15.3 Decrease in Reactor Coolant System Flow Rate

15.3.1 Reactor Internal Pump Trip

15.3.1.1 Identification of Causes and Frequency Classification

15.3.1.1.1 Identification of Causes

Reactor internal pump (RIP) motor operation can be tripped off by design for intended reduction of other transient core and RCPB effects, as well as randomly by unpredictable operational failures. Intentional tripping will occur in response to:

(1) Reactor vessel water level L3 setpoint trip (4 RIPs)

(2) Reactor vessel water level L2 setpoint trip (the other 6 RIPs)

(3) TCV fast closure or stop valve closure (the same 4 RIPs as L3 trip)

(4) High pressure setpoint trip (the same 4 RIPs as L3 trip)

(5) Motor overcurrent protection (single pump)

(6) Motor overload and short circuit protection (single pump)

Random tripping will occur in response to:

(1) Operator error.

(2) Loss of electrical power source to the pumps.

(3) Equipment or sensor failures and malfunctions which initiate the above intended trip response. However, all trip logics use redundant digital designs. Single failures in the UAT or MPT and/or their protection circuits can result in loss of preferred power source to the plant.

Thus, the worst single-failure event is a loss of electrical power bus, which supplies power to the RIPs. Since four buses are used to supply power to the RIPs, the worst single failure can only cause three RIPs to trip.

A loss of AC power to station auxiliaries may cause RIPs to trip. However, not all RIPs would be tripped at the same time due to the M-G sets. Transients caused by a loss of AC power are discussed in Subsection 15.2.6.

The effect of an additional single failure on this event (i.e., trip of three RIPs) is the tripping of additional RIPs. For example, if an additional power bus fails at the same time, the number of RIPs tripped is five or six, instead of three. However, the probability of this occurring is low.

This event should be classified as a limiting fault. In this analysis, the trip of all RIPs is provided to bound the events of low probability.

When a rapid core flow reduction caused by a trip of all RIPs is sensed, a reactor scram is initiated to terminate the power generation. The core flow reduces rapidly due to the relatively small inertia of the RIPs. However, natural circulation is still available to keep the reactor core covered and cooled.

15.3.1.1.2 Frequency Classification

15.3.1.1.2.1 Trip of Three Reactor Internal Pumps

This transient event is categorized as one of moderate frequency.

15.3.1.1.2.2 Trip of All Reactor Internal Pumps

This event is categorized as an infrequent low probability event with special acceptance for fuel failure (see Subsection 15.3.1.5.2).

15.3.1.2 Sequence of Events and Systems Operation

15.3.1.2.1 Sequence of Events

15.3.1.2.1.1 Trip of Three Reactor Internal Pumps

Table 15.3-1 lists the sequence of events for Figure 15.3-1.

15.3.1.2.1.2 Trip of All Reactor Internal Pumps

Table 15.3-2 lists the sequence of events for Figure 15.3-2.

15.3.1.2.1.3 Identification of Operator Actions

15.3.1.2.1.3.1 Trip of Three Reactor Internal Pumps

Because no scram occurs for trip of three RIPs, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. The operator should also determine the cause of failure prior to returning the system to normal operation.

15.3.1.2.1.3.2 Trip of All Reactor Internal Pumps

The operator should ascertain that the reactor scram is initiated. If the main turbine and feedwater pumps are tripped resulting from reactor water level swell, the operator should regain control of reactor water level through RCIC operation, monitoring reactor water level and pressure after shutdown. When both reactor pressure and level are under control, the operator should secure RCIC as necessary. The operator should also determine the cause of the trip prior to returning the system to normal operation.

15.3.1.2.2 Systems Operation

15.3.1.2.2.1 Trip of Three Reactor Internal Pumps

Tripping of three RIPs requires no protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.3.1.2.2.2 Trip of All Reactor Internal Pumps

Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection and reactor protection systems.

If a trip of all RIPs is caused by an electrical power supply to the RIPs, a reactor scram will be initiated at time 0 due to load rejection or turbine trip at time 0. For other causes, a reactor scram will be initiated upon the condition of high simulated thermal power scram, turbine trip due to high water level, or rapid core flow coastdown. High system pressure is limited by the pressure relief valve system operation.

Since the event becomes more severe when the reactor scram is delayed, the analysis conservatively assumes that the reactor scram is initiated by the last signal (i.e., core flow rapid coastdown scram). It is also conservatively assumed that the event is caused by a common mode failure in all ASDs, which results in a trip of all RIPs.

15.3.1.3 Core and System Performance

15.3.1.3.1 Input Parameters and Initial Conditions

Pump motors and pump rotors are simulated with minimum specified rotating inertias.

15.3.1.3.1.1 Trip of Three Reactor Internal Pumps

The nuclear conditions for the beginning of cycle (BOC) are used to provide conservative bounding analysis.

15.3.1.3.1.2 Trip of All Reactor Internal Pumps

Since the Peak Clad Temperature (PCT) result is more limiting in the high core flow condition, the analysis is performed at 111.1% core flow condition.

15.3.1.3.2 Results

15.3.1.3.2.1 Trip of Three Reactor Internal Pumps

Figure 15.3-1 shows the results of losing three RIPs. MCPR remains above the safety limit; thus, the fuel thermal limits are not violated. During this transient, level swell is not sufficient to cause feedwater pumps to trip, and turbine trip and scram.

Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.1.3.2.2 Trip of all Reactor Internal Pumps

Figure 15.3-2 graphically shows this event with the minimum specified rotating inertia for the RIPs. The reactor is scrammed based on the rapid flow coastdown signal. Subsequent events, such as initiation of the RCIC System occurring late in this event, have no significant effect on the results. The peaking cladding temperature (PCT) during this event is calculated to be 487°C, which is below the applicable limit of 600°C. The cladding temperature during this event is shown in Figure 15.3-2d. The period of time the cladding is above normal temperatures is less than 5 seconds. Since the time that the cladding temperature is above the coolant saturated temperature is less than 60 seconds, and the peak cladding temperature is less than 600°C, no fuel failure is expected.

This event does not have to be reanalyzed for specific core configurations.

15.3.1.4 Barrier Performance

15.3.1.4.1 Trip of Three Reactor Internal Pumps

The results shown in Figure 15.3-1 indicate that peak pressures stay well below the 9.48 MPa limit allowed by the applicable code. Therefore, the barrier pressure boundary is not threatened.

15.3.1.4.2 Trip of All Reactor Internal Pumps

The results shown in Figure 15.3-2 indicate that peak pressures stay well below the limit allowed by the applicable code. Therefore, the barrier pressure boundary is not threatened.

15.3.1.5 Radiological Consequences

15.3.1.5.1 Trip of Three Reactor Internal Pumps

This event does not result in any fuel failures, nor any discharge to the suppression pool. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.3.1.5.2 Trip of All Reactor Internal Pumps

The approved procedures for radiological dose calculation for this event are as follows:

(1) For fuel rods with less than or equal to 20 GWd/MT exposure, fuel failures are assumed if the PCT stays above 600°C for more than 60 seconds.

(2) For fuel rods with greater than 20 GWd/MT exposure, rods that are in transition boiling shall be assumed to fail for radiological dose calculations.

(3) The radiological doses shall be less than 10% of 10CFR100 requirements.

As discussed in Subsection 15.3.1.3.2.2, the PCT during this event is significantly less than 600°C and the time at above normal temperature is less than 5 seconds. At the EOEC condition, about 20% of the fuel bundles in the core will have between 20 and 23 GWD/MT and experience some rod boiling transition, slightly greater than the 20 GWD/T criterion developed from reference 15.3-1. There is no evidence that fuel rods with this exposure will experience cladding failure as a result of the small temperature transient seen. Further, more recent fuel rod dryout testing has been performed at the Halden reactor, simulating conditions expected after an all-pump trip in BWRs having reactor internal pumps (References 15.3-2 through 15.3-4). Fresh fuel and fuel pre-irradiated from 22 to 40 GWD/MT with non-barrier and Zr barrier cladding were tested through repeated cycles with PCTs up to 800 °C and durations above 500 °C of up to 40 seconds. The fuel was placed into continued normal operation conditions for up to one month after the dryout tests. No fuel clad failures were observed.

In the current analysis, no fuel with exposure greater than 23 GWD/MT was calculated to go into boiling transition. Therefore since it is expected that there will be no fuel failures, the criterion of 10CFR 100 is met.

15.3.2 Recirculation Flow Control Failure—Decreasing Flow

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

The Recirculation Flow Control System (RFCS) uses a triplicated, fault-tolerant digital control system, instead of an analog system, as used in BWR/2 through BWR/6. The RFCS controls all 10 reactor internal pumps (RIPs) at the same speed. As presented in Subsection 15.1.2.1.1, no credible single failure in the control system will result in a minimum demand to all RIPs. A voter or actuator failure may result in an inadvertent runback of one RIP at its maximum drive speed (~40%/s). In this case, the RFCS will sense the core flow change and command the remaining RIPs to increase speeds and thereby automatically mitigate the transient and maintain the core flow.

As presented in Subsection 15.1.2.1.1, multiple failures in the control system might cause the RFCS to erroneously issue a minimum demand to all RIPs. Should this occur, all RIPs could reduce speed simultaneously. Each RIP drive has a speed limiter which limits the maximum speed change rate to 5%/s. However, the probability of this event occurring is very low, and, hence, the event should be considered as a limiting fault. However, criteria for moderately frequent incidents are conservatively applied.

15.3.2.1.2 Frequency Classification

15.3.2.1.2.1 Fast Runback of One Reactor Internal Pump

The failure rate of a voter or an actuator is very low. However, it is analyzed as an incident of moderate frequency.

15.3.2.1.2.2 Fast Runback of All Reactor Internal Pumps

This event should be classified as a limiting fault event. However, criteria for moderate frequent incidents are conservatively applied.

15.3.2.2 Sequence of Events and Systems Operation

15.3.2.2.1 Sequence of Events

15.3.2.2.1.1 Fast Runback of One Reactor Internal Pump

Table 15.3-3 lists the sequence of events for Figure 15.3-3.

15.3.2.2.1.2 Fast Runback of All Reactor Internal Pumps

Table 15.3-4 lists the sequence of events for Figure 15.3-4.

15.3.2.2.1.3 Identification of Operator Actions

15.3.2.2.1.3.1 Fast Runback of One Reactor Internal Pump

As soon as possible, the operator verifies that no operating limits are being exceeded. The operator determines the cause of failure prior to returning the system to normal.

15.3.2.2.1.3.2 Fast Runback of All Reactor Internal Pumps

As soon as possible, the operator verifies that no operating limits are being exceeded. If they are, corrective actions must be initiated. Also, the operator determines the cause of the failures prior to returning the system to normal.

15.3.2.2.2 Systems Operation

15.3.2.2.2.1 Fast Runback of One Reactor Internal Pump

Normal plant instrumentation and control is assumed to function.

15.3.2.2.2.2 Fast Runback of All Reactor Internal Pumps

Normal plant instrumentation and control is assumed to function.

15.3.2.3 Core and System Performance

15.3.2.3.1 Input Parameters and Initial Conditions

15.3.2.3.1.1 Fast Runback of One Reactor Internal Pump

Failure can result in the maximum speed of the RIP decreasing at a rate of 40%/s as limited by the pump drive.

15.3.2.3.1.2 Fast Runback of All Reactor Internal Pumps

A downscale failure of the master controller will generate a zero flow demand signal to all RIPs. Each individual RIP drive has a speed limiter which limits the maximum speed decrease to a rate of 5%/s. Core flow decreases to approximately 40% of rated. This is the flow expected when the RIPs are maintained at their minimum speeds.

15.3.2.3.2 Results

15.3.2.3.2.1 Fast Runback on One Reactor Internal Pump

Figure 15.3-3 illustrates the fast runback of one RIP event with the maximum rate which is limited by hydraulic means. The MCPR remains above the safety limit. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.2.3.2.2 Fast Runback of All Reactor Internal Pumps

Figure 15.3-4 illustrates the expected event. Design of limiter operation is intended to render this event to be less severe than the trip of all RIPs. No fuel damage is expected to occur. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.2.4 Barrier Performance

15.3.2.4.1 Fast Runback of One Reactor Internal Pump

Peak pressures are about the same as those for the “Fast Runback of All RIPs” presented in Subsection 15.3.2.4.2.

15.3.2.4.2 Fast Runback of All Reactor Internal Pumps

Pressure in the vessel bottom is not higher than its initial value and below the ASME code limit.

15.3.2.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.3.3 Reactor Internal Pump Seizure

15.3.3.1 Identification of Causes and Frequency Classification

The seizure of a reactor internal pump (RIP) is considered a design basis accident (DBA) event. It has been evaluated as being a very mild accident in relation to others DBAs such as the LOCA. (Refer to Section 5.1 for special mechanical considerations and Chapter 7 for electrical aspects.)

The seizure event postulated is not expected to be the mode failure of such a device. Safe shutdown components (e.g., electrical breakers, protective circuits) preclude an instantaneous seizure event.

15.3.3.1.1 Identification of Causes

The cause of RIP seizure represents the unlikely event of instantaneous stoppage of the pump motor shaft of one reactor internal pump. This event produces a very rapid decrease of pump flow as a result of the large hydraulic resistance introduced by the stopped rotor. Consequently, a decrease in core inlet flow and core cooling capability occurs. However, with only one out of ten RIPs seized, the core flow decrease is small (~10%), so the event is very mild.

15.3.3.1.2 Frequency Classification

This event is considered to be a limiting fault but results in effects which can satisfy an event of greater probability (i.e., infrequent incident classification).

15.3.3.2 Sequence of Events and Systems Operations

15.3.3.2.1 Sequence of Events

Table 15.3-5 lists the sequence of events for Figure 15.3-5.

15.3.3.2.1.1 Identification of Operator Actions

Because no scram occurs for one RIP seizure, no immediate operator action is required. As soon as possible, the operator verifies that no operating limits are being exceeded. Also, the operator determines the cause of failure and proceeds to shutdown the plant for repair.

15.3.3.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems. Acceptance Criterion II.8 of SRP Section 15.3.3 provides that only safety grade equipment is be used to mitigate the consequences of this event. It also provides that safety functions be accomplished assuming the worst single failure of a safety system active component. Acceptance Criterion II.10 of SRP Section 15.3.3 also provides that the analysis assume turbine trip and coincident loss of offsite power. Should a coincident loss of offsite power occur, the consequences would be similar to the consequences of the loss of offsite power (LOOP) transient described in Subsection 15.2.6.

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the plant instrumentation and controls, plant protection, and reactor protection systems will not cause this accident to be more severe than analyzed.

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

Refer to Table 15.0-1a.

15.3.3.3.2 Input Parameters and Initial Conditions

For the purpose of evaluating consequences to the fuel thermal limits, this event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at 102% NBR power. Also, the reactor is assumed to be operating at thermally limited conditions.

15.3.3.3.3 Results

This event produces a very rapid decrease of pump flow as a result of the large hydraulic resistance introduced by the stopped rotor. Consequently, a decrease in core inlet flow and core cooling capability occurs. However, with only one out of ten RIPs seized, the core flow decrease is small (~10%), so the event is very mild. The RFCS will sense the core flow change and command the remaining RIPs to increase speeds and thereby automatically mitigate the transient and maintain the core flow. Figure 15.3-5 shows the analysis results of this event. Table 15.3-5 lists the sequence of events for Figure 15.3-5.

This event does not have to be reanalyzed for specific core configurations.

15.3.3.4 Barrier Performance

As shown in Figure 15.3-5, system pressure during this event is not higher than the original values. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.3.5 Radiological Consequences

The consequences of the events identified do not result in any fuel failures or SRV actuation.

15.3.4 Reactor Internal Pump Shaft Break

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a RIP is considered a DBA event. It has been evaluated as a very mild accident in relation to other DBAs such as the LOCA (Refer to Chapter 5 for specific mechanical considerations and Chapter 7 for electrical aspects.).

This postulated event is bounded by the more limiting case of RIP seizure. Quantitative results for this more limiting case are presented in Subsection 15.3.3.

15.3.4.1.1 Identification of Causes

The case of RIP shaft breakage represents the extremely unlikely event of instantaneous stoppage of the pump motor operation of one reactor internal pump. This event produces a very rapid decrease of pump flow as a result of the break of the pump shaft. Consequently, it results in a small decrease in core inlet flow and core cooling capability.

15.3.4.1.2 Frequency Classification

This event is considered a limiting fault but results in effects which can easily satisfy an event of greater probability (i.e., infrequent incident classification).

15.3.4.2 Sequence of Events and Systems Operations

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the shaft of one RIP (Subsection 15.3.4.1.1) causes the core flow to decrease rapidly. The sequence of events is the same as that presented in Subsection 15.3.3.2.1.

15.3.4.2.1.1 Identification of Operator Actions

Same as Subsection 15.3.3.2.1.1.

15.3.4.2.2 Systems Operation

Same as Subsection 15.3.3.2.2.

15.3.4.3 Core and System Performance

The severity of this pump shaft break event is bounded by the pump seizure event (Subsection 15.3.3). This can be demonstrated easily by consideration of these two events. In either of these two events, the recirculation drive flow of the affected pump decreases rapidly. In the case of the pump seizure event, the pump flow decreases faster than the normal flow coastdown as a result of the large hydraulic resistance introduced by the stopped rotor. For the pump shaft break event, the hydraulic resistance caused by the broken pump shaft is less than that of the stopped rotor for the pump seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump seizure event. Thus, it can be concluded that the potential effects of the hypothetical pump shaft break accident are bounded by the effects of the pump seizure event and this event does not have to be reanalyzed for specific core configurations.

15.3.4.4 Barrier Performance

The reactor coolant pressure boundary is not threatened by overpressure.

15.3.4.5 Radiological Consequences

The consequences of this event do not result in any fuel failures or SRV actuation.

15.3.5 References

15.3-1 R. Van Houten, “Fuel Rod Failure as a Consequence of Departure from Nucleate Boiling or Dryout”, NUREG-0562, June 1979.

15.3-2 M. Limbäck and R. Ianiri, “Dryout Fuel Behaviour Tests in IFA-613.1”, Halden Reactor Project report HWR-493, February 1997.

15.3-3 M. Limbäck and R. Ianiri, “Dryout Fuel Behaviour Tests in IFA-613.2”, Halden Reactor Project report HWR-494, February 1997.

15.3-4 R. Ianiri, “The Third Dryout Fuel Behavior Test Series in IFA-613”, Halden Reactor Project report HWR-552, February 1998.

Table 15.3-1 Sequence of Events for Figure 15.3-1 (Three Pump Trip)

Time (s)	Event
0.00	Simulation starts
1.00	Three RIPs are tripped
1.46	Max steam dome pressure
12.9	Max neutron flux
~30	The reactor settles in a new steady state

Table 15.3-2 Sequence of Events for Figure 15.3-2 (All Pump Trip)

Time (s)	Event
0.00	Simulation starts
1.00	Trip of all RIP initiated
1.47	Max steam dome pressure
2.82	Reactor scram initiated by Rapid Core Flow Coastdown
2.82	Feedwater pump trip on scram signal
3.07	Control rods start to move
6.59	Control rods fully inserted
20.0	Vessel water level (L2) setpoint reached. RCIC started..
~50 (est.)	RCIC flow enters vessel (not simulated).

Table 15.3-3 Sequence of Events for Figure 15.3-3 (Fast Runback of One RIP)

Time (s)	Event
0.00	Simulation starts
1.00	Fast runback of one RIP initiated
2.36	The runback RIP reaches minimum speed
~30	Reactor power settles in a new steady state condition

Table 15.3-4 Sequence of Events for Figure 15.3-4 (Fast Runback of All RIPs)

Time (s)	Event
0.00	Simulation starts
1.00	Fast runback of all RIPs initiated
11.4	The RIP reach their minimum speed
~55	The reactor power settles in a new steady state condition

Table 15.3-5 Sequence of Events for Figure 15.3-5 (One RIP Seizure)

Time (s)	Event
0.00	Simulation starts
0.00	Seizure of one reactor internal pump
5.66	Max neutron flux
9.07	Max steam dome pressure
~30	Reactor settles in a new steady state

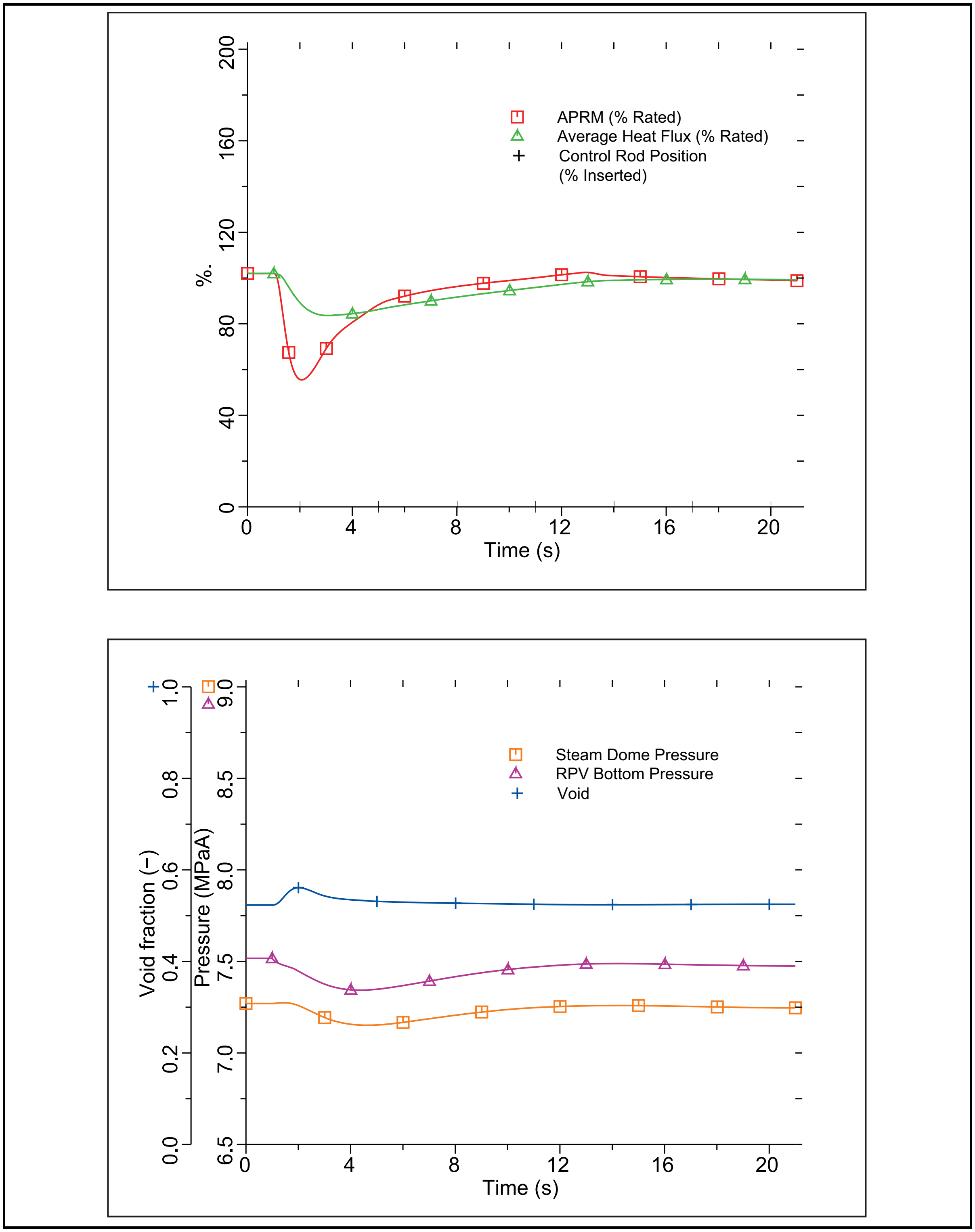


Figure 15.3-1a Three Pump Trip

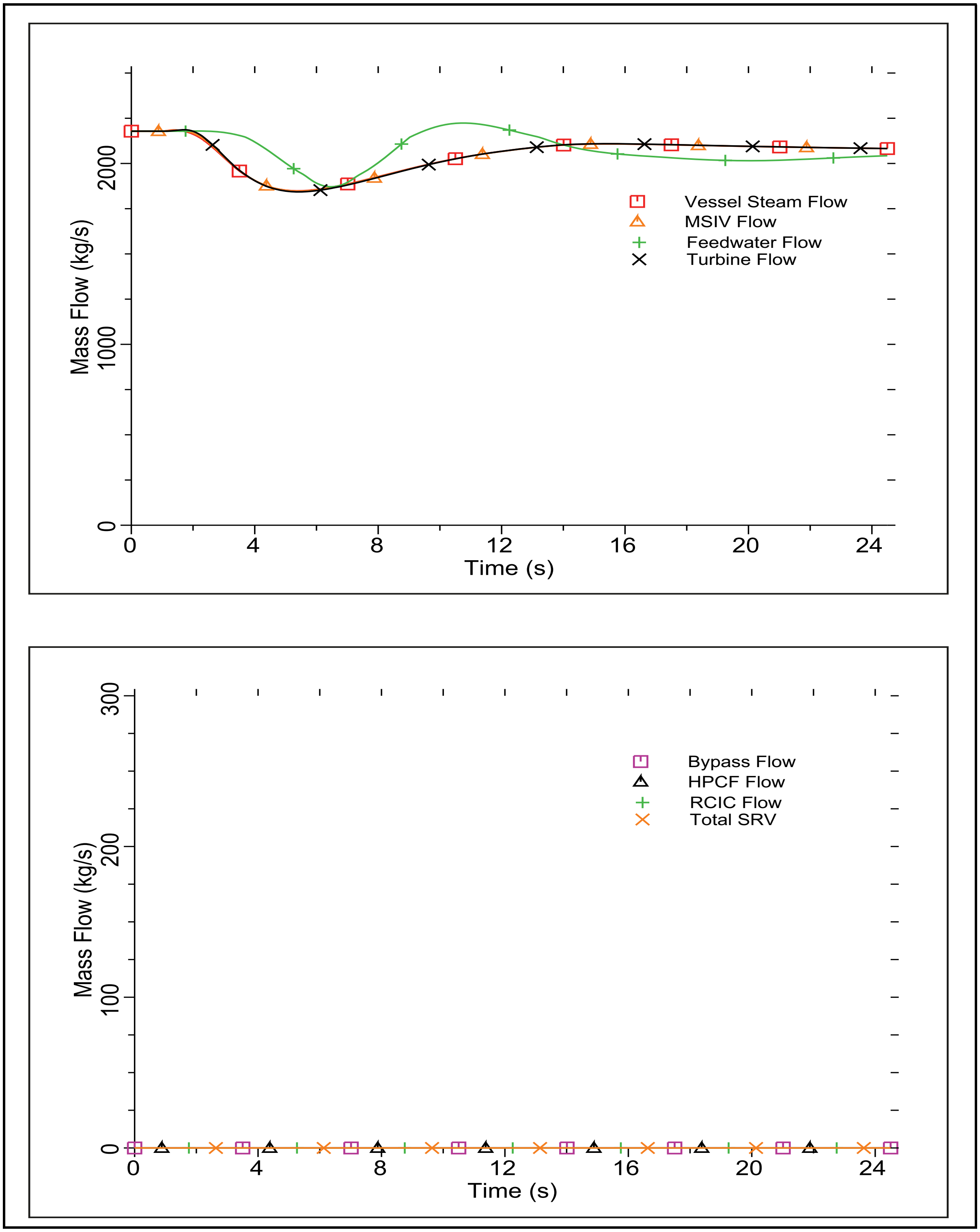


Figure 15.3-1b Three Pump Trip

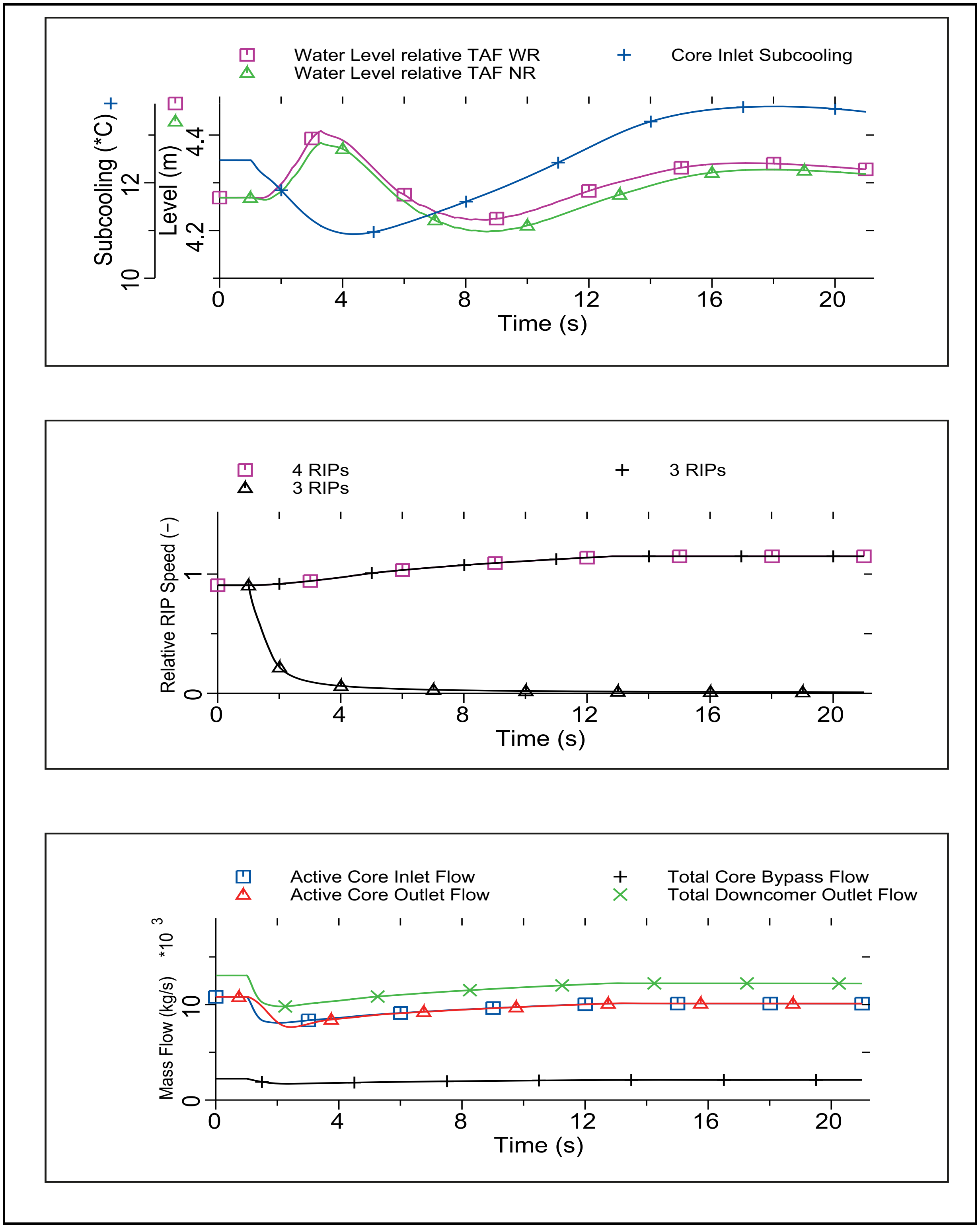


Figure 15.3-1c Three Pump Trip

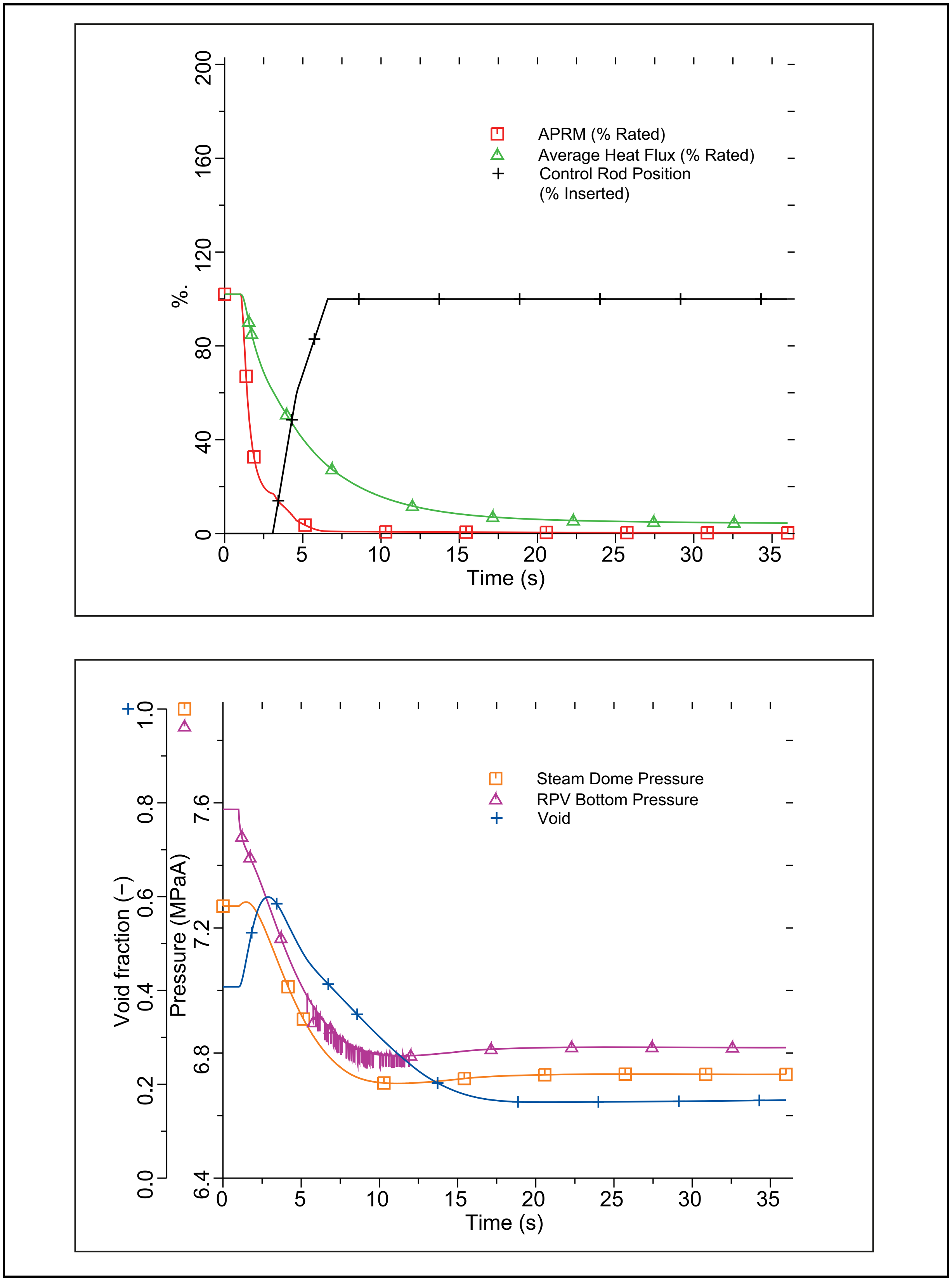


Figure 15.3-2a All Pump Trip

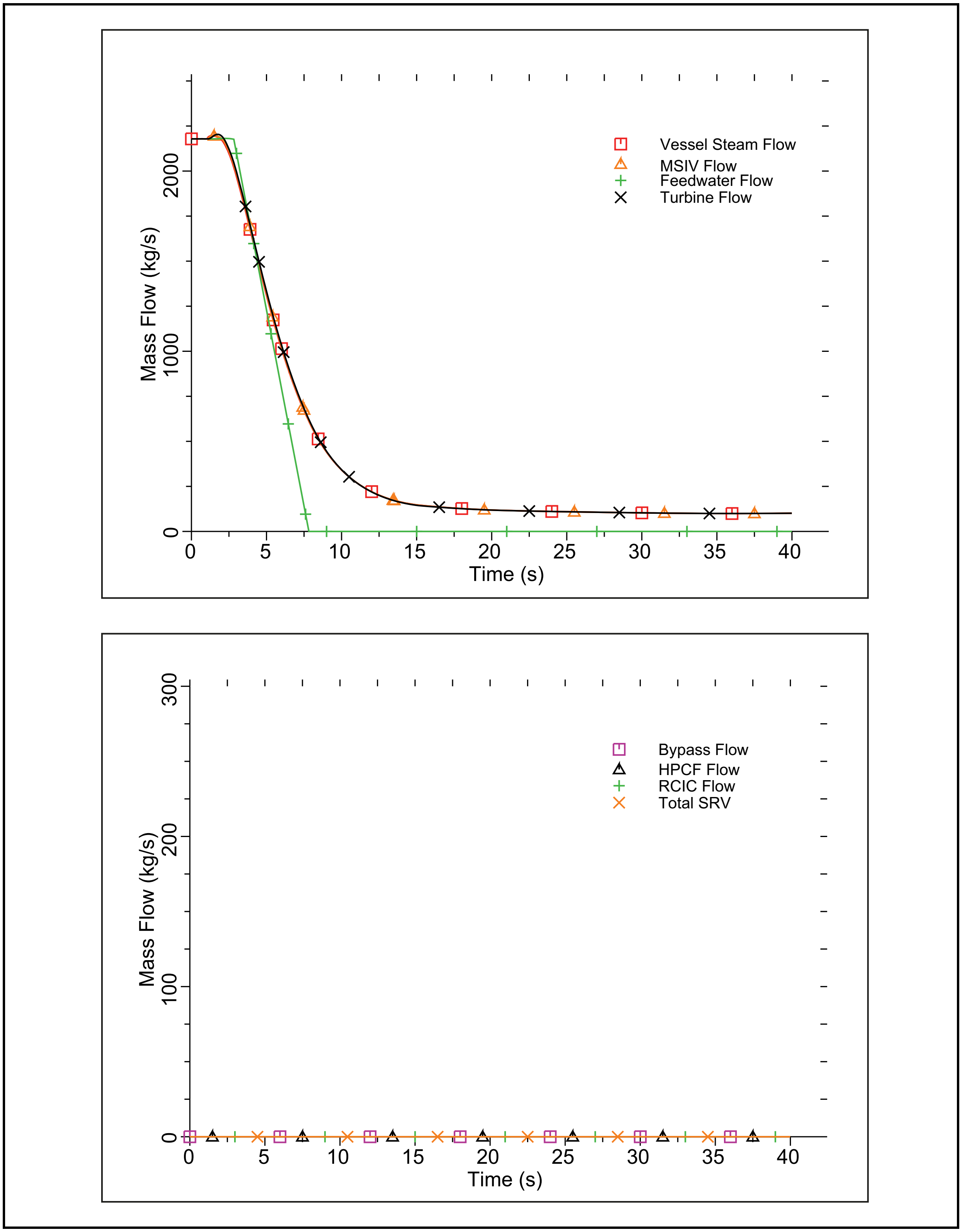


Figure 15.3-2b All Pump Trip

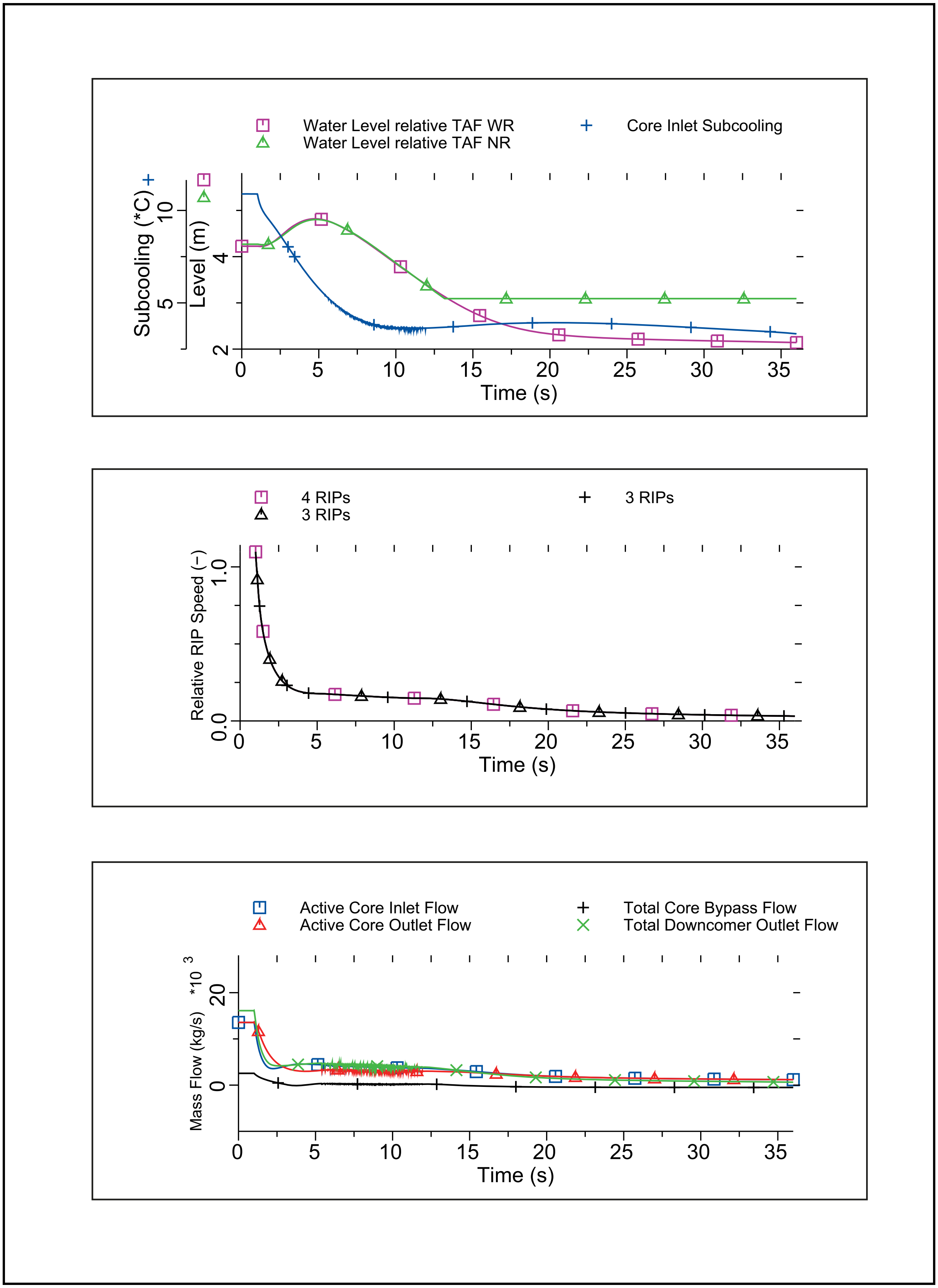


Figure 15.3-2c All Pump Trip

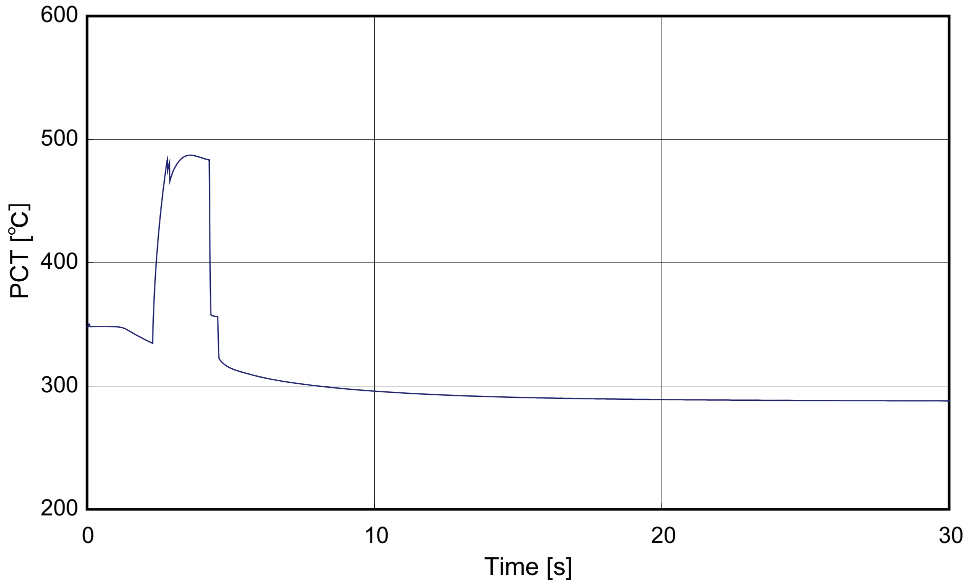


Figure 15.3-2d Cladding Temperature During All Pump Trip

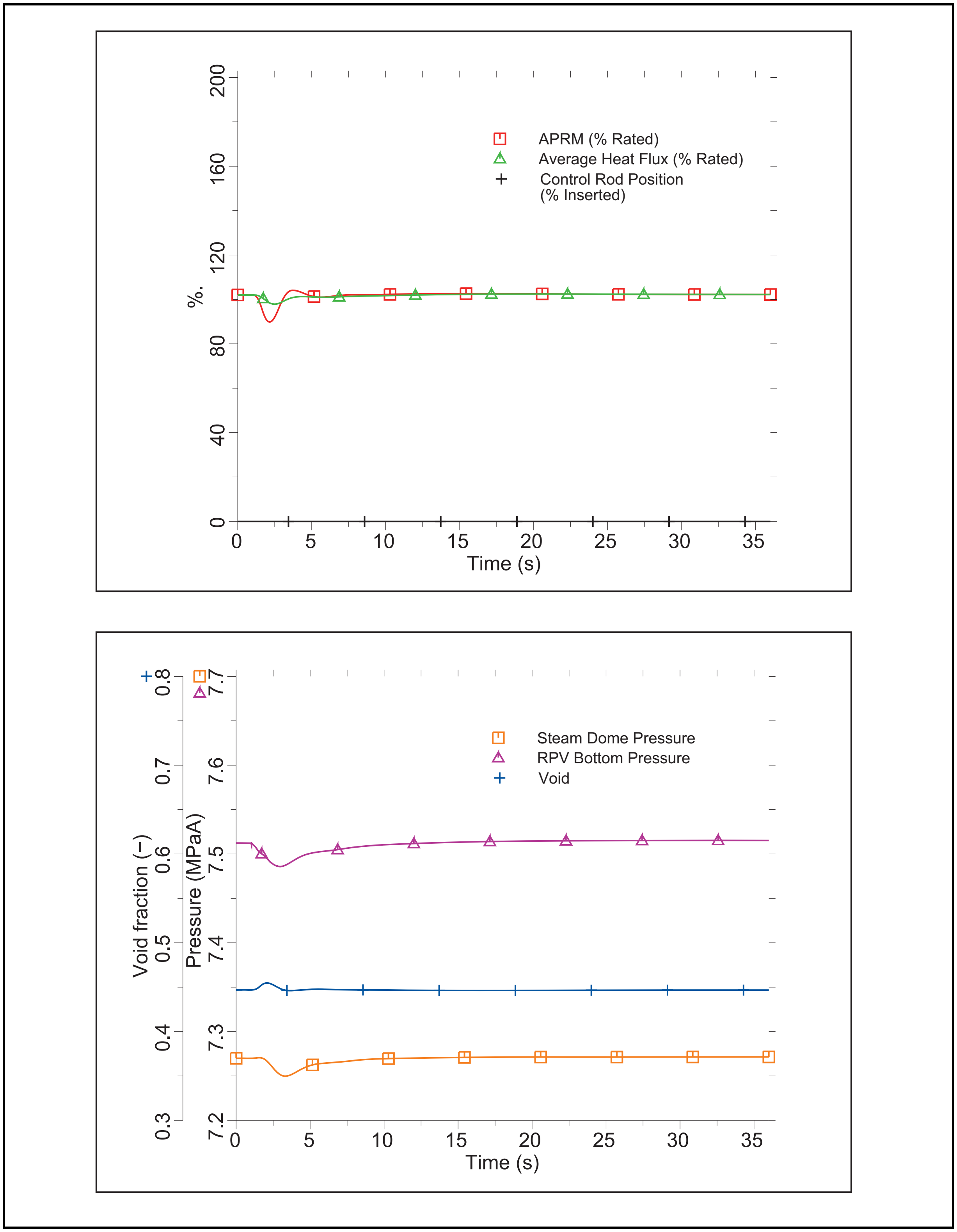
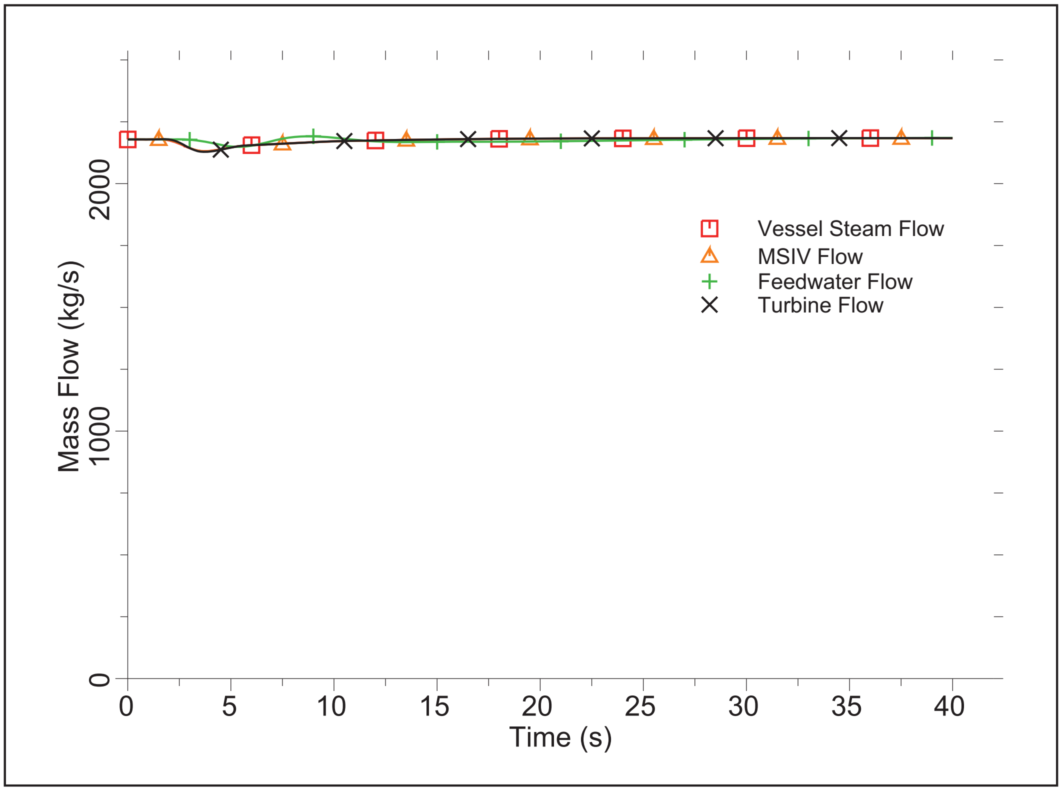


Figure 15.3-3a Fast Runback of One RIP



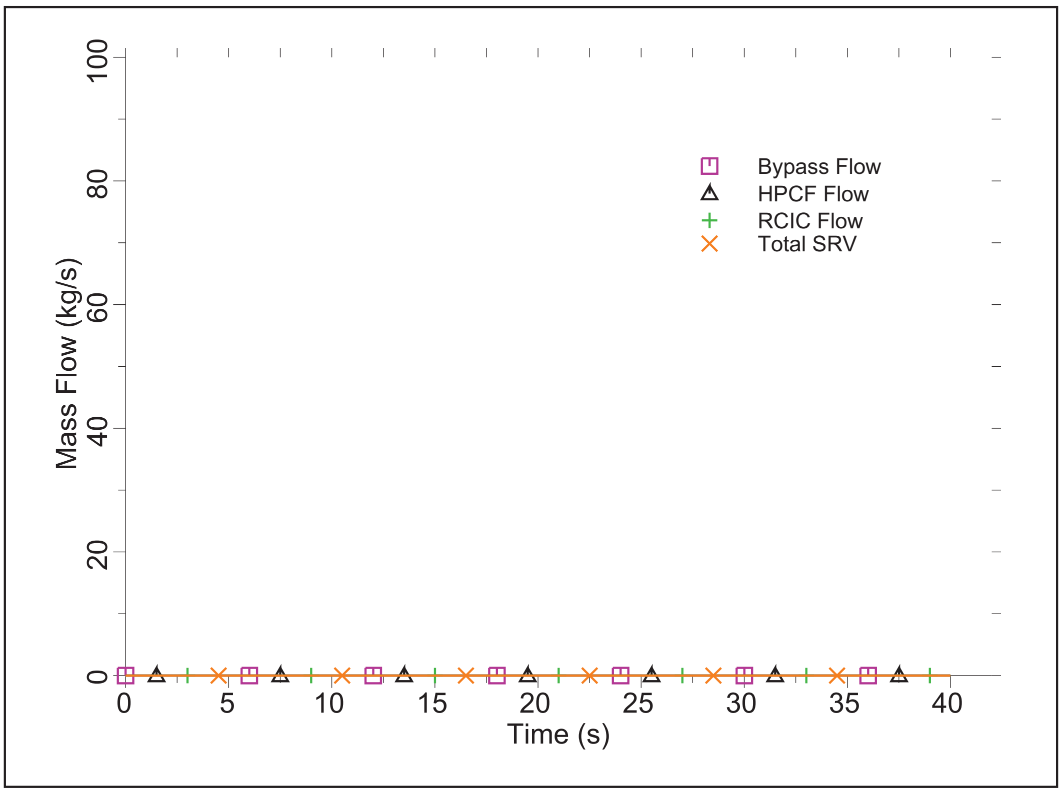


Figure 15.3-3b Fast Runback of One RIP

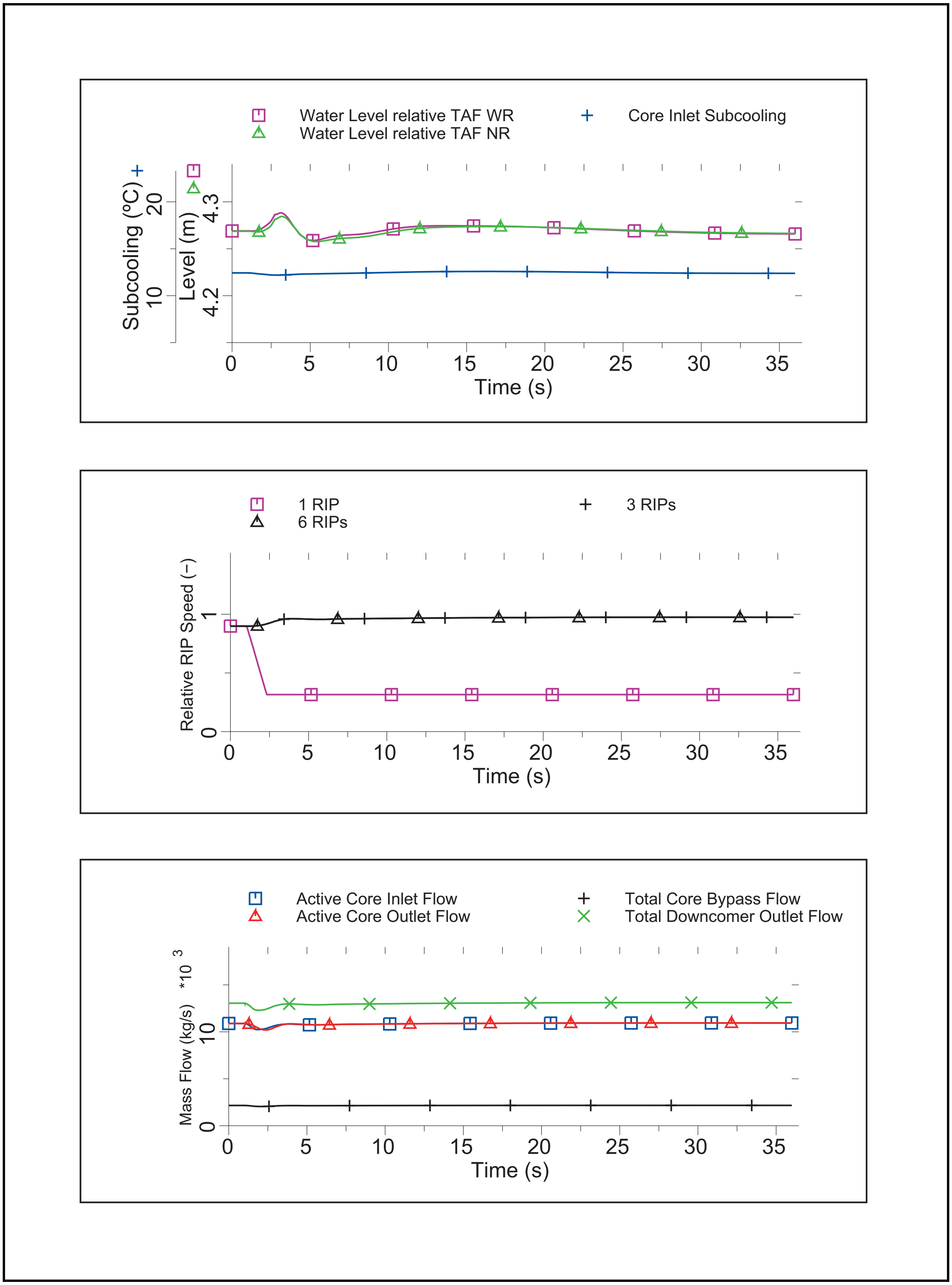


Figure 15.3-3c Fast Runback of One RIP

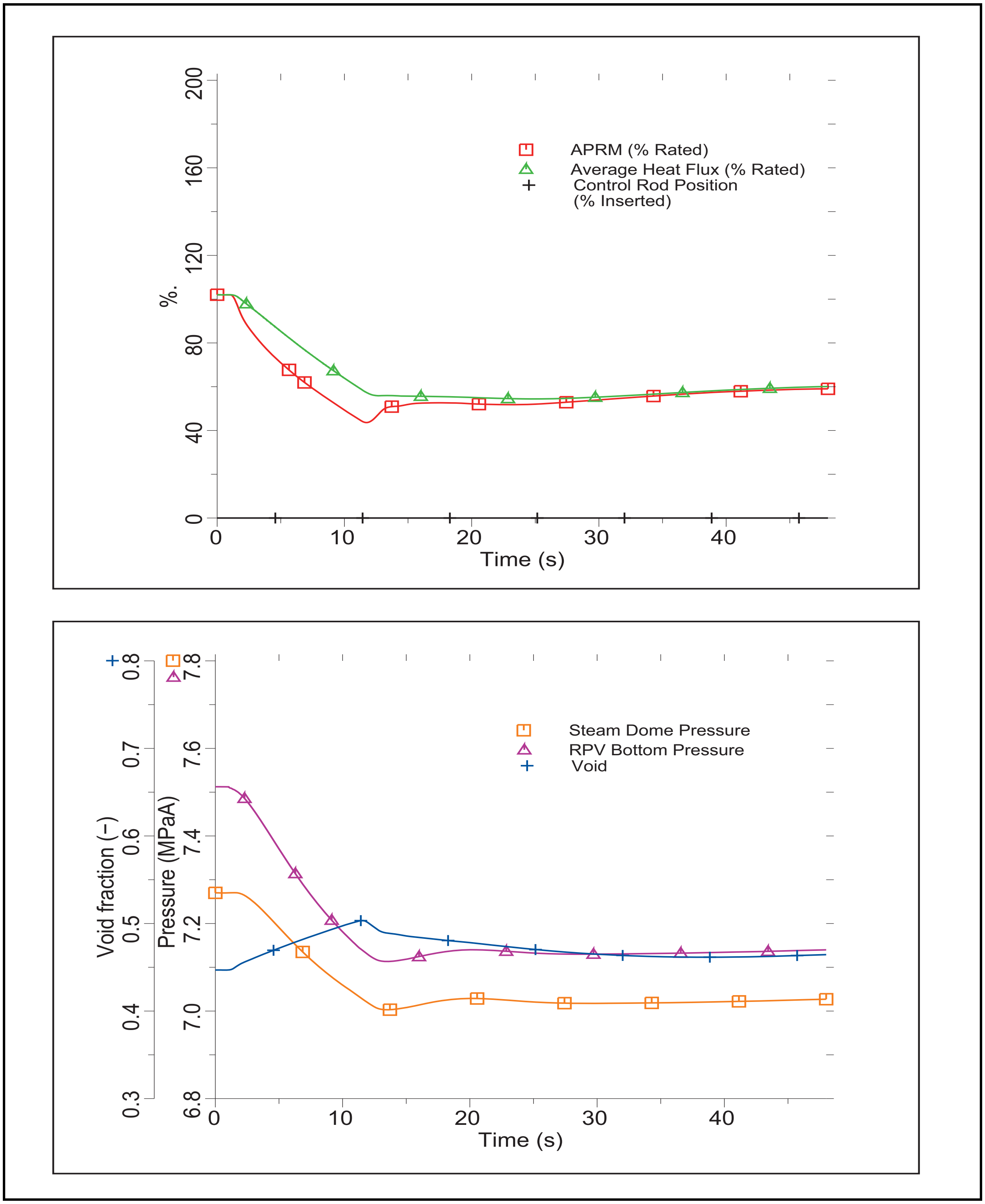


Figure 15.3-4a Fast Runback of All RIPs

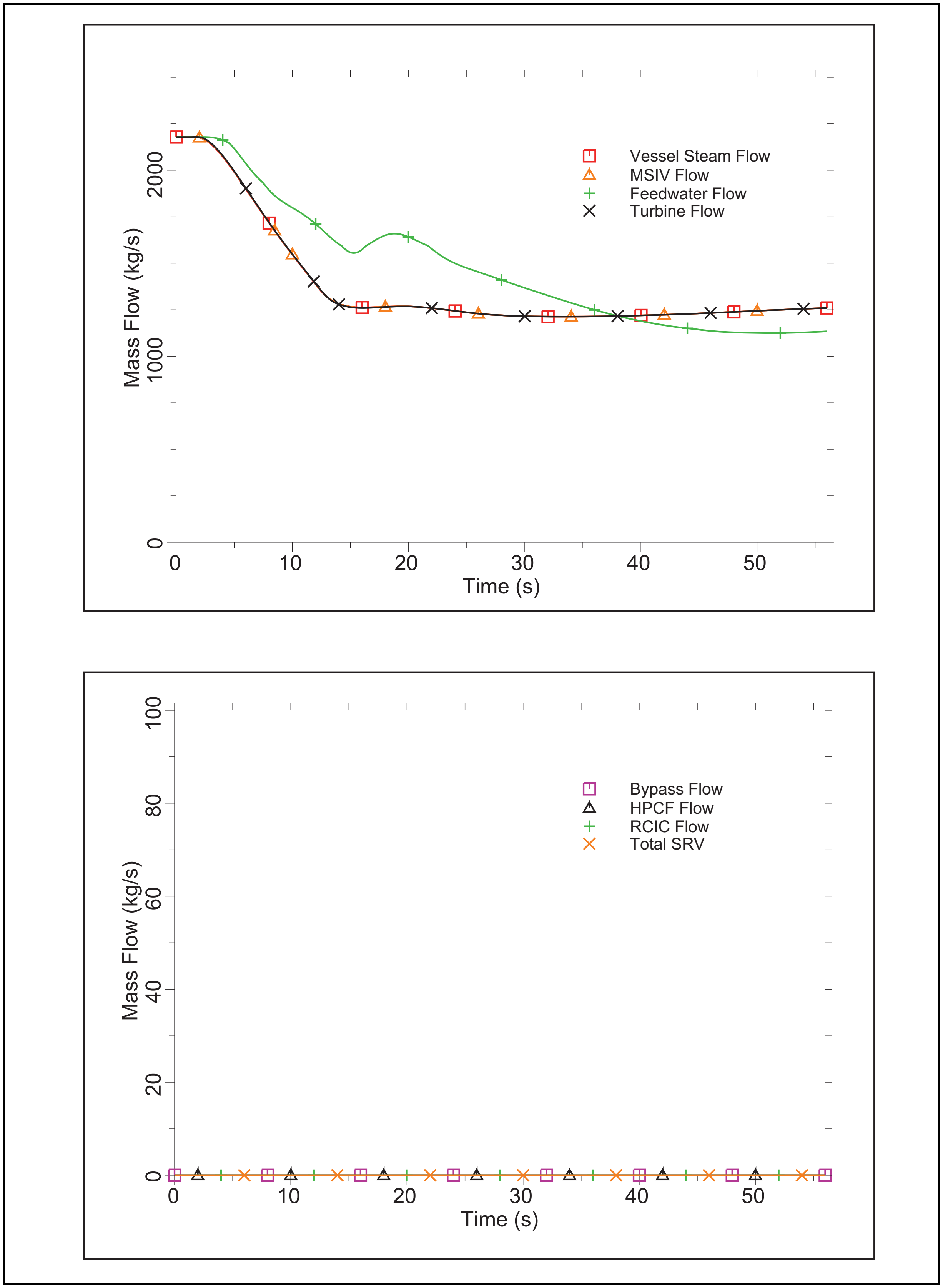


Figure 15.3-4b Fast Runback of All RIPs

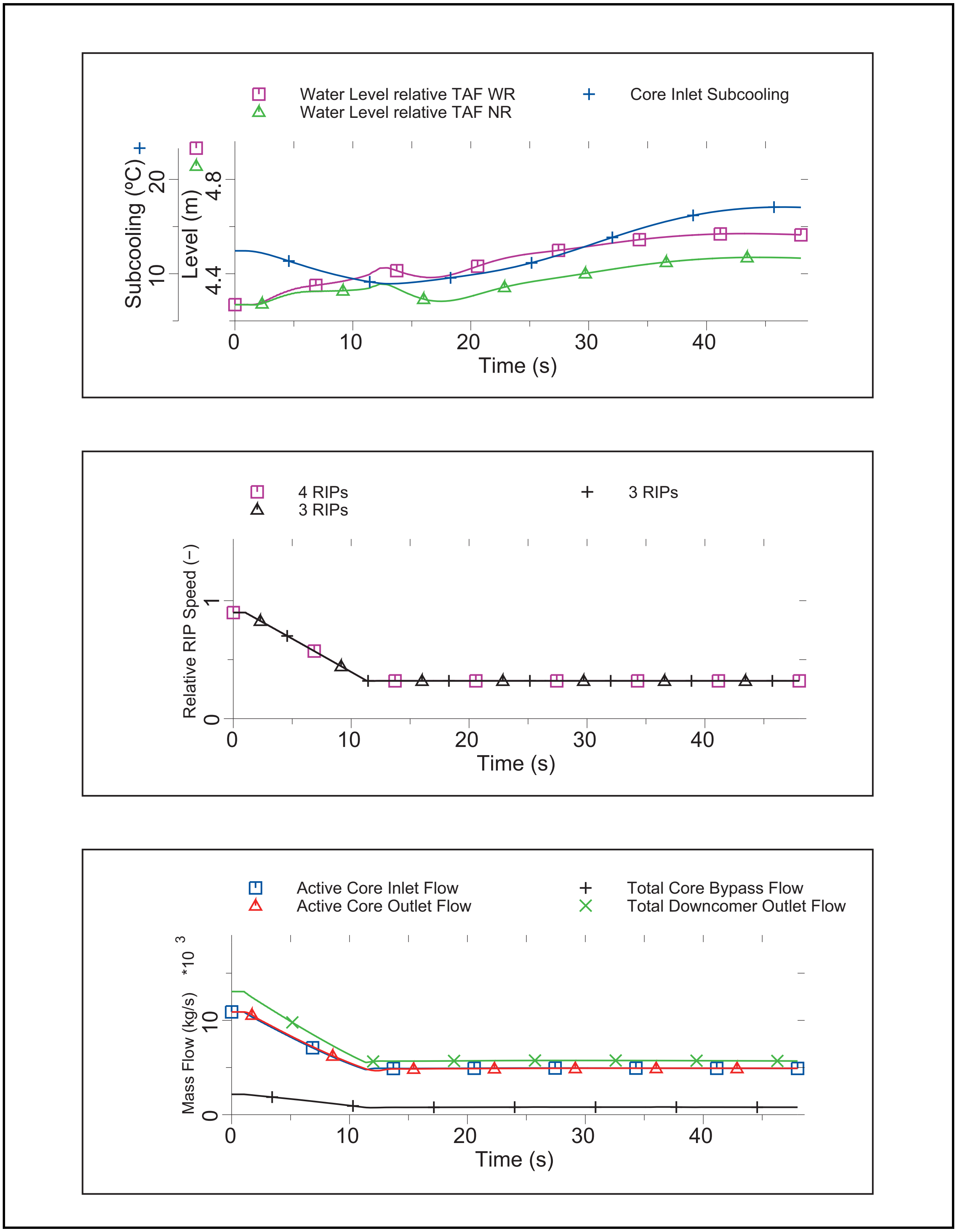


Figure 15.3-4c Fast Runback of All RIPs

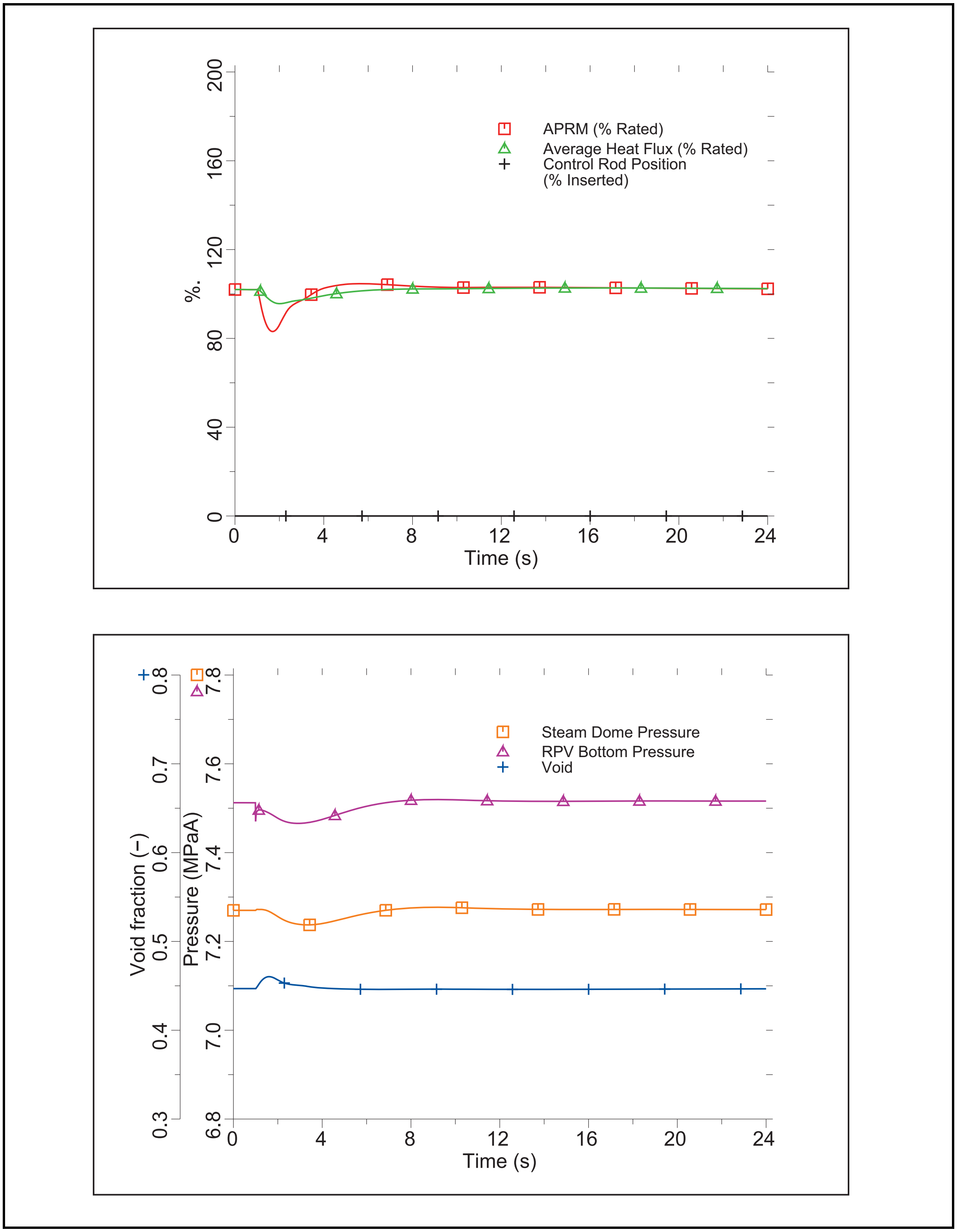


Figure 15.3-5a One RIP Seizure

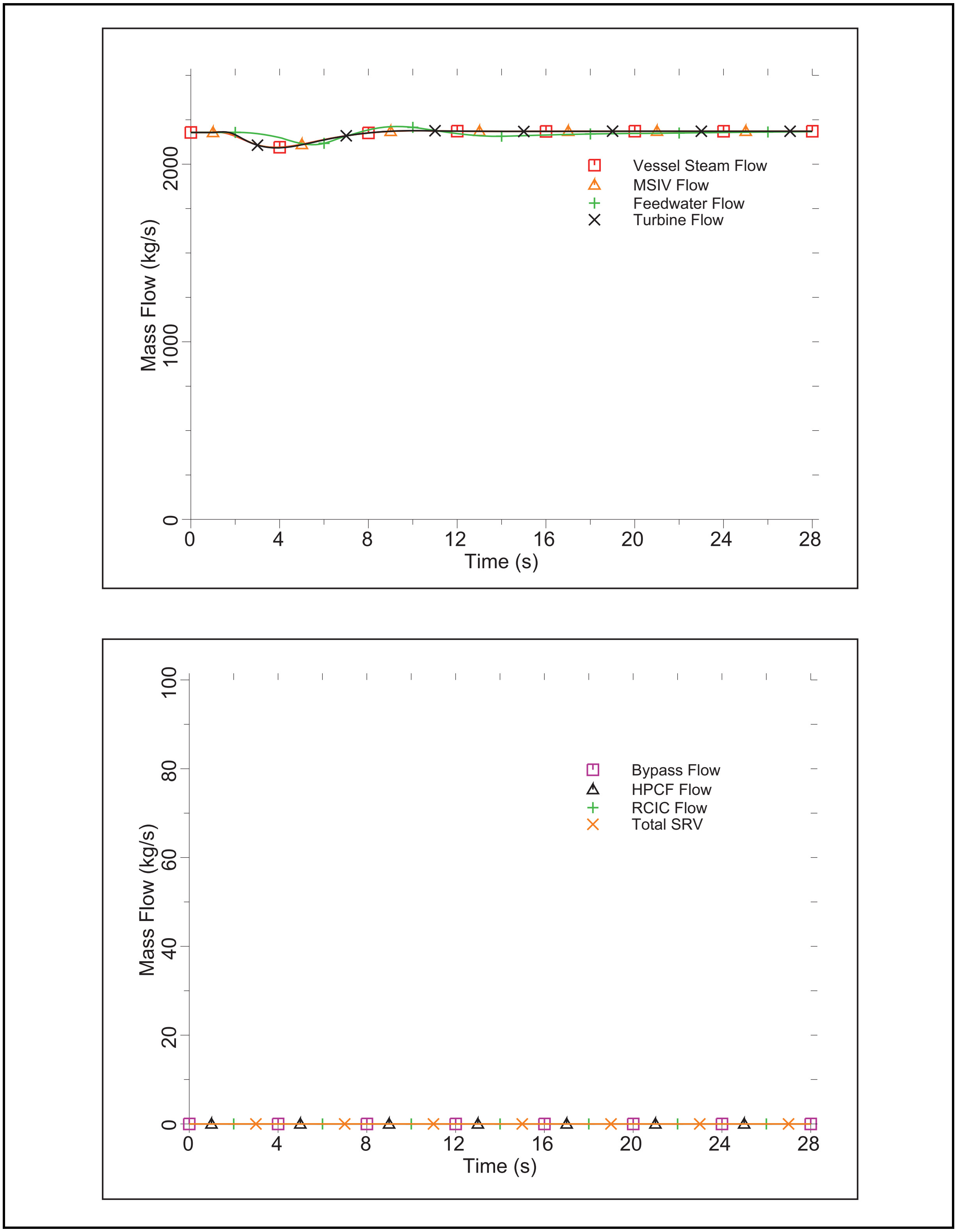


Figure 15.3-5b One RIP Seizure

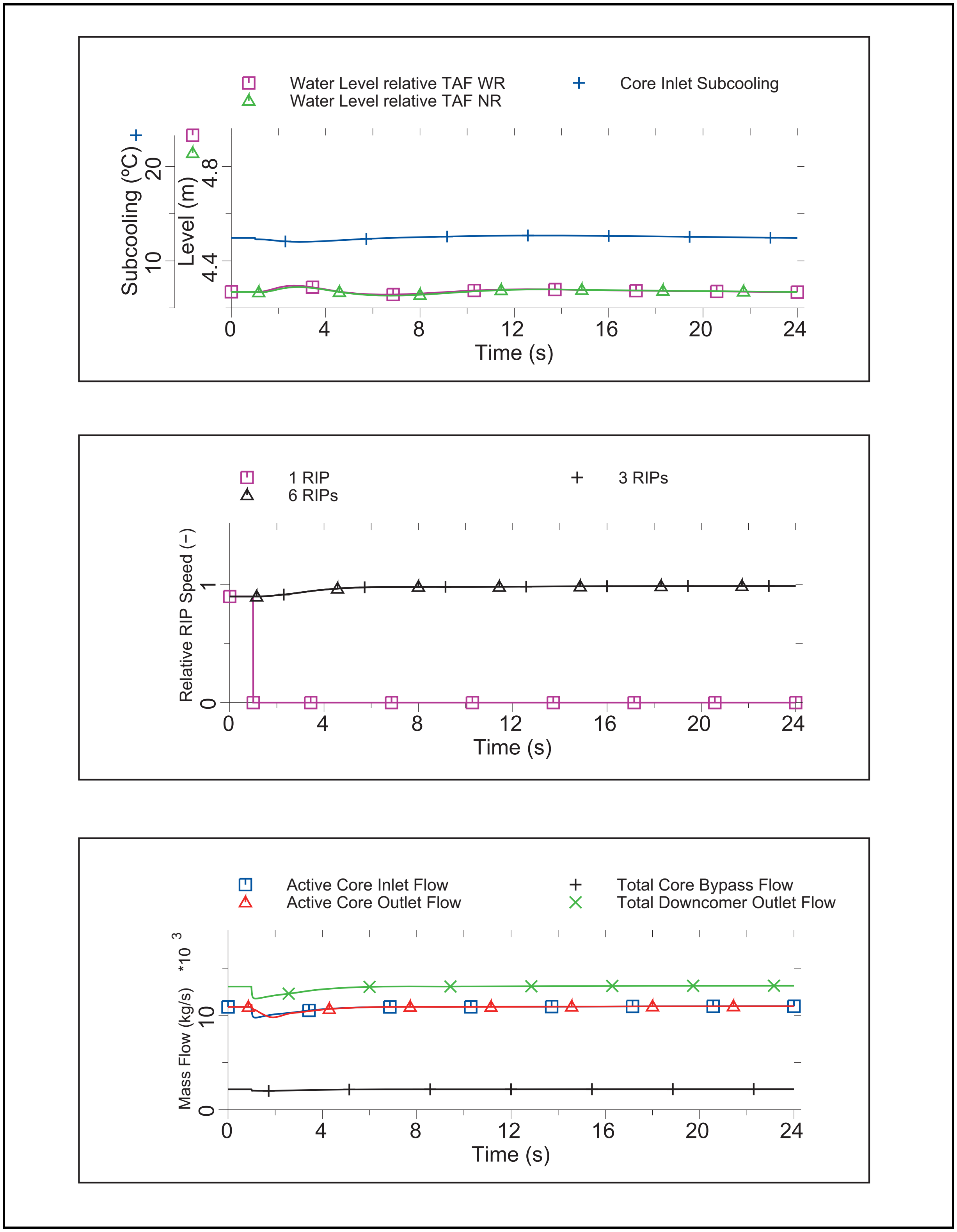


Figure 15.3-5c One RIP Seizure