CONF-880601--16

INTERFACING SYSTEM LOCAs AT PWRs - A NEW ASSESSMENT*

BNL-NUREG--40754

DE88 006291

FEB 29 1988

G. Bozoki, P. Kohut, and R. Fitzpatrick

Brookhaven National Laboratory

Upton, New York 11973

Interfacing system loss-of-coolant accidents (ISL) were first identified and analyzed in the Reactor Safety Study (called V-events). ISLs were found to be significant contributors to the risk estimates for the pressurized water reactor (PWR) analyzed in the RSS. Some evaluations of ISL sequences in subsequent studies have also found that their contribution to risk is also significant. ISLs are potentially important to risk because a direct path is opened from the damaged reactor core which bypasses the containment building. Thus, these sequences can potentially result in a significant fission product release to the environment.

In spite of the numerous analyses conducted in various Probabilistic Risk Assessment (PRA) Studies during recent years, the models and assumptions used to describe ISLs are subject to substantial uncertainties. In addition, recent studies[1] performed within the framework of the Industry Degraded Core Rulemaking Program (IDCOR) indicate that previous ISL analyses may result in overly conservative results.

In order to provide more realistic estimates for the core damage frequencies (CDFs) and a reduction in the magnitude of the uncertainties, a reexamination of the ISL sequences at PWRs has been performed at Brookhaven National Laboratory. The aims of the study were 1) to examine principal dependencies

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

involved in the ISL accident sequences including any spatial interaction effects and 2) to identify effective methods that would significantly reduce the frequency of these events. This paper focuses on the new and innovative modelling techniques applied to this study and a companion paper discusses the analysis of the risk reduction methods.

Three plants representative of the different PWR designs were selected for the analysis and all major ISL pathways were analyzed. The configuration of the pressure isolation valves and their failure modes were identified. Valve testing and maintenance methods were reviewed and used in the calculation of the initiator frequencies. For the isolation check valves, the initiator models involve multiple failure modes including a flow-dependent leak failure mode and a number of separate failure modes for the MOVs.

Dependencies due to common or shared pathways with the accumulators were specifically treated. This was prompted by the results of a "root cause" analysis of experienced accumulator inleakage events, which revealed that the accumulator outlet check valve has an exceedingly high probability for the "failure to operate (reseat) on demand" failure mode. Previous analyses did not consider the fact that the shared pathways with the accumulator may affect the ISL initiator frequencies.

The analytical models for the initiators made it possible to incorporate the effects of different test and/or leak monitoring practices for a sensitivity study. The initiator frequencies were evaluated by searching the data bases[2] for ISL precursor events or using the generic valve failure data.

Specific event trees were developed for each pathway, which included the pipe rupture probabilities and the plant and operator responses. The timing of expected operator actions and the sequences of the plant responses were determined using the results of various thermal-hydraulic studies. The rupture probabilities were estimated by constructing a failure probability distribution as a function of pipe stress using information theory principles. The end states of the event trees were connected to plant specific event trees (contained in the plant PRAs) through a conditional core damage frequency multiplier that also included the effect of any spatial interaction.

The overpressurization and core damage frequencies were determined for all the interfacing pathways and a summary of results are listed in Table 1.

In general, two groups of lines, the residual heat removal suction and the low pressure injection, were the dominant contributors to CDF due to ISL bypassing the containment. Plant specific features such as the location of the isolation devices, testing practices and interconnection with other safety systems account for most of the differences in the numerical values between the three reference plants. The factor that dominated the results was found to be the frequency of leak testing at the individual plants.

As indicated in Table 1, the total ISL CDF (including events resulting in LOCAs inside or outside the containment) are less than a few percent of the overall CDF (as given in the plant PRAs). One of the major findings of this study was that, for most plants, the preferred direction of an ISL is expected to be through the accumulators representing a LOCA inside containment. However, it would be expected that the major contribution to the risk would be from ISLs that bypass the containment. The results of the present study indicate that the CDF due to these events may range from 3×10^{-6} to as low as 3×10^{-9} depending on plant specific features. The low range of the CDF generally corresponds to a frequent leak testing program, which indicates the importance of the leak testing program to the frequency estimates of ISLs leading to core damage.

References

1. Fauske & Associates, Inc., "Evaluation of Containment Bypass and Failure to Isolate Sequences for the IDCOR Reference Plants," Draft Report FAI/84-9, July 1984.

2. DOE/RECON, Nuclear Safety Information Center, File 8, 1963 to present, "Nuclear Power Experience," (NPE), published by the S. M. Stoller Corp.

Table 1
Core Damage Frequency Due to ISLs
(Overpressurization)

System	Plant 1	Plant 2	Plant 3
LPI	2.64-08	7.43-07	2.31-09
RHR	4.46-09	1.08-07	3.36-06
Accumulator	1.63-06	1.08-06	2.67-06
Total CDF	1.75-06	1.93-06	6.03-06
Total CDF With Containment Bypass	2.78-09	8.42-07	3.38-06
CDF in PRA*	1.18-04	1.59-05	3.34-05

*Due to LOCA