B.22 LER Number 374/92-012

Event Description: Reactor Trip With Degraded Reactor Core Isolation Cooling

Date of Event: August 27, 1992

Plant: LaSalle 2

B.22.1 Summary

The reactor scrammed from 80% power because of a main turbine trip. The main turbine tripped due to a thrust bearing failure indication. The reactor core isolation cooling system (RCIC) auto-started, and the motor-driven feed pump (MDFP) was started in preparation for tripping the turbine-driven feed pumps (TDFPs). However, when the TDFPs failed to trip, the reactor water level rose, resulting in a trip of the MDFP and RCIC. In an attempt to prevent flooding of the steam lines, the outboard main steam isolation valves (MSIVs) were manually closed, resulting in a TDFP shutdown. Later, the operators experienced difficulty in starting RCIC for reactor pressure control. Water that had accumulated in the steam line passed through the pump turbine and into the exhaust header. Flashing of that water to steam prevented RCIC startup due to high exhaust pressure trip signals. The conditional probability of subsequent core damage estimated for the event is 6.1×10^{-6}. The relative significance of the event compared to other postulated events at LaSalle 2 is shown in Fig. B.46.

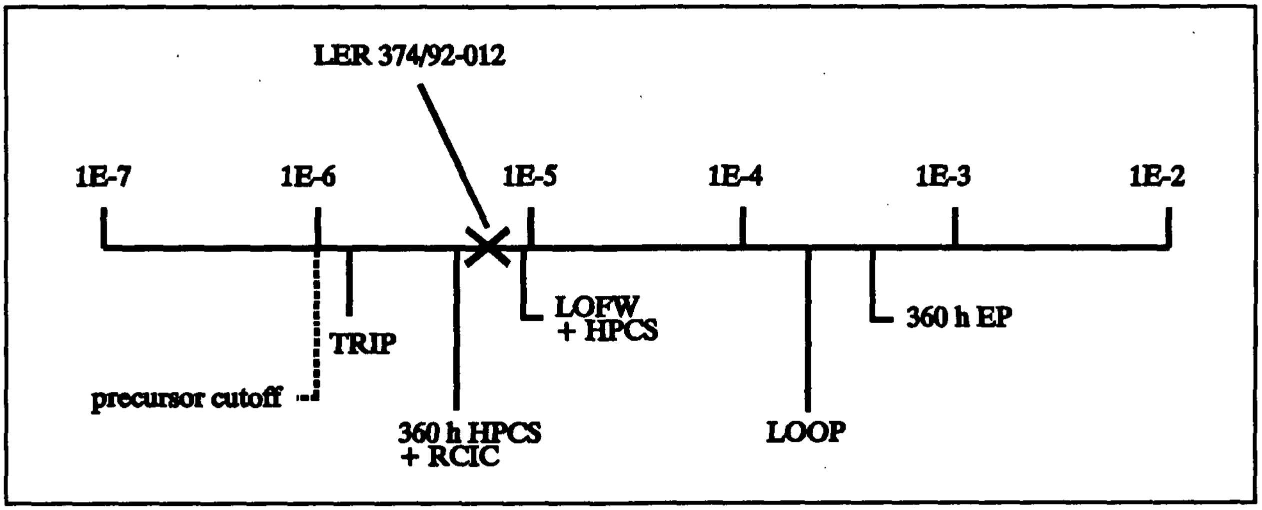

Fig. B.46. Relative event significance of LER 374/92-012 compared with other potential events at La Salle.

B.22.2 Event Description

On August 27, 1992, while reactor power was being reduced to 80%, LaSalle 2 scrammed because of a spurious thrust-bearing wear detector turbine trip signal. The spurious signal was caused by a shift in the trip setpoint due to manufacturer error. Within seconds, RCIC auto-started on a spurious low reactor water level signal caused by pressure oscillations induced by closure of the turbine stop valves.

The MDFP was successfully started to control water level; however, the TDFPs then failed to trip because of oil contamination and blockage in the turbine oil system (both TDFPs failed to trip on high vessel level and after multiple attempts from the control room and locally at the pump). The increasing water level in the reactor eventually resulted in a trip of RCIC and the MDFP. The MISVs were manually closed 3 min into the event when the 73-in administrative limit was reached; to prevent flooding outboard of the MISVs. However, the RCIC steam line is inboard of the MSIVs and the transient water level rose to 130 in. which is 22 in. above the bottom of the main steam lines. Closure of the MSIVs resulted in a trip of the TDFPs and loss of the main condenser as a heat sink. The safety relief valves (SRVs) were then required for control of reactor pressure. Although the SRVs were used successfully for this function, corrosion-caused instrumentation failures prevented direct confirmation of closure of two SRVs.

Attempts were made to use RCIC for reactor pressure control. Two start-up attempts failed as a result of high-exhaust-pressure trips. The cause was water accumulation in the steam lines which passed into the exhaust header via the pump turbine. There, flashing of water to steam resulted in pressure peaks which triggered the RCIC trips. The RCIC steam line drains had operated as designed, but the time available for water drainage was insufficient. The third attempt to start the RCIC (approximately 5 min after the first trip) was successful.

B.22.3 Additional Event-Related Information

LaSalle is equipped with high pressure core spray (HPCS) and RCIC, either of which can provide adequate reactor vessel makeup following a loss of feedwater (LOFW) or a loss of inventory from a stuck open relief valve. In addition, the MDFP can be used for reactor vessel makeup.

B.22.4 Modeling Assumptions

The event was modeled as a LOFW with failed RCIC. Potential sequences associated with the event are described in Appendix A, section A.3.2, BWR Nonspecific Reactor Trip, and shown on the event tree included with this analysis documentation. The plant response observed during the event impacted the following branches on the event tree:

- TRANSIENT (reactor trip occurs). The reactor tripped because of main turbine stop-valve closure.
- Power conversion system provides core cooling. The MSIVs were manually closed during the event in an attempt to prevent flooding of the main steam lines. This resulted in unavailability of the PCS.

- Feedwater provides reactor pressure vessel (RPV) makeup. The turbine-driven feedwater pumps shut down when the MSIVs were closed. The motor-driven feedwater pump tripped on high RPV water level. The motor-driven pump was assumed to be recoverable with a non-recovery probability of 0.12 (ASP non-recovery class R3, see Appendix A, sect. A.1). This value was chosen because the tripped pump appeared recoverable in the required period from the control room, but, because of the main steam line flooding and the problems with the turbine-driven feedwater and RCIC pumps, recovery was considered to be non-routine and burdened.

- RCIC provides reactor pressure vessel makeup. RCIC tripped twice on high exhaust pressure because of water accumulation in the steam lines. RCIC was assumed to be recoverable with a non-recovery probability of 0.12 (ASP non-recovery class R3), for the same reasons as FW. This non-recovery probabilty for RCIC may be conservative, since the steam line drain valves operated as intended and the third RCIC startup attempt was successful.

The current ASP event trees for LaSalle do not model the potential use of RCIC to provide RPV makeup in the event of a single stuck-open SRV. The use of RCIC for this purpose was included in the NUREG-1150 PRAs and utility-sponsored IPEs. To address this, the conditional probabilities for applicable sequences (sequences 25, 26 and 28) were reduced by the probability of failing to successfully use RCIC for this purpose. This is the probability that either RCIC fails, two or more SRVs fail to close given one or more fail to close, or long-term core cooling fails given RCIC is successful and only one SRV fails open. Since long-term core cooling is reliable, this probability can be approximated by

p(RCIC) + p(2 or more valves fail open | 1 or more valves fail open).

The failure probability for RCIC during this event was estimated above as 0.12. A value of 0.027 was estimated for p(2 or more valves fail open | 1 or more valves fail open), based on an estimated probability for two or more SRVs stuck open of 0.0015 (see NUREG/CR-4550, Vol. 1, Rev. 1, Analysis of Core Damage Frequency: Internal events Methodology, january 1990, p. 6-10) and an estimated probability of one or more SRVs stuck open of 0.056 (developed as described in Appendix A, sect. A.4).

The probability multiplier used to adjust sequences 25, 26 and 28 to account for the potential use of RCIC to mitigate the effects of a single stuck-open SRV is therefore 0.12 [p(RCIC)] + 0.027 [p(2 or more SRVs fail open | 1 or more SRVs fail open)] = 0.15. The conditional probability for sequence 28 (the only dominant sequence of the three sequences − 25, 26, and 28) was manually revised from 5.8×10^{-7} to 8.7×10^{-8} to reflect this. This reduces the core damage probability estimated for the event from 6.6×10^{-6} indicated on the calculational sheets to 6.1×10^{-6}.

B.22.5 Analysis Results

The estimated conditional core damage probability associated with the event is 6.1×10^{-6}. This probability was calculated by reducing the core damage probability shown in the calculations (6.6×10^{-6}) by the change in sequence 28 (from 5.8×10^{-7} to 8.7×10^{-8}, a factor of 0.15) as discussed in the last paragraph in the modeling assumptions section. This has been reduced from the value shown on the calculational sheets to reflect the potential use of RCIC to mitigate a single stuck open relief valve, as

discussed in the modeling assumptions section. The dominant core-damage sequence, highlighted on the event tree in Fig. B.47, involves an effective LOFW with successful reactor vessel makeup and failure to remove decay heat in the long term. Note that failure of RCIC does not contribute to the dominant sequences associated with the event.

This analysis addressed the potential loss of core cooling caused by failures of systems associated with transient mitigation. If the MISVs had not been closed, failure of the main steam line could have resulted. The potential for core damage from this sequence was not addressed in this analysis due to the difficulty in estimating the required steam line failure probabilities.

Additional information concerning this event is included in Augmented Inspection Team report 50-374/92020(DRS).

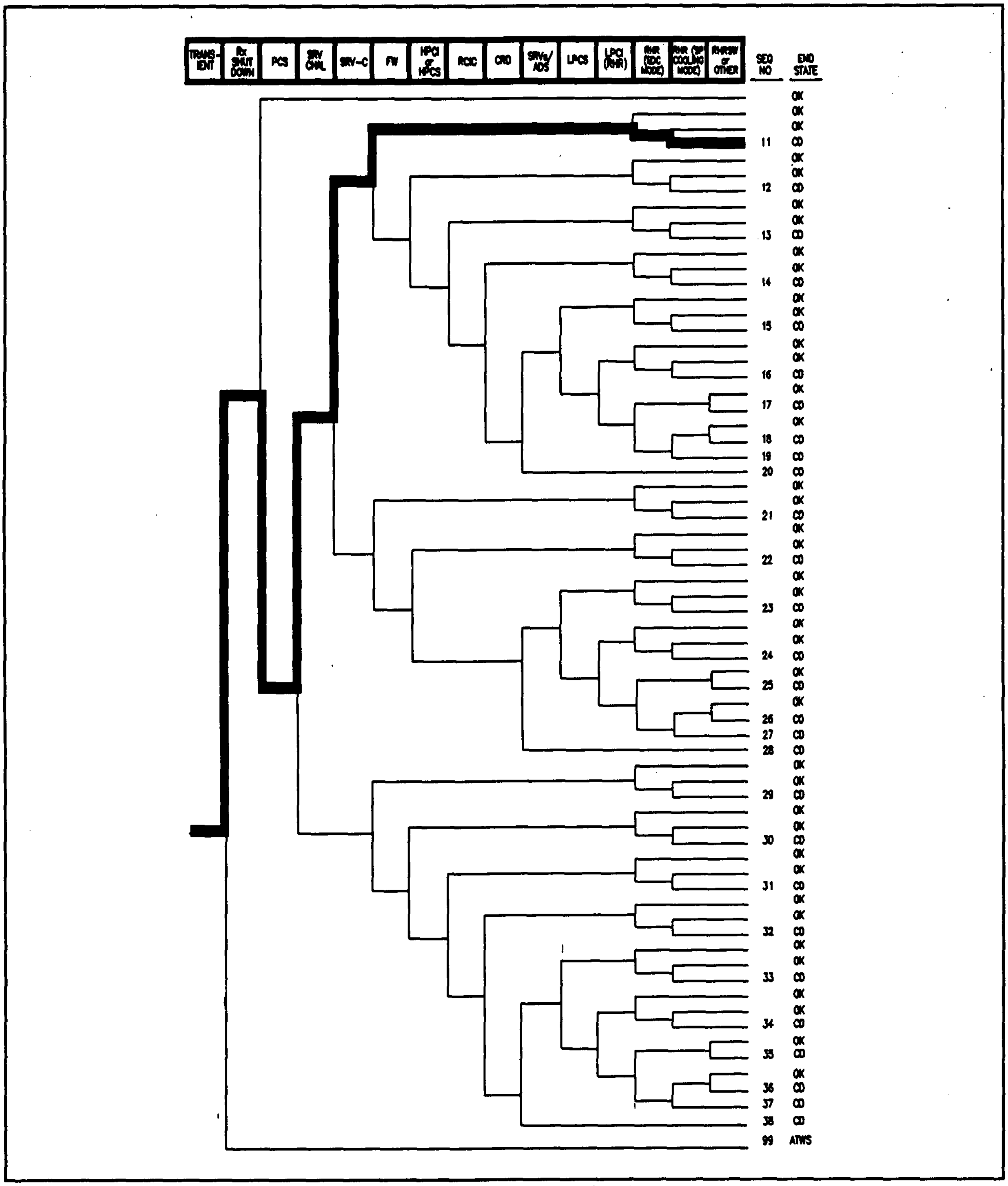

Fig. B.47. Dominant core damage sequences for LER 374/92-012

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 374/92-012
Event Description: Reactor trip and vessel overfill with degraded RCIC
Event Date: 08/27/92
Plant: LaSalle 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	6.6E-06[1]
Total	6.6E-06[1]
ATWS	
TRANS	3.0E-05
Total	3.0E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
11	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close -FW/PCS.TRANS rhr(sdc) rhr(spcool)/rhr(sdc)	CD	5.0E-06	1.0E-01
12	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	6.7E-07	1.4E-02
28	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS hpci srv.ads	CD	5.8E-07[1]	2.9E-02
21	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close -FW/PCS.TRANS rhr(sdc) rhr(spcool)/rhr(sdc)	CD	3.0E-07	1.0E-01
99	trans rx.shutdown	ATWS	3.0E-05	1.0E+00

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
11	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close -FW/PCS.TRANS rhr(sdc) rhr(spcool)/rhr(sdc)	CD	5.0E-06	1.0E-01
12	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram -srv.close FW/PCS.TRANS -hpci rhr(sdc) rhr(spcool)/rhr(sdc)	CD	6.7E-07	1.4E-02
21	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close -FW/PCS.TRANS rhr(sdc) rhr(spcool)/rhr(sdc)	CD	3.0E-07	1.0E-01
28	trans -rx.shutdown PCS/TRANS srv.chall/trans.-scram srv.close FW/PCS.TRANS hpci srv.ads	CD	5.8E-07[1]	2.9E-02
99	trans rx.shutdown	ATWS	3.0E-05	1.0E+00

** non-recovery credit for edited case

Event Identifier: 374/92-012

SEQUENCE MODEL: c:\asp\1989\bwrcseal.cmp
BRANCH MODEL: c:\asp\1989\lasalle.sl1
PROBABILITY FILE: c:\asp\1989\bwr_csl1.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	7.4E-05	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	3.3E-06	5.0E-01	
rx.shutdown	3.0E-05	1.0E+00	
rx.shutdown/ep	3.5E-04	1.0E+00	
PCS/TRANS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.7E-01 > Unavailable		
srv.chall/trans.-scram	1.0E+00	1.0E+00	
srv.chall/loop.-scram	1.0E+00	1.0E+00	
srv.close	5.6E-02	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
ep.rec	1.7E-01	1.0E+00	
FW/PCS.TRANS	4.6E-01 > 1.0E+00	3.4E-01 > 1.2E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.6E-01 > Unavailable		
FW/PCS.LOCA	1.0E+00 > 1.0E+00	3.4E-01 > 1.2E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	1.0E+00		
hpci	2.0E-02	3.4E-01	
RCIC	6.0E-02 > 1.0E+00	7.0E-01 > 1.2E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	6.0E-02 > Failed		
crd	1.0E-02	1.0E+00	1.0E-02
srv.ads	3.7E-03	7.1E-01	1.0E-02
lpcs	2.0E-02	3.4E-01	
lpci(rhr)/lpcs	6.0E-04	7.1E-01[1]	
rhr(sdc)	2.3E-02	3.4E-01	1.0E-03
rhr(sdc)/-lpci	2.0E-02	3.4E-01	1.0E-03
rhr(sdc)/lpci	1.0E+00	1.0E+00	1.0E-03
rhr(spcool)/rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/-lpci.rhr(sdc)	2.0E-03	3.4E-01	
rhr(spcool)/lpci.rhr(sdc)	9.3E-02	1.0E+00	
rhrsw	2.0E-02	3.4E-01	2.0E-03

* branch model file

Event Identifier: 374/92-012

Notes:

[1]See Modeling Assumptions for a discussion of changes made to this probability value.