15.1 Decrease in Reactor Coolant Temperature

15.1.1 Loss of Feedwater Heating

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

A feedwater heater can be lost in at least two ways:

(1) Steam extraction line to heater is closed.

(2) Feedwater is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the feedwater bypasses the heater and no heating of that feedwater occurs. In either case, the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations.

The ABWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C feedwater heating. The reference steam and power conversion system shown in Figures 10.1-1 to 10.1-3 meets this requirement. In fact, the feedwater temperature drop based on the reference heat balance (Figure 10.1-1) is less than 30°C . Therefore, the use of 55.6°C temperature drop in the transient analysis is conservative.

This event has been conservatively estimated to incur a loss of up to 55.6°C of the feedwater heating capability of the plant and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) includes a logic intended to mitigate the consequences of a loss of feedwater heating capability. The system will be constantly monitoring the actual feedwater temperature and comparing it with a reference temperature. When a loss of feedwater heating is detected (i.e., when the difference between the actual and reference temperatures exceeds a Δ T setpoint, which is currently set at 16.7°C), the FWCS sends an alarm to the operator. The operator can then take actions to mitigate the event. This will avoid a scram and reduce the Δ CPR during the event. The same signal is also sent to the RCIS to initiate the SCRRI (selected control rods run-in) to automatically reduce the reactor power and avoid a scram. This will prevent the reactor from violating any thermal limits.

Because this event is very slow, the operator action or automatic SCRRI will terminate this event. Therefore, the worst event is the loss of feedwater heating resulting in a temperature difference just below the ΔT setpoint. However, a loss of 55.6°C feedwater temperature is analyzed to bound this event.

15.1.1.1.2 Frequency Classification

The probability of this event is considered low enough to warrant it being categorized as an infrequent incident. However, because of the lack of a sufficient frequency database, this transient disturbance is analyzed as an incident of moderate frequency.

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

Table 15.1-1 lists the sequence of events for this transient.

15.1.1.2.1.1 Identification of Operator Actions

Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The high simulated thermal power trip (STPT) scram is the primary protection system trip in mitigating the consequences of this event. However, the power increase in this event is not high enough to initiate this scram. Operation of engineered safety features (ESF) is not expected for this transient.

15.1.1.3 Core and System Performance

15.1.1.3.1 Input Parameters and Initial Conditions

The transient is simulated by programming a change in feedwater enthalpy corresponding to a 55.6°C loss in feedwater heating. Another case with the Δ T setpoint in FWCS of 16.7°C is also analyzed.

15.1.1.3.2 Results

Because the power increase during this event is relatively slow, it can be treated as a quasi steady-state transient. The 3-D core simulator, POLCA, has been used to evaluate this event for the equilibrium cycle. The results are summarized in Tables 15.1-2 and 15.1-2a.

The MCPR response of this event is small due to the mild thermal power increase with shifting axial shape. The worst Δ CPR response is 0.09, and occurs at 102% power, 90% flow BOC conditions.

No scram is initiated in this event. The increased core inlet subcooling aids thermal margins. Nuclear system pressure does not change significantly and, consequently, the RCPB is not threatened.

15.1.1.4 Barrier Performance

As noted previously, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.1.5 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.

15.1.2 Feedwater Controller Failure—Maximum Demand

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow.

The ABWR FWCS uses a triplicated digital control system, instead of a single-channel analog system, as used in current BWR designs (BWR 2-6). The digital systems consist of a triplicated fault-tolerant digital controller, the operator control stations and displays. The digital controller contains three parallel processing channels, each containing the microprocessor-based hardware and associated software necessary to perform all the control calculations. The operator interface provides information regarding system status and the required control functions.

Redundant transmitters are provided for key process inputs, and input voting and validation are provided such that faults can be identified and isolated. Each system input is triplicated internally and sent to the three processing channels (Figure 15.1-1). The channels will produce the same output during normal operation. Interprocessor communication provides self-diagnostic capability. A two-out-of-three voter compares the processor outputs to generate a validated output to the control actuator. A separate voter is provided for each actuator. A “ringback” feature feeds back the final voter output to the processors. A voter failure will thereby be detected and alarmed. In some cases, a protection circuit will lock the actuator into its existing position promptly after the failure is detected.

Table 15.1-3 lists the failure modes of a triplicated digital control system and outlines the effects of each failure. Because of the triplicated architecture, it is possible to take one channel out of service for maintenance or repair while the system is online. Modes 2 and 5 of Table

15.1-3 address a failure of a component while an associated redundant component is out of service. This type of failure could potentially cause a system failure. However, the probability of a component failure during servicing of a counterpart component is considered to be so low that these failure modes will not be considered incidents of moderate frequency, but, rather, limiting faults.

Adverse effects minimization is mentioned in the effects of Mode 2. This feature stems from the additional intelligence of the system provided by the microprocessor. When possible, the system will be programmed to take action in the event of some failure which will reduce the severity of the transient. For example, if the total steam flow or total feedwater flow signals fail, the FWCS will detect this by the input reasonability checks and automatically switch to one-element mode (i.e., control by level feedback only). The level control would essentially be unaffected by this failure.

The only credible single failures which would lead to some adverse effect on the plant are Modes 6 (failure of the output voter) and 7 (control actuator failure). Both of these failures would lead to a loss of control of only one actuator (i.e., only one feedwater pump with increasing flow). A voter failure is detected by the ringback feature. The FWCS will initiate a lockup of the actuator upon detection of the failure. The probabilities of failure of the variety of control actuators are very low based on operating experience. In the event of one pump run-out, the FWCS would then reduce the demand to the remaining pump(s), thereby automatically compensating for the excessive flow from the failed pump. Therefore, the worst single failure in the FWCS causes a runout of one feedwater pump to its maximum capacity. However, the demand to the remaining feedwater pump(s) will decrease to offset the increased flow of the failed pump. The effect on total flow to the vessel will not be significant. The worst additional single failure would cause all feedwater pumps to run out to their maximum capacity. However, the probability of this to occur is extremely low.

15.1.2.1.2 Frequency Classification

15.1.2.1.2.1 Runout of One Feedwater Pump

Although the frequency of occurrence for this event is very low, this event is conservatively evaluated as an incident of moderate frequency.

15.1.2.1.2.2 Feedwater Controller Failure—Maximum Demand

The frequency of occurrence for this event is estimated to be so low that it should be classified as a limiting fault as specified in Chapter 15 of Regulatory Guide 1.70. Nonetheless, the criteria of moderate frequent incidents are conservatively applied to this event.

15.1.2.2 Sequence of Events and Systems Operation

15.1.2.2.1 Sequence of Events

15.1.2.2.1.1 Runout of One Feedwater Pump

With momentary increase in feedwater flow, the water level rises and then settles back to its normal level. Table 15.1-4 lists the sequencing of events for Figure 15.1-2.

15.1.2.2.1.2 Feedwater Controller Failure—Maximum Demand

With excess feedwater flow, the water level rises to the high-level reference point, at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. Table 15.1-5 lists the sequence of events for Figure 15.1-3. The figure shows the changes in important variables during this transient.

15.1.2.2.1.3 Identification of Operator Actions

15.1.2.2.1.3.1 Runout of One Feedwater Pump

Because no scram occurs for runout of one feedwater pump, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

15.1.2.2.1.3.2 Feedwater Controller Failure—Maximum Demand

The operator should:

(1) Observe that high feedwater pump trip has terminated the failure event

(2) Switch the feedwater controller from auto to manual control to try to regain a correct output signal

(3) Identify causes of the failure and report all key plant parameters during the event

15.1.2.2.2 Systems Operation

15.1.2.2.2.1 Runout of One Feedwater Pump

Runout of a single feedwater pump requires no protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.1.2.2.2.2 Feedwater Controller Failure—Maximum Demand

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. Important system operational actions for this event include (1) high level tripping of the main turbine and feedwater pumps, (2) scram and recirculation pump trip (RPT) of the 4

RIPs not connected to M-G sets due to turbine trip, and (3) low water level initiation of the RCIC System to maintain long-term water level control following tripping of feedwater pumps.

15.1.2.3 Core and System Performance

15.1.2.3.1 Input Parameters and Initial Conditions

The runout capacity of one feedwater pump is assumed to be 75% of rated flow at the design pressure of 7.35 MPaG. The total feedwater flow for all pumps runout is assumed to be 130% of rated at the design pressure of 7.35 MPaG.

15.1.2.3.2 Results

15.1.2.3.2.1 Runout of One Feedwater Pump

The simulated runout of one feedwater pump event is presented in Figure 15.1-2. When the increase of feedwater flow is sensed, the feedwater controller starts to command the remaining feedwater pump to reduce its flow immediately. The vessel water level increases slightly (about 20 cm) and then settles back to its normal level. The vessel pressures only increase about 0.01 MPaD. MCPR remains above the safety limit.

15.1.2.3.2.2 Feedwater Controller Failure—Maximum Demand

The simulated runout of all feedwater pumps is shown in Figure 15.1-3. The high water level turbine trip and feedwater pump trip are initiated at approximately 20 seconds after the start of the event. Scram occurs and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. It is calculated that the MCPR is not limiting. Therefore, the design limit for the moderate frequent incident is met. The Turbine Bypass System opens to limit peak pressure in the steamline near the SRVs to 8.04 MPaG and the pressure at the bottom of the vessel to about 8.18 MPaG.

The level will gradually drop to the Low Level reference point (Level 2), activating the RCIC System for long-term level control.

The COL applicant will provide reanalysis of this event for the specific core configuration.

15.1.2.4 Barrier Performance

As previously noted, the consequence of this event does not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.2.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences

identified in Subsection 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.1.3 Pressure Regulator Failure—Open

15.1.3.1 Identification of Causes and Frequency Classifications

15.1.3.1.1 Identification of Causes

The ABWR Steam Bypass and Pressure Control System (SB&PCS) uses a triplicated digital control system instead of an analog system as used in current BWR designs (BWR 2-6). The SB&PCS controls turbine control valves and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.1.2.1.1, no credible single failure in the control system will result in a maximum demand to all actuators for all turbine control valves and bypass valves. A voter or actuator failure may result in an inadvertent opening of one turbine control valve or one turbine bypass valve. In this case, the SB&PCS will sense the pressure change and command the remaining control valves to close, and thereby automatically mitigate the transient and maintain reactor power and pressure.

Because the effect of a sudden opening of one bypass valve, which bypasses about 11% of rated steam flow when full opened, is more severe than the sudden opening of one turbine control valve (which is almost wide open at rated power), it is assumed for purposes of this transient analysis that a single failure causes a single bypass valve to fail open.

As presented in Subsection 15.1.2.1.2, multiple failures might cause the SB&PCS to erroneously issue a maximum demand to all turbine control valves and bypass valves. Should this occur, all turbine control valves and bypass valves could be fully opened. However, the probability of this event is extremely low, and, hence, the event is considered as a limiting fault. However, the criteria of moderate frequency incidents are conservatively applied to this event.

15.1.3.1.2 Frequency Classification

15.1.3.1.2.1 Inadvertent Opening of One Turbine Bypass Valve

This transient disturbance, estimated to occur with very low frequency, is conservatively categorized as one of moderate frequency.

15.1.3.1.2.2 Inadvertent Opening of all Turbine Control Valves and Bypass Valves

The frequency of occurrences for this event is estimated to be extremely low. The event should thus be classified as a limiting fault as specified in Chapter 15 of Regulatory Guide 1.70. Nonetheless, since the consequence of this event has no significant impact on the operating CPR limit, the criteria of moderate frequent incidents are conservatively applied to this event.

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

15.1.3.2.1.1 Inadvertent Opening of One Turbine Bypass Valve

Table 15.1-6 lists the sequence of events for Figure 15.1-4.

15.1.3.2.1.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

Table 15.1-7 lists the sequence of events for Figure 15.1-5.

15.1.3.2.1.3 Identification of Operator Actions

15.1.3.2.1.3.1 Inadvertent Opening of One Turbine Bypass Valves

Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

15.1.3.2.1.3.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet (5.17 MPaG) in the run mode, the following sequence of operator actions is expected during the course of the event. Once isolation occurs, the pressure will increase to a point where the SRVs open. The operator should:

(1) Monitor that all rods are in

(2) Monitor reactor water level and pressure

(3) Observe turbine coastdown and break vacuum before the loss of steam seals. Check turbine auxiliaries

(4) Observe that the reactor pressure relief valves open at their setpoint

(5) Observe that RCIC initiated on low-water level

(6) Secure RCIC when reactor pressure and level are under control

(7) Monitor reactor water level and continue cooldown per the normal procedure

(8) Complete the scram report and initiate a maintenance survey of the SB&PCS before reactor restart

15.1.3.2.2 Systems Operation

15.1.3.2.2.1 Inadvertent Opening of One Turbine Bypass Valve

This event does not require any protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.1.3.2.2.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

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, except as otherwise noted.

Initiation of RCIC System functions occurs when the vessel water level reaches the L2 setpoint. Normal startup and actuation can take up to 30 seconds before effects are realized.

If these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.1.3.3 Core and System Performance

15.1.3.3.1 Input Parameters and Initial Conditions

A three second closure is assumed when the turbine pressure decreases below the turbine inlet low pressure setpoint for main steamline isolation initiation. This is within the specification limits of the valve.

15.1.3.3.2 Results

15.1.3.3.2.1 Inadvertent Opening of One Turbine Bypass Valve

The simulated inadvertent opening of one turbine bypass valve is presented in Figure 15.1-4. When the decrease in reactor pressure is sensed, the pressure control system starts immediately to command turbine control valves to close to maintain the reactor pressure. The vessel water level increases slightly (about 8 cm) and then settles back to its normal level. Reactor pressure decreases by about 0.07 MPaD. MCPR remains above the safety limit.

15.1.3.3.2.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

Figure 15.1-5 presents graphically how the isolation valve closure stops vessel depressurization and produces a normal shutdown of the isolated reactor.

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. The depressurization rate is large and causes the water level to swell.

The pressure continues to drop until turbine inlet pressure is below the low turbine pressure isolation setpoint when main steamline isolation and subsequent reactor scram finally terminates the depressurization. No significant reduction in fuel thermal margins occur; therefore, this event does not have to be analyzed for specific core configurations.

15.1.3.4 Barrier Performance

Barrier performance analyses were not required because the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed. During the event of inadvertent opening of all turbine control and bypass valves, peak pressure in the bottom of the vessel reaches 8.0 MPaG, which is below the ASME code limit of 9.48 MPaG for the reactor coolant pressure boundary. Vessel dome pressure reaches 7.91 MPaG.

15.1.3.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.1.4 Inadvertent Safety/Relief Valve Opening

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent opening is attributed to malfunction of the valve or an operator initiated opening. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

15.1.4.1.2 Frequency Classification

This transient disturbance should be categorized as an infrequent incident, however, the criteria for moderate frequency events are conservatively applied.

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-8 lists the sequence of events for this event.

15.1.4.2.1.1 Identification of Operator Actions

The plant operator must reclose the valve as soon as possible and check that reactor and T-G output return to normal. If the valve cannot be closed, plant shutdown should be initiated.

15.1.4.2.2 Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

15.1.4.3 Core and System Performance

The opening of one SRV allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

The SB&PCS senses the nuclear system pressure decrease and within a few seconds closes the turbine control valves far enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. MCPR is essentially unchanged and, therefore, the safety limit margin is unaffected and this event does not have to be reanalyzed for specific core configurations.

The discharge of steam to the suppression pool causes the temperature of the suppression pool to increase. When the pool temperature reaches the high temperature setpoint, the suppression pool cooling function of the RHR System is automatically initiated. The pool temperature continues to increase due to the mismatch of cooling capacity and steam discharged into the pool. When the pool temperature reaches the next setpoint of 43.3°C, a reactor scram is automatically initiated. In this analysis a conservative scram set point of 48.9°C was assumed.

15.1.4.4 Barrier Performance

As presented previously, the transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following and therefore has no significant effect on RCPB and containment design pressure limits.

15.1.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with the established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.5 Spectrum of Steam System Piping Failures Inside and Outside Containment in a PWR

This event is not applicable to BWR plants.

15.1.6 Inadvertent RHR Shutdown Cooling Operation

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

At design power conditions, no conceivable malfunction in the shutdown cooling system could cause a temperature reduction.

In startup or cooldown operation, if the reactor were critical or near critical, a very slow increase in reactor power could result, depending on the low temperature moderator coefficient of the fuel. A shutdown cooling malfunction leading to a moderate temperature decrease could result from misoperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease could cause a slow insertion of positive reactivity into the core, depending on the fuel design. If the operator did not act to control the power level, a high neutron flux reactor scram would terminate the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.1.6.1.2 Frequency Classification

Because no single failure could cause this event, it should be categorized as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for RHR heat exchangers. The resulting temperature decrease may cause a slow insertion of positive reactivity into the core. Scram occurs before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15.1-9.

15.1.6.2.2 System Operation

A shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered while at power operation because the nuclear system pressure is too high to permit operation of the Shutdown Cooling Mode (SDC) of the RHRs.

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers. In startup or cooldown operation, where the reactor is at or near critical, the slow power increase that may result from the cooler moderator temperature is controlled by the operator in the same manner normally used to control power in the startup range.

15.1.6.3 Core and System Performance

The increased subcooling caused by misoperation of the RHR SDC mode could result in a slow power increase due to a reactivity insertion. This power rise is terminated by a flux scram before fuel thermal limits are approached. Therefore, only qualitative description is provided here and this event does not have to be analyzed for specific core configuration.

15.1.6.4 Barrier Performance

As previously presented, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.6.5 Radiological Consequences

Because this event does not result in any fuel failures, no analysis of radiological consequences is required for this event.

Table 15.1-1 Sequence of Events for Loss of Feedwater Heating

Time (s)	Event
0	Initiate a 55.6°C (or 16.7°C) temperature reduction in the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level.
100 (est.)	Reactor variables settle into new steady state.

Table 15.1-2 Loss of 55.6°C Feedwater Heating

	BOC* to EOC*
Change in Core Power (%)	13.0
Change in MCPR	0.09

* BOC = Beginning of Cycle
EOC = End of Cycle

Table 15.1-2a Loss of 16.7°C Feedwater Heating

	BOC * to EOC*
Change in Core Power (%)	3.6
Change in MCPR	0.03

* BOC = Beginning of Cycle
EOC = End of Cycle

Table 15.1-3 Single Failure Modes for Digital Controls

Modes	Description	Effects
1.	Critical input failure	None—Redundant transmitter takes over—Operator informed of failure
2.	Input failure while one sensor out of service	Possible system failure. Adverse effects minimized when possible
3.	Operator switch single contact failure	None—Triplicated contacts
4.	Processor channel failure	None—Redundant processors maintain control; Operator informed of failure
5.	Processor failure while one channel out of service	System failure
6.	Voter failure	Loss of control of one actuator (i.e., one feedwater pump only). FWCS will lock up actuators.
7.	Actuator failure	Loss of one actuator (i.e., one feedwater pump only)

Table 15.1-4 Sequence of Events for Figure 15.1-2 (Runout of One Feedwater Pump)

Time (s)	Events
0.00	Simulation starts
1.00	Runout of one feedwater pump is initiated
1.27	The flow from the other feedwater pump(s) starts to reduce
2.98	Max steam dome pressure
14.2	Max neutron flux
~20	Vessel water level reaches its peak value and starts to return to its normal value
~60	The reactor settles at a new steady state

Table 15.1-5 Sequence of Events for Figure 15.1-3 (Feedwater Controller Failure –Maximum Demand)

Time (s)	Event
0.00	Simulation starts
1.00	Feedwater controller failure – maximum demand is initiated
20.3	High level L8 reached
20.3	TSV closure initiated by high level L8
20.3	Feedwater pump trip initiated by high level L8
20.4	Reactor scram initiated by TSV position
20.5	TSVs closed
20.5	RPT of 4 RIPs initiated by TSV position
20.5	Main turbine bypass valves start to open
20.6	Control rods start to move
20.8	Max neutron flux
22.1	Relief opening of SRVs according to their pressure setpoints
22.9	Max steam dome pressure
24.1	Control rods fully inserted
53.9	RPT of 6 RIPs initiated by low water level L2; RCIC start

Table 15.1-6 Sequence of Events for Figure 15.1-4 (Inadvertent Opening of One Bypass Valve)

Time (s)	Events
0.00	Simulation starts
1.00	One TBV open and bypass 11% of the rated steam flow
1.04	TCVs start to adjust their position to reduce steam flow from the reactor
~30	Reactor settles at a new steady state

Table 15.1-7 Sequence of Events for Figure 15.1-5 (Opening of All Control and Bypass Valves)

Time (s)	Events
0.00	Simulation starts
1.00	Opening of all control and bypass valves is initiated
4.95	Low pressure reached in the main steam lines, at turbine inlet. MSIV closure initiated
5.70	Scram initiated by MSIV closure
5.96	Control rods start to move
8.33	RPT of 4 RIPs activated by low level L3
9.48	Control rods fully inserted
11.25	RPT of 6 RIPs activated by low level L2
~21 (est)	SRVs open to control pressure
~41	RCIC flow into vessel (not included in simulation)

Table 15.1-8 Sequence of Events for Inadvertent Safety/Relief Valve Opening

Time (s)	Event
0	Initiated opening of one SRV.
0.5 (est.)	Relief flow reaches full flow.
15 (est.)	System establishes new steady-state operation.
750 (est.)	Suppression pool temperature reaches setpoint; suppression pool cooling function is initiated.
1200 (est.)	Suppression pool temperature reaches setpoint; reactor scram is automatically initiated.

Table 15.1-9 Sequence of Events for Inadvertent RHR Shutdown Cooling Operation

Approximate Elapsed Time	Event
0	Reactor at states B or D (of Appendix 15A) when RHR shutdown cooling inadvertently activated.
0–10 min.	Slow rise in reactor power.
+10 min.	Operator may take action to limit power rise. Flux scram will occur if no action is taken.

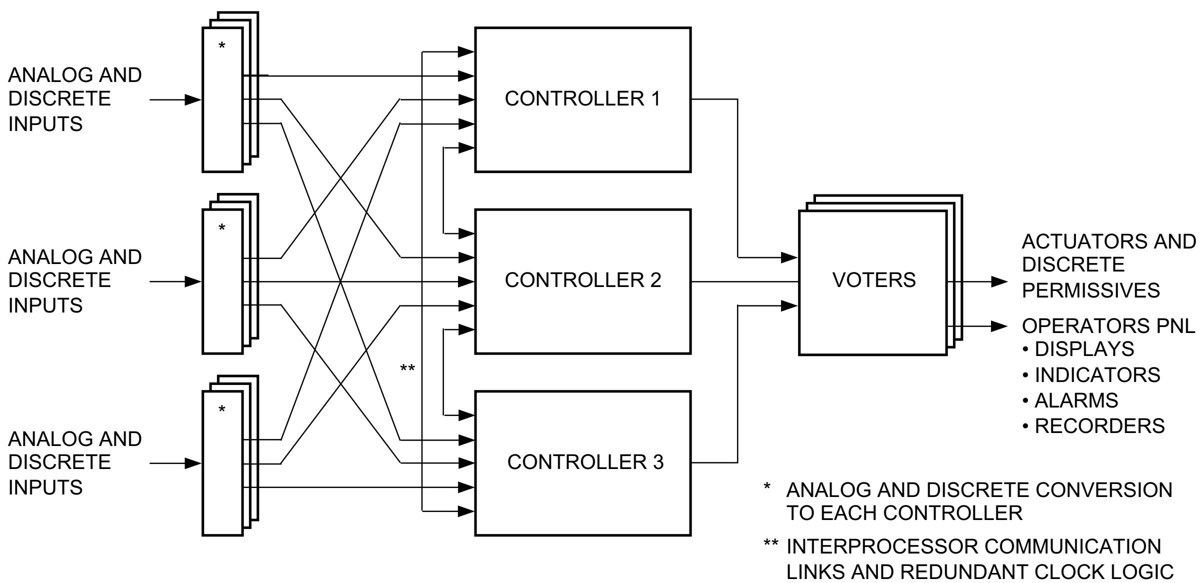


Figure 15.1-1 Simplified Block Diagram of Fault-Tolerant Digital Controller System

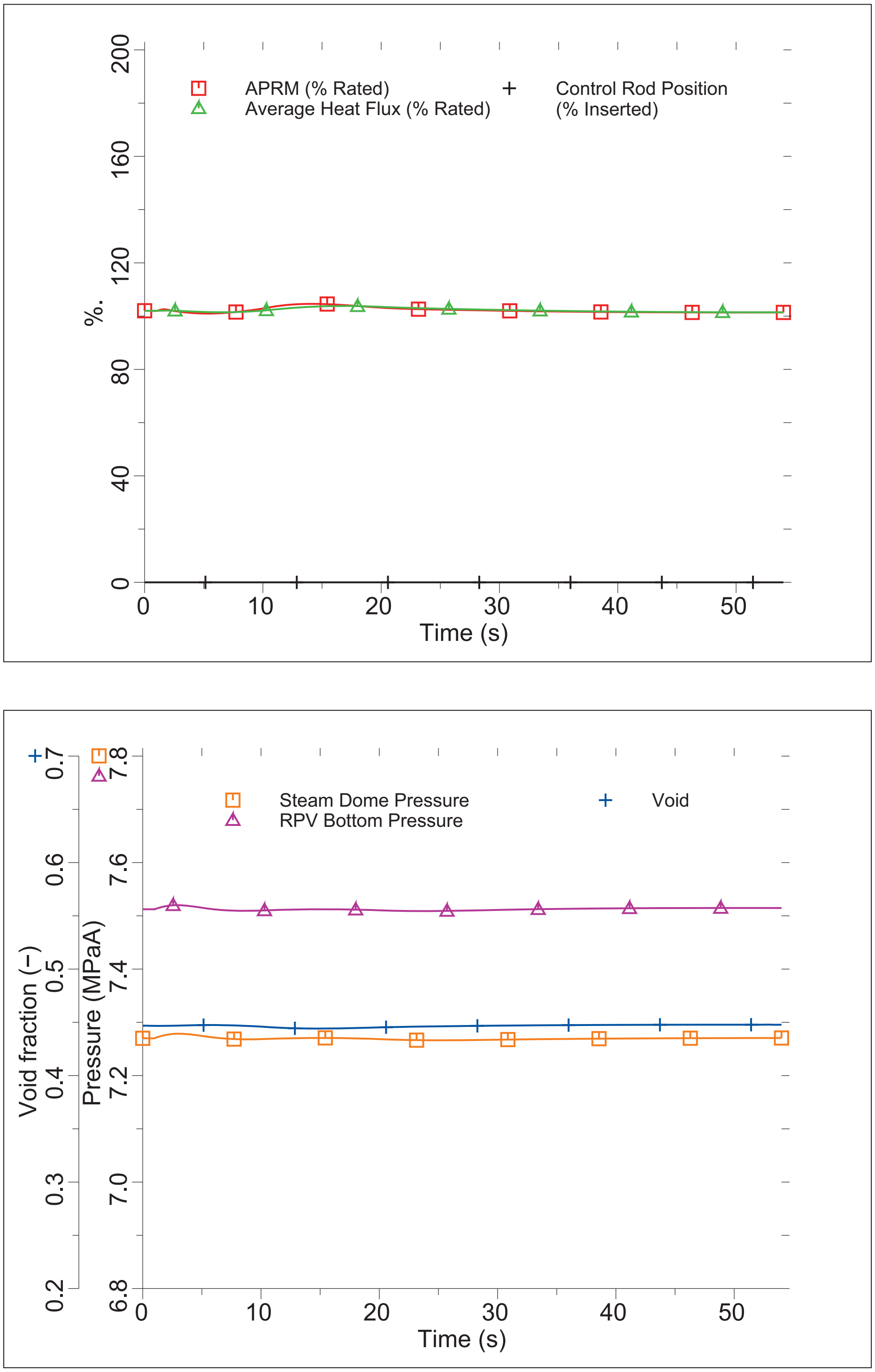
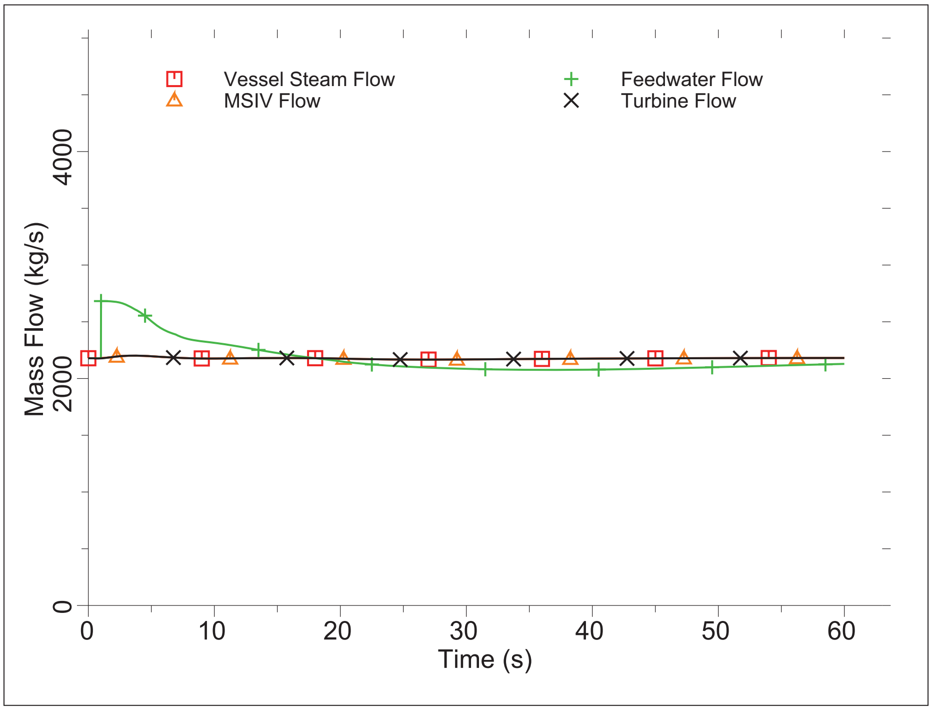


Figure 15.1-2a Runout of One Feedwater Pump



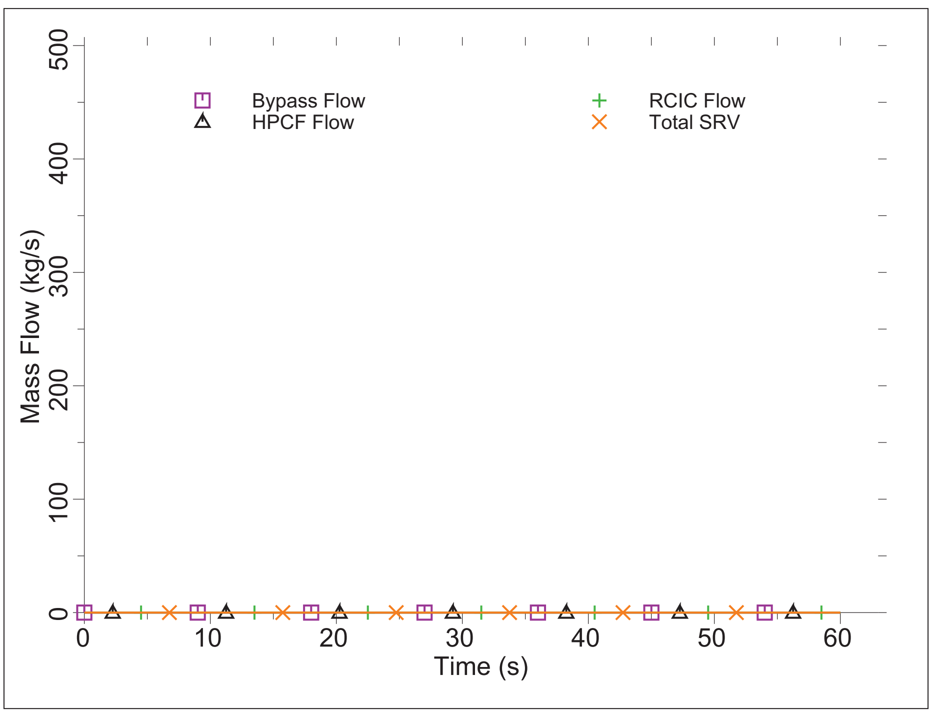


Figure 15.1-2b Runout of One Feedwater Pump

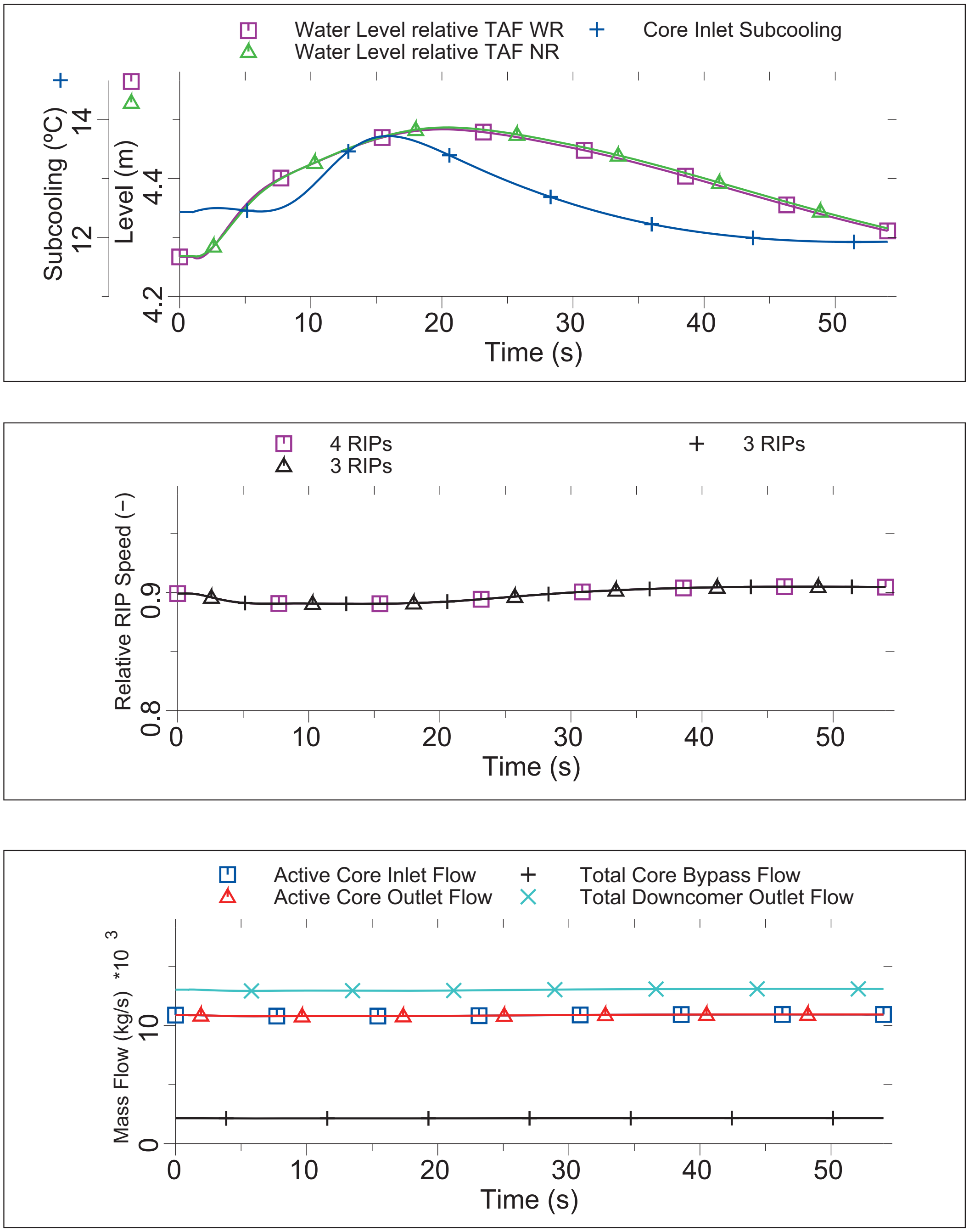
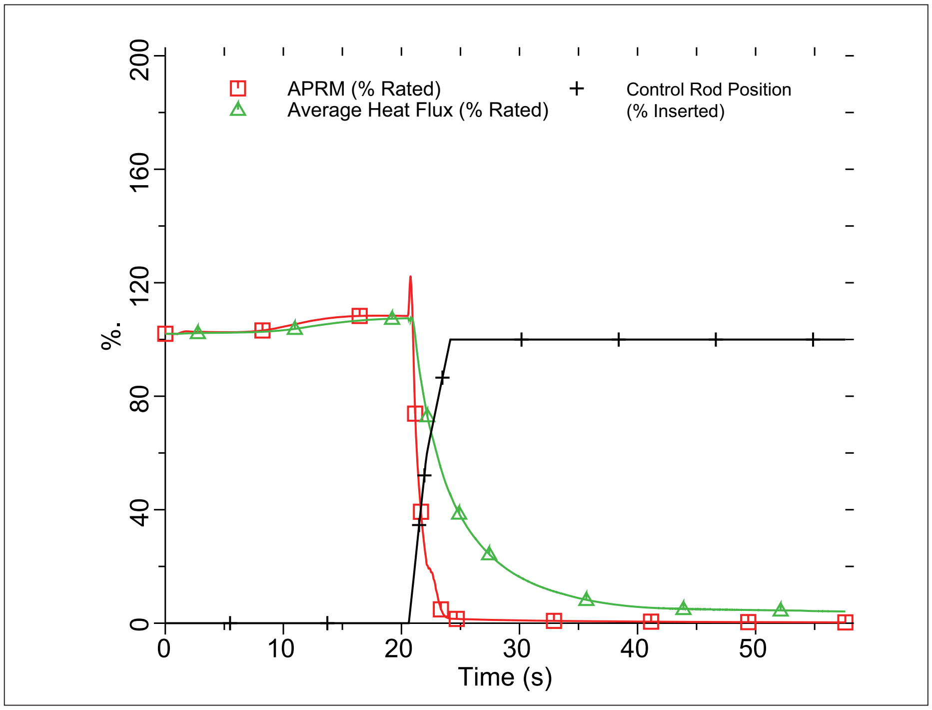


Figure 15.1-2c Runout of One Feedwater Pump



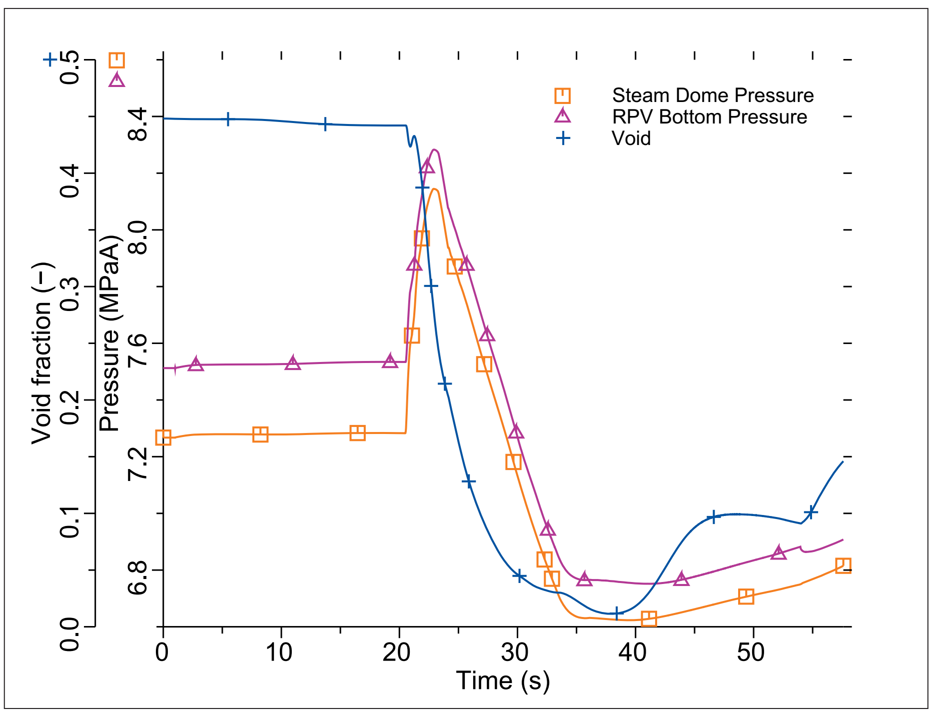


Figure 15.1-3a Feedwater Controller Failure – Maximum Demand

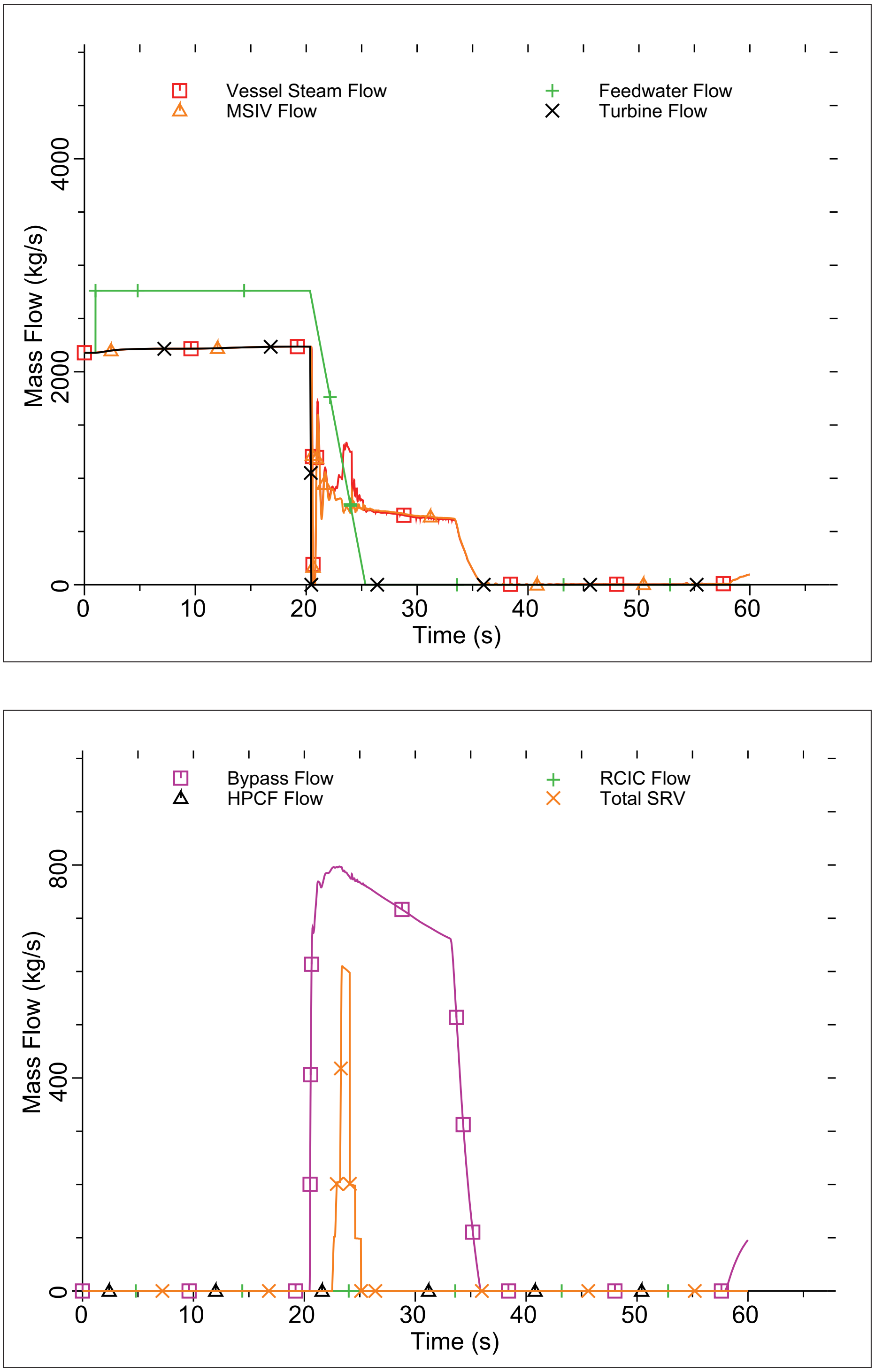


Figure 15.1-3b Feedwater Controller Failure – Maximum Demand

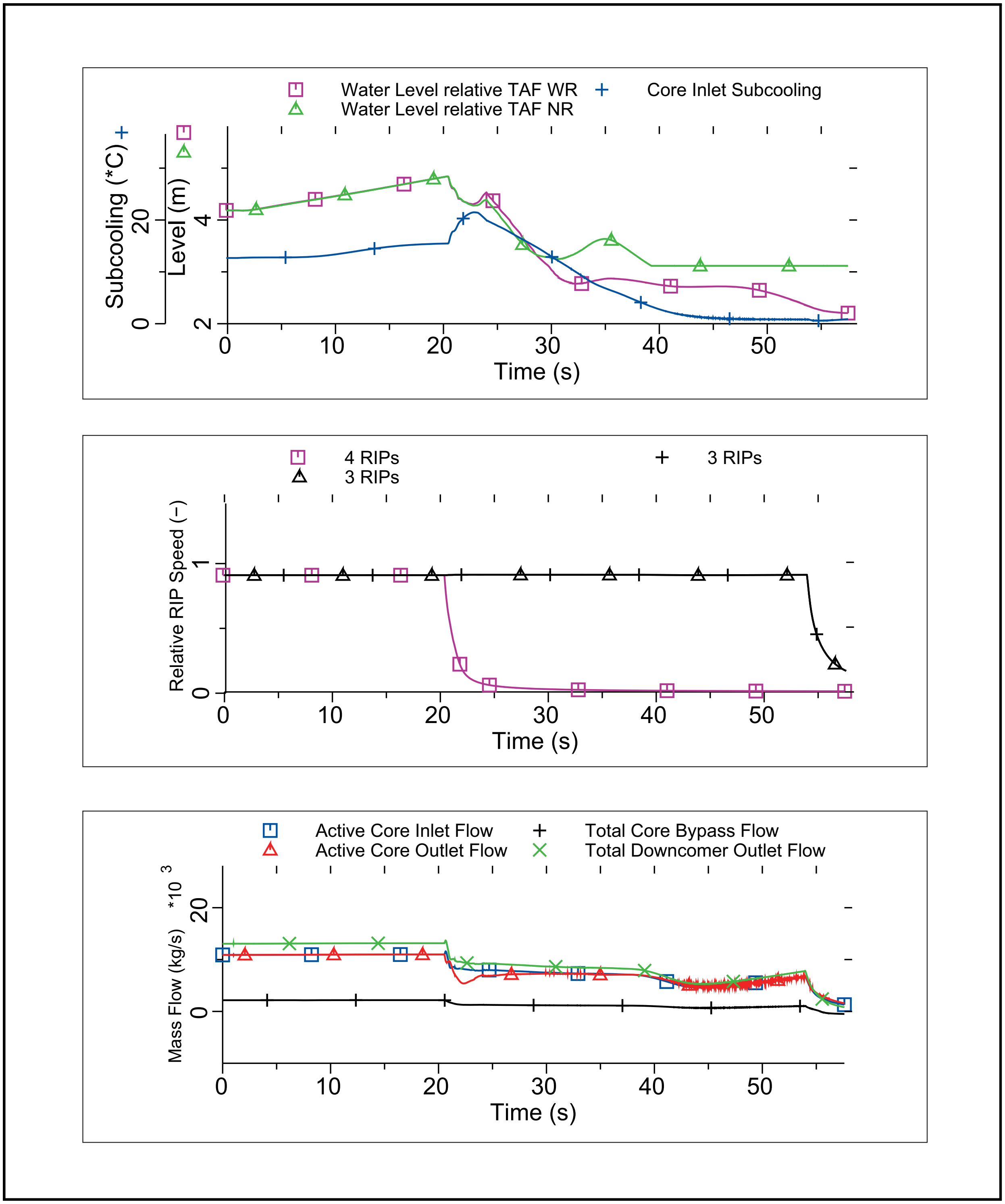


Figure 15.1-3c Feedwater Controller Failure – Maximum Demand

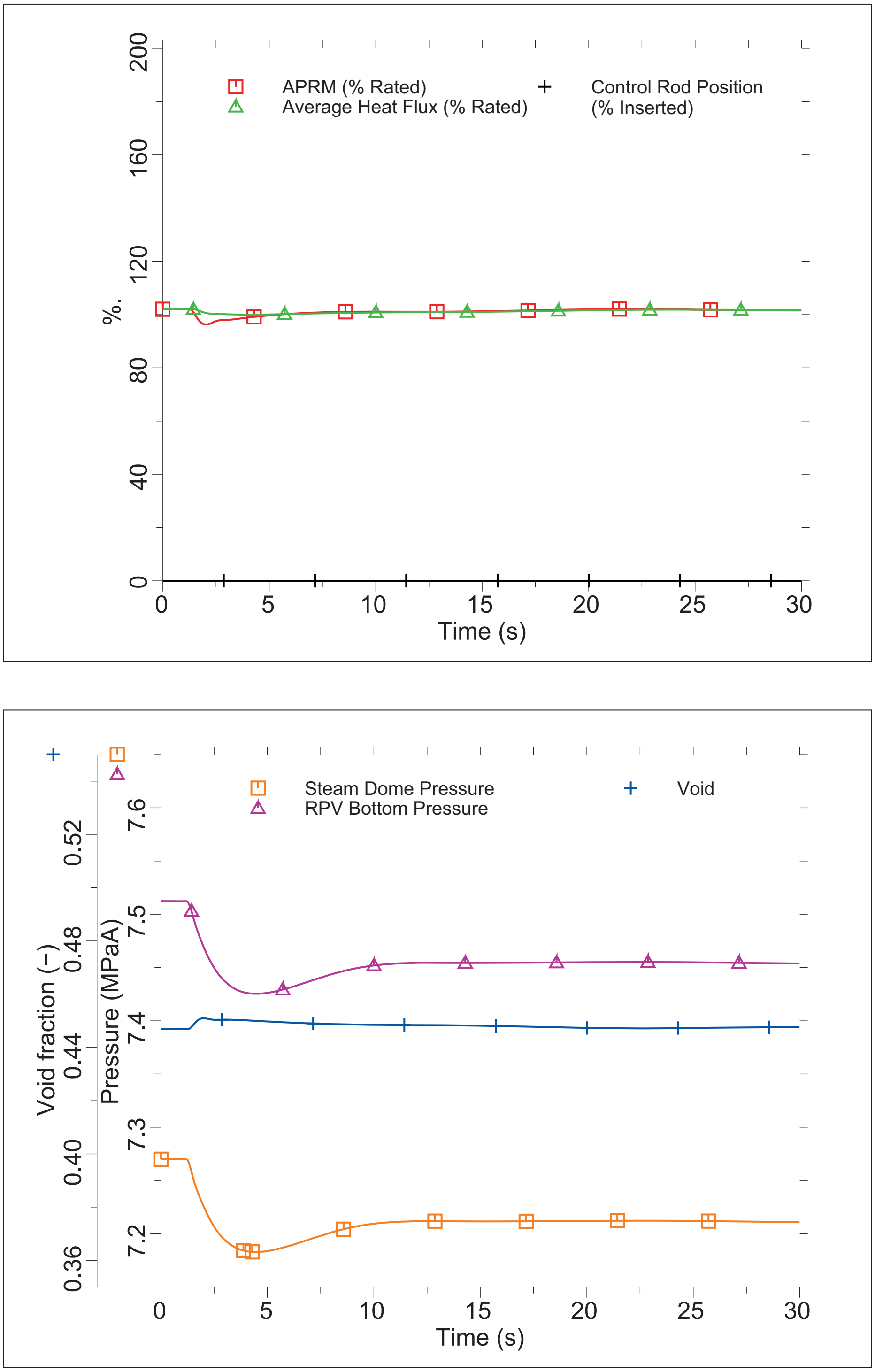


Figure 15.1-4a Inadvertent Opening of One Bypass Valve

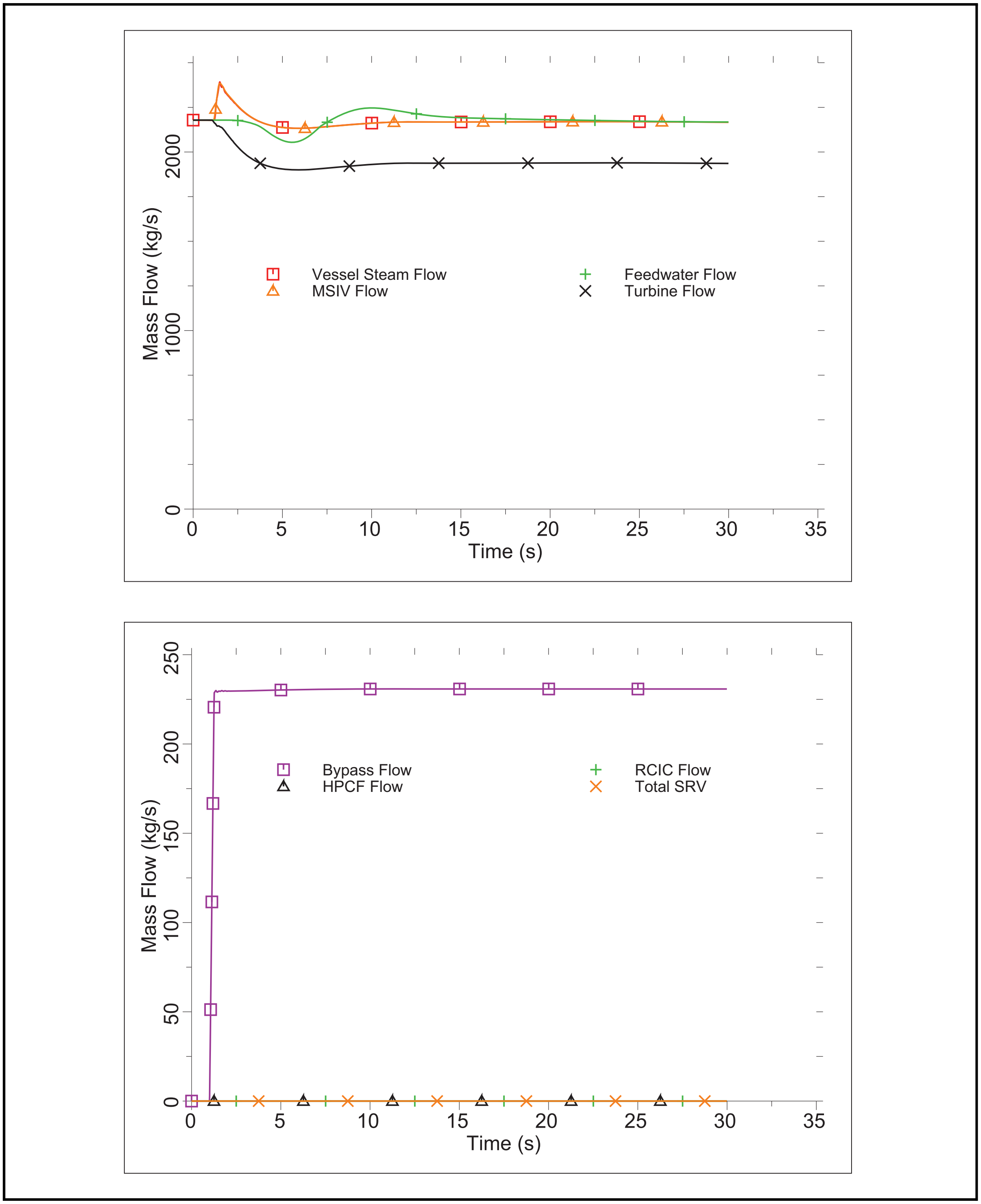


Figure 15.1-4b Inadvertent Opening of One Bypass Valve

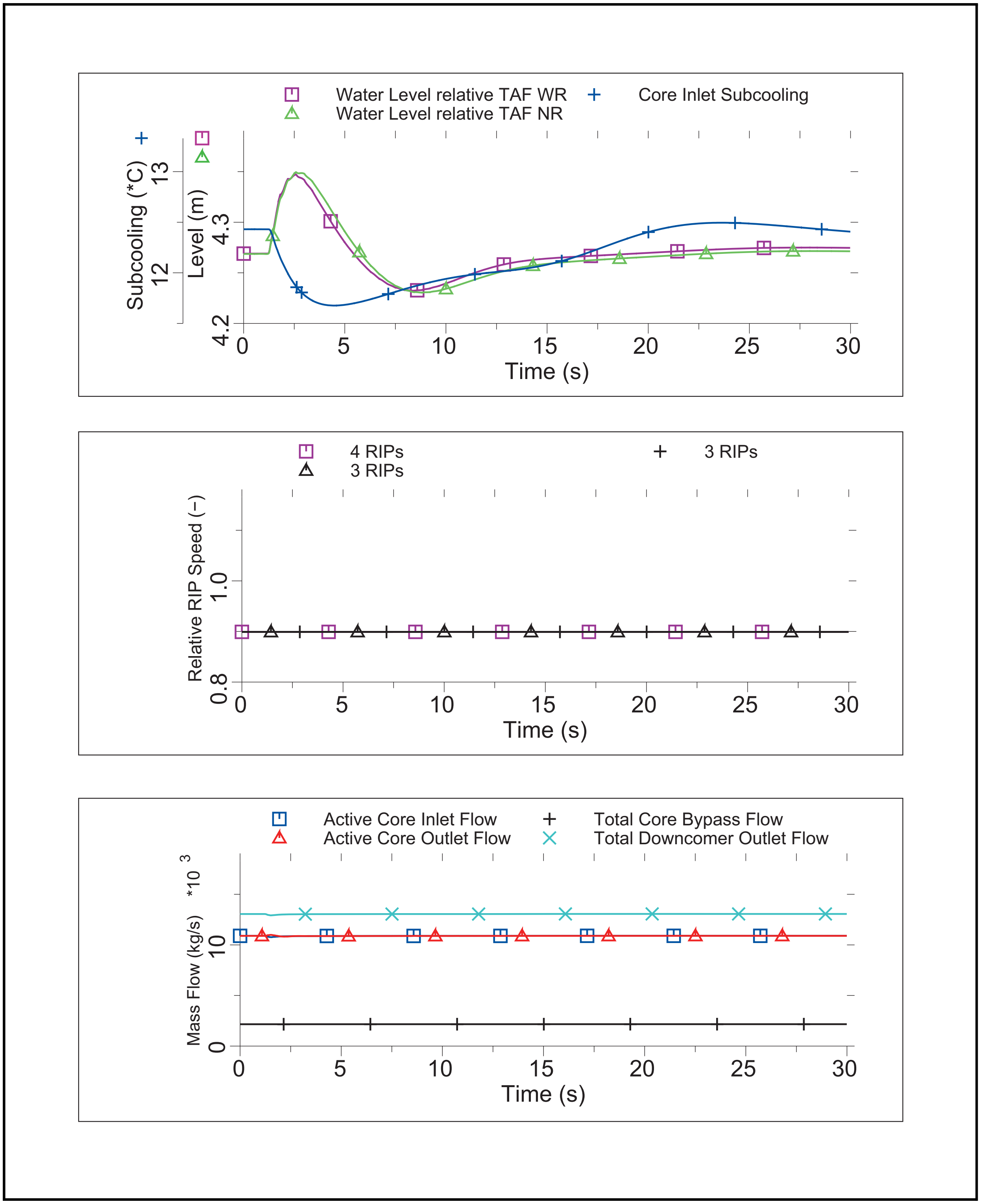


Figure 15.1-4c Inadvertent Opening of One Bypass Valve

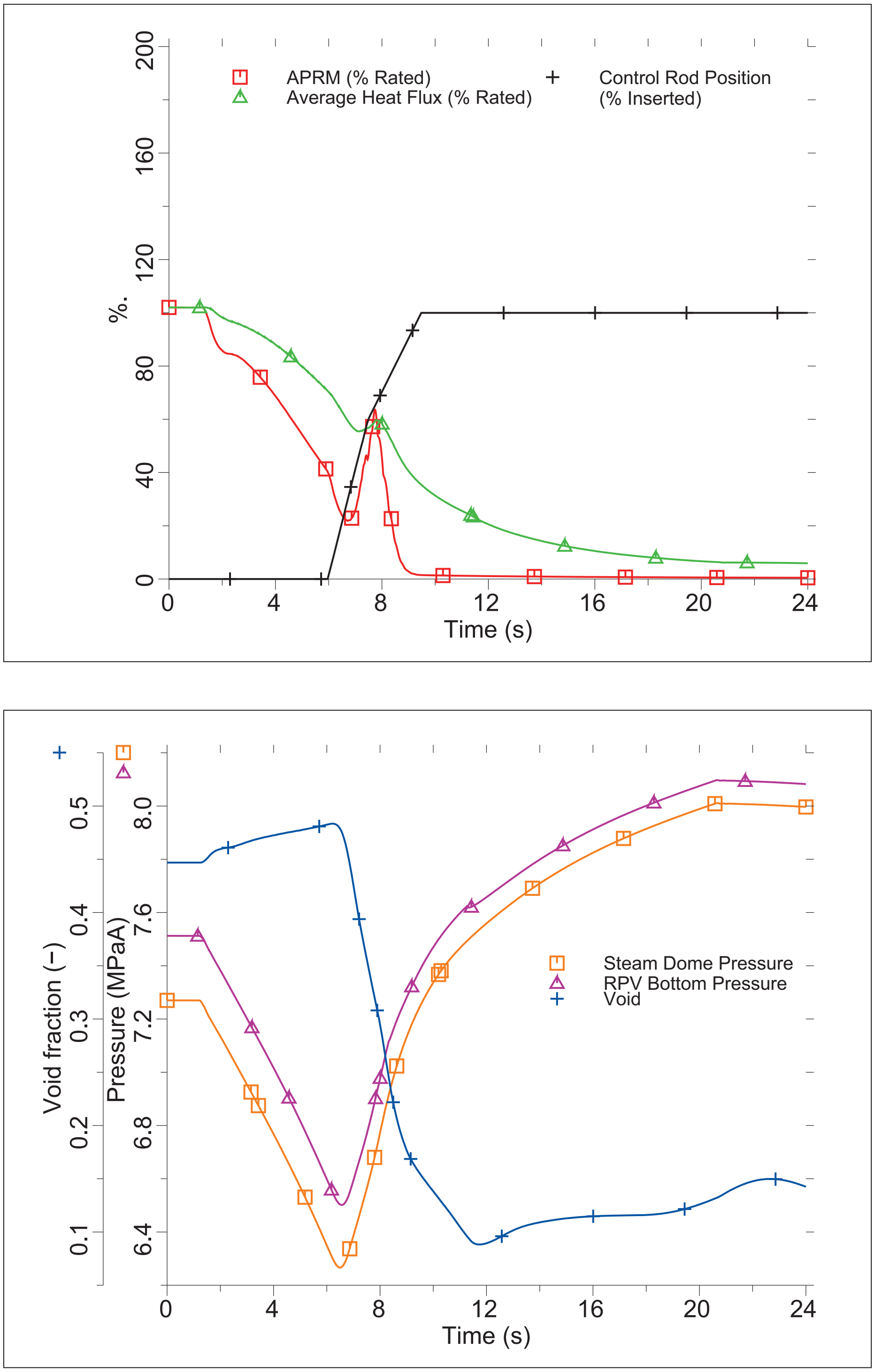


Figure 15.1-5a Opening of all Control and Bypass Valves

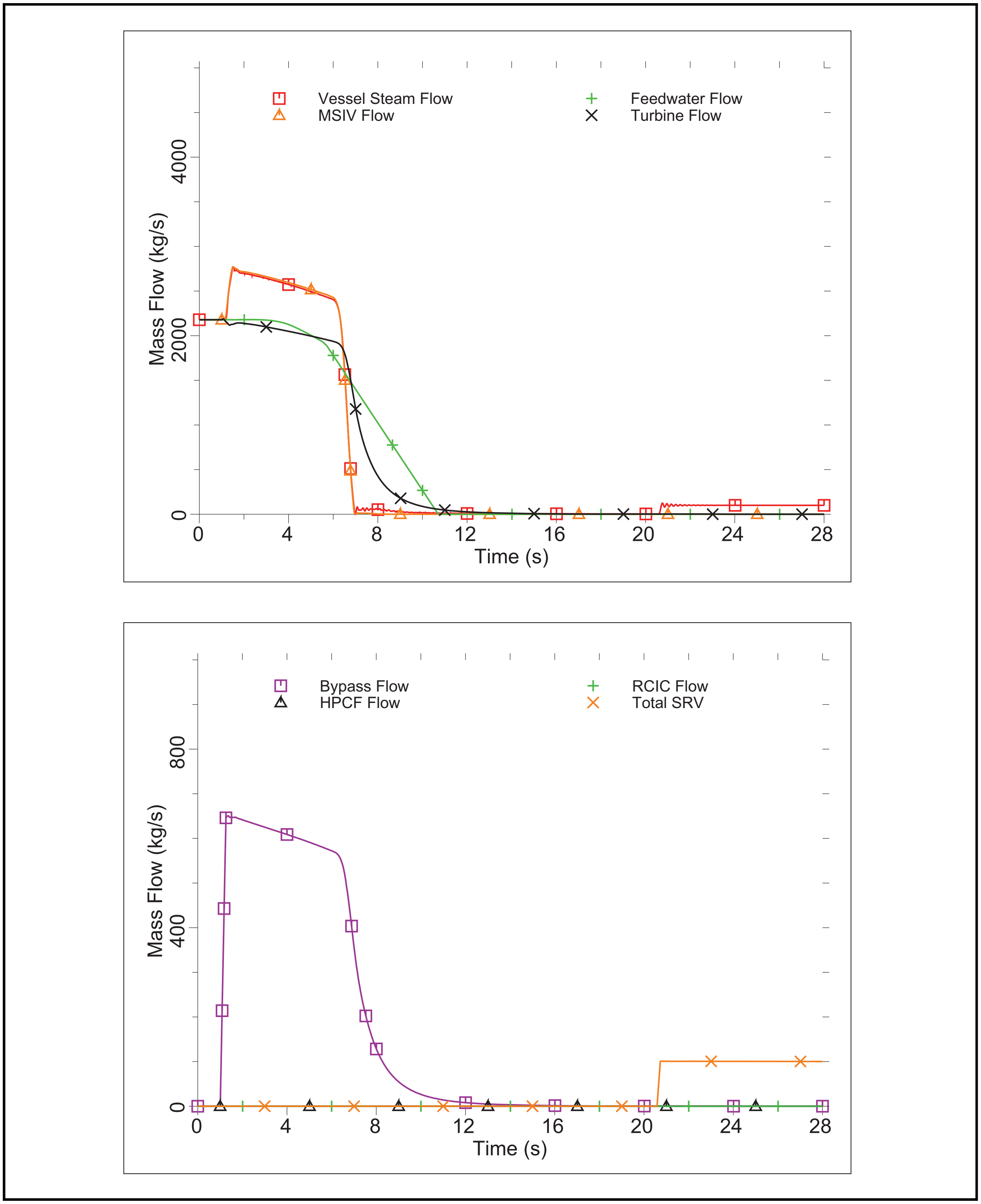


Figure 15.1-5b Opening of all Control and Bypass Valves

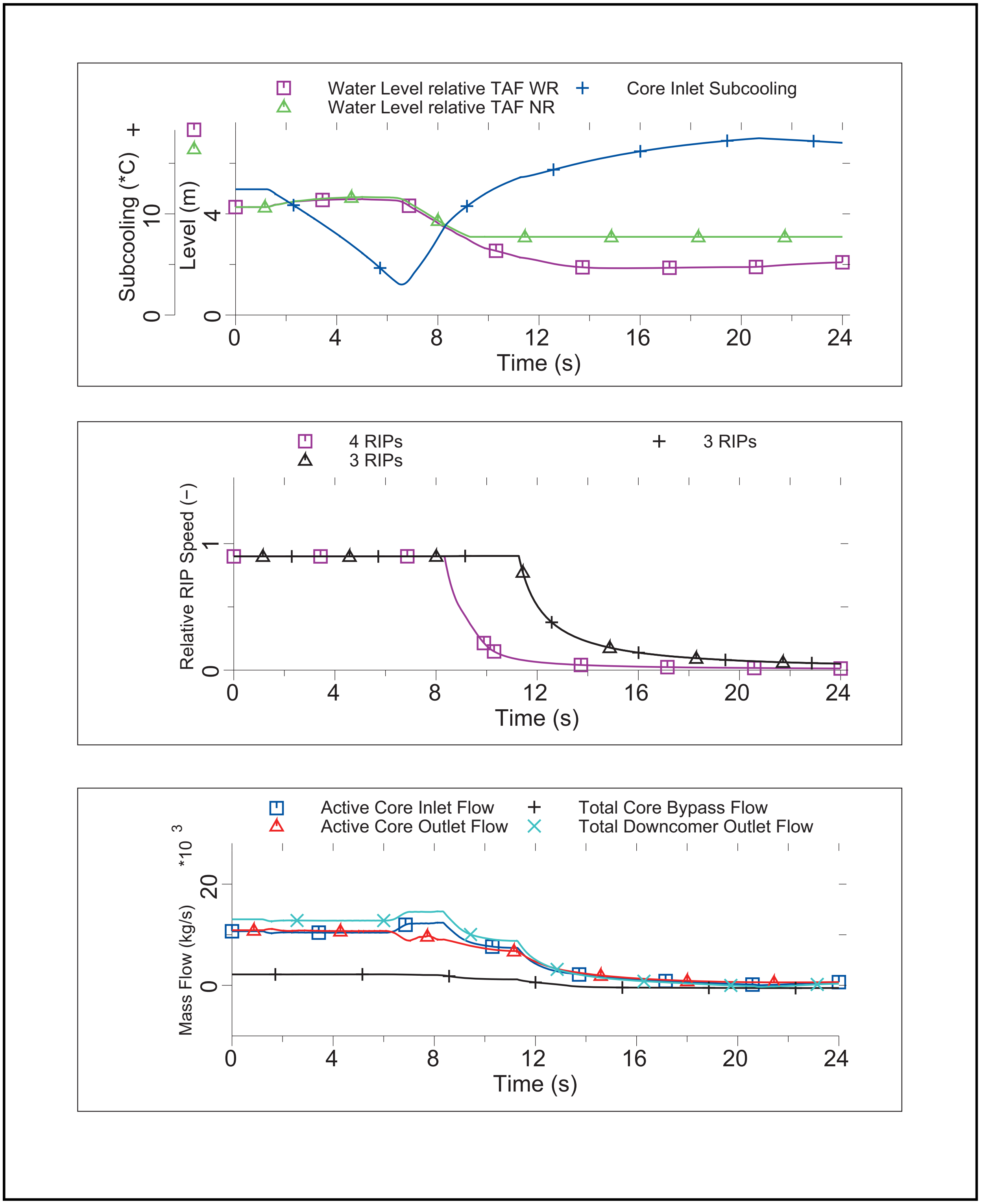


Figure 15.1-5c Opening of all Control and Bypass Valves