B.15 LER Number 285/92-023 and 285/92-028

Event Description: Reactor Trip with Faulty Pressurizer Safety Valve

Date of Event: July 3, 1992

Plant: Fort Calhoun

B.15.1 Summary

Fort Calhoun tripped from 100% power on July 3, 1992. The reactor tripped on high pressure following the closure of all turbine control valves. Two pressurizer power-operated relief valves and one pressurizer safety valve opened to relieve reactor coolant system (RCS) pressure. After an initial pressure decrease in the RCS, the safety valve opened again. When RCS pressure reached 1000 psia the valve closed but continued to leak. The conditional core damage probability estimated for this event is 2.5×10^{-4}. The relative significance of this event compared to other postulated events at Fort Calhoun is shown in Fig. B.27.

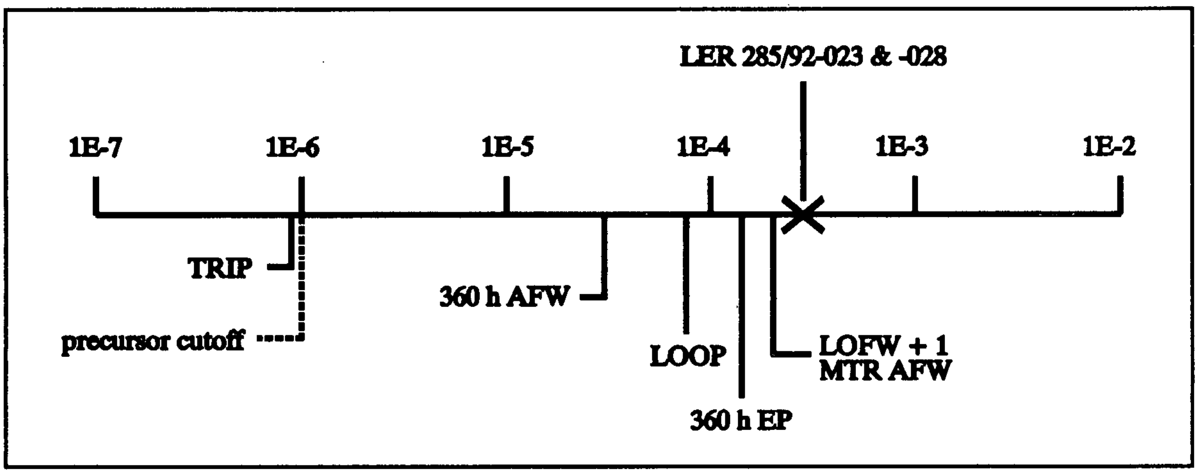

Fig. B.27. Relative significance of LER 285/92-023 and -028 compared with other potential events at Fort Calhoun.

B.15.2 Event Description

With Fort Calhoun at 100% power, nonsafety-related inverter no. 2 switched to its bypass mode three times on July 3, 1992. In the first two instances, the cause could not be determined and the inverter was returned to service. In the third instance, two circuit boards in the inverter were replaced before returning it to service at 2335 hours. However, one of the boards did not have the required jumper in place before installation. When the inverter was placed back in service, a voltage oscillation between 0

and 120-Vac was observed on the output of the inverter. The voltage oscillations caused the normal electrohydraulic control (EHC) system power supply to de-energize which caused four pressure transmitters powered by the EHC cabinet to also de-energize.

The four deenergized transmitters provide control signals to the turbine control valves. When power was lost to these transmitters, the control valves were sent a signal to close; however, closure of the control valves does not generate a turbine or reactor trip signal. With the control valves closed, a large mismatch developed between primary power production and secondary heat removal (steam dump has a 5% capacity when the turbine is not tripped). As a result, RCS pressure increased to about 2400 psia at which point a reactor trip occurred, both pressurizer power-operated relief valves (PORVs) opened, and one of the two pressurizer safety valves lifted. When RCS pressure decreased to 2350 psia, the PORVs reclosed; however, the safety valve remained open until RCS pressure decreased to 1745 psia. After the safety valve closed, RCS pressure increased for the next 8 min until it reached 1925 psia. At this point the safety valve opened again, and RCS pressure decreased rapidly. During this depressurization, safety injection (SI), emergency boration, containment isolation (CI), and containment ventilation isolation (CVI) were automatically actuated on low pressurizer pressure. The operator closed the PORV block valves in response to the loss of RCS pressure and the increasing pressure and temperature in the pressurizer quench tank; however, the safety valve still did not close. All four reactor coolant pumps (RCPs) were manually tripped as required when pressure dropped below 1350 psia. Throughout the remainder of the event, the SI flow was throttled to maintain 20°C subcooling. The rupture disk on the pressurizer quench tank actuated because of the sustained flow from the pressurizer safety valve. As a result, containment temperature, pressure, and radiation levels increased. RCS pressure decreased to $\approx$ 1000 psia, at which time the pressurizer safety valve finally closed but did not properly seat. This resulted in a continuous leak rate of approximately 200 gpm.

Following the shutdown of all RCPs, natural circulation was established at 0004 hours on July 4, 1992. By 1840 hours shutdown cooling was placed in service for cooldown to cold shutdown conditions. Approximately 21,500 gal of coolant was released though the pressurizer PORVs and safety valve during the event.

On August 22, 1992 with the plant operating at 100% power, the plant tripped following a partial closure of all the main turbine control valves. The partial closure was the result of a failure of an ac to dc power converter for two of the four pressure transmitters that provide control signals for the turbine control valve. These are the same transmitters that initiated the July 3 event. Due to the loss of turbine load, RCS pressure increased. The pressure increase was terminated when one of the pressurizer safety valves actuated 100 psi below its normal setpoint. This was the same pressurizer safety valve that lifted prematurely during the July 3, 1992 event. The reactor tripped when RCS pressure decreased to 1750 psia. RCS pressure stabilized at 1721 psia.

B.15.3 Additional Event-Related Information

Inverter no. 2 is a nonsafety-related inverter that supplies various nonsafety-related instrumentation and components in the plant. Among the loads supplied from this bus is the EHC power cabinet. All EHC components, except for the four transmitters that sense turbine pressures, receive backup power from the

permanent magnet generator (PMG). The PMG is driven by the main turbine shaft. Normally inverter no. 2 converts 125-Vdc from battery bus 2, to 120-Vac to supply the instrument bus. However, the inverter is equipped with a 480-Vac/120-Vac transformer to allow the inverter to be bypassed. The inverter automatically switches to this bypass mode when a problem is detected with the inverter.

There are two pressurizer code safety valves on the pressurizer that are set to actuate at 2500 and 2545 (+/− 25) psia. Each valve has a blowdown of ≈ 20% and therefore would be expected to shut at ≈ 2000 psia. During the event on July 3, 1992 it appears that the safety valve lifted the first time at about 2430 psia, which is below its normal setpoint. It remained open until pressure decreased to 1745 psia, which is below its normal blowdown setpoint of 2000 psia. The valve reopened when pressure increased to 1925 psia and then reclosed at about 1000 psia. The safety valve did not reset properly following the second cycle. From post-event inspection of the valve, it was concluded that the valve setpoint had changed during the event because of valve chatter. Valve chattering occurs when a safety valve oscillates off its seat (i.e., opens and closes rapidly). During the time of the valve chatter, vibration and torque caused the adjusting bolt to turn and reduced the valve's setpoint. Thus the valve had a blowdown of > 20% each time it lifted and actuated. To minimize valve chattering, the valve's blowdown was increased in 1990 from 5% to about 20%. The valve chattering also caused damage to the valve disc and disc holder. This prevented the valve from properly seating after the second cycle and resulted in a leak rate of approximately 200 gpm through the valve.

The utility determined that the root cause of the premature opening of the pressurizer safety valve during the August 22, 1992 event was improper calibration. The valve's setpoint was found to be sensitive to the temperature of the valve body and bonnet. The valve was calibrated while it was below its normal operating temperature. This resulted in a lower setpoint than anticipated.

B.15.4 Modeling Assumptions

The July 3, 1992 event was modeled as small break loss-of-coolant accident (LOCA). It could also have been modeled as a challenged pressurizer safety valve with no recovery possible. The results of both calculations are the same. The existing event model was modified to include the potential for RCS cooldown and use of RHR following successful initiation of HPI. Once the unit is placed on RHR (with limited HPI for RCS makeup), the transfer to HPR can be avoided. To do this, the HPR event was replaced with the results of the event tree in Fig. B.28 for sequences where AFW or MFW were successful (sequences 71 and 73). The probabilities for the additional events are shown in Table B.8. The August 22, 1992 event was not modeled and does not contribute to the calculated conditional probability of core damage.

B.15.5 Analysis Results

The conditional probability of core damage estimated for the July 3, 1992 event is 2.5×10^{-4}. The dominant core damage sequence, highlighted on the event tree in Fig. B.29 involves a failure of the high-pressure injection system following the LOCA.

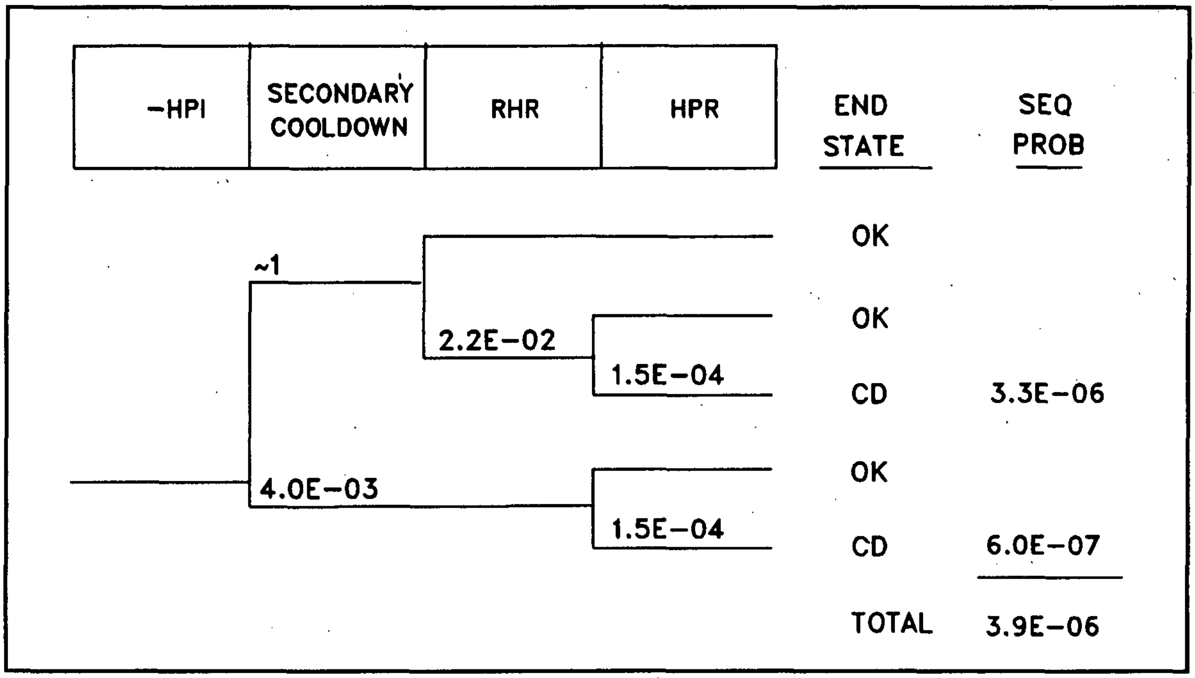

Fig. B.28. Modification for HPR event to include RHR as an alternative to HPR with successful HPI and secondary side cooling available.

Table B.8. Probability values used for modification of high-pressure recirculation event for 285/92-023 and -028.

Event	Model Probability	×	Non Recovery	+	Operator Action	=	Branch Probability
Secondary cooldown	3.0E-03*		1.0		1.0E-03		4.0E-03
RHR = VLV1+VLV2+ (PMP1×PMP2)+ (VLV3×VLV4× VLV5×VLV6) = 0.01+0.01+(0.01×0.1) +(0.01×0.1×0.3×0.5) = 0.021	2.1E-02		1.0		1.0E-03		2.2E-02
HPR = VLV1×VLV2 = 0.01×0.015 = 0.00015	1.5E-04		1.0				1.5E-04

*See NRR Daily Events Evaluation Manual, 1-275-03-336-01, January 31, 1992 (Preliminary).

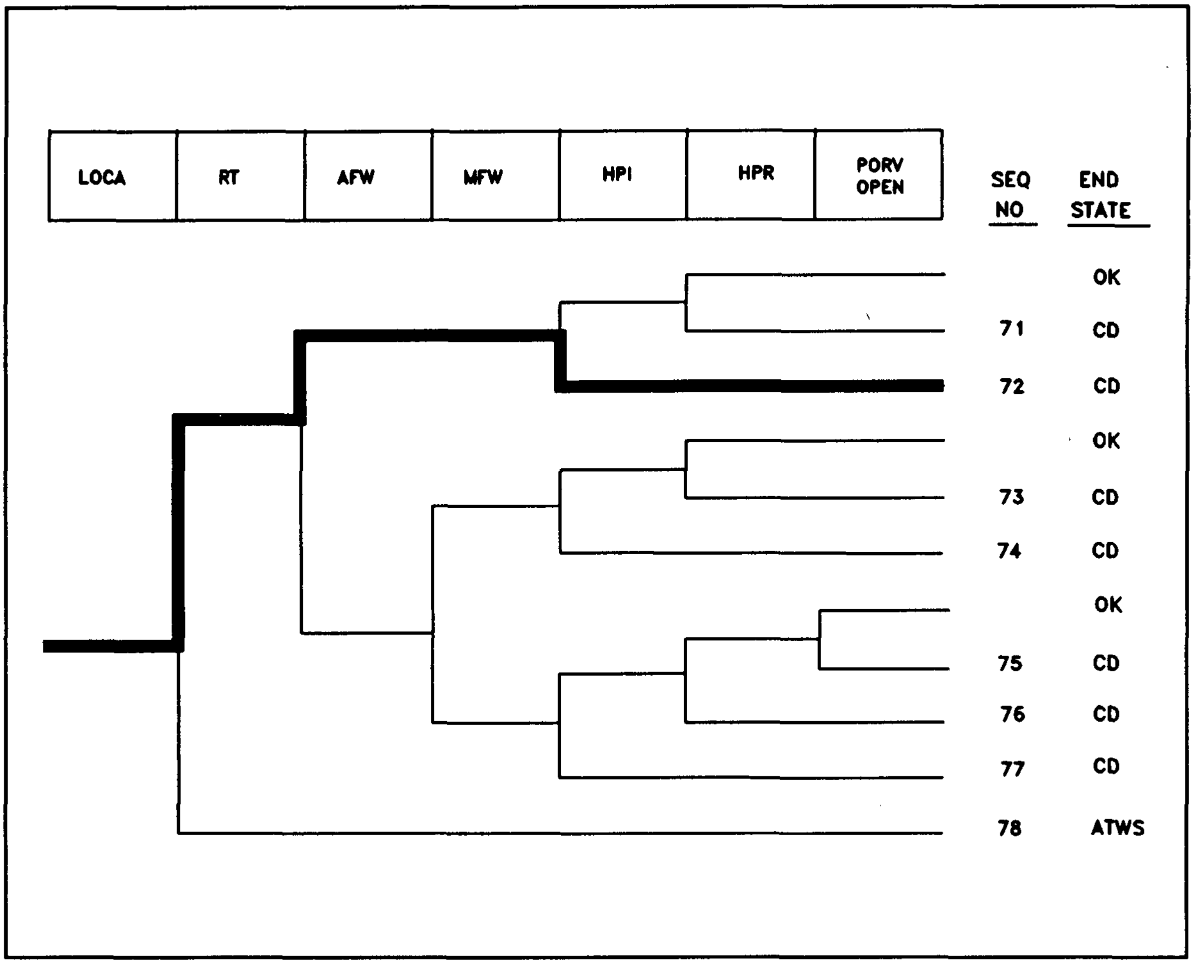

Fig. B.29. Dominant core damage sequence for LER 285/92-023.

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 285/92-023
Event Description: Trip Followed by Stuck Open PRZR Safety Valve
Event Date: 07/03/92
Plant: Fort Calhoun

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOCA	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOCA	2.5E-04
Total	2.5E-04
ATWS	
LOCA	3.4E-05
Total	3.4E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
72	LOCA -rt -afw hpi	CD	2.5E-04	8.4E-01
71	LOCA -rt -afw -hpi HPR/-HPI	CD	3.9E-06	1.0E+00
78	LOCA rt	ATWS	3.4E-05	1.2E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
71	LOCA -rt -afw -hpi HPR/-HPI	CD	2.1E-05	1.0E+00
72	LOCA -rt -afw hpi	CD	2.5E-04	8.4E-01
78	LOCA rt	ATWS	3.4E-05	1.2E-01

** non-recovery credit for edited case

SEQUENCE MODEL: s:\asp\prog\models\pwrgseal.cmp
BRANCH MODEL: s:\asp\prog\models\calhoun.sl2
PROBABILITY FILE: s:\asp\prog\models\pwr_prob.pro

No Recovery Limit

Event Identifier: 285/92-023

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	6.0E-05	1.0E+00	
loop	1.6E-05	5.3E-01	
LOCA	2.4E-06 > 2.4E-06	4.3E-01 > 1.0E+00	
Branch Model: INITOR			
Initiator Freq:	2.4E-06		
rt	2.8E-04	1.2E-01	
rt/loop	0.0E+00	1.0E+00	
emerg.power	2.9E-03	8.0E-01	
afw	1.3E-03	2.6E-01	
afw/emerg.power	5.0E-02	3.4E-01	
mfw	1.9E-01	3.4E-01	
porv.or.srv.chall	4.0E-02	1.0E+00	
porv.or.srv.reseat	2.0E-02	1.1E-02	
porv.or.srv.reseat/emerg.power	2.0E-02	1.0E+00	
seal.loca	4.6E-02	1.0E+00	
ep.rec(sl)	5.7E-01	1.0E+00	
ep.rec	1.4E-02	1.0E+00	
hpi	3.0E-04	8.4E-01	
hpi(f/b)	3.0E-04	8.4E-01	1.0E-02
porv.open	2.0E-02	1.0E+00	4.0E-04
HPR/-HPI	1.5E-04 > 3.9E-06 **[1]	1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.5E-02		
csr	2.0E-03	3.4E-01	1.0E-03

* branch model file
** forced

Notes:
1. Probability was modified to account for the possible use of RHR as an alternative to HPR. See the modeling assumptions sectio for a description of this modification.

Event Identifier: 285/92-023