Table of Contents

System Special Event Summaries

1 HPCS Special Events Data Sheet

If a LOCA should occur, a low reactor water level signal or high drywell pressure signal initiates the HPCS and its support equipment. The system can also be placed in operation manually. If the leak rate is less than the HPCS system flow rate, the HPCS system automatically stops when a high reactor water level signal shuts the HPCS injection valve. The injection valve will automatically reopen upon a subsequent low water level signal. Suction piping for the HPCS pump is provided from the condensate storage tank (CST) and the suppression pool. Such an arrangement provides the capability to use reactor-grade water from the CST when the HPCS system functions to back up the RCIC system. In the event that the CST water supply becomes exhausted or is not available, automatic switchover to the suppression pool water source ensures a cooling water supply for long-term operation of the system.

1.1 Special Event Description

The HPCS special events are listed in Table 1-1.

Table 1-1. HPCS special events.

Special Event	Parameter	Units	Description
SUC-FTFR	*p*	-	Failure to transfer (to the suppression pool)
SUC-FRFTFR	*p*	-	Failure to recover transfer failure

1.2 Data Collection and Review

Using the process outlined in Section C.1.2, the optimized baseline period for each special event is listed in Table 1-2. Results include total number of events and either total demands or total hours for the U.S. commercial nuclear power plant industry. The table summarizes the data used in the HPCS special event analysis.

Table 1-2. HPCS special event data.

Special Event	Data Source	Data After Review	
		Events	Demands or Hours
SUC-FTFR	System Study (1988 - 2002)	1	478
SUC-FRFTFR	System Study (1988 - 2002)	1	1

1.3 Industry-Average Baselines

Table 1-3 lists the industry-average distributions for the HPCS special events.

Table 1-3. Selected industry distributions of *p* for HPCS (before rounding).

Event	5%	Median	Mean	95%	Distribution		
					Type	α	β
SUC-FTFR	1.23E-05	1.43E-03	3.13E-03	1.20E-02	Beta	0.500	1.592E+02
SUC-FRFTFR	3.26E-02	9.51E-01	7.50E-01	1.00E+00	Beta	0.500	1.667E-01

For use in the SPAR models, the industry-average event probabilities and rates were rounded to 1.0, 1.2, 1.5, 2.0, 2.5, 3.0, 4.0, 5.0, 6.0, 7.0, 8.0, or 9.0 times the appropriate power of ten. Similarly, the α parameter was rounded. In order to preserve the mean value, the β parameter is presented to three significant figures. Table 1-4 shows the rounded values for the HPCS special events.

Table 1-4. Selected industry distributions of *p* for HPCS (after rounding).

Event	5%	Median	Mean	95%	Distribution		
					Type	α	β
SUC-FTFR	1.2E-05	1.5E-03	3.0E-03	1.2E-02	Beta	0.50	1.66E+02
SUC-FRFTFR	5.0E-02	1.0E+00	8.0E-01	1.0E+00	Beta	0.50	1.25E-01

2 HPCI Special Events Data Sheet

The HPCI system is actuated by either a low reactor water level or a high drywell pressure. Initially the system operates in an open loop mode, taking suction from the condensate storage tank (CST), and injecting water into the reactor pressure vessel (RPV) via one of the main feedwater lines. When the level in the CST reaches a low-level setpoint, the HPCI pump suction is aligned to the suppression pool. To maintain RPV level after the initial recovery, the HPCI system is placed in manual control, which may involve controlling turbine speed, diverting flow through minimum flow or test lines, cycling the injection motor-operated valve (MOV), or complete stop-start cycles.

The HPCI system is also used manually to help control RPV pressure following a transient. In this mode, the turbine-driven pump is operated manually with the injection valve closed and the full-flow test line MOV open. Turbine operation with the injection line isolated and the test line open allows the turbine to draw steam from the RPV, thereby reducing RPV pressure. Operation of the system in the pressure control mode may also occur with intermittent injection of coolant to the RPV. As steam is being drawn off the RPV, the RPV water inventory is reduced, resulting in the need for level restoration. When level restoration is required, the injection valve is opened and the test line MOV is closed. Upon restoration of RPV water inventory, the system is returned to the pressure control line-up. This cycling between injection and pressure control can be repeated as necessary.

2.1 Special Event Description

The HPCI special events are listed in Table 2-1.

Table 2-1. HPCI special events.

Special Event	Parameter	Units	Description
MOV-PMINJ	*p*	-	Injection valve probability of multiple injections
MOV-FTRO	*p*	-	Injection valve fails to reopen
MOV-FRFTRO	*p*	-	Failure to recover injection valve failure to reopen
SUC-FTFR	*p*	-	Failure to transfer (to the suppression pool)
SUC-FRFTFR	*p*	-	Failure to recover transfer failure

2.2 Data Collection and Review

Using the process outlined in Section C.1.2, the optimized baseline period for each special event is listed in Table 2-2. Results include total number of events and either total demands or total hours for the U.S. commercial nuclear power plant industry. The table summarizes the data used in the RCIC special event analysis.

Table 2-2. HPCI special event data.

Special Event	Data Source	Data After Review	
		Events	Demands or Hours
MOV-PMINJ	System Study (1995 - 2002)	2	17
MOV-FTRO	System Study (1988 - 2002)	1	8
MOV-FRFTRO	System Study (1988 - 2002)	1	1
SUC-FTFR	System Study (1989 - 2002)	0	1270
SUC-FRFTFR	System Study (1989 - 2002)	0	0

2.3 Industry-Average Baselines

Table 2-3 lists the industry-average distributions for the HPCI special events.

Table 2-3. Selected industry distributions of *p* for HPCI (before rounding).

Event	5%	Median	Mean	95%	Distribution		
					Type	α	β
MOV-PMINJ	6.88E-04	7.65E-02	1.39E-01	4.88E-01	Beta	0.500	3.100E+00
MOV-FTRO	8.70E-04	9.58E-02	1.67E-01	5.70E-01	Beta	0.500	2.500E+00
MOV-FRFTRO	3.26E-02	9.51E-01	7.50E-01	1.00E+00	Beta	0.500	1.667E-01
SUC-FTFR	1.55E-06	1.79E-04	3.93E-04	1.51E-03	Beta	0.500	1.271E+03
SUC-FRFTFR	6.16E-03	5.00E-01	5.00E-01	9.94E-01	Beta	0.500	5.000E-01

For use in the SPAR models, the industry-average event probabilities and rates were rounded to 1.0, 1.2, 1.5, 2.0, 2.5, 3.0, 4.0, 5.0, 6.0, 7.0, 8.0, or 9.0 times the appropriate power of ten. Similarly, the α parameter was rounded. In order to preserve the mean value, the β parameter is presented to three significant figures. Table 2-4 shows the rounded values for the HPCI special events.

Table 2-4. Selected industry distributions of *p* for HPCI (after rounding).

Event	5%	Median	Mean	95%	Distribution		
					Type	α	β
MOV-PMINJ	8.0E-04	8.0E-02	1.5E-01	5.0E-01	Beta	0.50	2.83E+00
MOV-FTRO	8.0E-04	8.0E-02	1.5E-01	5.0E-01	Beta	0.50	2.83E+00
MOV-FRFTRO	5.0E-02	1.0E+00	8.0E-01	1.0E+00	Beta	0.50	1.25E-01
SUC-FTFR	1.5E-06	2.0E-04	4.0E-04	1.5E-03	Beta	0.50	1.25E+03
SUC-FRFTFR	6.0E-03	5.0E-01	5.0E-01	1.0E+00	Beta	0.50	5.00E-01

3 RCIC Special Events Data Sheet

Following a normal reactor shut down, core fission product decay heat causes steam generation to continue, albeit at a reduced rate. During this time, the turbine bypass system diverts the steam to the main condenser, and the RCIC system supplies the makeup water required to maintain reactor pressure vessel (RPV) inventory. (Note that the RCIC system is just one of a number of systems capable of performing this function.) The turbine-driven pump (TDP) supplies makeup water from the condensate storage tank (CST) to the reactor vessel. An alternate source of water is available from the suppression pool. The turbine is driven by a portion of the steam generated by the decay heat and exhausts to the suppression pool. This operation continues until the vessel pressure and temperature is reduced to the point that the residual heat removal (RHR) system can be placed into operation.

Operation of RCIC for long-term missions involves providing adequate RPV water level for periods up to several hours. For these long-term missions, either the control room operator would manually initiate the RCIC system, or the system would automatically start at the predetermined low reactor water level setpoint. At this point, the system would inject until the system was shut down by the operator or the high level trip setpoint was reached, at which time the RCIC turbine steam supply and coolant injection valves would close. With the continued steam generated by decay heat and corresponding lowering of vessel level (as a result of safety relief valve or turbine bypass valve operation), the system would be restarted during the event and the cycle repeated one or more times.

3.1 Special Event Description

The RCIC special events are listed in Table 3-1.

Table 3-1. RCIC special events.

Special Event	Parameter	Units	Description
TDP-PRST	*p*	-	TDP probability of restart
TDP-FRST	*p*	-	TDP restart failure per event
TDP-FRFRST	*p*	-	Failure to recover TDP restart failure
SUC-FTFRI	λ	1/h	Failure to transfer back to injection mode (pump recirculation valve)
SUC-FRFTFR	*p*	-	Failure to recover transfer failure
MOV-PMINJ	*p*	-	Injection valve probability of multiple injections
MOV-FTRO	*p*	-	Injection valve fails to reopen
MOV-FRFTRO	*p*	-	Failure to recover injection valve failure to reopen

3.2 Data Collection and Review

Using the process outlined in Section C.1.2, the optimized baseline period for each special event is listed in Table 3-2. Results include total number of events and either total demands or total hours for the U.S. commercial nuclear power plant industry. The table summarizes the data used in the RCIC special event analysis.

Table 3-2. RCIC special event data.

Special Event	Data Source	Data After Review	
		Events	Demands or Hours
TDP-PRST	System Study (1996 - 2002)	6	47
TDP-FRST	System Study (1991 - 2002)	1	17
TDP-FRFRST	System Study (1991 - 2002)	0	1
SUC-FTFRI	System Study (1988 - 2002)	1	198 h
SUC-FRFTFR	System Study (1988 - 2002)	0	1
MOV-PMINJ	System Study (1998 - 2002)	14	28
MOV-FTRO	System Study (1988 - 2002)	1	38
MOV-FRFTRO	System Study (1988 - 2002)	1	1

3.3 Industry-Average Baselines

Table 3-3 lists the industry-average distributions for the RCIC special events. The SCNID was used for six of the eight events. TDP-PRST uses the Jeffreys distribution because the empirical Bayes analysis (looking for year-to-year variation) failed but indicated little variation between years. Finally, MOV-PMINJ uses the empirical Bayes results.

Table 3-3. Selected industry distributions of p and λ for RCIC (before rounding).

Event	5%	Median	Mean	95%	Distribution		
					Type	α	β
TDP-PRST	6.43E-02	1.30E-01	1.35E-01	2.23E-01	Beta	6.500	4.150E+00
TDP-FRST	3.74E-04	4.23E-02	8.33E-02	3.06E-01	Beta	0.500	5.500E+00
TDP-FRFRST	1.54E-03	1.63E-01	2.50E-01	7.71E-01	Beta	0.500	1.500E+00
SUC-FTFRI	2.98E-05	3.45E-03	7.58E-03	2.91E-02	Gamma	0.500	6.598E+01
SUC-FRFTFR	1.54E-03	1.63E-01	2.50E-01	7.71E-01	Beta	0.500	1.500E+00
MOV-PMINJ	2.32E-01	5.03E-01	5.03E-01	7.73E-01	Beta	4.180	4.130E+00
MOV-FTRO	1.54E-04	1.77E-02	3.85E-02	1.40E-01	Beta	0.500	1.300E+01
MOV-FRFTRO	3.26E-02	9.51E-01	7.50E-01	1.00E+00	Beta	0.500	1.667E-01

For use in the SPAR models, the industry-average event probabilities and rates were rounded to 1.0, 1.2, 1.5, 2.0, 2.5, 3.0, 4.0, 5.0, 6.0, 7.0, 8.0, or 9.0 times the appropriate power of ten. Similarly, the α parameter was rounded. In order to preserve the mean value, the β parameter is presented to three significant figures. Table 3-4 shows the rounded values for the RCIC special events.

Table 3-4. Selected industry distributions of p and λ for RCIC (after rounding).

Event	5%	Median	Mean	95%	Distribution		
					Type	α	β
TDP-PRST	7.0E-02	1.5E-01	1.5E-01	2.5E-01	Beta	6.00	3.40E+01
TDP-FRST	4.0E-04	4.0E-02	8.0E-02	3.0E-01	Beta	0.50	5.75E+00
TDP-FRFRST	1.5E-03	1.5E-01	2.5E-01	8.0E-01	Beta	0.50	1.50E+00
SUC-FTFRI	3.0E-05	4.0E-03	8.0E-03	3.0E-02	Gamma	0.50	6.20E+01
SUC-FRFTFR	1.5E-03	1.5E-01	2.5E-01	8.0E-01	Beta	0.50	1.50E+00
MOV-PMINJ	2.5E-01	5.0E-01	5.0E-01	8.0E-01	Beta	4.00	4.00E+00
MOV-FTRO	1.5E-04	2.0E-02	4.0E-02	1.5E-01	Beta	0.50	1.25E+01
MOV-FRFTRO	5.0E-02	1.0E+00	8.0E-01	1.0E+00	Beta	0.50	1.25E-01

4 References

1. J.P. Poloski et al., *Reliability Study: Reactor Core Isolation Cooling System, 1987 – 1993*, U.S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 7, September 1999.

2. J.P. Poloski et al., *Reliability Study: High-Pressure Core Spray System, 1987 – 1993*, U.S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 8, September 1999.

3 G.M. Grant et al., *Reliability Study: High-Pressure Coolant Injection (HPCI) System, 1987 – 1993*, U.S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 4, September 1999.

4 U.S. Nuclear Regulatory Commission, "Reactor Operational Experience Results and Databases, System Studies," http://nrcoe.inel.gov/results.