BNL-NUREG--36009

TI85 007483

# ANALYSIS OF A SBLOCA INITIATED BY AN ATWS EVENT

MASTER

Jilong Pu,* David J. Diamond and William G. Shier

Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

Submitted for presentation at the
Specialists' Meeting on Small Break LOCA Analyses in LWRs,"
Pisa, Italy, June 1985



*Guest Engineer from the Peoples Republic of China.

# Analysis of a SBLOCA Initiated by an ATWS Event*

Jilong Pu, David J. Diamond and William G. Shier

Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

## ABSTRACT

The response of a four-loop Westinghouse pressurized water reactor to SBLOCAs initiated as a result of an anticipated transient without scram (ATWS) has been analyzed using the RELAP5 computer code. The ATWS is initiated by a loss-of-feedwater, and the small breaks were due to either one or three stuck-open safety valves or reactor coolant pump seal failure. For the cases analyzed, the results show that a LOF-ATWS followed by a SBLOCA does not have more safety significance than that found when each accident is analyzed independently of one another.

*Work performed under the sponsorship of the U.S. Nuclear Regulatory Commission.

Analysis of a SBLOCA Initiated by an ATWS Event*

Jilong Pu, David J. Diamond and William G. Shier

Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

A transient in a pressurized water reactor (PWR) in which there is a failure of the reactor trip (an "ATWS" event) may lead to a rapid rise in pressure. This in turn may challenge the reactor coolant system pressure boundary due to the failure to close of pressurizer safety valves that open during the transient or due to split steam generator tubes, reactor coolant pump seal failure or small pipe breaks. The last three "small breaks" are most likely if the pressure exceeds Service Level C of the ASME Boiler and Pressure Vessel Code. The objective of the present work (1) was to analyze several of these ATWS-initiated small break loss of coolant accidents (SBLOCAs) in order to see how they differed from accidents in which there was reactor trip. The intent was to determine the sequence of events and understand how various safety systems and operator actions might mitigate the consequences of the accident.

The calculations were done for a typical Westinghouse four-loop plant (based on the Zion plant) with the RELAP5/MOD1 code (2). The initiating event was a loss-of-feedwater (LOF), one of the worst ATWS initiators in terms of peak pressure. In all calculations there was a turbine trip (at 30 seconds) and initiation of auxiliary feedwater (at 60 seconds). Three different SBLOCAs were calculated assuming that either one or all three safety valves stuck open or that there was an RCP seal failure at the time of peak

*Work performed under the sponsorship of the U.S. Nuclear Regulatory Commission.

pressure. The seal failure was a "worst" case based on complete seal failure in all four pumps and was the cold-leg equivalent of one stuck-open valve in terms of the equivalent break size which was 0.00174 $m^2$ (0.0187 $ft^2$). For all three cases the emergency core cooling (ECC) charging flow was initiated 30 seconds after the low pressure setpoint of 12.8 MPa (1860 psia) and safety injection at 11.2 MPa (1618 psia). The water was assumed to be from the refueling water storage tank at a temperature of 311 K (100°F) with a boron concentration of 2000 ppm.

With one stuck-open valve, two cases were run to determine the effect of different RCP trip criteria. In one case the trip was due to the hot-leg subcooling falling below 9.4 K (17°F), whereas in the other case trip occurred when ECC began. With three stuck valves, it was assumed that trip occurred on the hot-leg subcooling signal and with RCP seal failure, the assumption was that trip would occur 30 seconds after the failure.

Results for the calculated pressurizer pressure are shown in Figure 1 for a case without a SBLOCA and a case with one stuck-open valve in order to demonstrate the effect of a small break. Without the break, after the initial pressurization the primary pressure decreases. This depressurization from 160 to 600 seconds occurs because the reactor power is shutting down (due to negative moderator density feedback, it is at 8% of steady state power at 200 seconds) and because cold ECC water is entering the system (at 350 seconds). With the break, the depressurization is obviously more pronounced.

After 600 seconds with no break, the increase in primary coolant inventory repressurizes the system, whereas with the open valve the pressure only levels off. Because of the lower pressure with the open valve, more cold ECC water is entering the system helping to keep the pressure low and resulting in cooler clad temperatures than without the break.

In either case, although steam voids formed in the core, no departure from nucleate boiling is predicted. Furthermore, a stable hot shutdown condition can be achieved because of the borated ECC water, the removal of energy via the auxiliary feedwater (and stuck-open valve) and the maintenance of inventory with the ECC system.

These results and the results with three stuck-open safety valves and with RCP seal failure show that a LOFW-ATWS may not be any worse if followed by a SBLOCA (of the type analyzed herein) provided all other systems are operating as designed; specifically, one expects turbine trip, RCP trip, auxiliary feedwater and ECC system actuation. Indeed, the small break provides an additional heat sink that may increase the margin to a power-cooling mismatch. The results with a single stuck valve and different RCP trip criteria result in trips 250 seconds apart, but do not change the outcome of the accident significantly. From all cases studied it is seen that the consequences of a SBLOCA following a LOF-ATWS are not significantly worse than the accident with reactor trip since reactor power is shut down due to moderator density feedback initially and then due to the injection of borated water.

## Acknowledgements

The authors would like to thank M. W. Hodges and C. C. Graves of the USNRC for their encouragement and technical support during the course of this work.

## References

1. J. Pu, D.J. Diamond and W.G. Shier, "Analysis of ATWS in a Westinghouse PWR Including Scenarios with a Small-Break LOCA," NUREG/CR to be published, Brookhaven National Laboratory (1984).
2. V. Ransom et al., "RELAP5/MOD1 Code Manual," NUREG/CR-1826, Idaho National Engineering Laboratory (1982).

FIGURE 1
WESTINGHOUSE PWR LOFW ATWS WITH/WITHOUT BREAK
PZR PRESSURE