15.4 Reactivity and Power Distribution Anomalies

15.4.1 Rod Withdrawal Error—Low Power

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The probability of the initial causes, alone, is considered low enough to warrant its being categorized as an infrequent incident, because there is no postulated set of circumstances which results in an inadvertent rod withdrawal error (RWE) while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal or Withdrawal

During refueling operation, safety system interlocks provide assurance that inadvertent criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

15.4.1.1.2.2 Fuel Insertion with Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell with a withdrawn control rod, it is required that all control rods are fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the REFUEL position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

15.4.1.1.2.3 Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn when the RCIS SINGLE/GANG switch is in the SINGLE position. When the RCIS switch is in the GANG position, only one control rod pair with the same HCU may be withdrawn. Any attempt to withdraw an additional rod results in a rod block by the RCIS interlock. Because the core is designed to meet shutdown requirements with any one control rod pair (with the same HCU) withdrawn, the core remains subcritical.

15.4.1.1.2.4 Control Rod Removal Without Fuel Removal

The design of the control rod, incorporating the bayonet coupling system, does not physically permit the upward removal of the control rod without decoupling by rotation and the simultaneous or prior removal of the four adjacent fuel bundles.

15.4.1.1.2.5 Identification of Operator Actions

No operator actions are required to preclude this event, because the protection system design as previously presented will prevent its occurrence.

15.4.1.1.3 Core and System Performance

Because the possibility of inadvertent criticality during refueling is precluded, the core and system performances are not analyzed. The withdrawal of the highest worth control rod during refueling does not result in criticality. This is verified experimentally by performing shutdown margin checks (see Subsection 4.3 for a description of the methods and results of the shutdown margin analysis). Additional reactivity insertion is precluded by refueling interlocks. Because no fuel damage can occur, no radioactive material will be released from the fuel. Therefore, this event is not reanalyzed for specific core configurations.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance is not made for this event because there is no postulated set of circumstances for which this event could occur.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences is not made for this event because no radioactive material is released from the fuel.

15.4.1.2 Continuous Control Rod Withdrawal Error During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

It is postulated that during a reactor startup, a gang of control rods or a single control rod is inadvertently withdrawn continuously due to a procedural error by the operator or a malfunction of the automated rod movement control system.

The Rod Control and Information System (RCIS) has a dual channel rod pattern control function that prevents withdrawal of any out-of-sequence rods from 100% control rod density (CRD) to 50% CRD (i.e., for Group 1 to Group 4 rods). It also has bank position withdrawal sequence constraints such that, if the withdrawal sequence constraints are violated, the rod pattern control function of the RCIS will initiate a rod block. The bank position constraints are in effect from 50% CRD to the low power setpoint.

The startup range neutron monitor (SRNM) has a period-based trip function that stops continuous rod withdrawal by initiating a rod block if the flux excursion, caused by rod withdrawal, generates a period shorter than 20 seconds. The period-based trip function also initiates a scram if the flux excursion generates a period shorter than 10 seconds. Any single SRNM rod block trip initiates a rod block. Any two divisional scram trips out of four divisions initiates a scram. The SRNM also has upscale rod block and upscale scram functions as a

double protection for flux excursion. A detailed description of the period-based trip function is presented in Chapter 7.

For this transient to happen, a large reactivity addition must be introduced. The reactor must be critical, with control rod density greater than 50%. Additionally, the BPWS logic must fail such that a gang of rods can be continuously withdrawn. The causes of the event are summarized in Table 15.4-1. The probability for this event to occur is considered low enough to warrant its being categorized as an infrequent incident.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

The sequence of events of a typical continuous control rod withdrawal error during reactor startup is shown in Table 15.4-2.

15.4.1.2.2.2 Identification of Operator Actions

No operator actions are required to terminate this event, since the SRNM period-based trip functions will initiate and terminate this event.

15.4.1.2.3 Core and System Performance

15.4.1.2.3.1 Analysis Method and Analysis Assumptions

The analysis uses the reactivity insertion analysis code described in Reference 15.4-2.

15.4.1.2.3.2 Analysis Conditions and Results

(1) Analysis Conditions

(a) The reactor is assumed to be in the critical condition before the control rod withdrawal, with an initial power of 0.001% rated, and a temperature of 286°C at the fuel cladding surface.

(b) The highest worth of the withdrawn rods (gang) is 2.9%Δ k from full-in to full-out. Gang rod withdrawal is used as during a normal startup.

(c) The control rod withdrawal speed is 30 mm/s, the nominal FMCRD withdrawal speed.

(d) With the gang rod withdrawal, the reactor period monitored by any SRNM is relatively the same. Any single channel bypass of the SRNM does not affect the result.

(2) Analysis Result

With this 2.9%Δk reactivity insertion, the flux excursion generates a period of approximately 4 seconds. The rod block trip is initiated at 18 seconds after the start

of the transient. The scram is initiated at 17.9 seconds. The reason that scram happens first is due to the longer instrument delay time for the rod block. The event is terminated by the scram. The peak fuel enthalpy reached is approximately 50.4 J/g, which is 0.045 J/g higher than the initial fuel enthalpy. The result is illustrated in Figure 15.4-1.

15.4.1.2.3.3 Evaluation Based On Criteria

Due to the effective protection function of the period-based trip function, the fuel enthalpy increase is small. The criterion of 711 J/g (170 cal/g) for fuel enthalpy increase under RWE event is satisfied.

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

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no fuel damage in this event and only with mild change in gross core characteristics.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.4.2 Rod Withdrawal Error at Power

15.4.2.1 Features of the ABWR Automatic Thermal Limit Monitoring System (ATLM)

In the ABWR, the Automatic Thermal Limit Monitoring (ATLM) System performs the rod block monitoring function. The ATLM System is a dual channel subsystem of the Rod Control and Information System (RCIS). In each ATLM channel there are two independent thermal limit monitoring devices. One device monitors the MCPR limit and protects the operating limit of the MCPR, and the other device monitors the APLHGR limit and protects the operating limit of the APLHGR. The rod block algorithm and setpoint of the ATLM System are based on actual online core thermal limit information. If any one of the two limits is reached, either due to control rod withdrawal or recirculation flow increase, control rod withdrawal permissive is removed. Detailed description of the ATLM System is presented in Chapter 7.

15.4.2.2 Identification of Causes and Frequency Classification

The causes of a potential RWE transient are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. But in either case, the operating thermal limits rod block function will block any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods will be stopped before the operating thermal limit is

reached. Because there is no operating limit violation due to the preventive function of the ATLM, there is no RWE transient event.

15.4.2.3 Sequence of Event and System Operation

Due to an operator error or a malfunction of the automated rod withdrawal sequence control logic, a single control rod or a gang of control rods is withdrawn continuously. The ATLM operating thermal limit protection function of either MCPR or MLHGR protection algorithm stops further control rod withdrawal when either operating limit is reached. There is no basis for occurrence of the continuous control rod withdrawal error event in the power range.

No operator action is required to preclude this event, because the plant design as described above prevents its occurrence.

15.4.2.4 Core and System Performance

The performance of the ATLM System of the RCIS prevents the RWE event from occurring. The core and system performance are not affected by such an operator error or control logic malfunction. There is no need to analyze this event.

15.4.2.5 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no postulated set of circumstances for which this event could occur.

15.4.2.6 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

This event is covered with evaluations presented in Subsections 15.4.1 and 15.4.2 and does not have to be reanalyzed for specific core configurations.

15.4.4 Abnormal Startup of Idle Reactor Internal Pump

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from manual action by the operator to initiate pump operation. It is assumed that the remaining nine RIPs are already operating.

The normal restart procedure requires the operator to reduce the pump speeds of running RIPs to, at or near, their minimum speeds (i.e., 31% of rated speed) before the restart of the idle RIP. Plant operating procedures specify the maximum allowable speed for the nine operating RIPs, for a normal restart of one RIP. Therefore, an abnormal restart occurs only when an operator

error (i.e., operator ignoring the procedure) occurs. Should an abnormal restart occur, the much higher reverse flow at the idle RIP requires the inverter to provide electrical current much higher than the normal. This overcurrent requirement activates the overcurrent protection logic of the adjustable speed drive (ASD) which supplies the power to the idle RIP. This ASD is tripped by the protection logic. Therefore, an abnormal restart of the idle RIP becomes a trip of one RIP, which is presented in Subsection 15.3.1.

15.4.4.1.1.1 Normal Restart of Reactor Internal Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Reactor Internal Pump at High Power

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

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15.4.3 lists the sequence of events for an abnormal startup of an idle RIP.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator should:

(1) Adjust rod pattern, as necessary, for new power level following idle RIP start

(2) Reduce the speed of the running RIPs to, at or near, their minimum speeds

(3) Start the idle loop pump and adjust speed to match the running RIPs (monitor reactor power)

(4) Readjust power, as necessary, to satisfy plant requirements per standard procedure

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls. No protection systems action is anticipated. No ESF action occurs as a result of the event.

15.4.4.3 Core and System Performance

An abnormal restart of an idle RIP becomes a trip of one RIP event, which is presented in Subsection 15.3.1.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event because no significant pressure increases are incurred during this transient (Subsection 15.3.1).

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel, nor any steam discharged to the suppression pool.

15.4.5 Recirculation Flow Control Failure with Increasing Flow

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

The ABWR Recirculation Flow Control System (RFCS) uses a triplicated, fault-tolerant digital control system. The RFCS controls all ten 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 results in a maximum demand to all RIPs. A voter or actuator failure may result in an inadvertent runout of one RIP at its maximum drive speed (~40%/s). In this case, the RFCS senses the core flow change and commands the remaining RIPs to decrease speed and thereby automatically mitigate the transient and maintains 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 maximum demand to all RIPs. Should this occur, all RIPs could increase speed simultaneously. Each RFCS processing channel has a speed demand 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.

15.4.5.1.2 Frequency Classification

15.4.5.1.2.1 Fast Runout 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.

15.4.5.1.2.2 Fast Runout of All Reactor Internal Pumps

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

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

15.4.5.2.1.1 Fast Runout of One Reactor Internal Pump

Table 15.4.4 lists the sequence of events for Figure 15.4-2.

15.4.5.2.1.2 Fast Runout of All Reactor Internal Pumps

Table 15.4.5 lists the sequence of events for Figure 15.4-3

15.4.5.2.1.3 Identification of Operator Actions

The operator should:

(1) Transfer flow control to manual and reduce the flow to minimum

(2) Identify cause of the failure

Reactor pressure is controlled as required, depending on whether scram occurs and, if scram occurs, whether a restart or cooldown is planned. In general, following a scram, the corrective action is to hold reactor pressure and condenser vacuum for restart after the malfunction has been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator should:

(1) Observe that all rods are in

(2) Check the reactor water level and maintain above low level (L2) trip to prevent RCIC initiation

(3) Switch the reactor mode switch to the SHUTDOWN position

(4) Maintain vacuum and turbine seals

(5) Transfer the recirculation flow controller to the manual position and reduce setpoint to zero

(6) Survey maintenance requirements and complete the scram report

(7) Monitor the turbine coastdown and auxiliary systems

(8) Establish a restart of the reactor per the normal procedure

15.4.5.2.2 Systems Operation

The analysis of this transient assumes and takes credit for normal functioning of plant instrumentation and controls and the reactor protection system. Operation of engineered safeguards is not expected.

15.4.5.3 Core and System Performance

15.4.5.3.1 Input Parameters and Initial Conditions

In each of these events, the most severe consequences result when initial conditions are established for operation at the low end of the rated flow control rod line. Specifically, this is

64.5% NBR power and 42% core flow. The maximum speed increasing rate of 40%/s is assumed for one RIP runout.

For all RIPs runout, 5%/s is assumed for the speed limit. The maximum core flow achieved by all RIPs runout is conservatively assumed to be 120% of rated.

15.4.5.3.2 Results

15.4.5.3.2.1 Fast Runout of One Reactor Internal Pump

Figure 15.4-2 presents the analysis of a fast runout of one RIP with its maximum speed increase rate of 40%/s. Table 15.4.4 provides the sequence of events of this failure.

The increase in core flow causes a rise in neutron flux. The peak neutron flux reached is 76% of NBR value, which is below the high neutron flux scram setpoint. The accompanying average fuel surface heat flux reaches 69% of NBR (107% of initial) at approximately 3.2 s. Because this event does not result in a significant increase in pressure and it is initiated from a low power condition, no MCPR is reported.

Reactor pressure is presented in Subsection 15.4.5.4.

15.4.5.3.2.2 Fast Runout of All Reactor Internal Pumps

Figure 15.4-3 illustrates the fast runout of all RIPs with a maximum speed increase rate of 5%/s. Table 15.4.5 shows the sequence of events for this failure. Flux scram occurs at 9.83 s, peaking at 130% of NB rated, while the average surface heat flux reaches 110% of NB rated (171% of initial) at approximately 9.9 s. No fuel failure is expected. Because this event does not result in a significant increase in pressure and it is initiated from a low power condition, no MCPR is reported.

15.4.5.4 Barrier Performance

15.4.5.4.1 Fast Runout of One Reactor Internal Pump

This transient results in a slight increase in reactor vessel pressure (Figure 15.4-2) and therefore represents no threat to the RCPB.

15.4.5.4.2 Fast Runout of All Reactor Internal Pumps

This transient results in a slight increase in reactor vessel pressure (Figure 15.4-3) and therefore represents no threat to the RCPB.

15.4.5.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event because no radioactive material is released from the fuel, nor any steam discharged to the suppression pool.

15.4.6 Chemical and Volume Control System Malfunctions

Not applicable to BWRs. This is a PWR event.

15.4.7 Mislocated Bundle Accident

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

The event discussed in this section is the loading of a fuel bundle in an improper location and subsequent operation of the core. Three errors must occur for this event to take place in the equilibrium core loading. First, a bundle must be placed into a wrong location in the core. Second, the bundle which was supposed to be loaded where the error occurred is also put in an incorrect location or discharged. Third, the mislocated bundles are overlooked during the core verification process performed following core loading.

Provisions to prevent potential fuel loading errors are included in the plant Operating Procedures/Technical Specification.

15.4.7.1.2 Frequency Classification

This unlikely event occurs when a fuel bundle is loaded into the wrong location in the core. It is assumed that the bundle is misplaced in the worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as a limiting fault, because the expected frequency is very low based upon past experience.

15.4.7.2 Sequence of Events and Systems Operation

15.4.7.2.1 Sequence of Events

The postulated sequence of events for the mislocated bundle accident (MBA) is presented in Table 15.4-6.

15.4.7.2.2 Systems Operation

A mislocated bundle error, undetected by incore instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

15.4.7.3 Core and System Performance

It is permitted to consider undetected Fuel Loading Errors (FLE) as infrequent incidents that meet 10 % of 10CFR100 dose criteria (References 15.4-6 and 15.4-7), provided that certain design features, together with approved plant operating procedures are in place. Even using very conservative assumptions, it has been shown that the required X/Q to meet 10 % of 10CFR100 was higher than the standard plant X/Q used in Table 2.0-1. Therefore, the COL

applicant will provide an analysis to confirm that the consequences of a fuel bundle mislocated event meet all requirements approved by the NRC. See Subsection 15.4.11.1 for COL license information.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because it is a mild and highly localized event. No perceptible change in the core pressure is observed.

15.4.7.5 Radiological Consequences

An evaluation of the radiological consequences has shown that they are a small fraction of the 10CFR100 limit, i.e. less than 10% (see Reference 15.4-7).

15.4.8 Misoriented Fuel Bundle Accident

15.4.8.1 Identification of Causes and Frequency Classification

15.4.8.1.1 Identification of Causes

The misoriented fuel bundle (MOFB) event discussed in this section is the situation in which a bundle has been loaded in the correct location but is rotated by 90 or 180 degrees. The rotation could result in non-uniform water gaps which could cause an increase in local rod power through increased moderation. The initiator for a reactor with a MOFB is an operator placing the bundle into the core in a misoriented position. The next step in the accident progression is failure to detect the MOFB. A verification procedure is recommended to detect a MOFB. This verification procedure requires two core scans. One scan is with an underwater TV camera positioned close enough to read the bundle serial numbers on top of the lifting bail (first attribute) and to check the orientation of the bosses on the bail (second attribute). The other scan is with a TV camera positioned sufficiently above the core to allow viewing one complete 4 bundle cell for the following four attributes: boss on lifting bail, channel fasteners, channel buttons, and “cell look alike”. Two independent reviewers (checkers A and B) are recommended to verify tapes from the above procedure.

15.4.8.1.2 Frequency Classification

This unlikely event occurs when a bundle is loaded with the wrong orientation (90 or 180 deg) in the core. It is assumed that the misoriented bundle is located in the worst possible location and the plant is operated with the misoriented bundle. This event is categorized as a limiting fault because the expected frequency is very low based on past experience.

15.4.8.2 Sequence of Events and Systems Operation

15.4.8.2.1 Sequence of Events

The postulated sequence of events for the misoriented fuel bundle accident (MOFB) is presented in Table 15.4-7.

15.4.8.2.2 Systems Operation

A misoriented fuel bundle accident, undetected by in-core instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

15.4.8.3 Core and System Performance

As discussed in 15.4.7.3 for misloaded fuel bundles it is now permitted to consider undetected Fuel Loading Errors (FLE) as infrequent incidents that meet 10% of 10CFR100 dose criteria (References 15.4-6 and 15.4-7), provided that certain design features, together with approved plant operating procedures are in place. Therefore, the COL applicant will provide an analysis to confirm that the consequences of a fuel bundle misoriented event meet all requirements approved by the NRC. See Subsection 15.4.11.2 for COL license information.

15.4.8.4 Barrier Performance

An evaluation of the barrier performance is not made for this event because it is a mild and highly localized event. No perceptible change in the core pressure is observed.

15.4.8.5 Radiological Consequences

An evaluation of the radiological consequences has shown that they are a small fraction of the 10CFR100 limit, i.e. less than 10%.

15.4.9 Rod Ejection Accident

15.4.9.1 Identification of Causes and Frequency Classification

The rod ejection accident is caused by a major break on the FMCRD housing, outer tube or associated CRD pipe lines. Due to a break of this type, the reactor pressure exerted on the CRD spud pushes down the hollow piston and the ballnut with a large force. The shaft screw and the motor are forced to unwind. A passive brake mechanism is installed in the FMCRD system to prevent the control rod from moving. The design of the brake is presented in Section 4.6.1. The probability of the initial causes (i.e., a CRD pipe line break or housing break) is considered low enough to warrant its being categorized as a limiting fault. Even if this accident does happen, the brake prevents the control rod from ejection. Should the brake fail, the check valve will serve as a backup brake to prevent the rod ejection.

15.4.9.2 Sequence of Events and Systems Operation

If a major break occurs on the FMCRD housing, the reactor pressure will provide forces that could cause the shaft screw to unwind. The FMCRD brake mechanism prevents the rod from moving. Therefore, no rod ejection can occur.

15.4.9.3 Core and System Performance

The FMCRD brake mechanism prevents this event from occurring. There is no need to analyze this event.

15.4.9.4 Barrier Performance

An evaluation of the barrier performance is not made for this accident since there is no circumstance for which this event would occur.

15.4.9.5 Radiological Consequences

The radiological analysis is not required.

15.4.10 Control Rod Drop Accident

15.4.10.1 Features of the ABWR Fine Motion Control Rod Drives

As presented in Subsection 4.6.1, the Fine Motion Control Rod Drive(FMCRD) System has several new features that are unique compared with locking piston control rod drives.

In each FMCRD, there are dual Class 1E separation-detection devices that will detect the separation of the control rod from the CRD if the control rod is stuck and separated from the ballnut of the CRD. The control rods are normally inserted into the core and withdrawn with the hollow piston, which is connected with the control rod, resting on the ballnut. The separation-detection device is used at all times to ascertain that the hollow piston and control rod are resting on the ballnut of the FMCRD. The separation-detection devices sense motion of a spring-loaded support for the ball screw and, in turn, the hollow piston and the control rod. Separation of either the control rod from the hollow piston or the hollow piston from the ballnut will be detected immediately. When separation has been detected, the interlocks preventing rod withdrawal will operate to prevent further control rod withdrawal. Also, an alarm signal will be initiated in the control room to warn the operator.

There is also the unique highly reliable bayonet type coupling between the control rod blade and the control rod drive. With this coupling, the connection between the blade and the drive cannot be separated unless they are rotated 45 degrees. This rotation is not possible during reactor operation. There are procedural coupling checks to assure proper coupling. There is also the automated overtravel check in the RCIS logic during automated operation. Finally, there is the latch mechanism on the hollow piston part of the drive. If the hollow piston is separated from the ballnut and rest of the drive due to a stuck rod, the latch will limit any subsequent rod drop to a distance of 8 inches. More detailed descriptions of the FMCRD System are presented in Subsection 4.6.1.

15.4.10.2 Identification of Causes and Frequency Classification

For the rod drop accident to occur, it is necessary for such highly unlikely events as failure of both Class 1E separation-detection devices, or the failure of the rod block interlock, and the

failure of the latch mechanism to occur simultaneously with the occurrence of a stuck rod on the same FMCRD. This would permit hollow piston separation from the ballnut.

Alternatively, separation of the blade from the hollow piston would require either that the control rod was installed without coupling and the coupling checks failed, or there is structural failure of this coupling. Under such circumstances of this coupling failure, the rod drop accident can only occur with the simultaneous failure of both separation-detection devices (or the failure of the rod block interlock), together with the occurrence of a stuck rod on the same FMCRD.

In either case, because of the low probability of such simultaneous occurrence of these multiple independent events, there is no basis to postulate this event to occur.

15.4.10.3 Sequence of Events and System Operation

15.4.10.3.1 Sequence of Events

The bayonet coupling and procedural coupling checks will preclude the uncoupling of the control rod from the hollow piston of the FMCRD. If the control rod is stuck, the separation-detection devices will detect the separation of the control rod and hollow piston from the ballnut of the FMCRD, and rod block interlock will prevent further rod withdrawal. The operator will be alarmed for this separation.

There is no basis for the control rod drop event to occur.

15.4.10.3.2 Identification of Operator Actions

No operator actions are required to preclude this event. However, the operator will be notified by the separation-detection alarm if separation is detected.

15.4.10.4 Core and System Performance

The performance of the separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident.

15.4.10.5 Barrier Performance

An evaluation of the barrier performance is not made for this accident, since there is no circumstance for which this event could occur.

15.4.10.6 Radiological Consequences

The radiological analysis is not required.

15.4.11 COL License Information

15.4.11.1 Mislocated Fuel Bundle Accident

COL applicants will provide an analysis to confirm that the consequences of a fuel bundle mislocated event meet all requirements approved by the NRC (Subsection 15.4.7.3).

15.4.11.2 Misoriented Fuel Bundle Accident

COL applicants will provide an analysis to confirm that the consequences of a fuel bundle misoriented event meets all requirements approved by the NRC (Subsection 15.4.8.3).

15.4.12 References

15.4-1 Not Used.

15.4-2 ABB Atom Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors. The RAMONA-3B Computer Code", RPA 890112, November 1989.

15.4-3 Not Used

15.4-4 Not Used

15.4-5 Not Used

15.4-6 Standard Review Plan, section 15.4.7, “Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position,” Rev.2, March 2007.

15.4-7 Final Safety Evaluation of GESTAR II Amendment 28, “Misloaded Fuel Bundle Event Licensing Basis Change to Comply With Standard Review Plan 15.4.7,” March, 2007.

Table 15.4-1 Causes of Control Rod Withdrawal Error

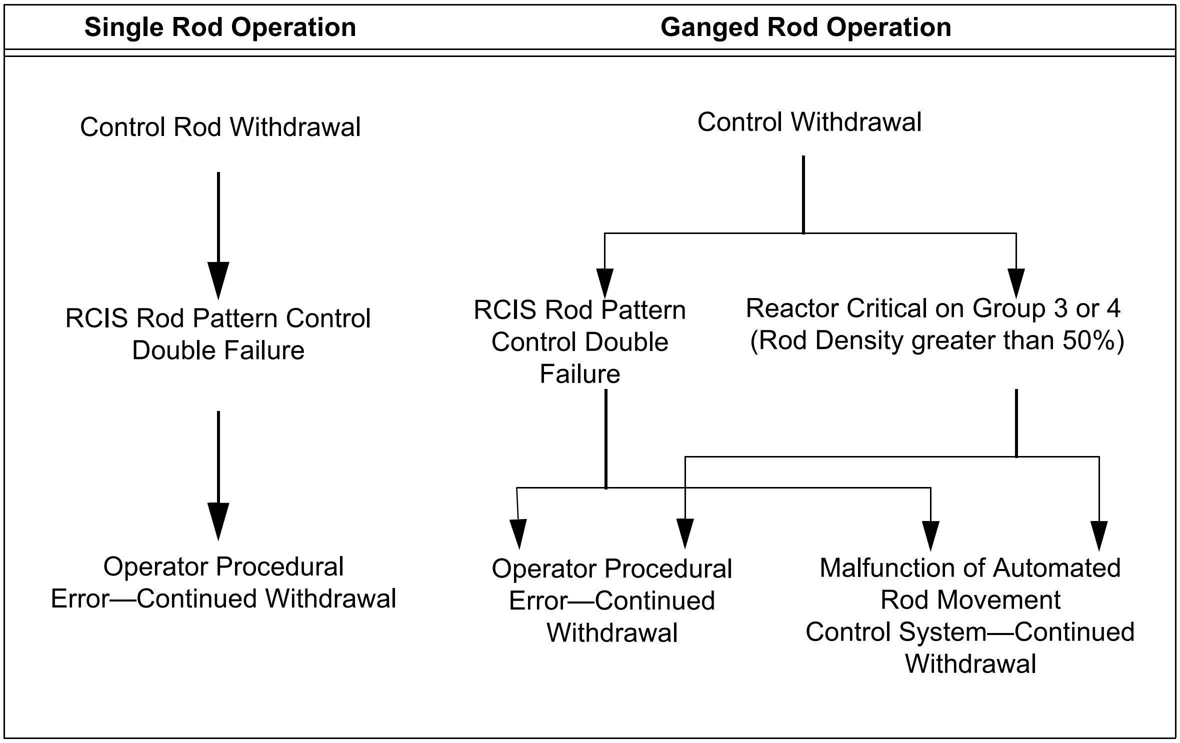


Table 15.4-2 Sequence of Events for Figure 15.4-1 (Continuous Control Rod Withdrawal Error During Reactor Startup)

Time (s)	Events
—	Rod Control & Information System (RCIS) logics to prevent continuous control rod withdrawal fail (from both channels).
0	Operator withdraws a gang of rods (or a single rod) continuously; or a gang of rods (or single rod) is withdrawn continuously due to a malfunction of the Automated Rod Movement Control System.
~6	Neutron flux increases rapidly due to the continuous reactivity addition, with a very short period.
17.9	The SRNM Period-Based Scram Trip initiates reactor scram due to short period (less than the 10-second setpoint).
18.0	The SRNM Period-Based Rod Block Trip initiates rod block due to short period (less than the 20-second setpoint).
~21	Reactor is scrammed and the event is terminated.

Table 15.4-3 Sequence of Events for Abnormal Startup of Idle RIP

Time (s)	Events
0	Operator starts idle RIP with running RIPs at higher than minimum speeds.
0	Interlock fails to prevent restart.
0.1 (estimated)	Overcurrent protection logic trips the electrical bus.
0.1 (estimated)	One or two RIPs are tripped due to the bus trip.
For Other Sequences of Events, see Table 15.3-1.	

Table 15.4-4 Sequence of Events for Figure 15.4-2 (Fast Runout of One RIP)

Time (s)	Events
0.00	Simulation starts
1.00	Fast runout of one reactor internal pump initiated
2.78	Max neutron flux
4.14	Max steam dome pressure
~15	Reactor settles in a new steady state

Table 15.4-5 Sequence of Events for Figure 15.4-3 (Fast Runout of All RIPs)

Time (s)	Event
0.00	Simulation starts
1.00	Runout of all RIPs to maximum flow
9.58	Reactor scram initiated by high neutron flux
9.83	Control rods start to move
9.83	Max neutron flux
10.1	Max steam dome pressure
13.3	RPT of 4 RIPs initiated by low level L3
13.4	Control rods fully inserted
18.6 sec	RPT of 6 RIPs initiated by low level L2. RCIC started
~49 sec	RCIC flow enters vessel (not simulated).

Table 15.4-6 Sequence of Events of the Mislocated Bundle Accident

(1)	During the core loading operation, a bundle is loaded into the wrong core location.
(2)	Subsequently, the bundle designated for this location is incorrectly loaded into the location of the previous bundle.
(3)	During the core verification procedure, the two errors are not observed.
(4)	The plant is brought to full power operation without detecting misplaced bundles.
(5)	The plant continues to operate throughout the cycle.

Table 15.4-7 Sequence of Events of the Misoriented Fuel Bundle Accident

(1)	During the core loading operation, a bundle is rotated 90 or 180 degrees.
(2)	During the core verification procedure, this error goes undetected.
(3)	The plant is brought to full power operation without detecting the misoriented bundle.
(4)	The plant continues to operate throughout the cycle.

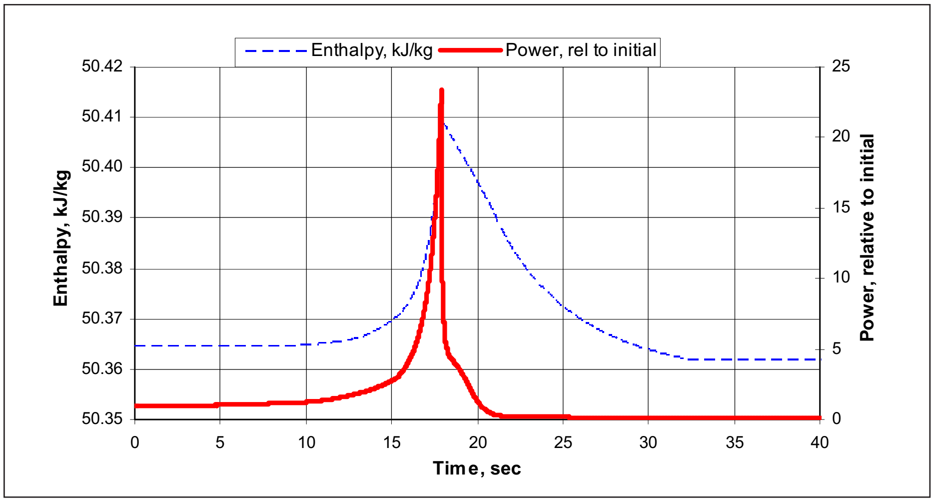
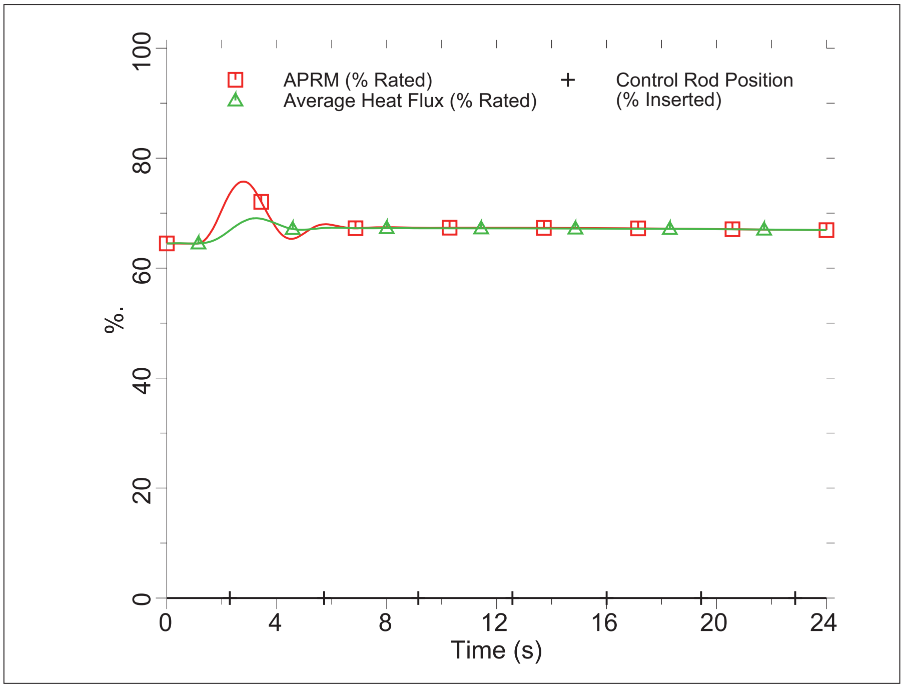


Figure 15.4-1 Transient Changes for Control Rod Withdrawal Error During Startup



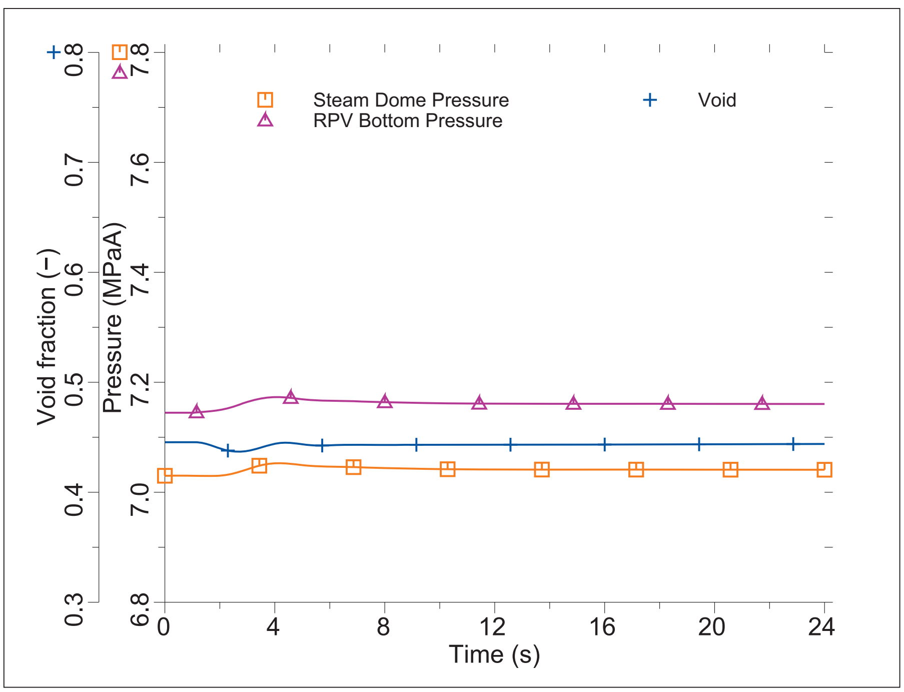


Figure 15.4-2a Fast Runout of One RIP

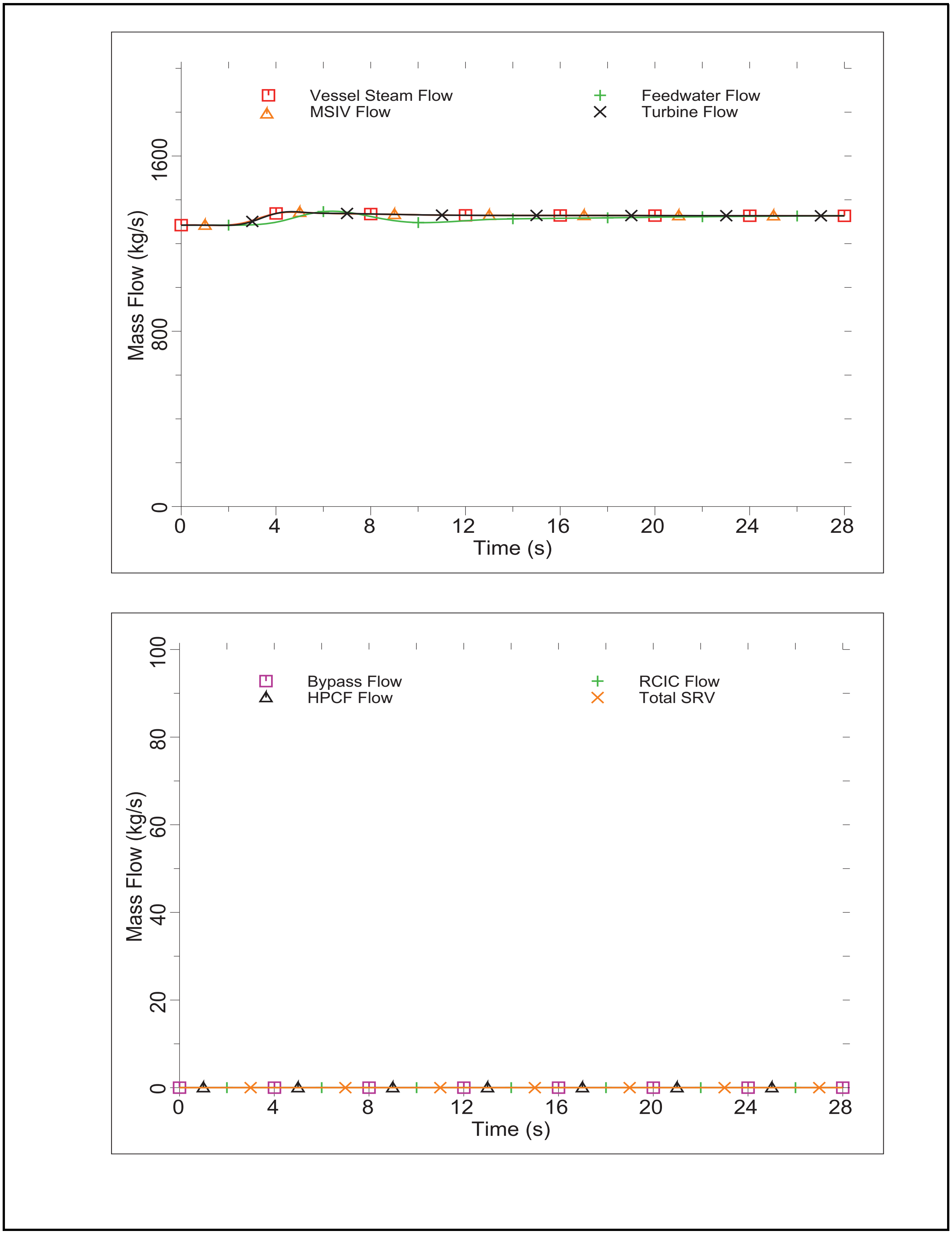


Figure 15.4-2b Fast Runout of One RIP

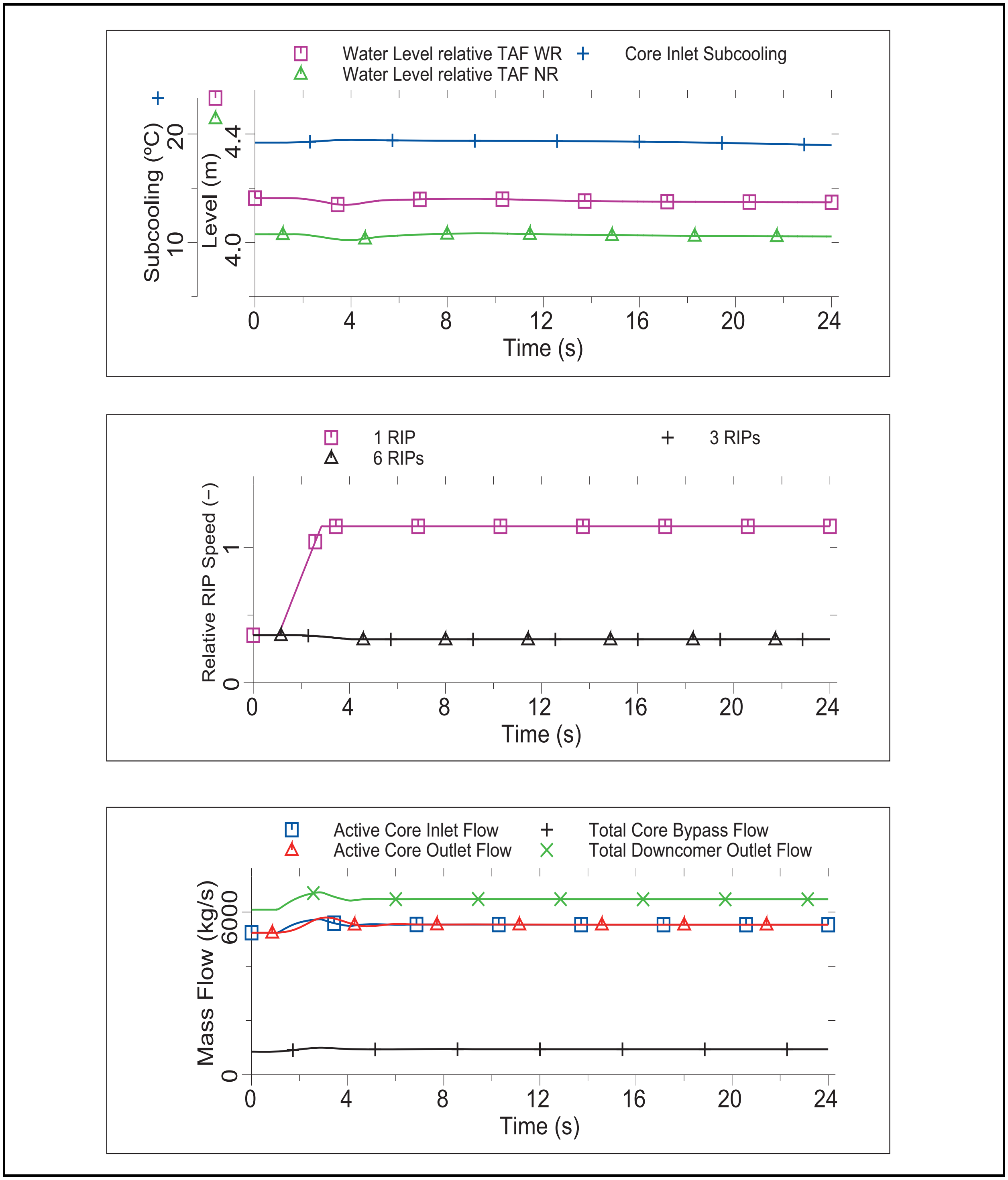


Figure 15.4-2c Fast Runout of One RIP

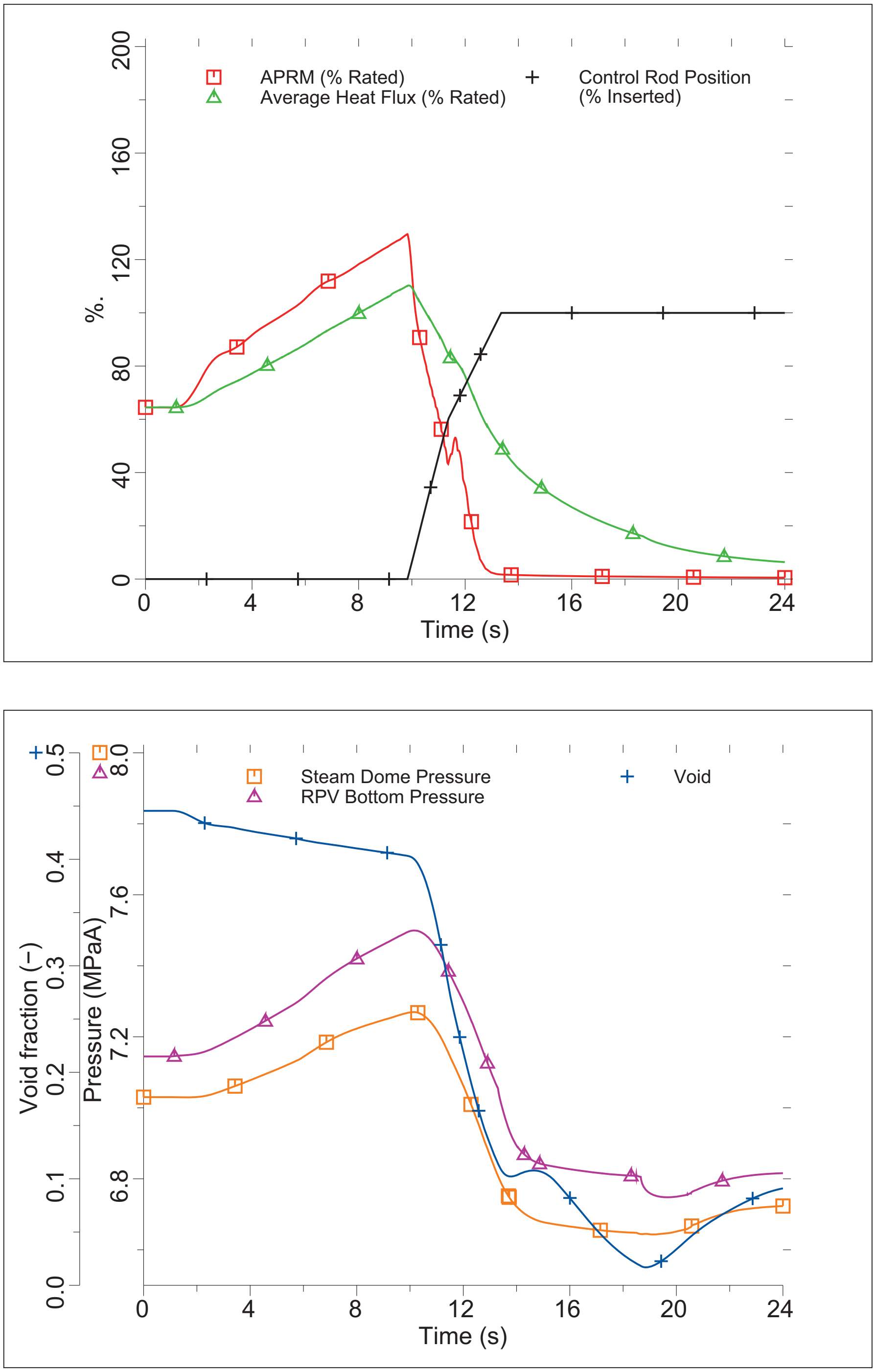


Figure 15.4-3a Fast Runout of All RIPs

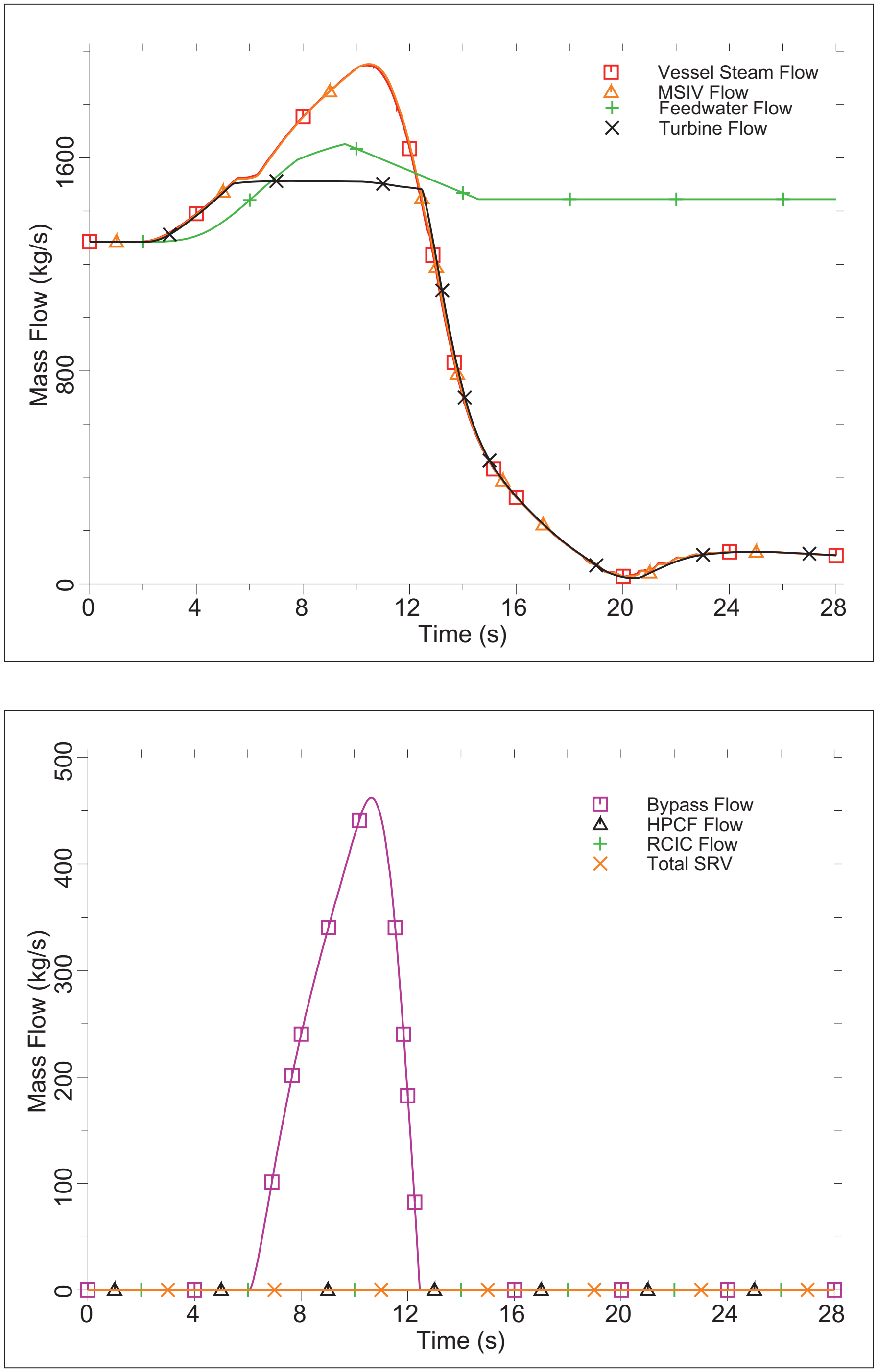


Figure 15.4-3b Fast Runout of All RIPs

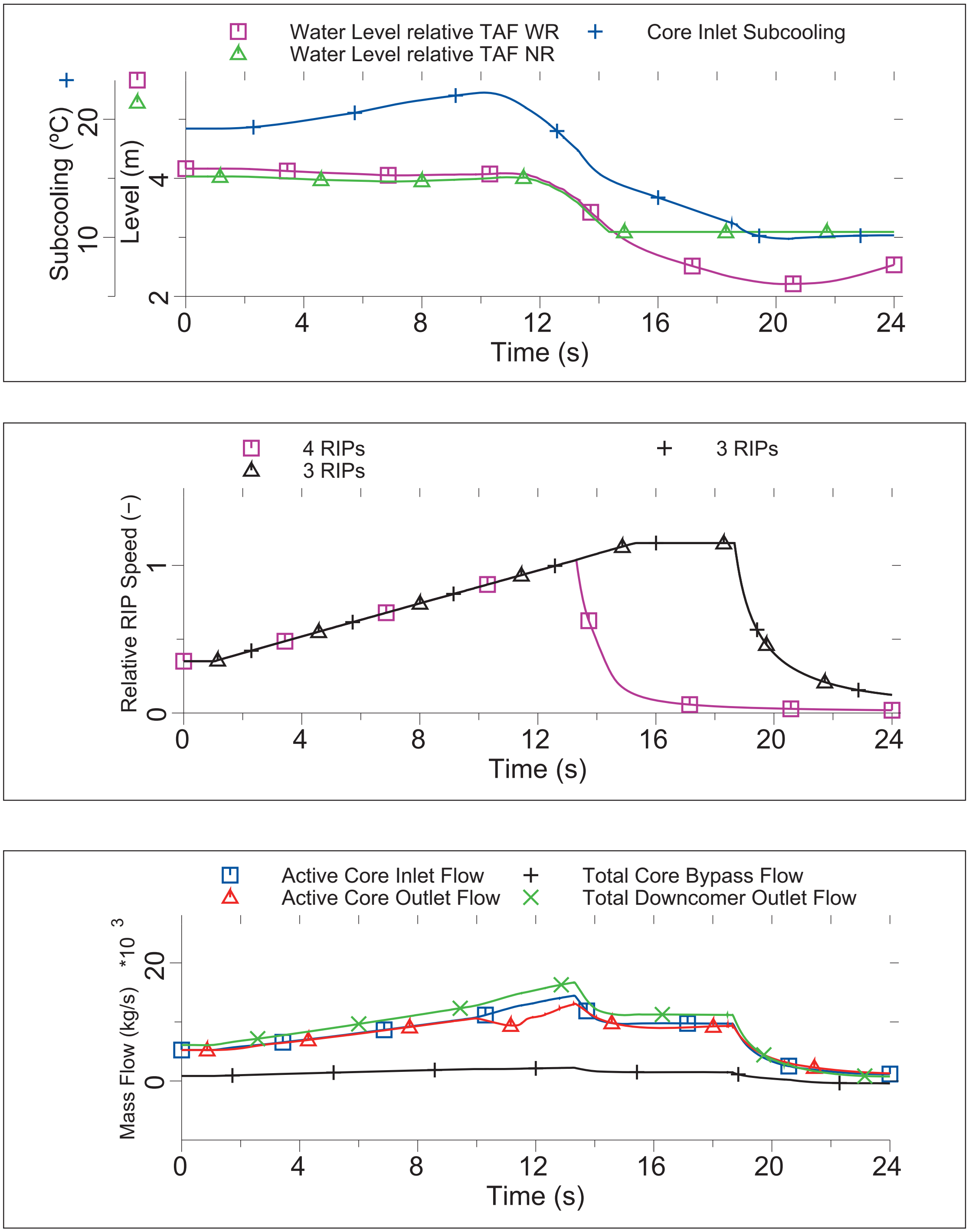


Figure 15.4-3c Fast Runout of All RIPs