTS AND DISCUSSION

Results and discussion of the two flow reversal events calculated using the RELAP5/MOD3/VERSION 1.1.1 input model are presented here.

Single Pump Failure with a Stuck Open Check Valve (Loop Flow Reversal)

At time zero, a main circulating pump was assumed to lose power and began coasting down. The two remaining functional pumps were tripped on low primary system pressure at 13.72 s and then coasted down to pony motor speed. The pony motors continued running for 30 min (until the batteries expired) and then coasted down so that all forced flow was lost. Subsequently, natural circulation developed within the primary system removing the decay heat from the core.

The pressure response at the upper core inlet and exit is shown in Fig. 4 (the upper core is limiting for this transient). After the initial pump trip, the core inlet pressure declines due to the diminishing primary pump head and flow rate, and coolant shrinkage after reactor scram. The core exit pressure changes only slightly because it is controlled by closing the letdown valves in response to the depressurization. The core pressure drop decreases as the core flow decreases and stabilizes once natural circulation is established.

Pump flow and total core flow (sum of three pump flows) are shown in Fig. 5. As the failed pump coasts down, the discharge (core inlet) pressure decreases, causing the functional pump flows to increase. At 1.95 s the failed pump developed head becomes low enough that flow reverses in the failed loop. Flow reversal in the failed loop is possible only because the check valve is also assumed to fail. Much of the flow supplied by the two remaining functional pumps is thus diverted back through the failed loop, bypassing the core. After the functional pump main motors are tripped (13.72 s), the loop flows diminish, but the same flow configuration continues with backflow through the failed loop. After the pony motor batteries expire at 30 min, a free convection flow pattern is established with forward flow through all three loops.

Temperature and coolant velocity transients are shown in Fig. 6 for the 95% hot channel exit. Three minima in the fuel channel subcooling occur, corresponding to three coolant temperature maxima (the saturation temperature remains relatively constant). Initially, the coolant temperature rises with declining core flow. The first peak occurs at the time of reactor scram since the drop in power causes the bulk temperature to fall sharply. After the functional pump main motors are tripped, core flow declines again, causing another increase in the core exit bulk temperature. However, decay heat is also decreasing so that once the flow rate stabilizes, the core exit bulk temperature begins to decline; thus, the second peak is observed. The final peak in the bulk temperature occurs during the transition to natural circulation, again because of reduced core flow. These three subcooling minima are responsible for the three thermal limit ratio minima shown in Fig. 7. Throughout the transient, the fuel surface temperature stays below the saturation temperature, indicating no boiling at the hot spot.

Figure 7 shows a comparison of various thermal limit ratios in the upper (most limiting) core hot spot for a 95% non-exceedance probability level. The thermal limit ratio is the ratio between the thermal limit criterion (e.g. $T_{wall} = T_{sat}$, IB, Costa or Modified Saha-Zuber) and the hot spot heat flux. All three thermal limit ratio minima (other than the original and the modified Saha-Zuber) are greater than unity, indicating that none of the thermal limits are exceeded. As expected, the $T_{wall} = T_{sat}$ and the IB criteria are the most limiting. Before scram, Costa is the most limiting OSV correlation. Discontinuities in both the original and the modified Saha-Zuber thermal limit ratios occur when the velocity falls below 8 m/s. Discontinuities in those ratios again occur (this time the ratios plummet to zero) when the Peclet number falls below 70,000 (about 20 s in Fig. 7). These two sets of discontinuities are a direct consequence of the sudden increases in uncertainty that we are applying to the Saha-Zuber correlation as the velocity is decreasing (see Table 1). In addition, the modification of the Saha-Zuber correlation causes the modified version of the correlation to predict higher limits at low subcoolings and lower limits at higher subcoolings than the original version of the correlation. The Costa correlation, which is considerably more

conservative than either the Saha-Zuber or modified Saha-Zuber correlation at high velocities, becomes less so as the velocities decrease. In these calculations, the uncertainties for the Costa correlation were kept constant over all of the velocity range, while changing those of the modified Saha-Zuber and original Saha-Zuber correlations. Thus, the Costa correlation has a lower thermal limit ratio than the Saha-Zuber formulations early in the transient (at high velocities), and higher ratios at the end of the transient (at lower velocities). Since the absolute lower limit on boiling (and therefore OSV) is $T_{\text{wall}} = T_{\text{sat}}$, the OSV curves cannot fall below the $T_{\text{wall}} = T_{\text{sat}}$ curve. This transient would therefore not exceed any of the thermal limits that could lead to potential fuel damage.

Table 1. Peaking factors for hot spot heat fluxes.

Limit Model		95% Probability	99.9% Probability
$T_{\text{wall}} = T_{\text{sat}}$		1.16	1.25
Bergles-Rohsenow (IB)		1.30	1.59
Costa (OSV)		1.30	1.59
Saha-Zuber and Modified Saha-Zuber (OSV)	Peclet > 70000, Vel > 8 m/s	1.52	2.30
	Peclet > 70000, Vel < 8 m/s	2.08	**
	Peclet < 70000	**	**

** Presently the $T_{\text{wall}} = T_{\text{sat}}$ or IB limit is recommended- and is more conservative- for these conditions until the low flow correlation for the modified Saha-Zuber correlation can be better characterized.

102-mm-diam Instantaneous Core Inlet Break after Shutdown (Core Flow Reversal)

The pipe break was assumed to occur 90 s after a shutdown of the reactor and AC pump motors (LOSP). The pony motors were assumed to be available to provide 10% of the nominal flow after the coastdown.

The core exit pressures, shown in Figs. 8 and 8a, drop sharply when the pipe break occurs at 90 s, then recover due to the accumulator injection which causes the flow reversal in the core. After the break flow equilibrates with the accumulator flow (Fig. 9), the primary system pressure declines steadily approaching the reactor pool pressure as the accumulators are drained.

Thermal limit ratios at the 95% lower core hot channel entrance (the entrance is used here since it is the limiting location after flow reversal) are shown in Fig. 10 and indicate two minima. The first minimum occurs immediately after reactor scram and the second minimum occurs during core flow reversal immediately following the pipe break. Again, the Saha-Zuber based correlations show very low thermal limit ratios under low flow conditions because of the uncertainties imposed upon them. Fig. 10a shows a close-up of the remaining thermal limit ratios immediately following the break. The Costa limit is exceeded because as the velocity reaches zero, the Costa prediction of heat flux required to initiate voiding goes to zero. However, the $T_{\text{wall}} = T_{\text{sat}}$ limit is not exceeded even during the core flow reversal. The

violation of the Costa limit should be disregarded because OSV cannot physically occur if T_{wall} stays below T_{sat} .

The limiting location in each fuel element changes with the changing pressure, coolant bulk temperature, and coolant flow direction. When a flow reversal occurs, the limiting location moves from the core exit to the core inlet (referenced to the normal coolant flow direction). The limiting locations in each fuel element, immediately following the pipe break for the $T_{\text{wall}} = T_{\text{sat}}$ criterion occurs at the lower element inlet and the upper element exit (the absolute limit is at the upper element exit). Fig. 11 shows temperature and velocity responses at the lower core 95% hot channel entrance with a peaking factor of 1.16 (this peaking factor includes an assumed normal distribution of the error in the heat transfer coefficient with a standard deviation of 3.3%). Pressure oscillations following the break cause small changes in the local saturation temperature. Diminishing core flow results in lower heat transfer in the core, thus causing the wall temperature to approach the saturation temperature within 71°C. The wall temperature decreases as decay heat decreases and core flow stabilizes. Fig. 12 shows the transition from the turbulent Petukhov correlation to the laminar constant Nu correlation for the heat transfer coefficient in the lower core hot channel during the flow reversal. Fig. 13 shows the temperature responses at the upper core 95% hot channel exit. The same behavior is observed at this location as was noted in Fig. 11, however, the closest approach between T_{sat} and T_{wall} is lower (but still about 39°C) due to the higher heat flux levels in the upper core.

CONCLUSIONS

Although this work is done specifically for the ANSR geometry and operating conditions, the general conclusions can be applied to other highly subcooled reactor systems. Similar trends in the thermal limit margin should be observed in other systems for the pipe break and the single pump failure with stuck open check valve events.

For the single pump failure assuming a stuck open check valve, the results show that the core should not exceed any thermal limit criteria if the pony motor battery lasts 30 minutes, at a 95% non-exceedance probability level (used for extremely unlikely events).

In the event of the pipe break, the results show that the worst time for a post-scrum pipe break to occur is approximately 92 s after reactor and AC motor pump shutdown. Even this event, which causes a core flow reversal, the closest approach between the fuel surface temperature and the saturation temperature at the hot spot is about 39 °C (for 95% non-exceedance probability levels), indicating that no boiling should occur in the core.

During low core flow, the OSV correlations were not useful for judging the acceptability of these events. The Costa correlation is based on data obtained at higher flow rates, and the Saha-Zuber correlation has very large uncertainty at the low flows. Instead, it was shown that the fuel surface temperature, T_{wall} , would not exceed the saturation temperature in the hot channel, T_{sat} , during the periods of low flow, thus prohibiting the possibility of OSV.

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Fig. 1. Advanced Neutron Source Reactor core assembly and fuel plates arrangement.

SOME SAFETY FEATURES OF THE ADVANCED NEUTRON SOURCE REACTOR DESIGN

Fig. 2. ANSR system schematic showing general cooling system design characteristics.

Fig. 4. Pressure traces at the upper core 95% hot channel inlet and exit during the single pump failure with stuck open check valve event.

Fig. 5. Total core flow and pump flows during the single pump failure with stuck open check valve event.

Fig. 6. Temperature and velocity transient at the upper core 95% hot channel exit during the single pump failure with stuck open check valve event.

Fig. 7. Comparison in thermal limit ratio at upper core 95% hot channel exit during the single pump failure with stuck open check valve event.

Fig. 8. Pressure traces at the lower and upper core 95% hot channel exit for the 102-mm-diam pipe break, 90 s after shutdown.

Fig. 8a. Close-up of pressure traces during core flow reversal for the 102-mm-diam pipe break, 90 s after shutdown.

Fig. 9. Total core flow, accumulator flow, and break flow behavior for the 102-mm-diam pipe break, 90 s after shutdown.

Fig. 10. Comparison in thermal limit ratio at the lower core 95% hot channel entrance for the 102-mm-diam pipe break, 90 s after shutdown.

Fig. 10a. Close-up of thermal limit ratio comparison at the lower core 95% hot channel entrance for the 102-mm-diam pipe break, 90 s after shutdown.

Fig. 11. Temperature and velocity response at the lower core 95% hot channel entrance for the 102-mm-diam pipe break for the 102-mm-diam pipe break, 90 s after shutdown.

Fig. 12. Transition from the Petukhov turbulent correlation to the constant-Nu laminar in the lower core hot channel when Reynolds number declines to 1000 during core flow reversal for the 102-mm-diam pipe break, 90 s after shutdown.

Fig. 13. Temperature and velocity response at the upper core 95% hot channel exit for the 102-mm-diam pipe break for the 102-mm-diam pipe break, 90 s after shutdown.