General Electric Advanced Technology Manual

Chapter 5.1

Introduction to Transients

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5.1 INTRODUCTION TO TRANSIENTS

Learning Objectives:

1. Given a transient curve:
   - At identified points, explain what caused the parameter to change.
   - At identified areas of the curve, explain why the parameter is trending in that area.
   - State the cause of the transient (initiating event).

2. Given a plant transient scenario, explain the behavior of selected plant parameters, control systems, and equipment for the time designated in the scenario.

5.1.1 Introduction

The purpose of this manual chapter is to illustrate how BWR systems interact in response to abnormal events, equipment malfunctions and/or operator actions, and how specific plant parameters respond to changing plant conditions. It is important for NRC personnel to understand these relationships and expected system responses to ensure adequate review of licensee's analyses and to detect potential performance deficiencies. Following plant events, the licensees will obtain plant parameter information and evaluate that data to ensure the cause of the event is determined and any equipment deficiency or operator error is detected and corrected. Inspectors routinely review these evaluations as part of an event follow-up inspection (Inspection Procedure 71153) or a problem identification and resolution inspection (IP 71152).

In this manual chapter, specific parameters were chosen to demonstrate the plant response. Detailed discussion on each parameter and what affects it are contained later in this chapter. Also, only the first six minutes of the transients are included in the graphs of these parameters. The licensee's analysis will typically include additional parameters and time frames to fully understand the plant and operator response.

The parameter values are presented in the form of recorder graphs. These recorders are configured such that the pens move horizontally (left or right) depending on the value of the parameter while the chart paper moves vertically down across the pens onto a take-up reel as time progresses. Therefore, the parameter value at time zero will be at the bottom of the chart and each horizontal grid line moving upward represents a 30-second time period later than the previous grid line. Figure 5.1-1 shows a representation of this type of chart recorder. This technology is gradually being replaced with digital recorders that simulate the same type of format on the screen with the data recorded on a disk, instead of paper. These electronic recorders have the capability of displaying the data in different forms such as a bar chart or as digital numbers. Figure 5.1-2 shows a typical digital recorder configured with a "waterfall" display.

These parameter graphs were developed from the GE simulator which models a specific plant design which may have unique setpoints, system capabilities, and piping configurations. All transients are initiated from 100% rated thermal power, at a specific time in core life, and with no operator actions taken in response to the transient (some transients have manual action as the initiator of the event). The setpoints for automatic action on the GE simulator are listed in Table 5.1-1. These setpoints are not necessarily the same at each BWR reactor. Caution is advised when trying to apply these simulator curves to an operating BWR plant. Even relatively minor changes in set point values, power level, operating history, system capabilities, piping configuration or operator action could cause significant differences in the indicated plant responses.

During analysis and study of the curves, the student should concentrate on understanding changes in various parameters caused by the initiating event, subsequent automatic operation of associated systems, and system responses to the event.

5.1.2 Transients

In general, the term reactor transient applies to any significant deviation from the normal operating value of any of the key reactor operating parameters. There are normal operating transients that take place during a normal plant startup, shutdown, or load change. A manual power change is presented in this course as one of the transients. Other transients can occur as a result of unintentional equipment failures and/or external causes.

10 CFR 50, Appendix A defines an Anticipated Operational Occurrence (AOO) as “those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power”. These AOOs are further divided into categories as described in RG 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, and NUREG 0800, Standard Review Plan, in accordance with their impact on the plant. Table 5.1-2 contains a listing of typical AOOs that the licensee is required to evaluate as part of their Final Safety Evaluation Report (FSAR). The transients used in this course are taken from this list of AOOs. The transient graphs will be provided separately and will be discussed individually during this course. Student handouts will be provided at the end of each discussion describing the cause of each identified parameter inflection point and/or trend.

A third type of transient that is also required to be evaluated by the licensees, are those resulting from design basis accidents. Postulated accidents are unanticipated occurrences (i.e., they are postulated but not expected to occur during the life of the nuclear power plant). Examples of these include loss of coolant accidents (LOCAs), main steam or feedwater line rupture, rod drop accident or a recirculation pump seizure. These types of transients are not presented in this course.

The Standard Review Plan requires licensees to present certain parameters when analyzing these transients. The graphs used in this course contain a subset of these parameters which are readily available to the operators in the control room. Others are derived from complex computer modeling and are not provided (e.g., Critical Power Ratio, hot channel exit temperatures, void fraction, etc…). The parameters listed below are provided in the graphs and each will be discussed in detail in the subsequent sections of this manual chapter:

- Neutron power (“A” APRM in % of rated power)
- RCS pressure (narrow range pressure in psig)
- Reactor steam flow rate (sum or the four steam line flow detectors in Mlbm/hr)
- Turbine steam flow rate (turbine first stage pressure converted to steam flow in Mlbm/hr)
- Feedwater flow rate (sum of the two feedwater flow detectors in Mlbm/hr)
- Reactor vessel level (narrow range in inches)
- Total core flow (sum of the flows through the 10 jet pumps in Mlbm/hr)

5.1.3 Reactor Power

It is important to recognize that the neutron power as indicated on the average power range monitor (APRM) is what is presented as reactor power in the graph. Thermal power is a measure of heat flux which takes 6 to 10 seconds to impact coolant temperature in the core following a reactivity change. This difference between neutron power and thermal power is discussed further in section 5.1.3.4 of this chapter.

Reactor power changes when reactivity changes. As discussed in R304B, Reactor Physics, reactivity is affected by several factors. Reactivity can change when control rods are inserted or withdrawn, fuel is depleted, fission product poisons build in or burnout, burnable poisons burnout, etc… Also, reactivity can change when other parameters change such as the amount of voiding, fuel temperature, and/or moderator temperature. These reactivity coefficients are all negative for the power level and time in core life for the transients presented. Therefore, they provide inherent feedback to mitigate power excursions. That is, if power increases (due to some positive reactivity being added), voiding and fuel temperature will rise which will add negative reactivity and limit the power rise. Each of the major causes of power changes are discussed below.

5.1.3.1 Control Rods

Control rods are designed to allow individual rod movement using the Reactor Manual Control System (RMCS) or to rapidly shutdown the reactor during abnormal conditions from a Reactor Protection System (RPS) actuation. In addition, failures in the Control Rod Drive (CRD) system can result in inadvertent insertion or withdrawal of an individual control rod.

A reactor scram is characterized by a rapid insertion of all control rods which results in a rapid shutdown of the reactor. The RPS signals and their setpoints are listed in the Technical Specifications (TS) and the bases for those trip signals are discussed in the TS Bases. In many cases the cause of a scram can be detected directly from the parameter graphs. For the cases of high power, high pressure, and low level scram signals, the parameter is shown directly on the graphs. In the case of main steam isolation valve (MSIV) closure scram signal, the cause of the Group I actuation on low steam line pressure with the mode switch in Run, may be inferred from the reactor pressure graph. In the case of the Turbine Stop Valve (TSV) closure and Turbine Control Valve (TCV) fast closure scram signals from a turbine trip on level 8, the cause is directly reflected on the reactor vessel level graph. However, there are other scram signals which cannot be determined from the graphs provided. These include a manual scram, a scram on high scram discharge volume level, and a scram on high drywell pressure.

5.1.3.2 Void Fraction (Core Flow and Pressure)

The amount of voiding in the core has a significant impact on power. If an event causes voids to increase, the density of the moderator decreases. Neutrons must travel further while they slow down and are more likely to leak out of the core or suffer resonance capture. This adds negative reactivity causing power to go down. Conversely, if voids collapse, density of the moderator increases and positive reactivity is added. There are two major causes for changes in the amount of voiding: reactor pressure changes and core flow changes.

When core flow changes, the amount of time the moderator is in the core changes which directly impacts the amount of voiding. When total core flow goes down (i.e., from a recirculation pump speed decrease or trip), the water entering the core remains in the core longer. This allows the voids to increase which adds negative reactivity and lowers reactor power. When total core flow goes up (i.e., when recirculation pump speed is increased) the water entering the core moves through the core faster, thus lowering the amount of heat that is absorbed. This lowers the amount of voiding, adding positive reactivity and raising reactor power.

Changes in reactor pressure also have a direct impact in the amount of voiding. If an event causes an increase in pressure, the voids collapse causing an increase in reactor power. Likewise, a drop in reactor pressure allows the voids to expand, which adds negative reactivity and lowers reactor power.

5.1.3.3 Core Inlet Subcooling

Water in the core is converted from subcooled liquid to a vapor/liquid mixture. The dry steam is separated from the liquid and leaves the vessel through the main steam lines. The saturated liquid drains from the moisture separators and steam dryer returns to the vessel annulus region. At 100% power, core flow is approximately 77 Mlbm/hr and steam (and feedwater) flow is approximately 11 Mlbm/hr. Therefore, the flow from the separator/dryer drains is approximately 66 Mlbm/hr or 6 to 7 times the amount of steam flow leaving the vessel or feed flow entering the vessel. Steam produced in the vessel is directed to the main turbine where the energy is converted to work. The heat is rejected to the circulating water system in the main condenser and the resultant condensate is subcooled. The amount of subcooling is referred to as condensate depression. A portion of the steam used to drive the main turbine is diverted to the extraction steam lines and is used to pre-heat the condensate and feedwater in the feedwater heaters, before it returns to the reactor. The subcooled (yet pre-heated) feedwater entering the vessel mixes with the drains from the moisture separators and steam dryer which are at saturation temperature. This water then enters the core where it is heated to saturation temperature and converted to wet steam. The amount of subcooling of this water entering the core has a direct impact on the amount of energy (i.e., reactor power) required to produce the steam. If core inlet subcooling increases (i.e., colder water enters the core), the amount of power needed to produce the same amount of steam must also increase. Likewise a reduction in core inlet subcooling (i.e., hotter water entering the core) will require less power to produce the same amount of steam. There are several factors that can impact the amount of subcooling of the water entering the core. These include the temperature of the feedwater entering the downcomer, the relative amounts of feedwater flow to dryer/moisture separator drain flows in the downcomer and, more subtly, the amount of carry-under of steam/vessel level in the downcomer.

At 100% power, the feedwater entering the downcomer is approximately 420°F and the water in the hotwell is approximately 104°F. When power is reduced, the amount of turbine steam flow is reduced. This reduces the amount of steam being condensed in the main condenser and increases the condensate depression (assuming circulating water flow remains the same) of the condensate. Likewise, this lower turbine steam flow reduces the amount of steam taken from the turbine that is used for extraction steam and therefore reduces the amount of pre-heating. Both of these affects will lower the temperature of the feedwater thus raising the amount of core inlet subcooling. For example, if power is lowered from 100% to 60%, the hotwell temperature goes from 104° to 88° and the feedwater temperature goes from 422°F to 375°F. This change in feedwater temperature takes several minutes to reach equilibrium following the power change. The impact of this feedwater temperature reduction on reactor power will be reflected as a gradual increase in power over several minutes following the initial power reduction.

Other events can impact the temperature of the incoming feedwater. If high pressure coolant injection (HPCI) or reactor core isolation cooling (RCIC) initiates at power, cold water from the condensate storage tank is injected into the feedwater lines thus reducing the incoming water temperature, increasing the core inlet subcooling and raising reactor power. Due to the large temperature difference between the feedwater and the CST, this change in reactor power occurs more rapidly than reduction in feedwater temperature from lowering turbine steam flow discussed above.

A reduction in feedwater flow event can also impact core inlet subcooling. When flow is initially lowered (e.g., from a feedwater control system failure or reactor feedwater pump trip), the amount of colder feedwater mixing with the saturated dryer/moisture separator drains is reduced and therefore the core inlet temperature rises. This reduction in subcooling and resultant reduction in reactor power occurs more rapidly than the change in feedwater temperature from a change in turbine steam flow discussed above.

Reactor vessel level also has an effect on core inlet subcooling. As level in the downcomer lowers, the amount of steam carry-under increases which raises the temperature of the water in the downcomer. The impact of lower level on power is relatively minor until level nears the bottom of the dryer skirt (0”). Since the reactor will automatically trip at level 3 before carry-under becomes significant, this affect on power is minimal for the transients discussed in this course. It does become significant for an Anticipated Transient Without Scram (ATWS) event discussed in Chapter 4.2. During an ATWS event, one of the strategies used to reduce power is to terminate and prevent make up flow and intentionally lower level. This acts to reduce the amount of core inlet subcooling and reduce power until the reactor can be shutdown.

5.1.3.4 Fuel Temperature (Fuel Time Constant)

As fuel temperature rises, resonance capture increases as more neutrons have the energies susceptible to capture and the self-shielding effect of the fuel is reduced (i.e., doppler broadening). Thus, an increase in fuel temperature will add negative reactivity and a lowering fuel temperature adds positive reactivity. At steady state conditions, the fuel temperature is constant and the heat generated is removed from the fuel by the water/steam flow in the core. If positive reactivity is added, the fuel temperature will increase as more fissions are occurring and a new higher steady state fuel temperature will be reached. However, this change in steady state fuel temperature does not occur instantaneously. A step increase in reactivity (e.g., due to a pressure increase that reduces the voids) will change the fission rate immediately, but it will take approximately 6 to 10 seconds for the fuel to heat up and reach equilibrium at the new flux level. As the fuel heats up from this increase in fission rate, the fuel temperature increase will add negative reactivity which will act to lower the power level. Likewise if negative reactivity is inserted (e.g., by the insertion of a control rod), the fission rate will immediately lower, but it will takes several seconds for the fuel to cool down as the heat is transferred to the moderator. While the fuel is cooling down, positive reactivity is being added and power will increase. The amount of time needed for the fuel to reach steady state temperature is known as the fuel time constant. The impact of the fuel temperature coefficient or fuel time constant is reflected on the APRM reactor power graph as an overshoot in power from the new steady state power level. That is to say, neutron power will initially drop to a lower level when negative reactivity is inserted, but will then rise to a steady state level over a period of approximately 6 to 10 seconds while the fuel cools down. Likewise, neutron power will immediately increase when positive reactivity is added, but then lower over 6 to 10 seconds to a new equilibrium level while the fuel heats up. This apparent “overshoot” in neutron power is indicative of rapid reactivity changes if the reactor remains critical, and is caused by this fuel time constant effect.

5.1.3.5 Poisons

Reactor poisons are atoms in the core that absorb thermal neutrons. The amount of poisons impacts power. Burnable poisons, such as Gadolinium, are added to the core to allow additional fuel to be loaded and still maintain adequate shutdown margin. Xenon and Samarium are fission product poisons whose concentration impacts reactor power. However, the transients presented in the course are limited to a six minute time frame. Changes in the amount of burnable poisons and fission product poisons are not significant in this short time period.

However, the impact on reactor power from boron added to the core during an ATWS event by the standby liquid control (SLC) system would be apparent in a six minute period. Reactor power would decrease relatively slowly (compared with a scram) for several minutes following SLC initiation and would eventually shutdown the reactor. The affect of SLC injection on reactor power will be seen in the R624B portion of the BWR technology series and is not one of the transients presented in this course.

5.1.4 Reactor Pressure

The key to understanding how reactor pressure changes with other parameters or events is to understand that in a boiling water reactor pressure is determined by the relationship between steam production (i.e., reactor power) and steam demand. If steam production in the core exceeds the steam demand from the main turbine and other steam loads, pressure will go up. If steam production in the core is lower than the steam demand, pressure will go down. Therefore changes in either steam production or steam demand will directly impact reactor pressure. Two of the AOO categories listed in Table 5.1-2, Increase in Heat Removal by the Secondary System and Decrease in Heat Removal by the Secondary System, deal directly with changes in steam demand.

5.1.4.1 Changes in Steam Production Rate (Reactor Power)

The electro-hydraulic control (EHC) system is specifically designed to regulate reactor pressure based on changes in steam production. The EHC system is designed to automatically match the steam demand to the steam production. Figure 5.1-3 shows an overview of BWR control systems including EHC and Figure 5.1-4 is a schematic of the EHC Logic. The EHC system senses steam line pressure at the turbine inlet and compares is to a pressure setpoint normally set at 920 psig. As reactor pressure goes up, steam line pressure will also increase, but to a lesser extent due to the pressure drop in the steam lines and turbine control valves. Figure 5.1-5 shows the relationship between reactor pressure and steam line pressure as a function of reactor power (and EHC steam flow demand). When steam line pressure goes up, the difference between the sensed pressure and pressure setpoint will increase resulting in an increased EHC steam flow demand signal. This increased EHC steam flow demand will cause the turbine control valves (TCVs) to throttle open to increase the steam demand. For example, if positive reactivity is added and power increases, steam production will increase and at that moment, it will be greater than steam demand. Since steam production is greater than steam demand, pressure will increase. As reactor pressure increases, steam line pressure starts to increase. The EHC regulator will sense this increase and will throttle open the TCVs to raise steam demand to match the steam production and pressure will stabilize at a higher pressure. The same is true for a power drop where the EHC system will sense the drop in steam line pressure and throttle the turbine control valves closed to lower steam demand to the lower rate of steam production, but at a lower pressure than before the power drop.

When the turbine is off line, the EHC system operates as before but will throttle the turbine bypass valves open or closed rather than the turbine control valves.

5.1.4.2 Increase in Steam Demand

Events that increase steam demand include a safety relief valve (SRV) failing open and a turbine bypass valve (BPV) failing open. When a BPV or SRV initially opens, steam demand exceeds steam production and reactor and steam line pressure go down. The EHC system senses this lowering steam line pressure and throttles closed the TCVs to reduce the steam demand to match steam production. Note that the final reactor pressure will be slightly lower than before the event because there is less head loss when steam is diverted directly to the main condenser (BPV) or suppression pool (SRV) than when the steam is directed to drive the main turbine. This reduction in line losses in affect shifts the reactor pressure curve on Figure 5.1-5 down slightly. Operation of the high pressure coolant injection pump (HPCI) or reactor core isolation cooling (RCIC) pump where steam is diverted from the steam lines to provide the motive force for the pumps has a similar affect on pressure, but to a lesser degree.

The other event listed in Table 5.1-2 for Increase in Heat Removal by the Secondary System is an EHC system failure that increases steam flow. Should the input from the steam line pressure detector to an EHC regulator fail high, the resulting large error signal will be selected by the high value gate (HVG) in the pressure control unit of the EHC system (refer to Figure 5.1-4). The EHC steam flow demand signal will increase to the point where the TCVs are full open at the Load Limit and the BPVs will throttle open until they reach the Maximum Combined Flow Limit (MCFL) which is normally set at 115%. This will increase steam demand to greater than steam production and pressure will start to drop. Due to the failure of the steam line pressure detector, the EHC system cannot respond to the pressure drop and the TCVs will remain full open and BPVs will remain throttled open. Without operator action, pressure will continue to decrease until main steam header pressure drops to 825 psig and an automatic RPS actuation occurs on MSIV closure.

5.1.4.3 Decrease in Steam Demand

Events that decrease steam demand include main steam isolation valve (MSIV) closure and turbine trip. These events rapidly decrease steam demand and result in a very large pressure spike. As discussed in the reactor power section of this chapter, increase in pressure causes collapse of the core voids and increase in power. Since these events will result in a power spike that could threaten thermal limits, the reactor protection system automatically scrams the reactor when MSIVs or turbine stop valve first start to go closed instead of relying on the automatic scram signal from high reactor pressure or high APRM power,

The other event listed in Table 5.1-2 for Decrease in Heat Removal by the Secondary System is an EHC system failure that decreases steam flow. Should the input from the steam line pressure detector to the controlling EHC regulator fail low, the backup regulator error signal will be selected by the high value gate (HVG) in the pressure control unit of the EHC system (refer to Figure 5.1-4). Since there is a 3 psig bias applied to the pressure setpoint in the backup regulator, the error signal will drop by 3 psig causing the EHC steam flow demand signal to drop and the TCVs will throttle closed. As the TCVs throttle close, the steam demand will be less than steam production and reactor pressure and steam header pressure will start to rise. The EHC regulator will sense this rise in steam line pressure and start to throttle the TCVs back open until steam demand returns to the steam production rate but at a slightly higher stable pressure.

5.1.5 Turbine Steam Flow

As discussed in the reactor pressure section of this chapter, turbine steam flow is automatically controlled by the EHC system in response to changes in steam line pressure. Since this is a measure of the steam going to the main turbine, the maximum turbine steam flow is determined by the limiters in the EHC system (e.g., Load Limit, Load Set, etc…). Turbine steam flow will go to zero on a turbine trip or if steam flow from the reactor is isolated (i.e., a Group 1 isolation).

5.1.6 Reactor Steam Flow

Reactor steam flow is the sum of the flows through the four main steam line venturi-type flow restrictors. In affect it is the sum of the turbine steam flow, steam flow through the bypass valves and steam flow to other steam loads including the reactor feedwater pumps, steam jet air ejectors, off-gas system, etc… These miscellaneous steam loads are collectively referred to as "house loads". Reactor steam flow does not include steam from the reactor through the safety relief valves or used by the HPCI and RCIC turbines (refer to Figure 5.1-6).

Since the major contributor to reactor steam flow is normally turbine steam flow, it will change as turbine steam flow changes based on EHC system operation. When the turbine is off-line, reactor steam flow will be the sum of the flow through the bypass vales and the house loads. Reactor steam flow will only stabilize and remain at zero if the MSIVs are closed.

5.1.7 Feedwater Flow

Feedwater flow is the sum of the flows through the two main feedwater lines from the discharge of the reactor feed pumps (RFPs). It does not include the flow from HPCI, RCIC or reactor water cleanup systems that penetrate the feedwater lines downstream of the feedwater flow detectors (see figure 5.1-6). Two of the AOO categories listed in Table 5.1-2, Increase in Reactor Coolant Inventory and Decrease in Reactor Coolant Inventory deal directly with changes in feedwater flow.

The feedwater control system (FWCS) is designed to regulate RFP speed to maintain reactor vessel level at the master controller level setpoint. Figure 5.1-3 shows an overview of BWR control systems including FWCS and Figure 5.1-7 is a schematic of the FWCS Logic. The FWCS is normally maintained in automatic control and in 3-element control. As such, the inputs to the FWCS are narrow range level, reactor steam flow and feedwater flow. The FWCS takes the level input, modifies it based on any mismatch between steam flow and feedwater flow, and compares this modified level to level setpoint (normally 37") and adjusts RFP speed to maintain level at setpoint. The control logic is setup to modify the level signal by 12 inches for a 25% mismatch between feedwater flow and reactor steam flow. If feedwater flow exceeds steam flow, the FWCS adds to indicated level in anticipation of level rise. If feedwater flow is less than steam flow, the FWCS subtracts from indicated level in anticipation of a level drop.

3-element control allows the system to anticipate level changes from changes in steam flow or feedwater flow. To maintain level stable, the steaming rate must be equal to the feed rate. If there is an increase in reactor steam flow or decrease in feedwater flow, more water is leaving the vessel then is entering the vessel and, without any change in feed rate, level would start to drop. The FWCS anticipates this level drop and increases the speed of the RFPs to maintain level at setpoint. Likewise, a decrease in reactor steam flow or increase in feedwater flow would result in a level increase and FWCS anticipates this level increase and lowers RFP speed.

In general as long as a RFP is operating, changes to feedwater flow are the result of changes to one of the input parameters to the FWCS. An increase in level, increase in feedwater flow, and/or a decrease in reactor steam flow will cause the FWCS to decrease feedwater flow. A drop in vessel level, decrease in feedwater flow, and/or an increase in reactor steam flow will cause the FWCS to increase feedwater flow.

Other factors that could impact feedwater flow are rapid reactor pressure changes and setpoint changes. Since feedwater flow is a function of the differential pressure between the RFP discharge pressure and the reactor pressure, rapid changes in reactor pressure will impact feedwater flow until the FWCS has time to respond to changes in the other input signals. A pressure spike will cause flow to decrease and a rapid pressure drop will cause feedwater flow to increase. Also if level reaches level 3 following a scram, the level setpoint automatically goes to Level Setpoint Setdown value of 18”. This can impact feedwater flow post-scram.

5.1.7.1 Increase in Reactor Coolant Inventory

Events that increase reactor coolant inventory include HPCI initiation and FWCS failure that increases feedwater flow. A HPCI initiation adds cold water to the vessel that is not measured by the FWCS and will result in an increase in vessel level. FWCS will reduce RFP speed in response to the rising level. If the reactor remains on line, feedwater flow will stabilize at a level where the feedwater flow and HPCI flow equals the reactor steam flow. Level will be above setpoint to compensate for the mismatch between steam flow and feedwater flow seen by the FWCS.

FWCS failures that result in increased reactor coolant inventory include steam flow detector failing high, feedwater flow detector failing low and level detector failing low. For the case where the level detector fails low, there is no feedback to the FWCS as level increases and the turbine and RFPs will eventually trip on a level 8 signal. For the feedwater detector failing low, FWCS will sense the mismatch with steam flow and raise RFP speed. At 100% power, the mismatch would equate to 50% or 24” of modification to the actual level before the modified level would be restored to setpoint. This level is above level 8 and the turbine and RFPs would trip on high level. For the case where a steam flow detector fails high, level would increase and stabilize when the level increase, modified by the mismatch between steam flow and feedwater flow, returns to setpoint.

5.1.7.2 Decrease in Reactor Coolant Inventory

Events that decrease reactor coolant inventory include SRV lifting, FWCS failure that decreases feedwater flow and partial or total loss of feed. As discussed in the reactor pressure section of this chapter, when an SRV opens, it lowers steam line pressure and

EHC throttles the TCVs closed. The FWCS senses this drop in reactor steam flow and reduces RFP speed in anticipation of a level increase. As feedwater flow decreases, the actual steam flow leaving the vessel (reactor steam flow plus SRV steam flow) exceeds the makeup flow into the vessel and level starts to lower. As level lowers, FWCS will start to restore the RFP speed and will eventually return to its original value. Level will stabilize when the lower level, modified by the mismatch between measured steam flow and feedwater flow, returns to setpoint.

FWCS failures that result in decreased reactor coolant inventory include steam flow detector failing low, feedwater flow detector failing high and level detector failing high. For the case where the level detector fails high, there is no feedback to the FWCS as actual level decreases and the reactor will eventually trip on level 3 signal. For the feedwater detector failing high or the steam flow detector failing low, FWCS will sense the mismatch between feed flow and steam flow and lower RFP speed. Level will decrease and stabilize when the level decrease modified by the mismatch between steam flow and feedwater flow, returns to setpoint.

There are several trips associated with the reactor feed pumps. A loss of all feedwater (i.e., both RFPs trip) can be caused by a loss of vacuum, level 8 trip, loss of suction pressure or a loss of steam flow to drive the pumps (e.g., MSIV closure). When the RFPs trip the feedwater flow graph will rapidly go to zero. For a single RFP trip, the feedwater flow will drop rapidly and level will start to drop as the steaming rate exceeds the feeding rate. The FWCS will sense this lower feedwater flow and lower level and increase the speed of the running RFP. If power is low enough (or is reduced low enough by the recirculation flow control operational limiter) that a single feed pump has the capacity to makeup for the steam demand, level will return to setpoint and feedwater flow will equal reactor steam flow.

5.1.8 Reactor Vessel Level

Reactor vessel level is measured in the downcomer or annulus region of the reactor vessel. It is normally maintained at or restored to level setpoint by the FWCS. If the steaming rate or feeding rate changes, level will change. For example, is there is an increase in the amount of steam leaving the vessel without an increase in the amount of feed coming into the vessel, the actual inventory of water in the vessel will go down and level will drop. As discussed in the feedwater flow rate section of this chapter, FWCS senses changes in reactor steam flow, feedwater flow, and level and automatically adjusts the feedwater flow rate accordingly. However, because the level is measured in the annulus, rapid changes in the amount of voiding in the core and recirculation pump speed can significantly impact indicated level during transients without any actual change in water inventory in the vessel.

The recirculation pumps take suction on the annulus of the vessel and discharge to the jet pumps which induce additional flow from the annulus into the core. When the speed of the recirculation pumps decrease or if one or both recirculation pumps trip, the rate of removal of water from the annulus drops and level in the annulus will go up. Likewise, an increase in recirculation pump speed will cause more water to be removed from the annulus and level will go down.

Changes in void fraction inside the core also have a large impact on level in the annulus during transients. During steady state conditions, there is a certain amount of voiding in the core. This steam bubble formation tends to impede the flow of the liquid entering the fuel assembly. This is known as 2-phase flow resistance. If voiding is increased in the core, this flow resistance goes up as additional voids form and this reduction in flow causes the level in the annulus to go up. Likewise, a reduction in voids reduces this 2-phase flow resistance, which increases the flow from the annulus and causes level in the annulus to lower. Since recirculation pump speed changes impact the amount of voiding in the core as well as the amount of water directly removed from the annulus as described above, speed changes have a profound impact on level. In summary, events that cause a reduction in voiding such as events that cause a pressure increase, control rod insertion events and an increase in recirculation pump speed, will cause a rapid lowering in level. Events that cause an increase in voiding such as events that cause a pressure drop, recirculation pump speed decrease, and a single or dual recirculation pump trip, will cause a rapid rise in level.

5.1.9 Total Core Flow

During normal operation with recirculation pumps running, power is affected by changes in recirculation pump speed and changes in 2-phase flow resistance. If forced circulation is stopped (recirculation pumps trip), core flow is a result of natural circulation (NC) which is affected by a number of factors that will be discussed later.

5.1.9.1 Recirculation Pump Speed Changes

Following reactor startup and power ascension, power is normally controlled by adjusting recirculation pump speed using the recirculation flow control (RFC) system. Figure 5.1-3 shows an overview of BWR control systems including RFC and Figure 5.1-8 is a schematic of the RFC Logic.

Core flow can be change by manual recirculation pump speed change using the RFC controllers (master or individual pump M/A stations). This change in core flow is relatively gradual (compared with a recirculation pump runback or trip). Figure 5.1-9 is the Power-to-Flow (N/F) map which reflects the relationship between reactor power and core flow. At 100% power, core flow is at 100% and control rods are withdrawn to the 100% load line. A decrease in recirculation pump speed will cause power to lower to the level for the new core flow as plotted on the 100% load line. Power will not be exactly on the 100% load line since the graph was developed for steady state conditions and immediately following the power change, feedwater temperatures and fission product poison concentrations will not be at equilibrium. Should a single recirculation pump trip, core flow lowers to approximately 60% and reactor power will stabilize around 70% as reflected on the N/F Map.

As shown on Figure 5.1-8, the recirculation pump Operational Limiter will limit pump speed if level reaches level 4 following a loss of a RFP, condensate booster pump or condensate pump. The Operational Limiter limits recirculation speed to 40%, which equates to approximately 50% core flow with reactor power lowing to the level indicated on the 100% load line. The limiter reduces power to the point where the capacity of a single RFP can maintain level and avoid an unnecessary reactor trip on low level. The Minimum Speed Limiter will reduce recirculation pump speed to 30%, however the plant conditions that cause a Minimum Speed Limiter would almost always also result in a reactor trip. Therefore, the flow will drop to the around 30% core flow which is the operating point along the 2-pump minimum speed line at < 3% power (i.e., decay heat following the scram).

Recirculation pumps can trip on various signals, including manual trips, loss of power or over-current to the motor generator set, high lube oil temperatures, low lube oil pressure, etc… There are also two recirculation pump trips (RPTs) that are required by Technical Specifications and open the output breakers of the MG sets. The end-of-cycle RPT (EOC-RPT) causes the pumps to trip on a turbine trip signal when power is greater than 30% (as indicated by turbine first stage pressure). The Anticipated Transient Without Scram RPT (ATWS-RPT) trips the pumps at level 2 or at 1120 psig reactor pressure. These setpoints are listed in Table 5.1-1 and also in the Technical Specifications. There is no direct signal from an RPT to scram the reactor, but abnormal operating procedures require a manual reactor trip if the plant transient does not cause an automatic RPS actuation (e.g., from the level, power or pressure transient resulting from the RPT). Therefore, the final core flow rate reflected on the N/F Map will be along natural circulation line depending on the amount of decay heat following the scram.

5.1.9.2 2-Phase Flow Resistance

As discussed in the reactor vessel level section of this chapter, the amount of voiding in the core has a direct impact on 2-phase flow resistance in the core. An increase in voiding causes more bubble formation, increasing the 2-phase flow resistance, and lowering core flow. Likewise, void collapse reduces the voids, lowers the 2-phase flow resistance, and raises core flow. The impact of these void changes are very pronounced in level changes where a 200 hundred gallon change in the amount of water in the downcomer causes an inch change in vessel level. Changes in void fraction have much less impact on core flow since a 200 gallon change (or 1500 lbm) is insignificant compared with the 77,000,000 lbm/hr core flow rate at 100% flow. However, for very large changes in voiding (e.g., void collapse caused by a reactor scram with a concurrent high pressure spike) without a change in recirculation pump speed, changes can be detected on the total core flow graph.

5.1.9.3 Natural Circulation Flow

Once the recirculation pumps have stopped, flow through the core continues through natural circulation (NC). Natural circulation is established due to the differences between the force exerted by the water in the downcomer (height of the water column times the density of the water) and force of the water (liquid and steam mixture) inside the core column (e.g., height time the density of the water in the core, in the upper plenum or shroud head, and in the steam separators/dryer).

The energy in the reactor core heats the water that enters from the lower plenum, transforming it into a water/steam mixture. As the liquid travels upwards through the core it is heated to saturation and some of the liquid is converted to steam until at the core exit it is a saturated water/steam mixture with a certain percentage of voids. This vapor/water mixture travels through the shroud head towards the steam separators where a centrifugal force separates the steam from the water. The saturated water, returns to the downcomer while the wet vapor travels upwards to the dryer. Further separation of the water from the steam occurs in the dryer with the steam exiting the vessel through the steam lines and the saturated liquid returning to the downcomer. The subcooled feedwater enters the vessel through the feed ring or sparger in the downcomer where it mixes with the saturated water from the steam separator and dryer drains. The resulting mixture, which is slightly sub-cooled, travels through the downcomer to the lower plenum where it enters the core and the cycle is repeated.

In general, NC flow is driven by the thermal driving head (TDH) between the water in the annulus and the water in the core column. The higher the density times the height of the water in the annulus, compared with the density times the height in the core column, the higher the flow. The water in the core column is both liquid and vapor such that the density of the water in the core column is highly dependent on the vapor fraction. Because NC involves 2-phase flow and due to the fact that the system is an open loop with feedwater coming in and steam leaving, NC flow is impacted by a number of factors including amount of decay heat, the amount of feedwater flow and the temperature of the incoming feedwater, the amount of voiding in the core, and the level in the downcomer. Complex computer models have been developed to calculate this flow under different conditions. A brief discussion of these factors is contained below. When analyzing the transients in this course, the student should be able to recognize when core flow is being driven by natural circulation only (i.e., that the recirculation pumps have tripped) and should understand that the magnitude of the flow is affected by a number of, often competing, factors.

5.1.9.3.1 Decay Heat

Following a reactor scram, the core will continue to generate heat as the fission products in the core at the time of the trip, decay away. The amount of fission products in the core is dependent on the power prior to the trip and the power history. The higher the power and the longer the core has been operating, the higher the decay heat. The most decay heat would be generated from a reactor that tripped from 100% rated thermal power after an extended run (e.g., two years of operation). For this power history, the decay heat being generated immediately following the trip would be approximately 7%, but would decrease exponentially as these fission products decay away. A thumb rule for the amount of decay heat being generated following a scram from 100% power versus the time since the reactor trip is: 6% power at 1 second after the trip, 3% power at 1 minute, 1,5% power at 1 hour and 0.75% at 1 day. Since this decay heat is the heat source that lowers the density of the water column in the core column and thus establishes the thermal driving head, NC flow, with all other factor being stable, would decrease exponentially as decay heat decreases following the trip.

5.1.9.3.2 Void Fraction

The TDH is directly impacted by the density of the water/steam mixture in the core/chimney/moisture separator area. The density is highly dependent on void fraction. That is, as voids increase, the density lowers and NC flow goes up. Likewise, as the void fraction goes down, the density goes up and, NC flow goes down. Reactor pressure will have an impact on NC flow based on its impact on void fraction. The core flow when pressure is being controlled on the SRVs at 1115 psig will be lower than core flow when pressure is being controlled on the turbine bypass valves at 920 psig (i.e., fewer voids and greater density of the water in the core column at the higher pressure).

5.1.9.3.3 Feedwater Flow Rate and Temperature

The density of the water in the annulus has a direct affect on the TDH. All other factors being equal, the colder the feedwater, the denser the mixture of feedwater and separator drains is in the downcomer, the higher the TDH, and the higher the core flow will be. Conversely, if feedwater flow is reduced or lost, the water in the downcomer will heat up reducing the density, lowering the TDH, and lowering core flow. However, the colder the water is that enters the core, the better able it is to cool the core and therefore reduce the voiding. As discussed above, the less voiding in the core, the higher the density in the core column and the lower the flow. Therefore, although the colder feedwater can raise the density of the water in the annulus, it can also raise the density in the core column and the overall impact is lower NC flow. An example of this affect is following a HPCI initiation. The cold water from the CST can initially raise NC flow as it increases the level and density of the downcomer water, however when this cold water enters the core, it drastically cools the reactor and causes the voids to collapse. This collapse of the voids greatly increases the density of the liquid/steam mixture in the core column and NC flow is reduced.

5.1.9.3.4 Level in the Downcomer

The thermal driving head is established by the difference in the height times the density of the water in the downcomer versus the height times the density of the liquid/water mixture in the core/shroud head/moisture separator area. Clearly, changing the level in the downcomer would impact the height of the water column in the downcomer, but other factors (e.g., amount of decay heat, void fraction, etc…) that can impact the density of the water/steam mixture in the core may have a greater affect on core flow. However, level in the downcomer has other impacts, which also affect natural circulation flow. As the level lowers to the level of the dryer skirt, more carry under of steam flow from the separators occurs into the downcomer. This increases the temperature of the downcomer water and lowers the density. Further reduction in level (at approximately -58”), will drop level to below the feed ring and allow steam to heat the incoming feedwater directly and again lower the density of the column of water in the downcomer. Also, lowering level is indicative of feedwater flow being less than steam flow. When feed flow is less than steam flow, the water coming into the downcomer is mostly from the separator drains at saturation temperature, which again raises the temperature and lowers the density of the column of water in the downcomer. This is especially important during at ATWS event when core heat is not limited to decay heat, but rather is a function of core power. The intentional lowering of level in the downcomer is a major strategy during an ATWS event to limit core flow and limit core inlet subcooling, both of which will reduce reactor power until the reactor can be shutdown.

5.1.10 Possible Strategies for Transient Analysis

As the student analyzes each transient and become more familiar with how these parameters and control system respond during these transients, they will develop their own techniques in evaluating the transient graphs. One possible approach is provided below. Students are encouraged to use whatever approach works best for them.

1. Do not try to immediately identify the initiating event.

2. Identify the first parameter that begins to change (or series of parameters that change at the same time. Due to how the charts are recorded, the first parameter to change is the lowest identified point or trend on the graphs.

3. Start with a parameter that you personally know the most about or are most comfortable with.

4. Make a list of what could cause the parameter of interest to change.

5. Start with the first item on the list and decide what direction and how much of a change you would expect; then look at the change on the curve and see if it is reasonable.

   Take reactor power as an example. The list of factors that could affect power would be similar to:

- Control rods (scram or individual rod movement)
- Change in voids due to recirculation pump speed change
- Change in voids due to pressure change
- Core inlet subcooling
- Fuel time constant
- Poisons

The expected direction and magnitude for some of these factors would include:

- A reactor scram would result in a rapid decrease in power to < 3%.
- A recirculation pump speed change would be reflected on the total core flow graph.
- A pressure change that causes a reactor power change (and not the result of the power change) would be reflected on the reactor pressure graph and EHC response would be reflected on steam flow graphs.
- A change in core inlet subcooling would also have a change in turbine steam flow or feedwater flow.
- A power change due to fuel time constant would be reflected on the reactor power graph as a power overshoot.

6. If you can show that the cause of the parameter change is not the one you are considering on the list or you are not sure, then continue down the list of causes until you find the possible cause.

7. Stay in the same time frame (i.e. do not continue on the same parameter trying to identify the causes of all the points during the transient on that graph prior to going to the next parameter).

8. Go to the parameter that is affected by the one you have chosen (i.e. power effects pressure, pressure effects turbine steam flow, reactor steam flow effects FWC, etc,).

9. Test to see if all points agree with the initiating event.

10. Move to the next time frame (next point or series of points higher on the graphs) and evaluate the points at this time frame using the process described above. Then continue the process until all points have been evaluated.

11. Use the “Rule of Arrows”. That is, where two arrows are close together on the graph of a parameter, the first point (occurred first or the lower of the two on the graph) will be a slight or subtle change and the next point will be the more significant change. For example, if power starts to decrease slightly (first point) and then rapidly drops (second point), the second point is the significant affect (scram or recirculation pump trip or runback) while the first point identifying the slight decrease would more likely

be a change in core inlet subcooling or slight pressure change.

5.1.11 Summary

The purpose of the subject is to provide NRC personnel with the skills needed to evaluate plant parameters following abnormal events for anticipated operational occurrences. It reinforces the concepts presented in the BWR technology course on how systems interact in a BWR.

The chapter presents seven plant parameters and details that can cause each to change. A summary is listed below:

Reactor power:

- Control rods (scram or individual rod movement)
- Change in voids due to recirculation pump speed change
- Change in voids due to pressure change
- Core inlet subcooling
- Fuel time constant
- Poisons

Reactor pressure:

- Change in steam production (change in power)
- Increase in steam demand (SRV/BPV fails open, EHC fails high)
- Decrease in steam demand (turbine trip, MSIV closure, EHC fails low)

Turbine steam flow:

- EHC throttling TCVs in response to steam line pressure
- Immediately goes to zero on a turbine trip
- Rapidly goes to zero on a scram or MSIV closure

Reactor steam flow:

- Sum of turbine steam flow, steam flow through the bypass valves, and house loads
- Only stabilizes at zero on a MSIV closure

Feedwater flow:

- Response to FWC system inputs (reactor steam flow, feedwater flow and level) to maintain or restore modified level to setpoint.
- Lose feedwater flow on RFP trip (including level 8, and loss of vacuum) or loss of steam to drive the pumps (MSIV closure)
- Rapid pressure drop in RPV will increase ΔP from RFP discharge and increase flow.

Reactor vessel level:

- Difference between makeup flow to the vessel and steam flow leaving the vessel.
- Normally restored to setpoint by FWC system (change in makeup flow rate)
- Rapid increase due to increasing voids (pressure reduction or recirculation pump trip or runback)
- Rapid decrease due to void collapse (scram or pressure increase)

Total core flow:

- Recirculation pump speed changes (RFC system operation and RPTs)
    - Manual speed change (gradual)
    - Single recirculation pump trip (60% core flow)
    - Operational Limiter (50% core flow)
    - Minimum Speed Limiter (30% core flow)
    - Dual recirculation pump trip, including EOC-RPT or ATWS-RPT (8 to 10% core flow with natural circulation flow affected by a number of factors)
- Change in 2-phase flow resistance due to large void collapse or increase (without a speed change)

Each of transients will be presented individually during the course. Handouts will be provided at the end of each lecture indentifying the cause of the identified inflection points and trends. During the examination, all the figures and tables contained in this chapter will be provided to the students to help evaluate the points of interest. The examination will only include questions on the transients covered during the course. Strategies have been presented in this chapter, and will be reinforced during the individual transient lectures, on how to approach transient analysis.

Table 5.1-1 Parameter Setpoint Aids

Reactor Vessel Level (inches)	
Level 8 (56.5)	Trip of main turbine, RFP, RCIC, **AND** HPCI
Level 7 (40.5)	High level alarm
Level 4 (33.5)	Low level alarm, **AND** permissive for Recirculation pump runback to 40%
Level 3 (12.5)	Reactor scram, Recirculation pump runback to 30%, ADS signal, RHR SDC Isolation signal **AND** FWC Setpoint Setdown
Level 2 (-38)	Initiate RCIC and HPCI, start RBSVS, RWCU **AND** other selected systems isolations, ATWS- RPT **AND** ATWS-ARI
Level 1 (-132.5)	Initiate CS **AND** LPCI, Start EDG, ADS signal, **AND** Isolate MSIVs
Reactor Pressure (psig)	
1120	ATWS - RPT **AND** ATWS-ARI
1115/1125/1135	4/4/3 SRVs Safety Mode Opening Pressures
1043	High Pressure Reactor Scram
1025	High Pressure Alarm
338 & 465	Permissive for LPCI **AND** CS Injection valve opening on a LOCA
310	Reactor Recirculation loop discharge valves close during a LOCA
125	RHR SDC Isolation clears
50 & 100	RCIC & HPCI low pressure isolations
Main Steam Line Pressure (psig)	
825	Closes MSIVs if in RUN mode
Condenser Vacuum (inches of Hg)	
25.0	Low Condenser Vacuum alarm
22.5	Turbine trip
20.0	RFP trip
8.5	MSIV closure
7	BPV closure
Turbine First Stage Pressure (%)	
30%	Bypasses EOC-RPT **AND** Reactor Scrams (due to TSV closure & TCV fast closure signals) when < 30% power as sensed by first stage pressure
Drywell Pressure (psig)	
0.75	High pressure alarm
1.69	LOCA Signal: SCRAM, Initiate HPCI, CS **AND** RHR, Start D/G, start RBSVS, **AND** Isolation signal for selected plant systems
FWCS Steam Flow (%)	
20%	RWM program restrictions bypassed at > 20% power as sensed by steam flow
30%	RWM Alarms bypassed at > 30% power as sensed by steam flow

Table 5.1-2 Types of Anticipated Operational Occurrences

Increase in heat removal by the secondary system
- EHC system failure that results in increased steam flow
- Turbine bypass valves fail open
- Safety relief valve lifting

Decrease in heat removal by the secondary system
- EHC system malfunction that results in decreased steam flow
- Generator load reject/Loss of off-site power
- Turbine trip
- Loss of condenser vacuum or condenser cooling
- MSIV closure

Decrease in RCS flow rate
- Recirculation pump trip (single and dual)
- Recirculation pump runbacks
- Recirculation Flow Control system failure that results in decreased flow

Increase in reactor coolant inventory
- HPCI Initiation
- FWC failure that results in increased feedwater flow

Decrease in reactor coolant inventory
- FWC failure that results in decreased feedwater flow
- SRV opening
- Loss of normal feedwater (total or partial loss)

Reactivity and power distribution anomalies
- Inadvertent control rod withdrawal or insertion
- Recirculation Flow Control system failure that results in increased flow
- Startup of an idle recirculation pump in a cold loop
- Loss of feedwater heating
- Operation with a fuel assembly in an improper position

Radioactive release from a subsystem or component
- Radioactive gas waste system leak or failure
- Radioactive liquid waste system leak or failure
- Radioactive releases due to liquid or gas storage tank failures

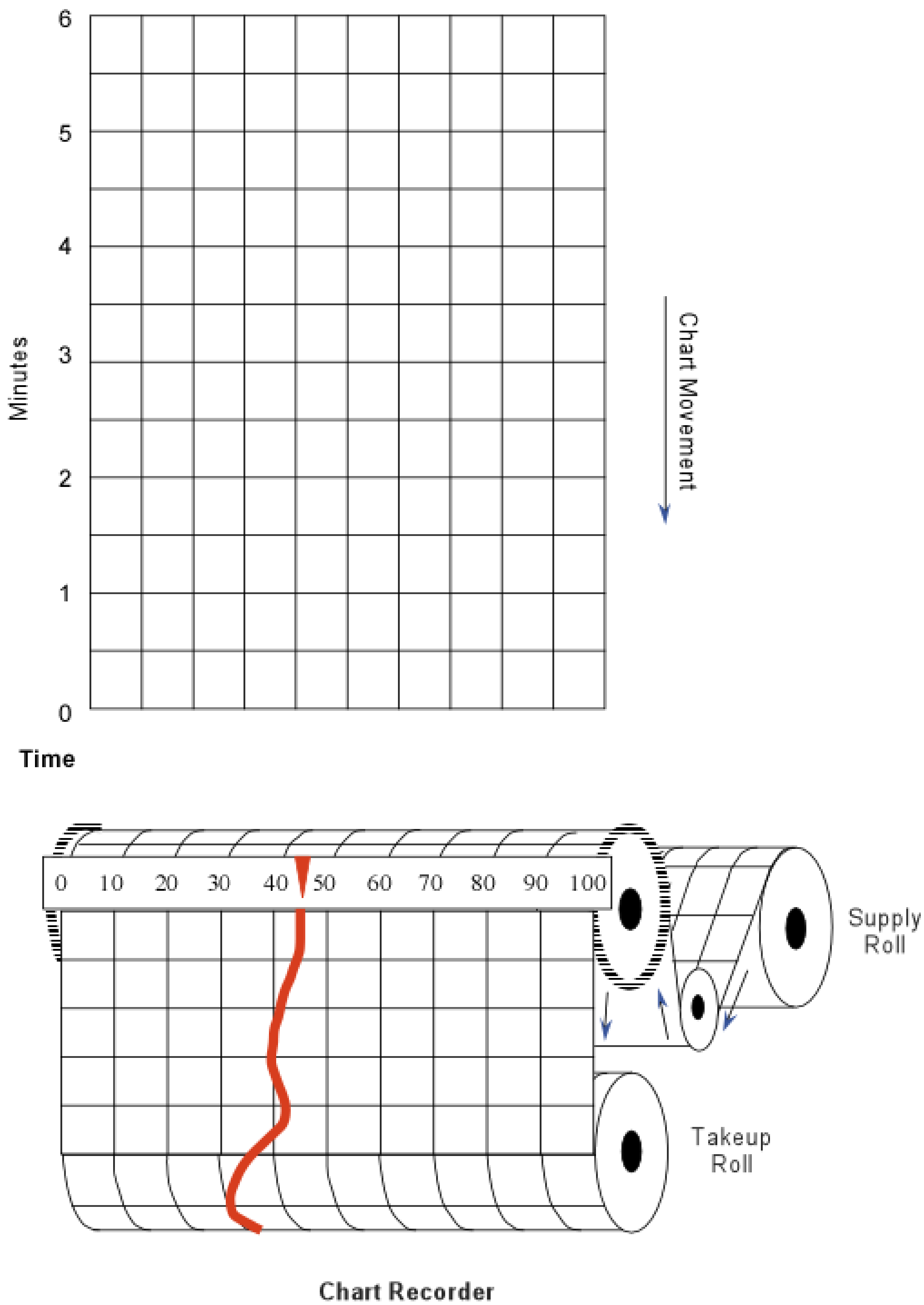

Figure 5.1-1 Chart Recorder

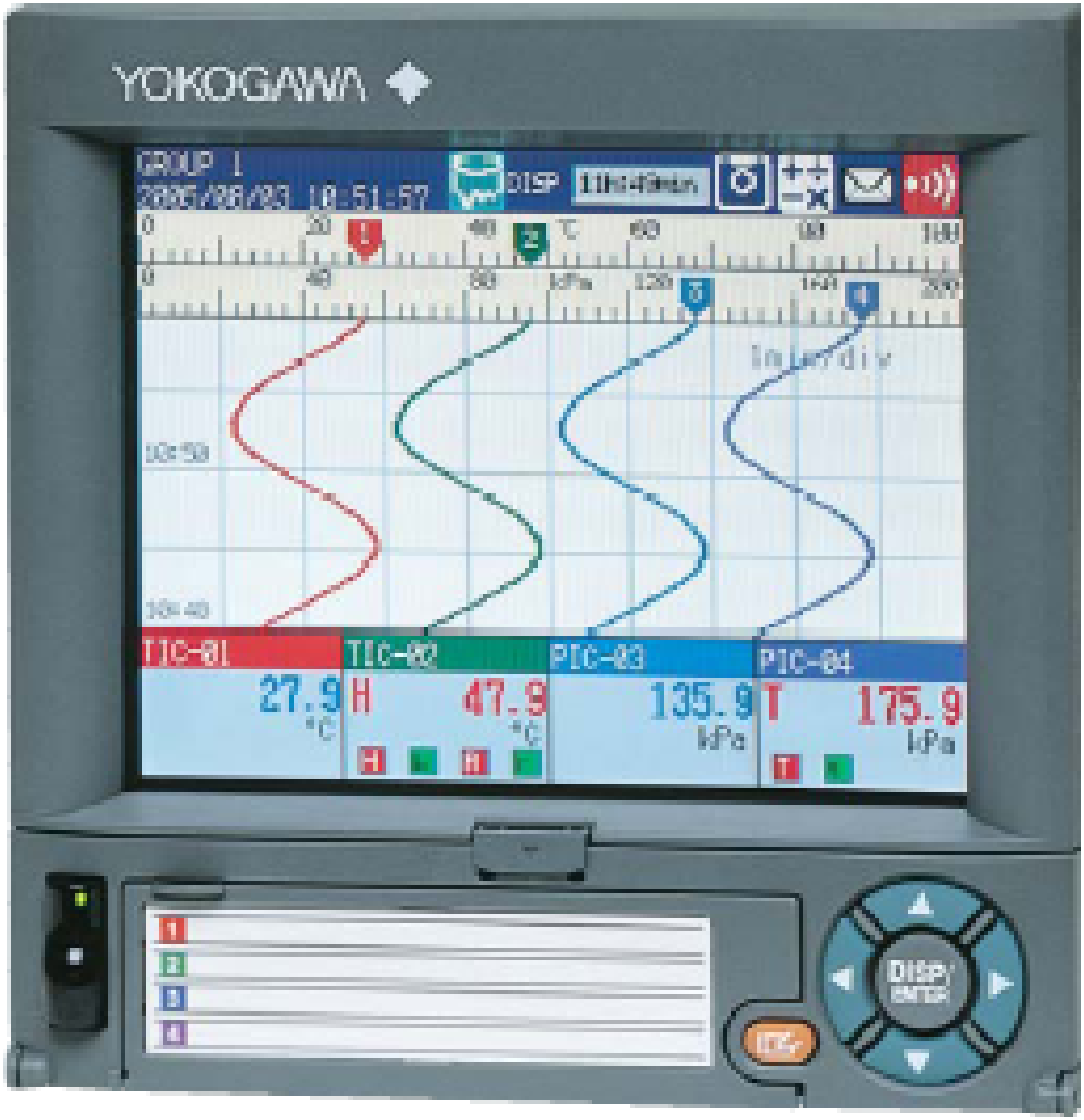

Figure 5.1-2 Electronic Recorder

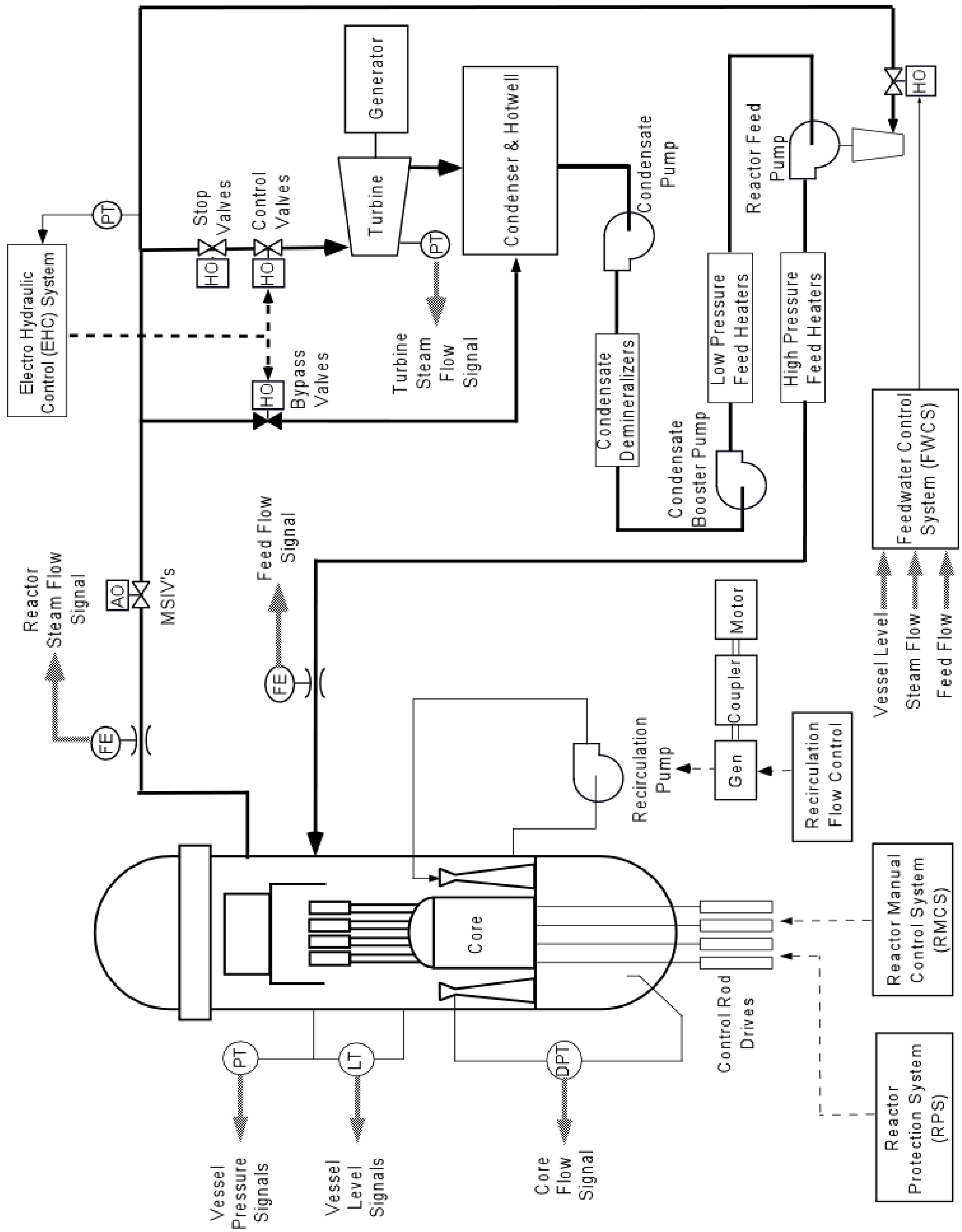

Figure 5.1-3 Process Instrumentation and Controls Summary

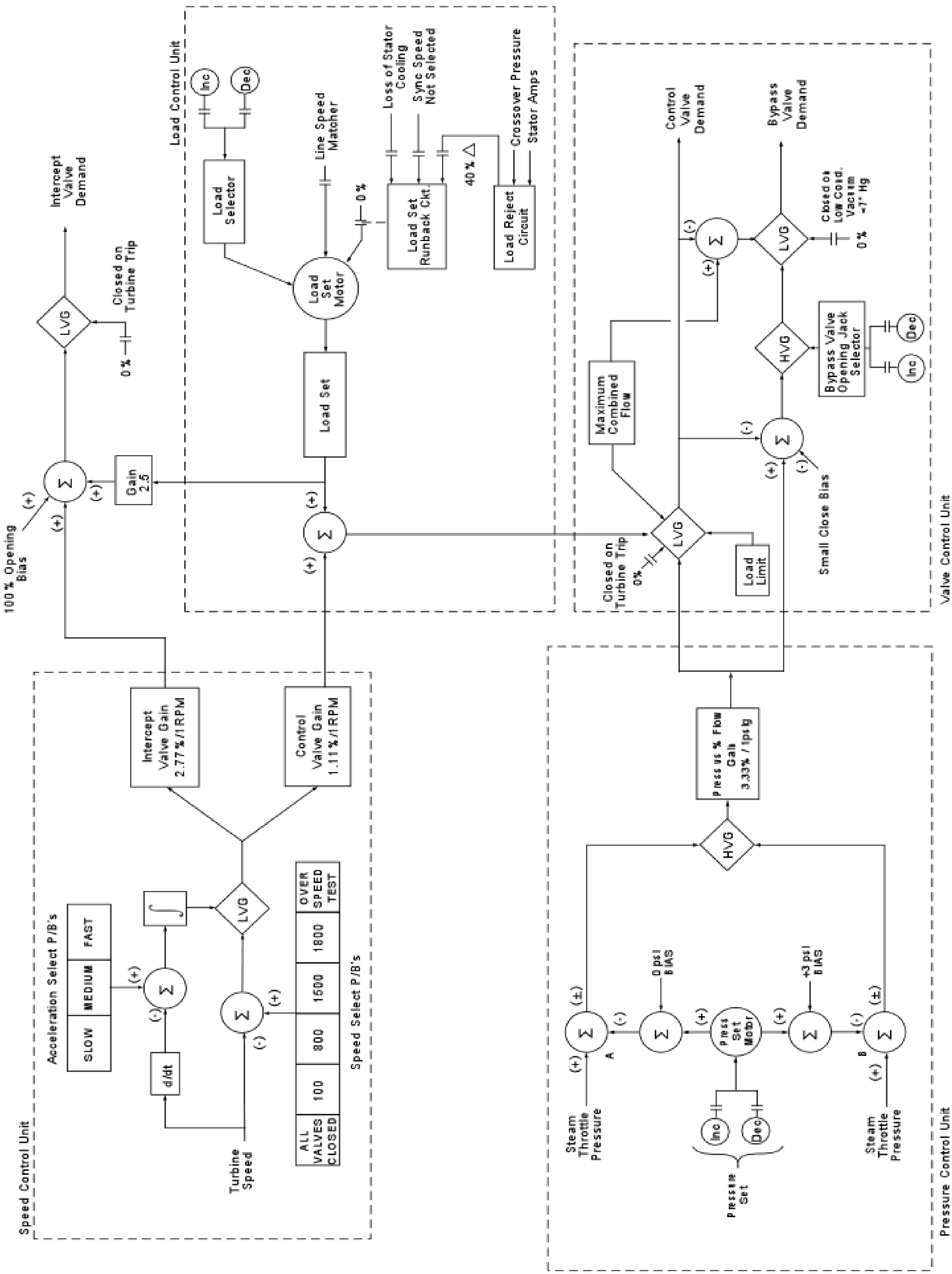

Figure 5.1-4 Electro Hydraulic Control System Logic

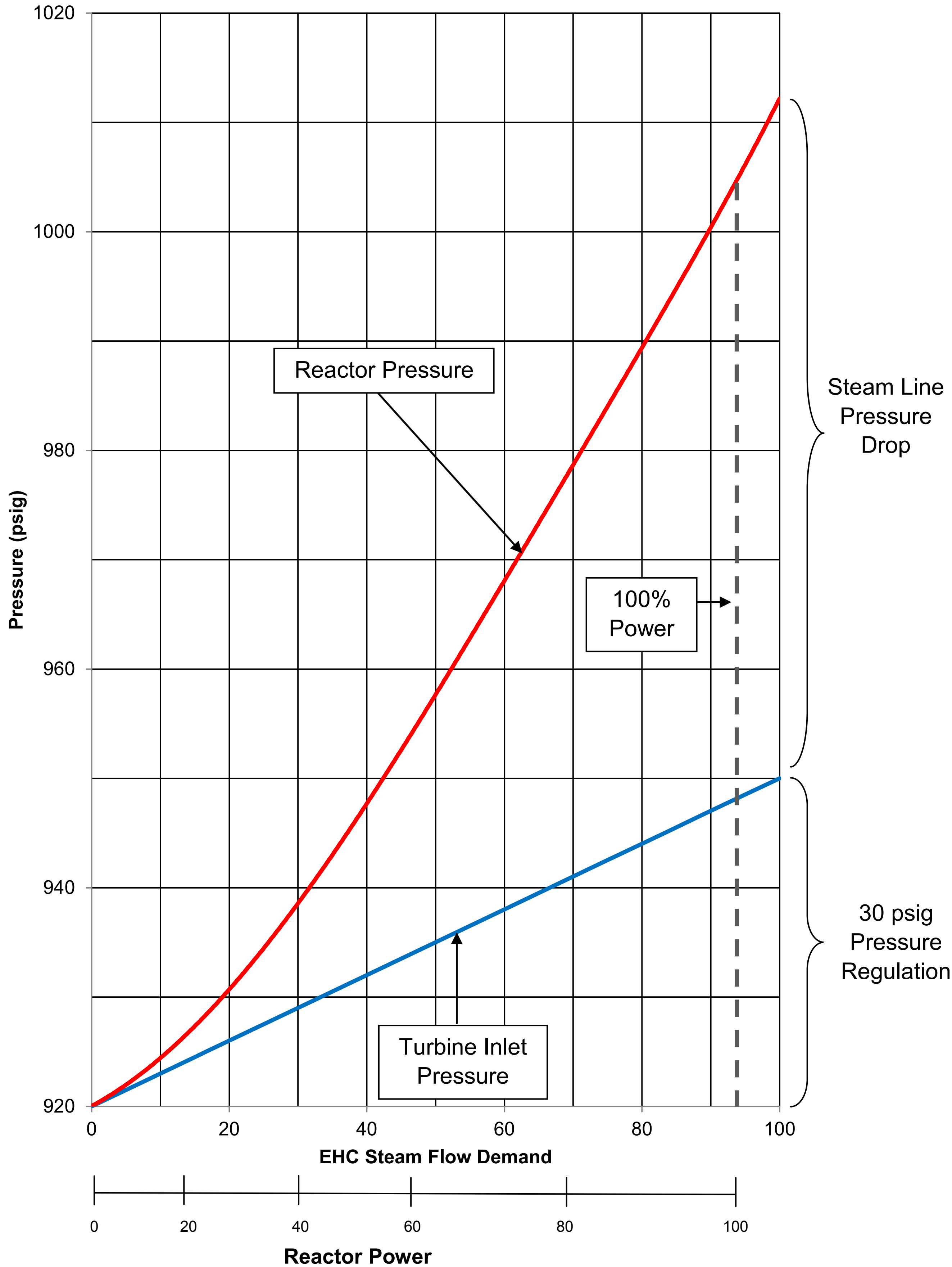

Figure 5.1-5 Pressure Control Spectrum

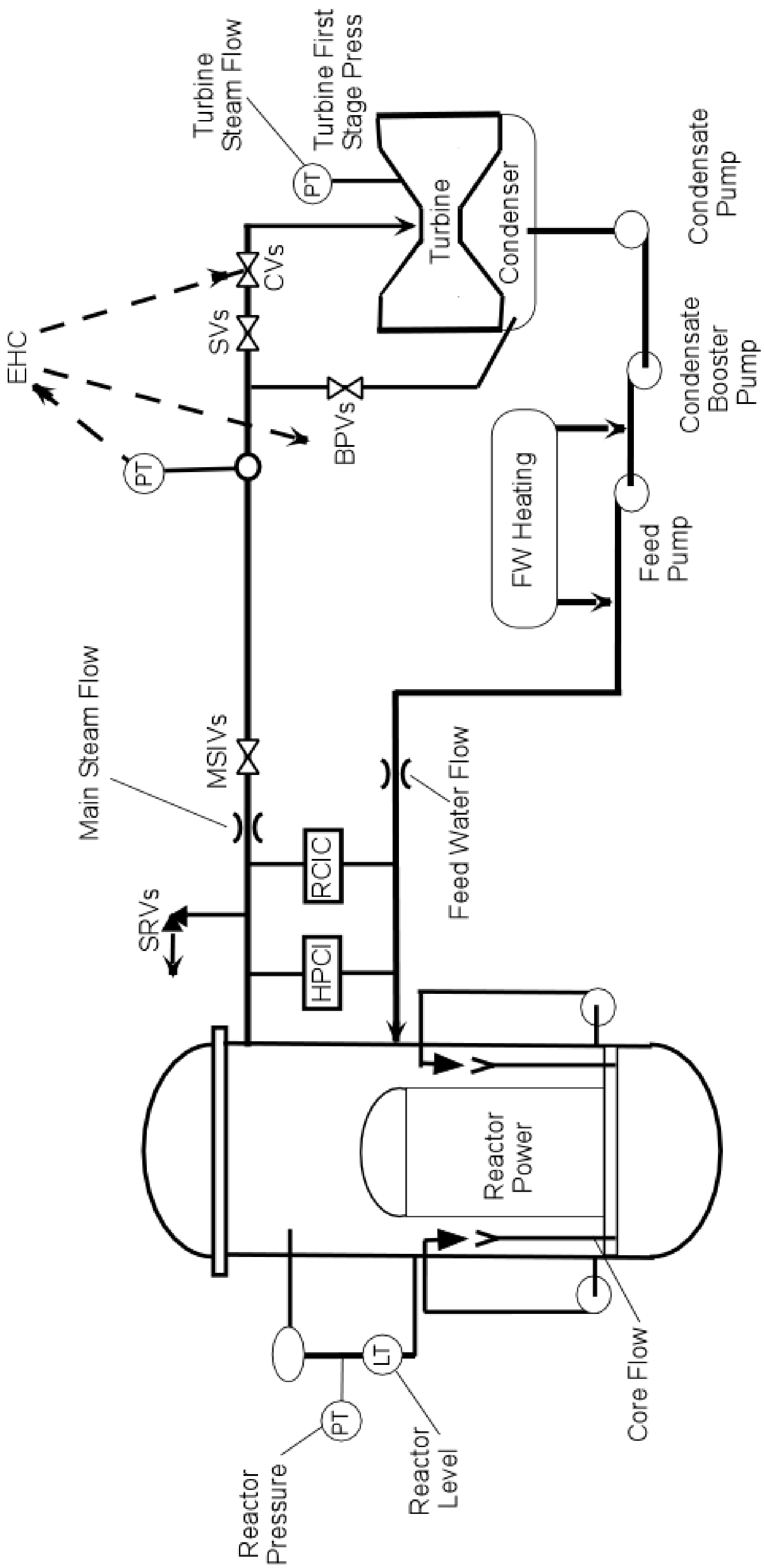

Figure 5.1-6 Balance of Plant

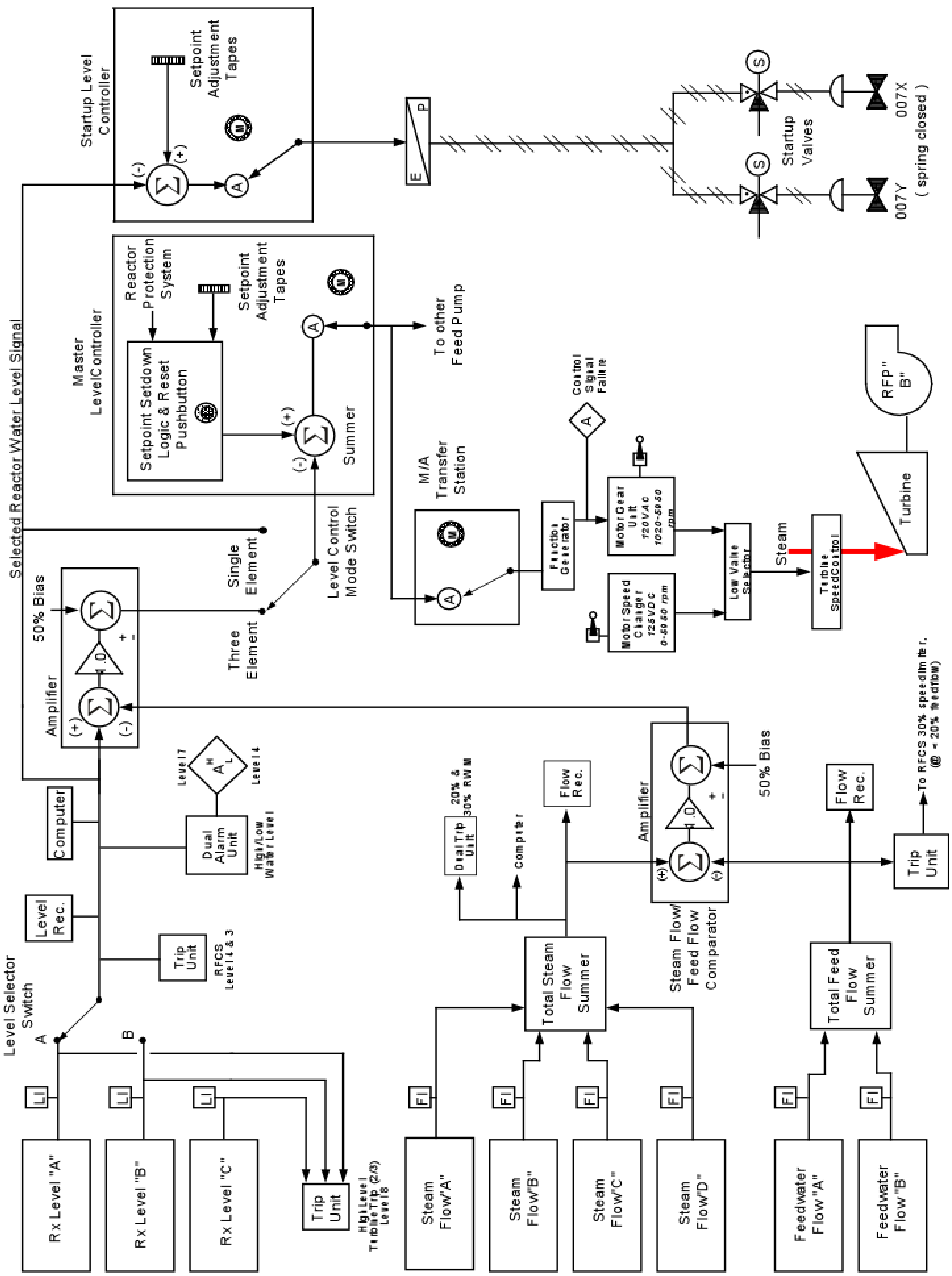

Figure 5.1-7 Feedwater Control System

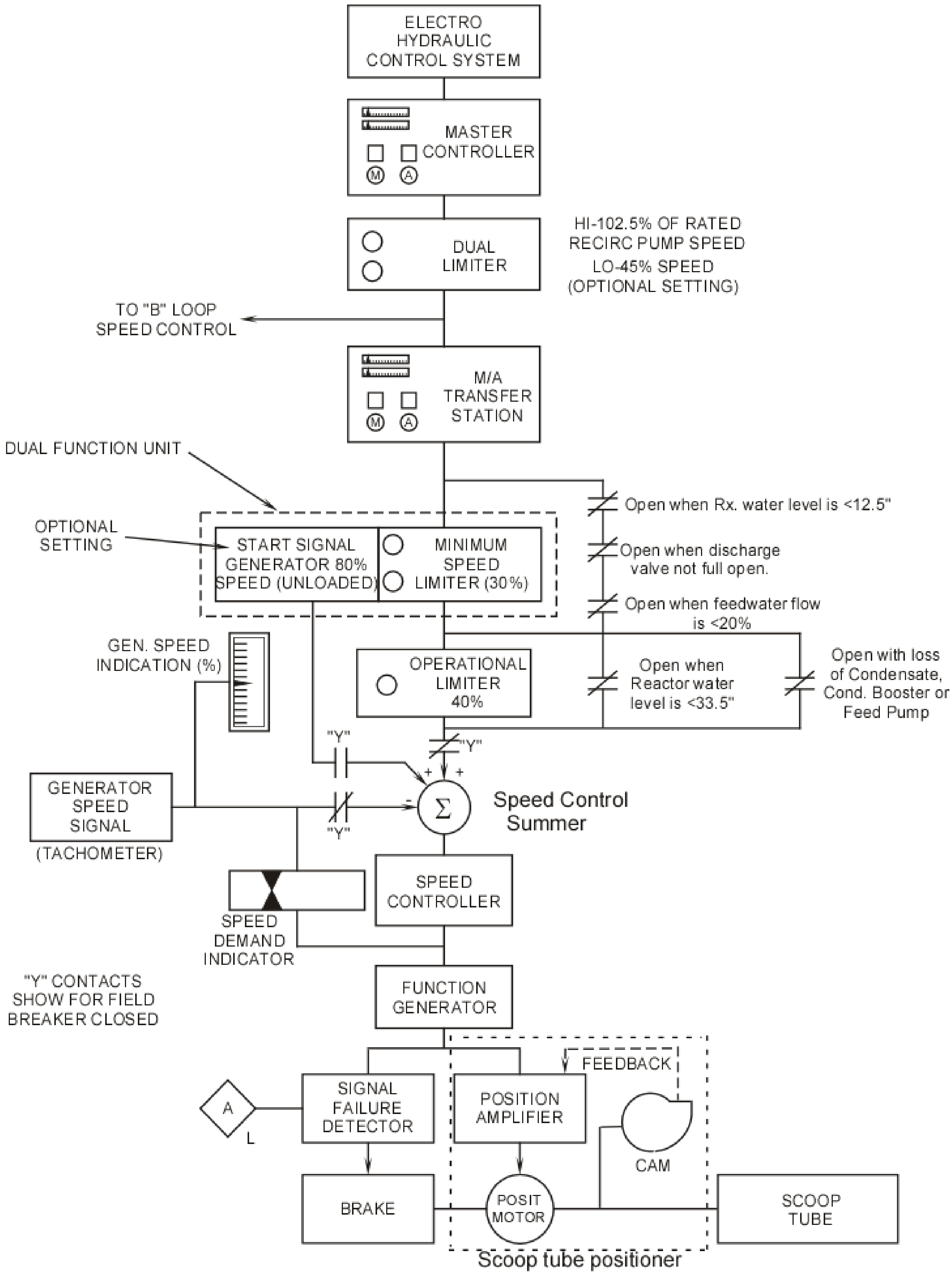

Figure 5.1-8 Recirculation Flow Control System

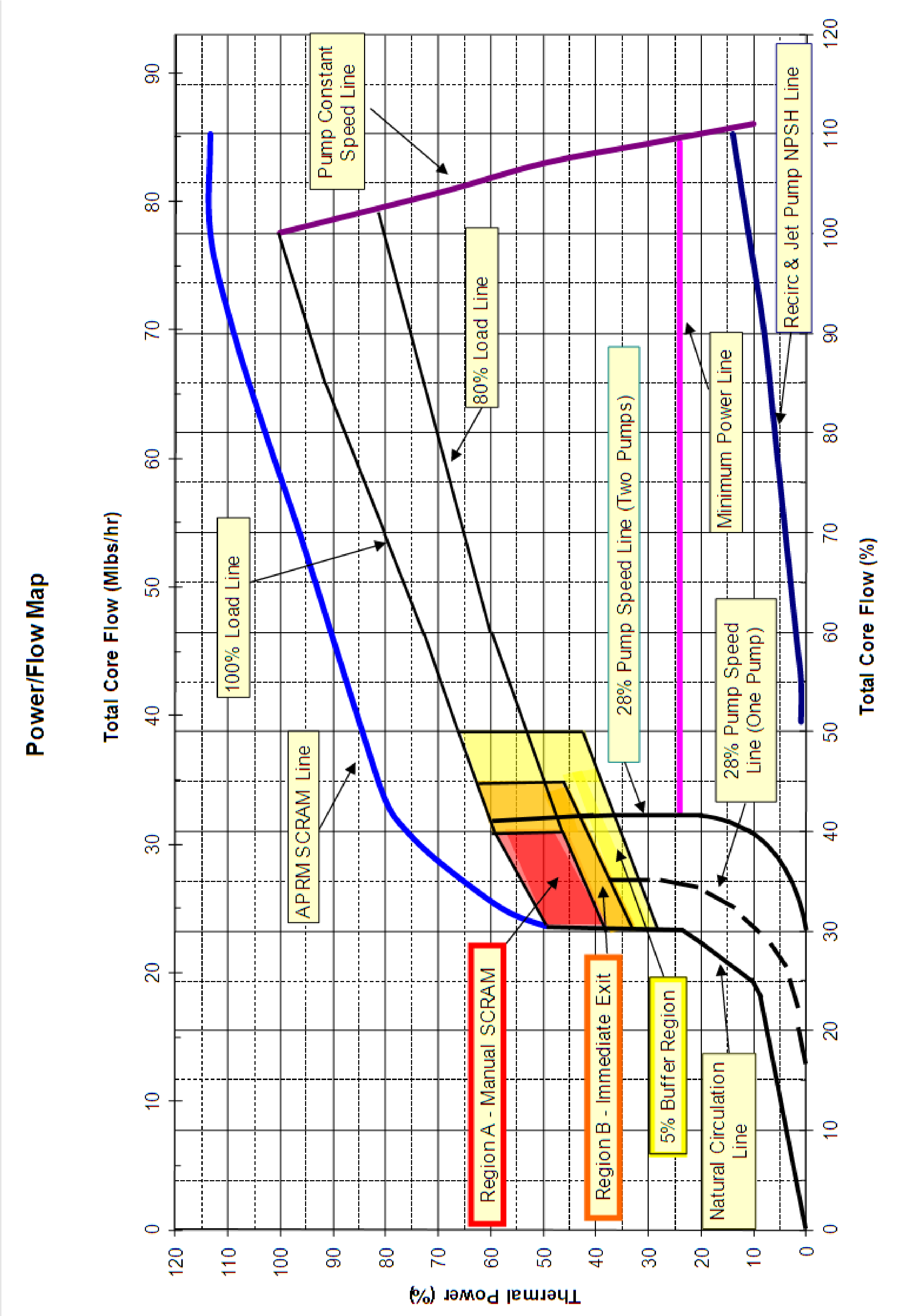

Figure 5.1-9 Power to Flow Map