CHAPTER 2: OVERALL UNIT CONTROL

MODULE A: NUCLEAR POWERT PLANT SAFETY

MODULE OBJECTIVES:

At the end of this module, you will be able to describe:

1. **The main energy conversions from fission to electricity in a nuclear generating unit;**
2. **State and explain the golden rule of reactor safety;**
3. **Define and explain the significance of the ALARA principle;**
4. **State what is meant by Defence in Depth, and describe the five parts of the Defence in Depth model;**
5. **Explain how each of the five barriers protect the public from fission products.**

1.0 INTRODUCTION

- The purpose of a nuclear plant is to allow the safe and economic conversion of the fission energy of fuel to electricity, with a minimum impact on the environment.

1.1 Conversion of fuel (mass) to electricity:

- input to the system is fuel, in the form of fissile material (Uranium);
- fission takes place in the reactor under suitable conditions: this ourse is restricted to thermal reactors, hence a moderator must be present;
- the energy liberated by fission appears in the form of heat: heat must be removed continuously to prevent fuel failure and the release of radioactivity;
- the heat is used to boil water as in any other thermal power plant: the steam generator is the primary heat sink for the energy relesed by the fuel;
- the steam energy is converted to kinetic energy by spinning a turbine;
- the kinetic energy of the turbine is converted to electricity in the generator;
- the output of the system is electricity to the electric power system.

1.2 Physical processes to be controlled:

- fission
- heat transfer
- flow
- pressure
- temperature
- change of state

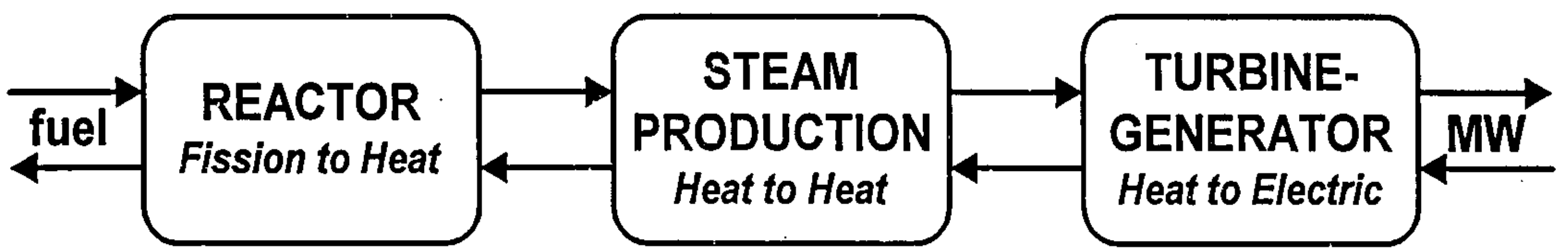


Figure 1: Main Energy Conversion Stages in a Nuclear Power Plant

2.0 SAFETY OF NUCLEAR POWER PLANTS

2.1 RADIATION HAZARD

- the unique safety concern with nuclear power plants is the exposure to radiation;
- the probable causes of radiation hazards are the same as for conventional hazards:
  - ⇒ design errors
  - ⇒ manufacturing flaws
  - ⇒ construction and installation mistakes
  - ⇒ operating and maintenance errors
  - ⇒ equipment failures.

2.2 ALARA Principle

- all hazards must be reduced to a level that is "As Low As Reasonably Achievable" this is known as the **ALARA Principle;**
- the main radiation hazard is from the fission products in irradiated fuel.

2.3 GOLDEN RULE OF REACTOR SAFETY

- there is a minimum risk to the public and the environment from reactor fuel, provided that at all times:
  - ⇒ the reactor power is controlled;
  - ⇒ the fuel is cooled;
  - ⇒ the radioactivity is contained.
- This rule is often shortened to **CONTROL, COOL, and CONTAIN,** and is referred to as the **GOLDEN RULE OF REACTOR SAFETY.**

2.4 SAFETY STANDARDS

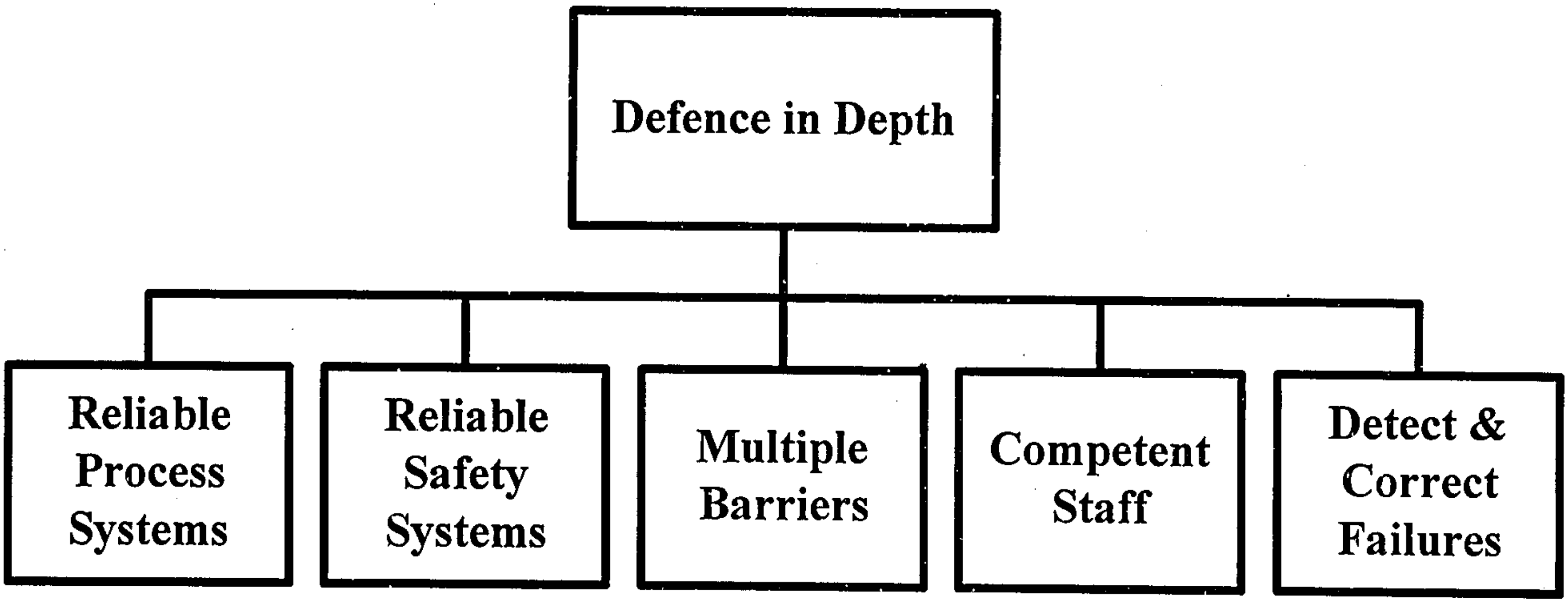
- employees at work should be safer than when they are not at work;
- employees at work in a nuclear generating station should be as safe as the average employee anywhere else in the same utility;
- employees at work in a nuclear generating station should be twice as safe as the average industrial worker in the same jurisdiction;
- employees at work in a nuclear generating station should be as safe as employees in all North American utilities.

2.5 NUMERICAL TARGETS

- employee fatalities < 2 per 100 million worker-hours;
- employee permanent disabilities < 2 per 100 million worker-hours;
- employee temporary disabilities < 0.4 per 200,000 hours worked;
- employee risk of disabling injury < 10 days lost per 200,000 hours worked.

3.1 DEFENCE IN DEPTH CONCEPTS

It is the principle way the Golden Rule (CONTROL, COOL and CONTAIN) is achieved in the design, construction, commissioning, operation and maintenance of a nuclear power plant.



3.2 Reliable Process Systems

They are the first line of defence, they operate continuously all the processes necessary to ensure that:

- **the fission is controlled,**
- **the fuel is cooled and**
- **radioactivity is contained**

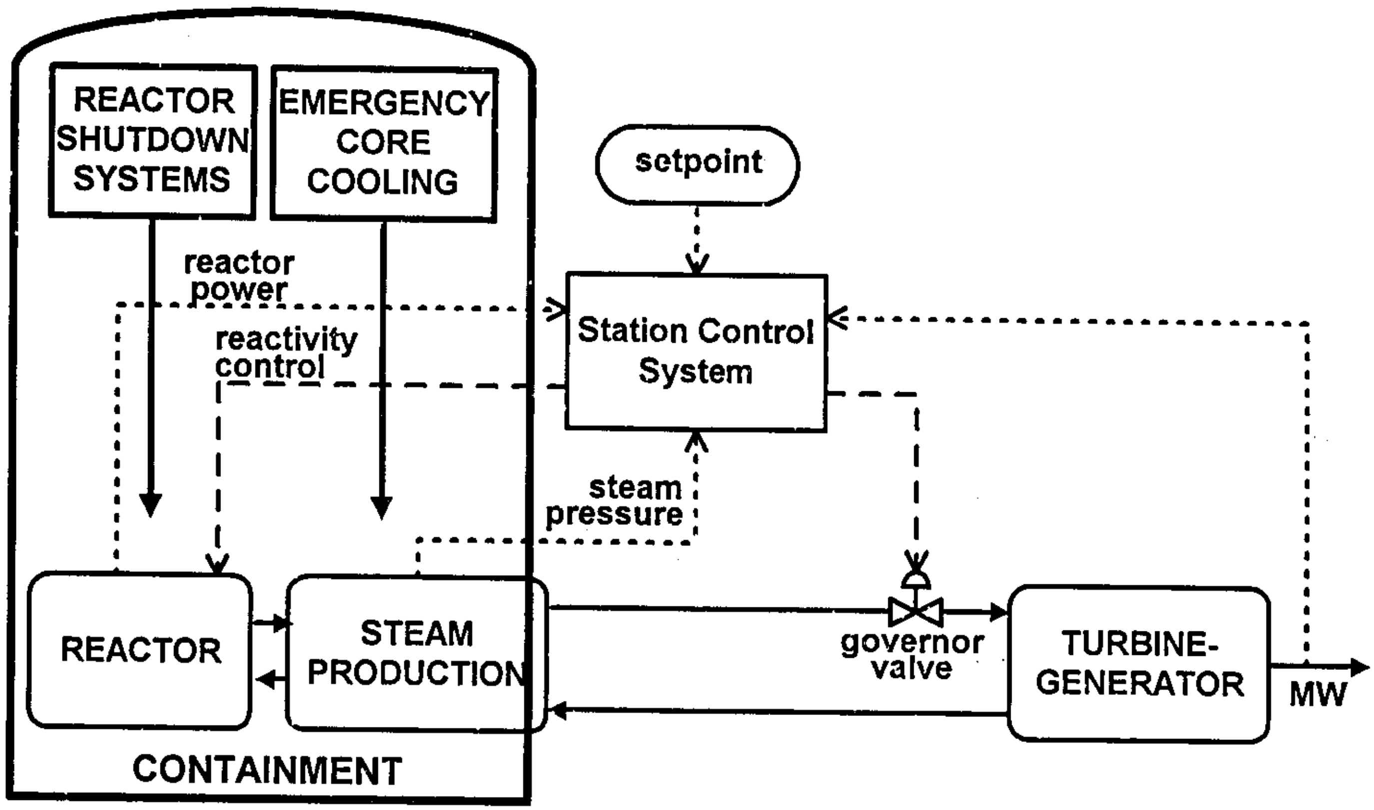
<u>i.e. their purpose is accident prevention;</u>

3.3 Reliable Safety Systems

They are the second level of ensuring safety, they are "poised" as a back-up in case there are process system failures; they will do one or more of the following:

- **shut down the reactor,**
- **ensure continued cooling of the fuel,**
- **containment of fission product release.**

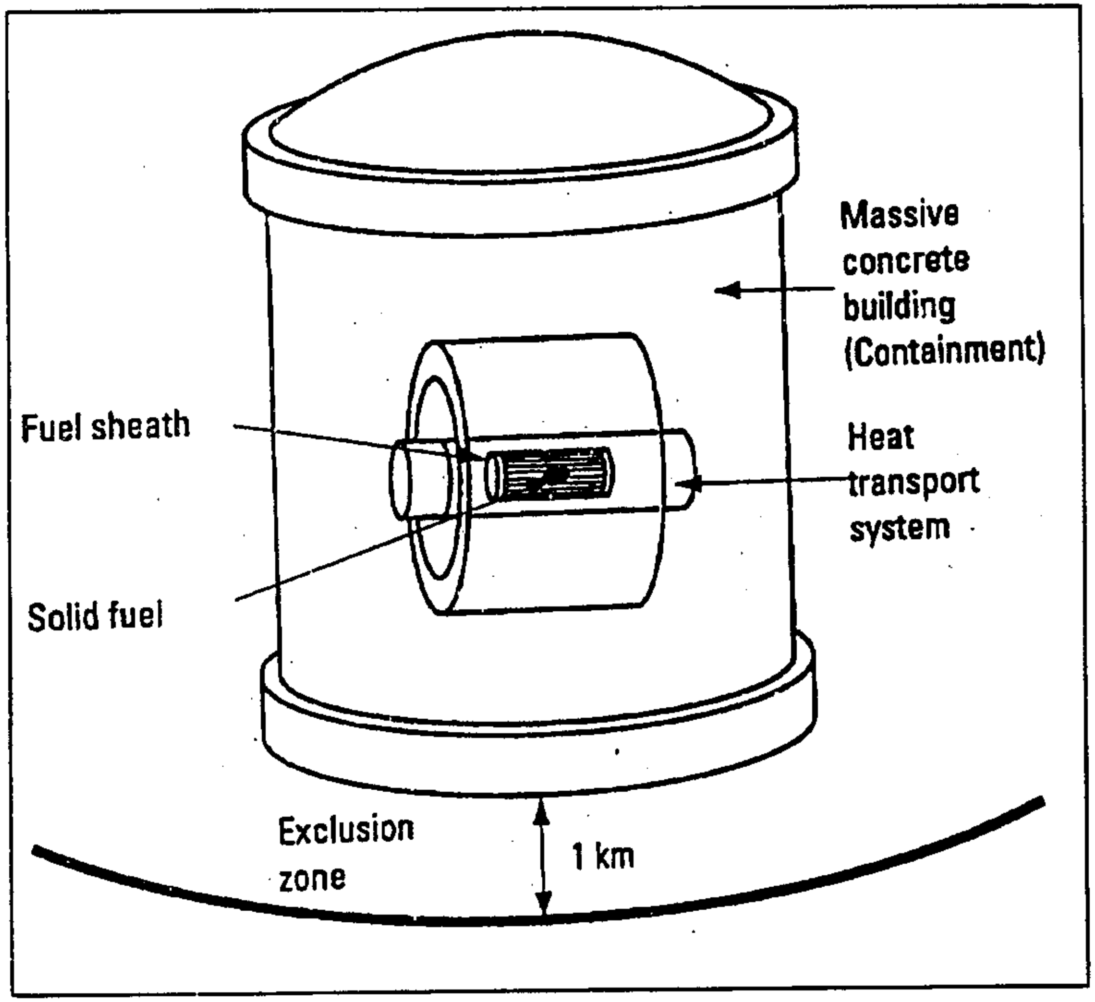
<u>i.e. they provide accident mitigation or accommodation</u>



3.4 Multiple Barriers

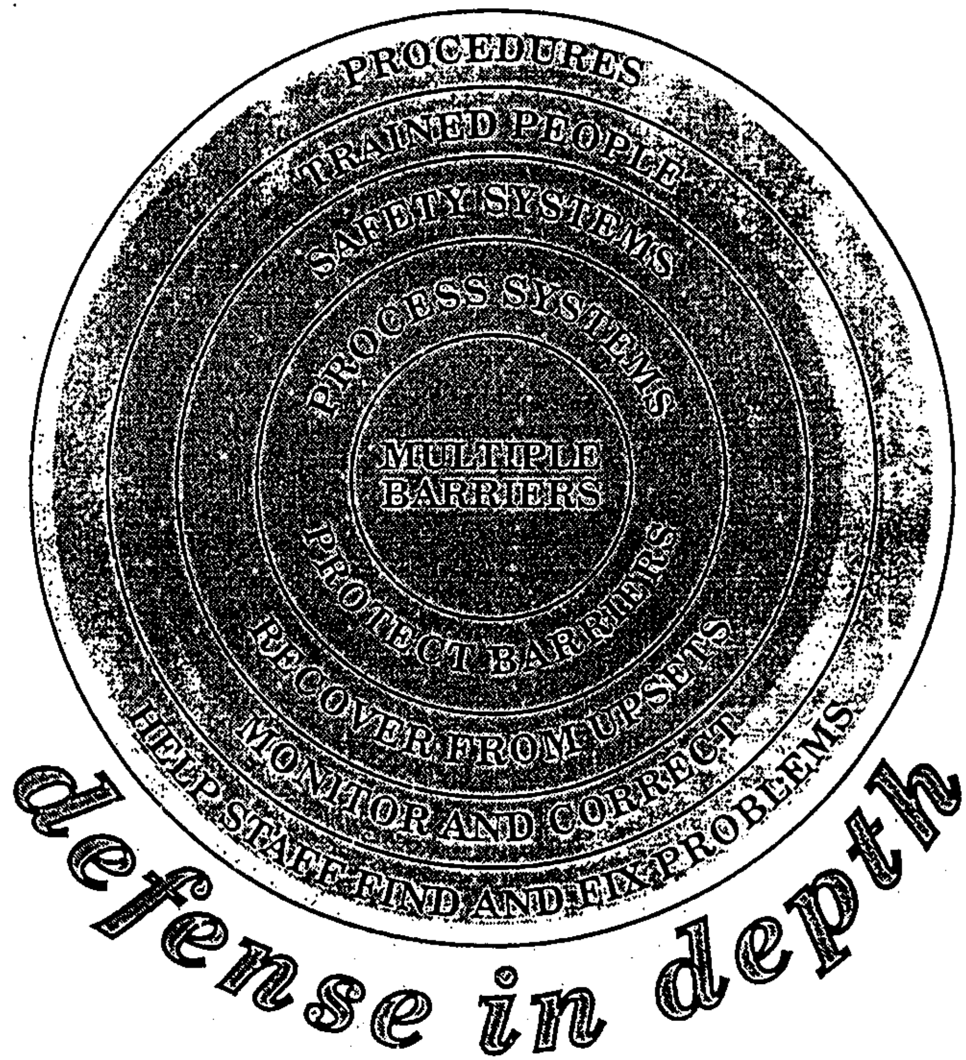
There are five barriers to radioactivity release from the fuel:

- **the uranium fuel is molded into ceramic pellets which have a high melting point and lock in most of the fission products;**
- **the fuel sheath is made of high integrity welded metal (zircaloy) and contains the ceramic fuel;**
- **the heat transport system is constructed of high strength pressure tubes, piping and vessels and contains the fuel bundles;**
- **the containment system provides a relatively leak tight envelope that is maintained slightly below atmospheric pressure;**
- **the exclusion zone of at least one kilometer radius around the reactor ensures any radioactive releases from the station are well diluted by the time they reach the boundary.**



3.5 Competent Operating and Maintenance Staff

- safety systems are designed to operate automatically
- the five passive barriers are always in place
- operating and maintenance staff monitor system conditions and act to prevent or minimize the consequences of any equipment or system failures.



3.6 Detect and Correct Failures

- processes and procedures for the staff to do their work in a systematic fashion
- routine testing programs for safety systems to meet the availability targets
- operational surveillance program
- planned preventive maintenance program
- failures, when they do occur, are thoroughly investigated and solutions applied through a rigorous change approval process.

3.7 Effectiveness of Defence in Depth as practiced in Canada:

- no fatality and no injury of any member of the public as a result of reactor operations;
- no release of radioactivity from a nuclear power plant that resulted in a measurable dose to the public;
- emission of radioactivity has always been far below the regulatory limits (typically < 1% of limits).

CHAPTER 2: OVERALL UNIT CONTROL

MODULE B: WATER COOLED AND MODERATED REACTORS

MODULE OBJECTIVES:

At the end of this module, you will be able to describe:

1. The main proceses and systems in a water moderated and cooled nuclear power plant;
2. Describe the main similarities and significant differences between BWR, PWR and PHWR type nuclear power plants;
3. Describe the main features and operating conditions of BWR type power plants;
4. Describe the main features and operating conditions of PWR type power plants;
5. Describe the main features and operating conditions of PHWR type power plants.

1.0 Common Characteristics of Commercial Water Cooled and Moderated Reactors:

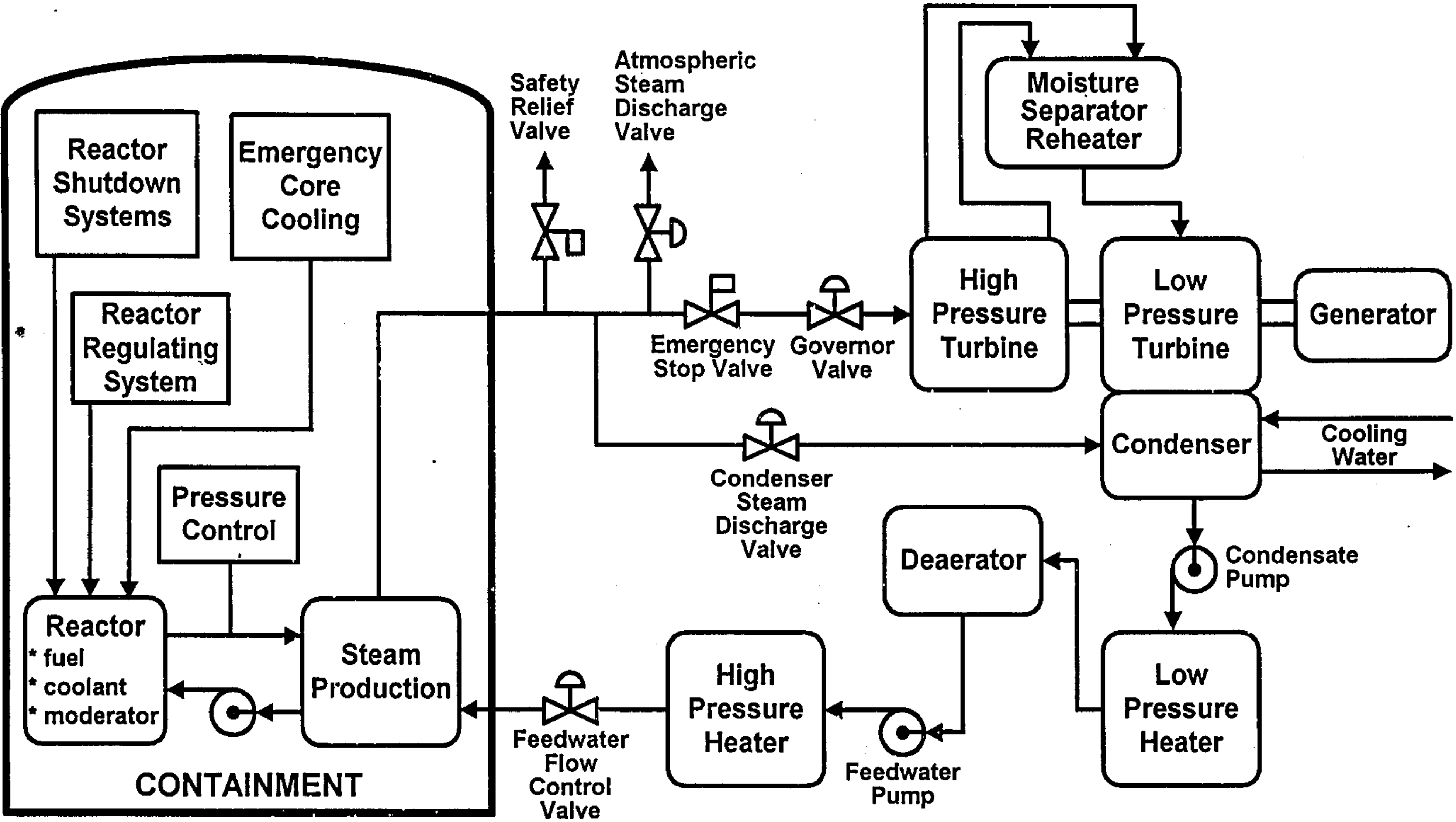


Figure 1: Major process and emergency systems common to commercial water cooled reactors.

1.1 Common Characteristics of Commercial Water Cooled and Moderated Reactors (continued):

- the secondary or steam utilization systems are very similar
- the differences arise due to the operating temperature of the reactor cooling system
- pressure and temperature of the steam
- design of the turbine and heat recovery systems
- note that Figure 2 is very much simplified, most of the equipment shown is duplicated, triplicated in some cases quadrupled
- extraction steam flows are not shown
- reheater drain flows are not shown
- feedwater recirculation flows are not shown

1.2 Significant Differences between Commercial Water Cooled and Moderated Reactors:

- Pressurized (light) Water Reactors (PWR) use pressure vessel for the light water coolant that is also the moderator and is not allowed to boil under normal operating conditions; separate primary cooling circuit and secondary steam generation systems; need enriched fuel.
- Boiling (light) Water Reactors (BWR) use pressure vessel for the light water coolant that is also the moderator, but the pressure is lower than in a PWR, allowing the coolant to boil; the coolant and steam generation systems are not physically separated; need enriched fuel.
- Pressurized Heavy Water Reactors (PHWR) use pressure tubes for the heavy water coolant that is allowed only a slight amount of boiling; the heavy water moderator is kept in a separate low pressure, slightly above ambient temperature vessel; it has a steam generation system that is physically separate from the reactor coolant system; normally operates on natural uranium fuel
- CANDU is the only PHWR commercially available, and will be used extensively in this course to illustrate the control principles presented.

2.1 Typical Boiling (Light) Water Reactor

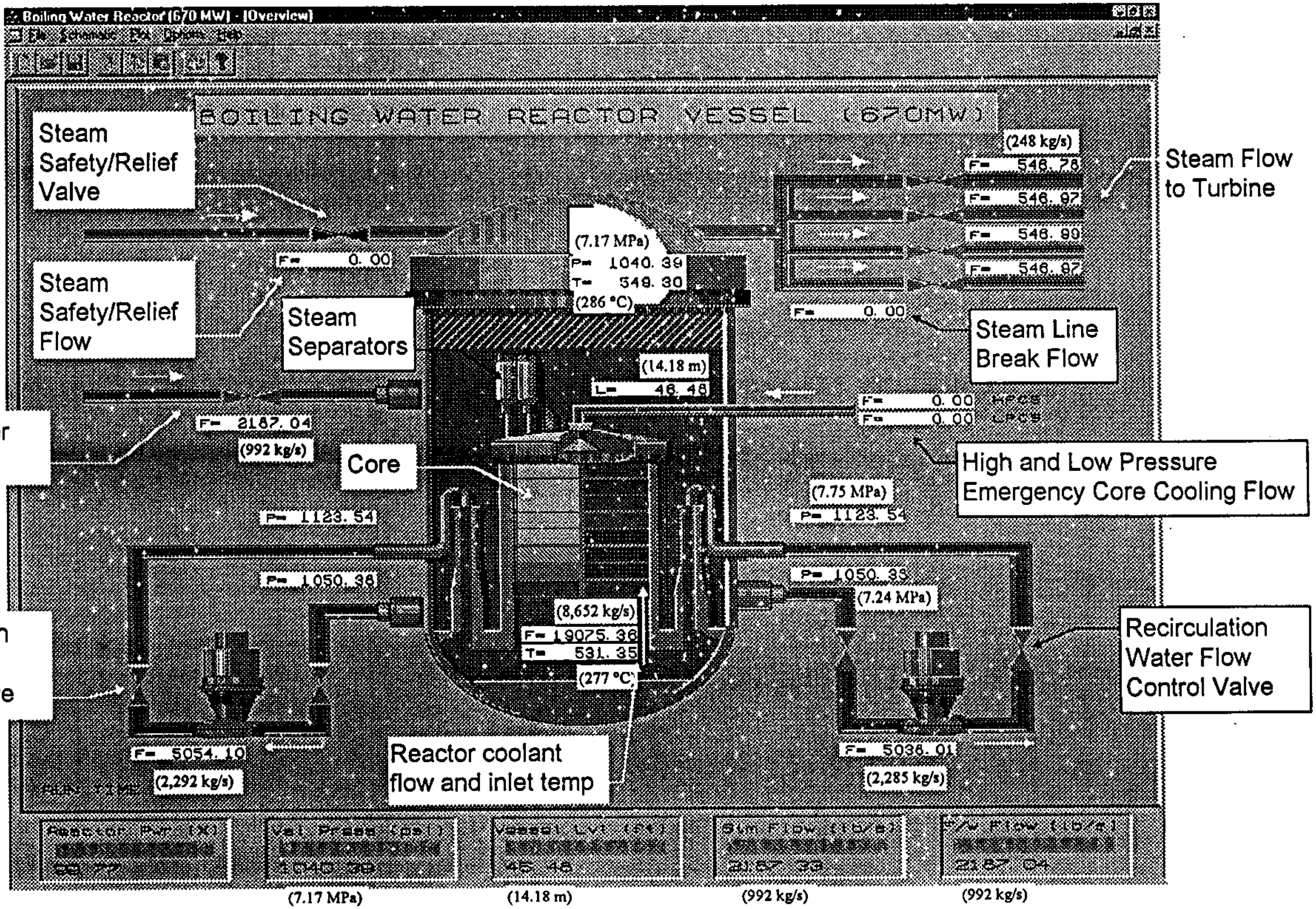
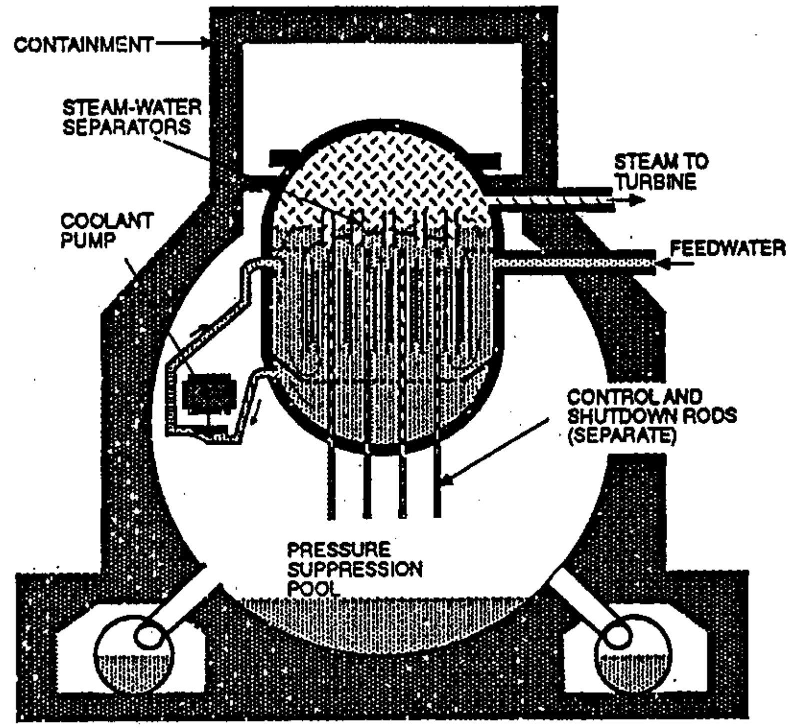


Figure 2: Reactor Vessel and Core for a 670 MW GE BWR/6 Generating Station.

2.2 Significant features of BWR steam supply systems



- enriched fuel, must be shutdown to refuel
- fuel rods are in vertical arrays
- light water moderator, light water coolant
- coolant and moderator are the same fluid
- coolant in the core is allowed to boil
- coolant circulation by convection and forced circulation
- circulating pumps inside and/or outside reactor vessel
- steam from reactor flows directly to the turbine
- larger reactor vessel but operates at lower pressure than PWR
- control rods located at bottom of reactor vessel

2.3 Operational aspects

- subcooled water enters the bottom of the core and flows past the fuel elements
- boiling takes place in the core, the coolant typically reaches 10% saturation at the top of the core
- steam separators pass the steam to steam dryers to remove any remaining water droplets, and the water from the separators and the dryers is recirculated back to the bottom of the vessel via the space between the core shroud and the pressure vessel wall, and the desired rate of flow is achieved with the aid of jet pumps
- the jet pumps are driven by flow of water provided by external loop pumps (the loop pump flow is taken from the bottom of the vessel, and typically one third of the total flow through the reactor)
- feedwater at or near saturation is returned to the reactor vessel above the core and mixes with the recirculating flow

3.1 Typical Pressurized (Light) Water Reactor

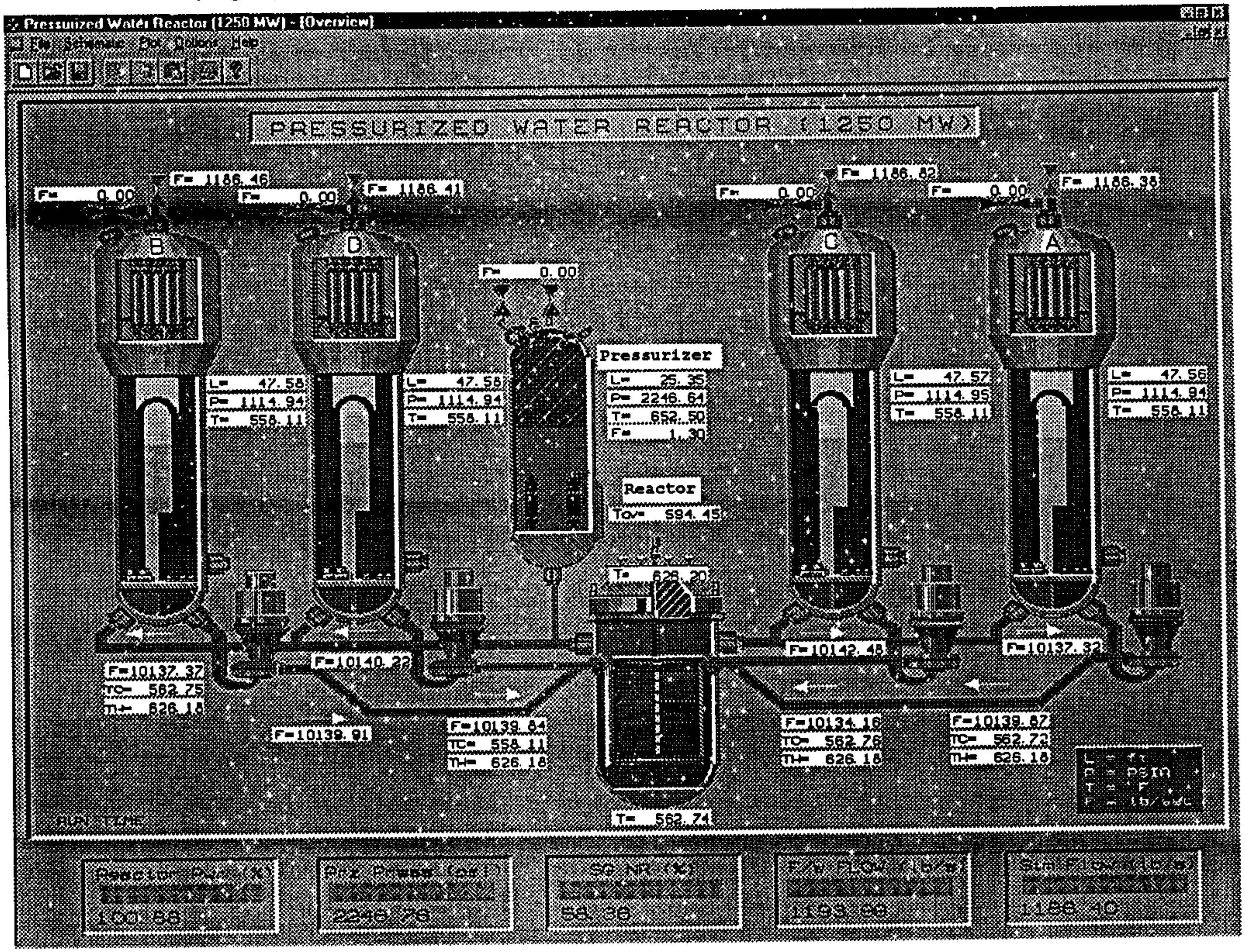
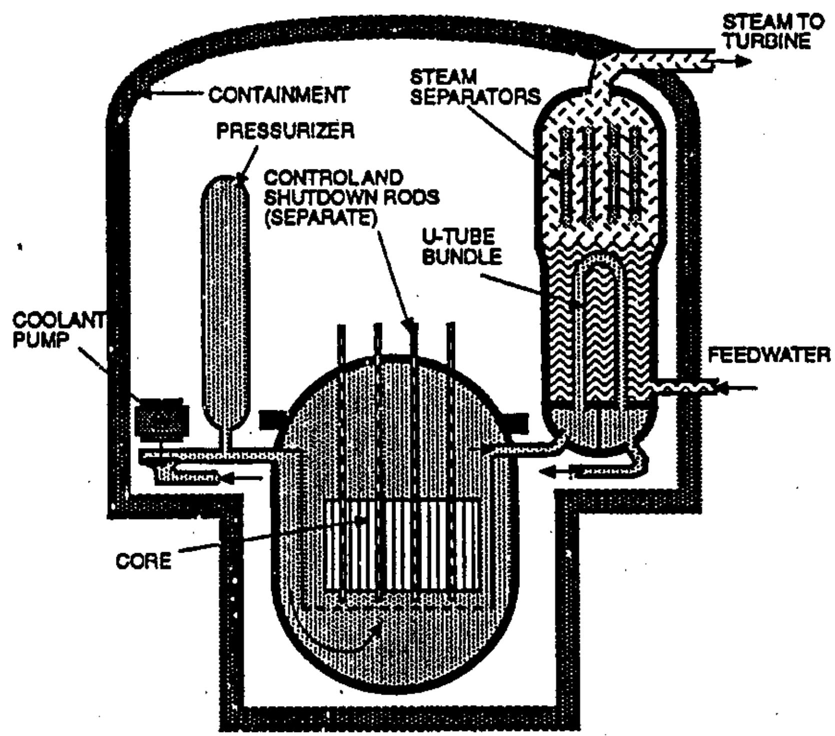


Figure 4: Reactor, Coolant Circuit and Steam Generators for 1250MW PWR.

3.2 Significant features of PWR steam supply systems

- enriched fuel
- fuel rods are in vertical arrays
- light water moderator, light water coolant
- coolant and moderator are the same fluid
- coolant in the core is normally not allowed to boil, or only to small degree
- heat from primary coolant produces steam in a separate secondary circuit
- requires large circulating pumps to provide required flow of coolant
- control rods located at top of reactor vessel



3.3 Operational aspects

- water enters the pressure vessel near the top of the core and flows downward in the annulus between the core shroud and the vessel wall, then it flows upward past the fuel elements and out to the pipes that connect the reactor to the boilers
- there are a number of external coolant circuit loops (from two to six depending on the particular design) each with a circulating pump and a steam generator
- a pressurizer connected to one of the loops maintains system pressure and allows for volume changes of the coolant as its average temperature changes with reactor power
- the steam generators consist of a boiling section, a steam drum and an in some designs include an integral preheater section

4.1 Typical Pressurized Heavy Water (CANDU) Reactor

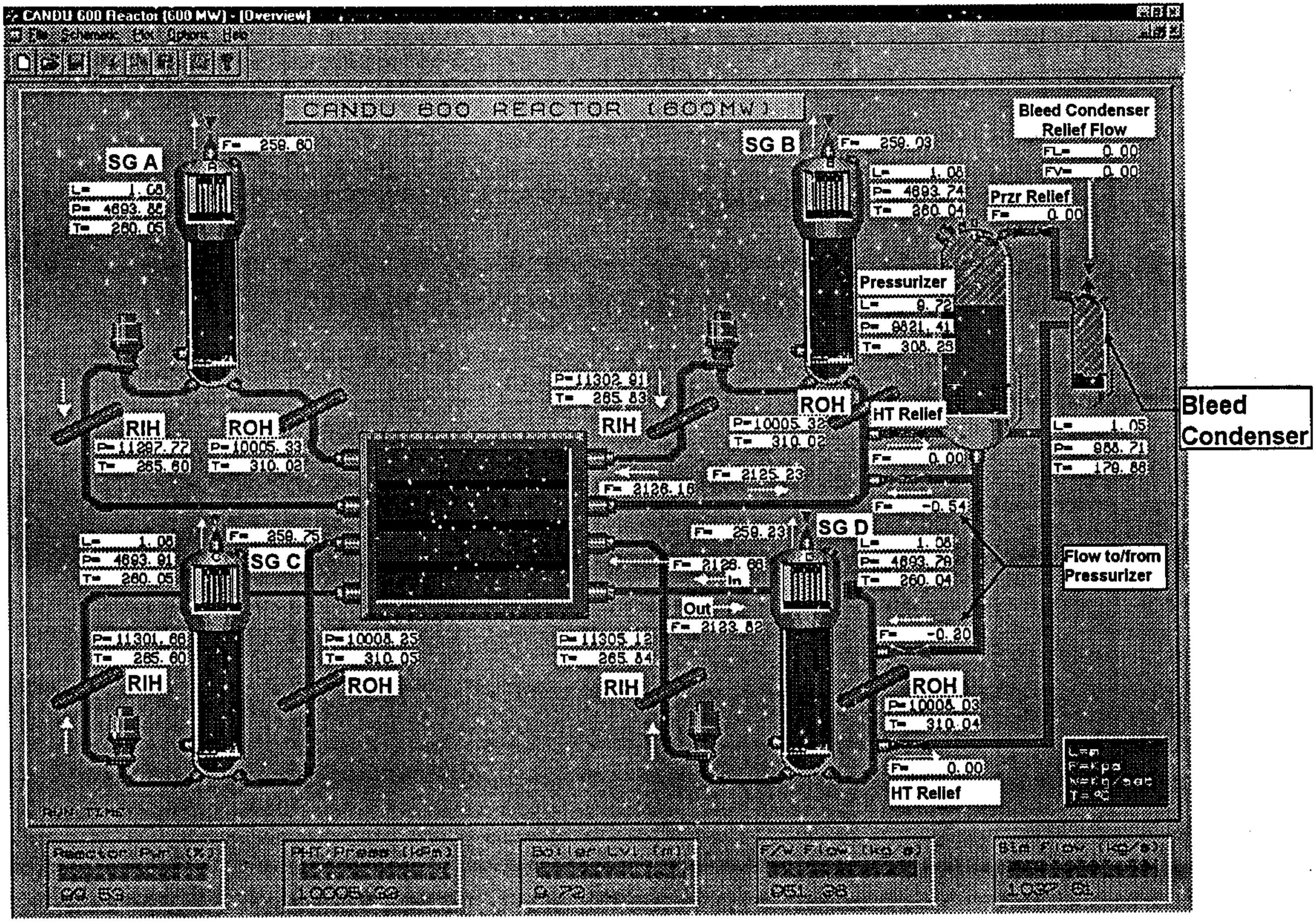
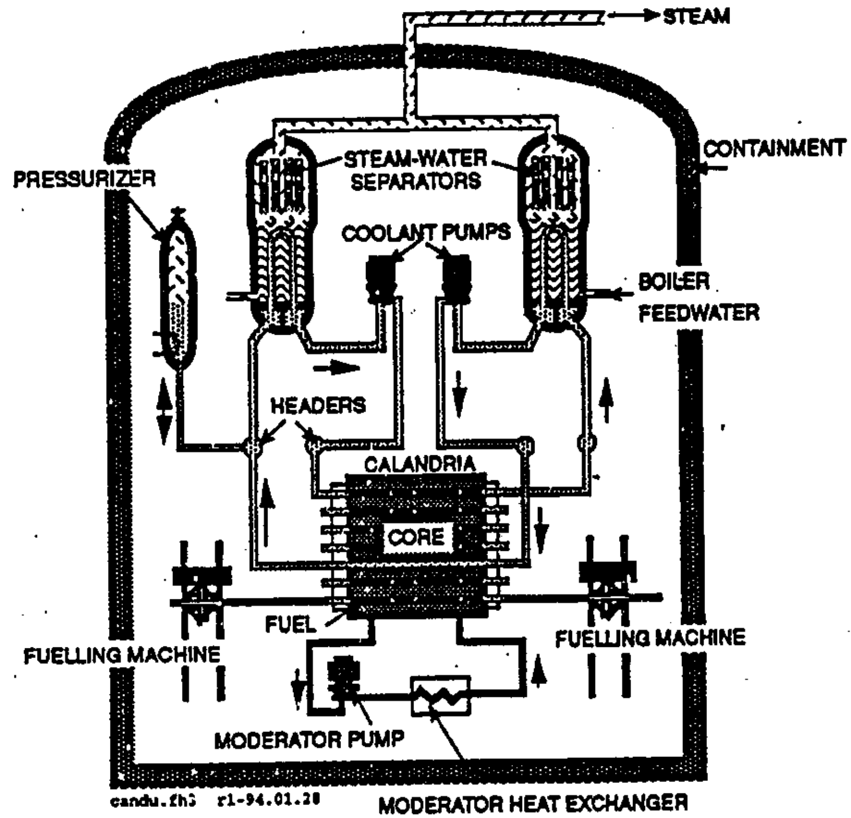


Figure 6: Reactor, Heat Transport and Steam Generators for CANDU 6.

4.2 Significant features of PHWR steam supply systems

- natural uranium fuel, on-power refuelling
- fuel bundles in horizontal fuel channels
- heavy water moderator, heavy water coolant
- coolant and moderator are in separate circuits: coolant at high pressure, moderator slightly above atmospheric
- fuel and coolant in pressure tubes
- coolant in the core is normally not allowed to boil, or only to small degree near the outlet of a fuel channel
- heat from primary coolant produces steam in a separate secondary circuit
- requires large circulating pumps to provide required flow of coolant
- control rods located at top of reactor vessel



4.3 Operational aspects

- separate moderator circuit, operating slightly above atmospheric conditions
- coolant and fuel contained in horizontal pressure tubes instead of pressure vessel, flow of coolant to and from each channel is via inlet and outlet headers outside the reactor
- coolant circuit typically consists of four circulating pump and four steam generators in two figure of 8 interconnected loops
- a pressurizer connected to one of the loops maintains system pressure and allows for volume changes of the coolant as its average temperature changes with reactor power
- the steam generators consist of a boiling section, a steam drum and an in some designs include an integral preheater section

CHAPTER 2: OVERALL UNIT CONTROL

MODULE C: CONTROLLING THE ENERGY CONVERSION PROCESS

MODULE OBJECTIVES:

At the end of this module, you will be able to describe:

1. The main energy conversions from fission to electricity in a nuclear generating unit;
2. How an energy balance is maintained between the reactor (or primary) and the conventional (or secondary) side of the station;
3. The two basic methods of controlling a nuclear generating unit;
4. The main control actions under "turbine-leading-reactor" control mode;
5. The main control actions under "reactor-leading-turbine" control mode.

1.0 THE CONTROL PROBLEM - MAINTAINING THE ENERGY BALANCE

1.1 Operating Requirements:

- assure safety of public, workers and equipment
- constant power generation at a specified level of output
- change power output to specified levels at specified rates
- maintain generation frequency within specified limits

1.2 Disturbances

- load voltage, current and frequency
- steam pressure, flow and quality
- condenser pressure & temperature, cooling water flow and temperature
- reactor coolant flow, pressure and temperature
- reactivity effects: fuel burn-up, fission products, control absorbers, moderator temperature, coolant temperature and void

1.3 Indications of Lack of Control

- generator output (MW)
- reactor power level
- coolant temperature and pressure
- steam flow and pressure

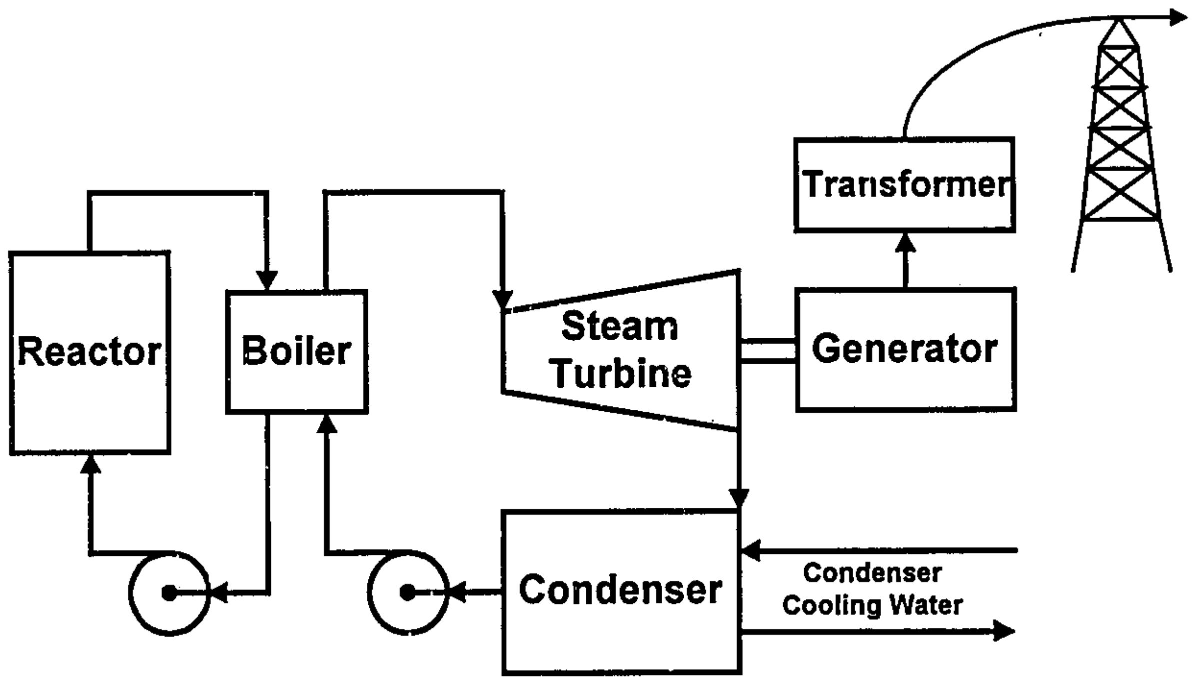


Figure 1: Simplified nuclear power plant block diagram

1.4 Control systems to be studied in this course:

Control System	BWR	PWR	PHWR
unit output	yes	yes	yes
reactor neutron flux - global	yes	yes	yes
- local	no	no	yes
reactor coolant pressure	no	yes	yes
steam generator pressure	yes	yes	yes
steam generator level	yes	yes	yes

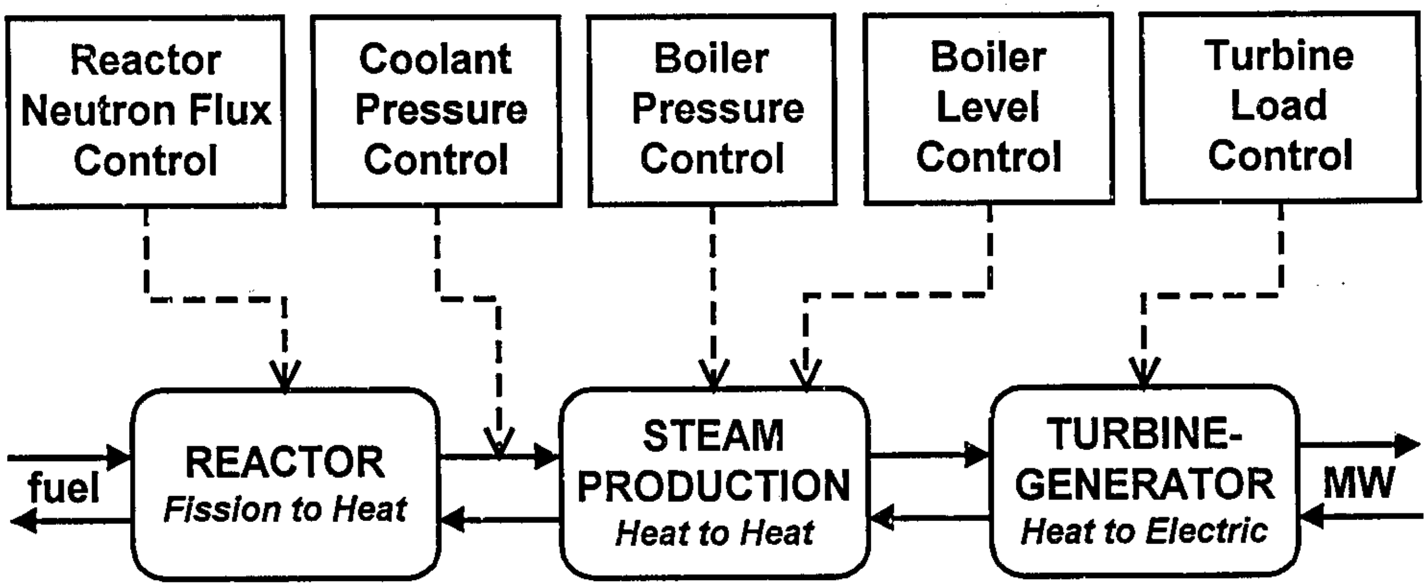


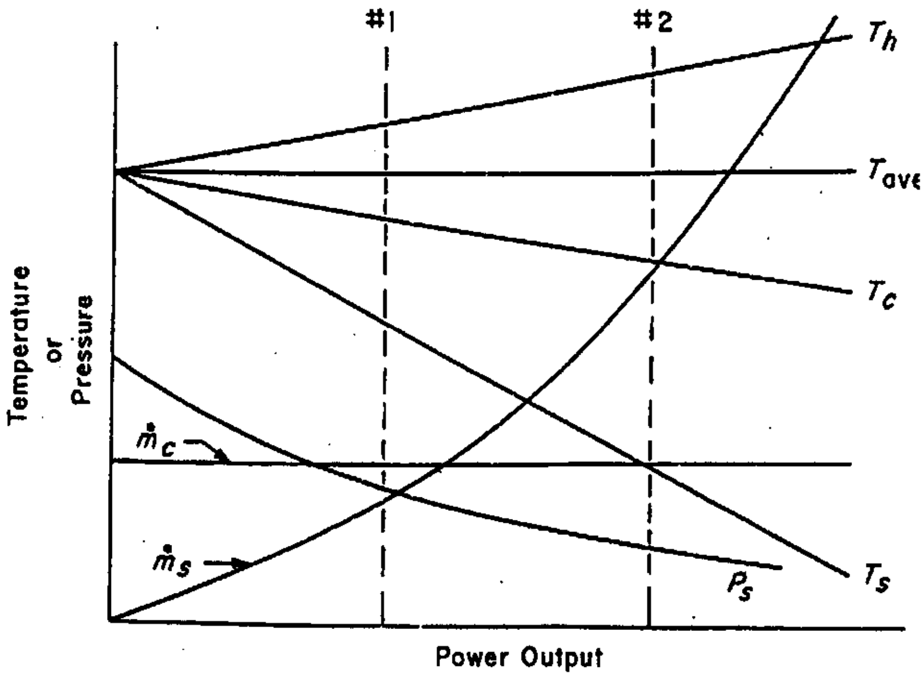
Figure 2: Nuclear Plant Main Control Systems

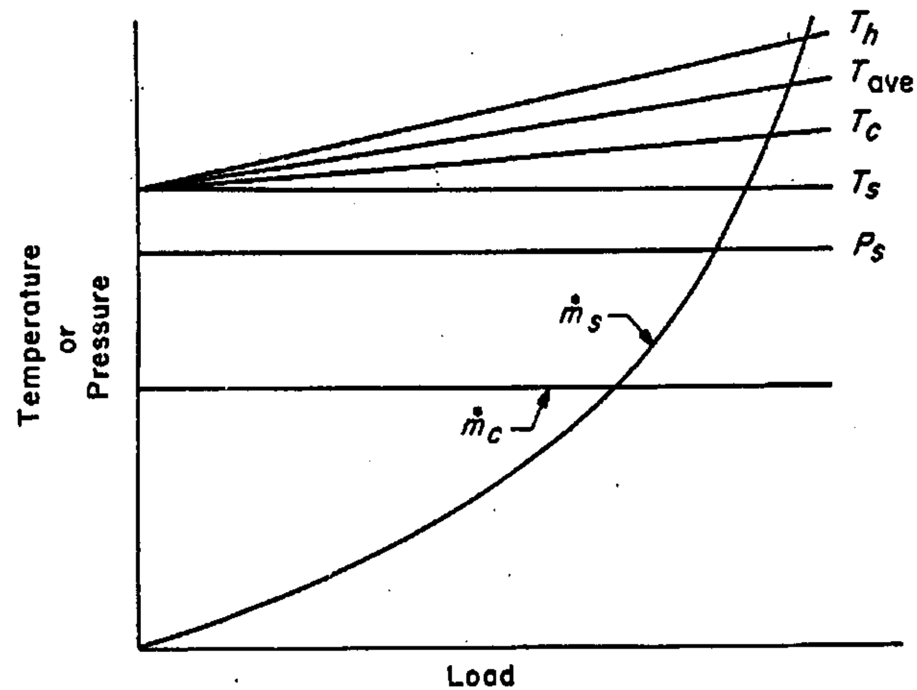
2.0 OVERALL UNIT CONTROL STRATEGIES

- constant specified amount of electrical power to be produced
- maximum electrical power to be produced as allowed by reactor and other plant systems

2.1 Steam generator pressure and average coolant temperature determine how unit control is implemented:

- constant steam generator pressure throughout power generation range
- constant primary coolant average temperature throughout power generation range
- both steam pressure and coolant average temperature vary as a function of reactor power





3.0 CONTROL OF UNIT POWER

- **if unit output power is specified as the setpoint, unit operates in "turbine-leads-reactor" (or "reactor lagging) mode**
- **if unit output power is determined by the setpoint of the reactor, unit operation is called "reactor-leads-turbine" (or "reactor leading") mode**
- **the choice of unit control mode depends on the operating status of the generating station and the requirements of the electrical power grid**

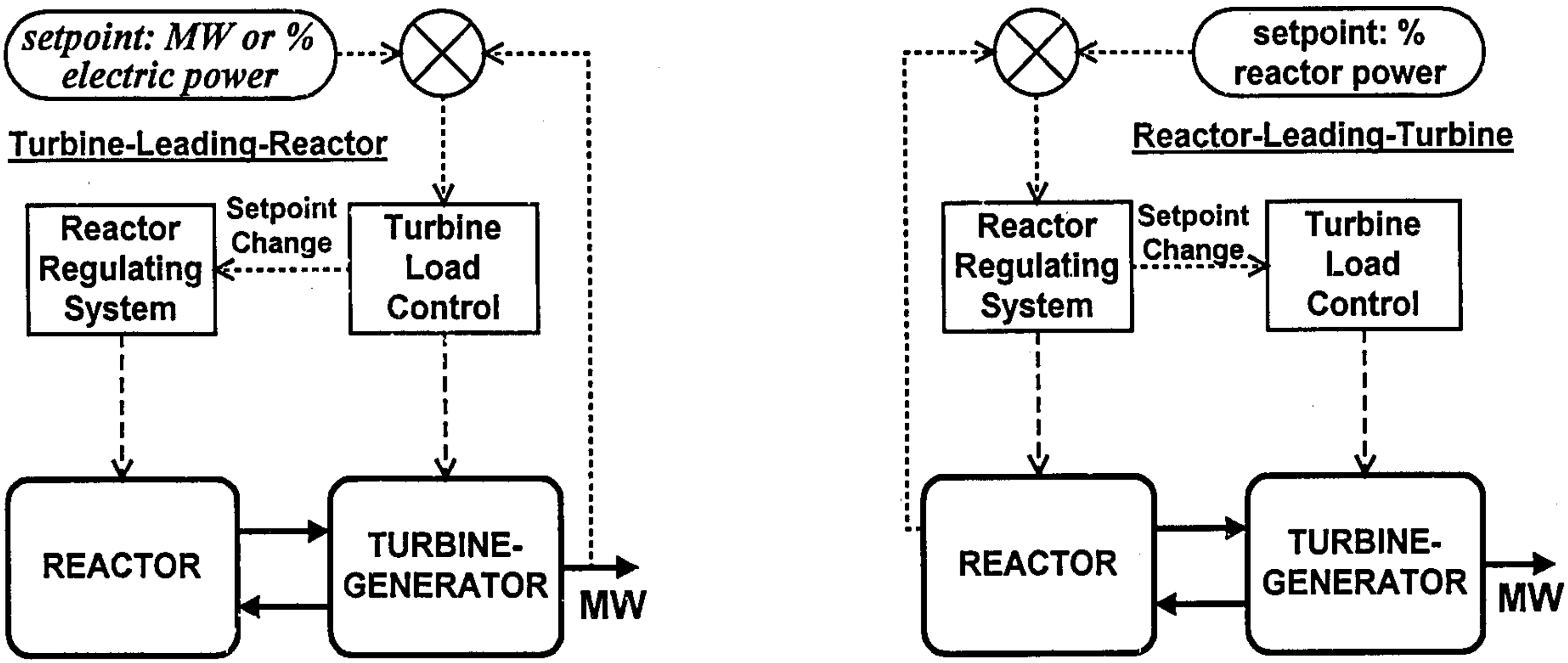


Figure 3: Two basic types of nuclear generating station overall unit control systems

3.1 Turbine-Leading-Reactor mode

- **the setpoint is the desired level of generator output (megawatts)**
- **if there is a difference between the setpoint and the actual power level, the control system makes a correction by altering the opening of the governor valve and hence the amount of steam flow going to the turbine**
- **changes in steam flow will change the steam generator pressure (in the opposite direction)**
- **the reactor control system adjusts reactor power by changing the position of the reactivity control devices to keep the steam generator pressure at its setpoint**

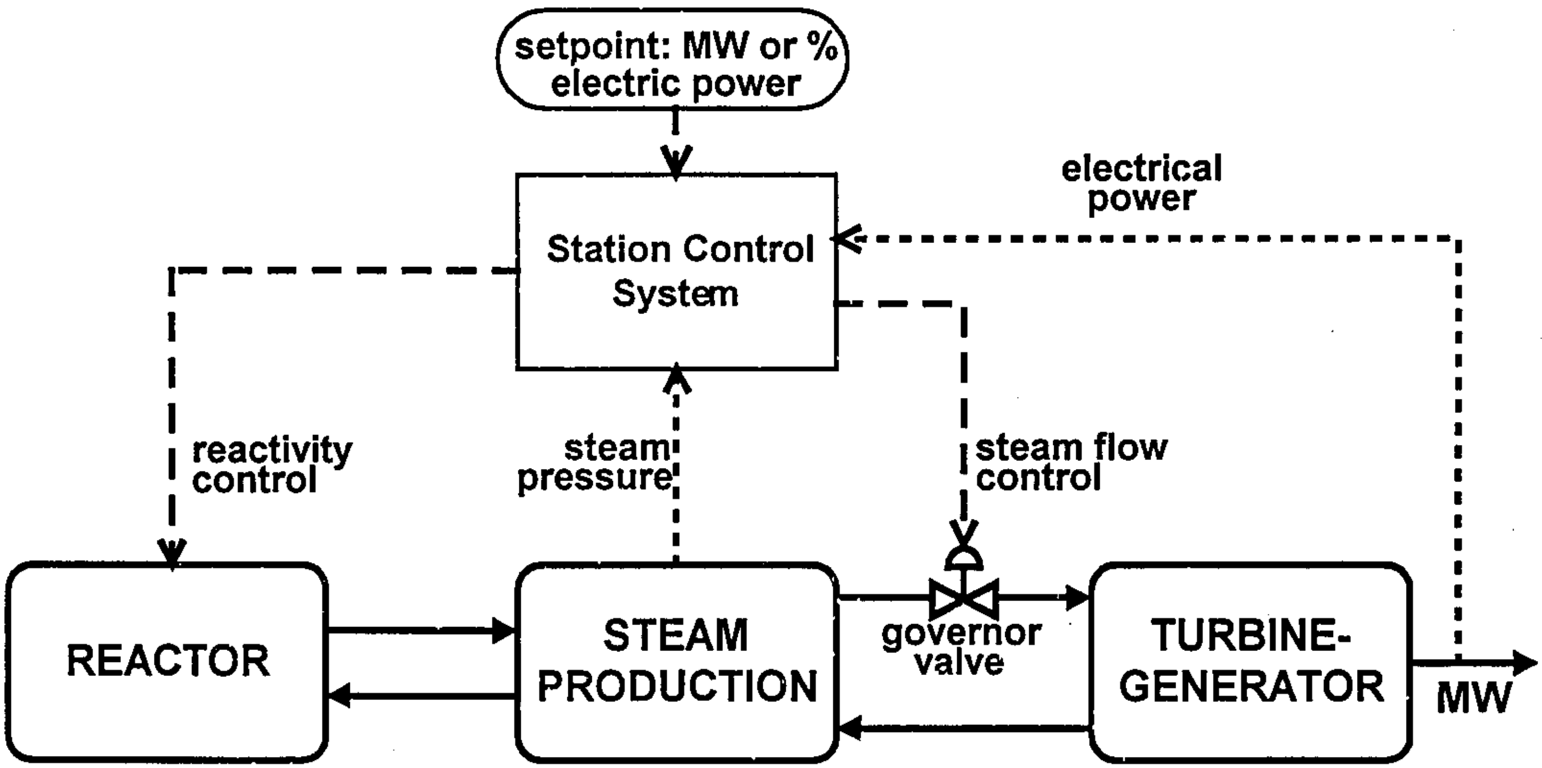


Figure 4: Simplified turbine-leads-reactor overall unit control system

3.2 Reactor-Leading-Turbine mode

- the setpoint is the desired level of reactor power output
- if there is a difference between the setpoint and the actual reactor power level, the control system makes a correction by altering the position of the reactivity control devices and hence the reactor neutron flux
- changes in reactor power will change the steam generator pressure (in the opposite direction)
- the steam generator pressure control system adjusts steam flow and hence turbine power by changing the position of the governor valve to keep the steam generator pressure at its setpoint

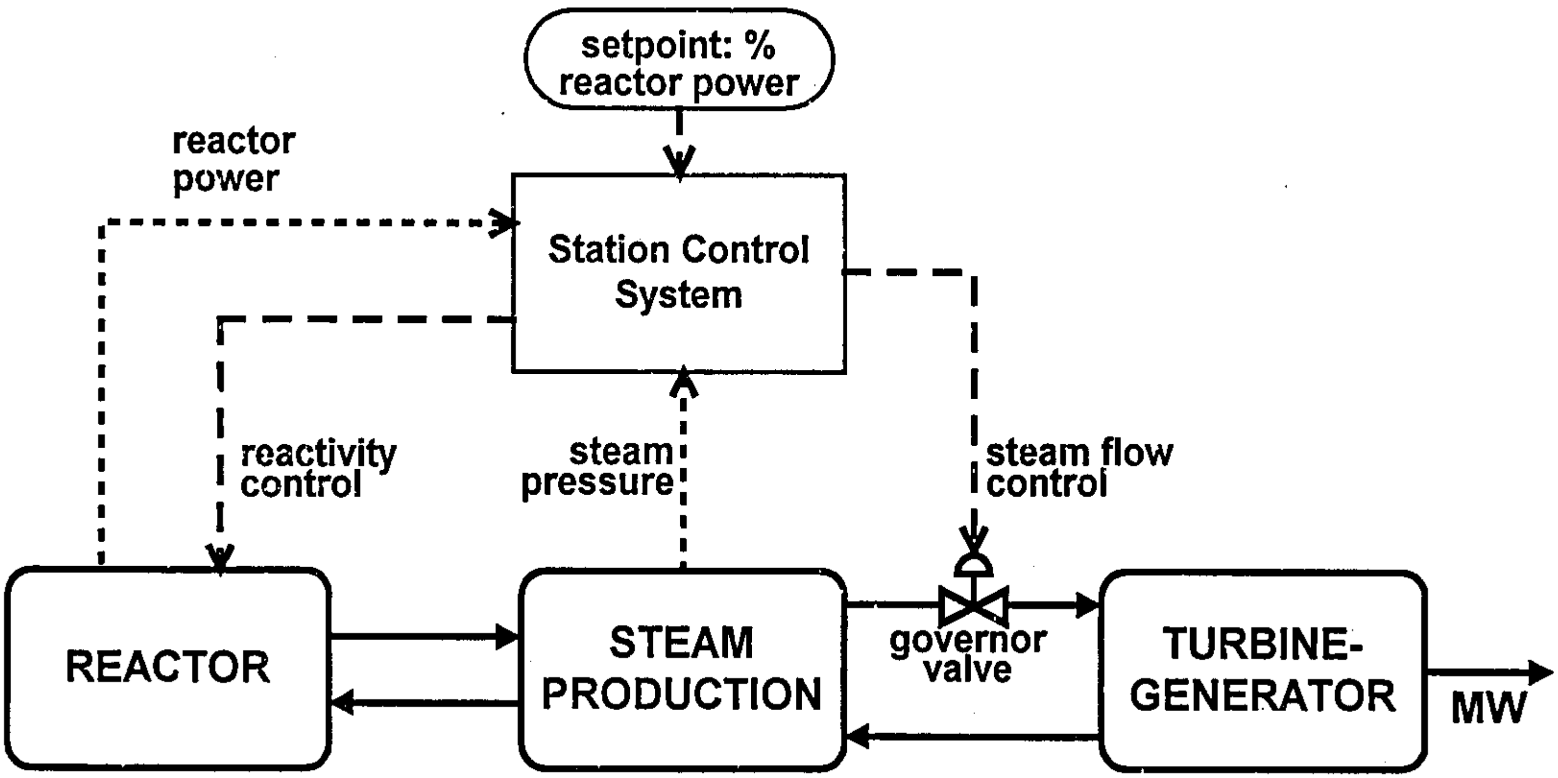
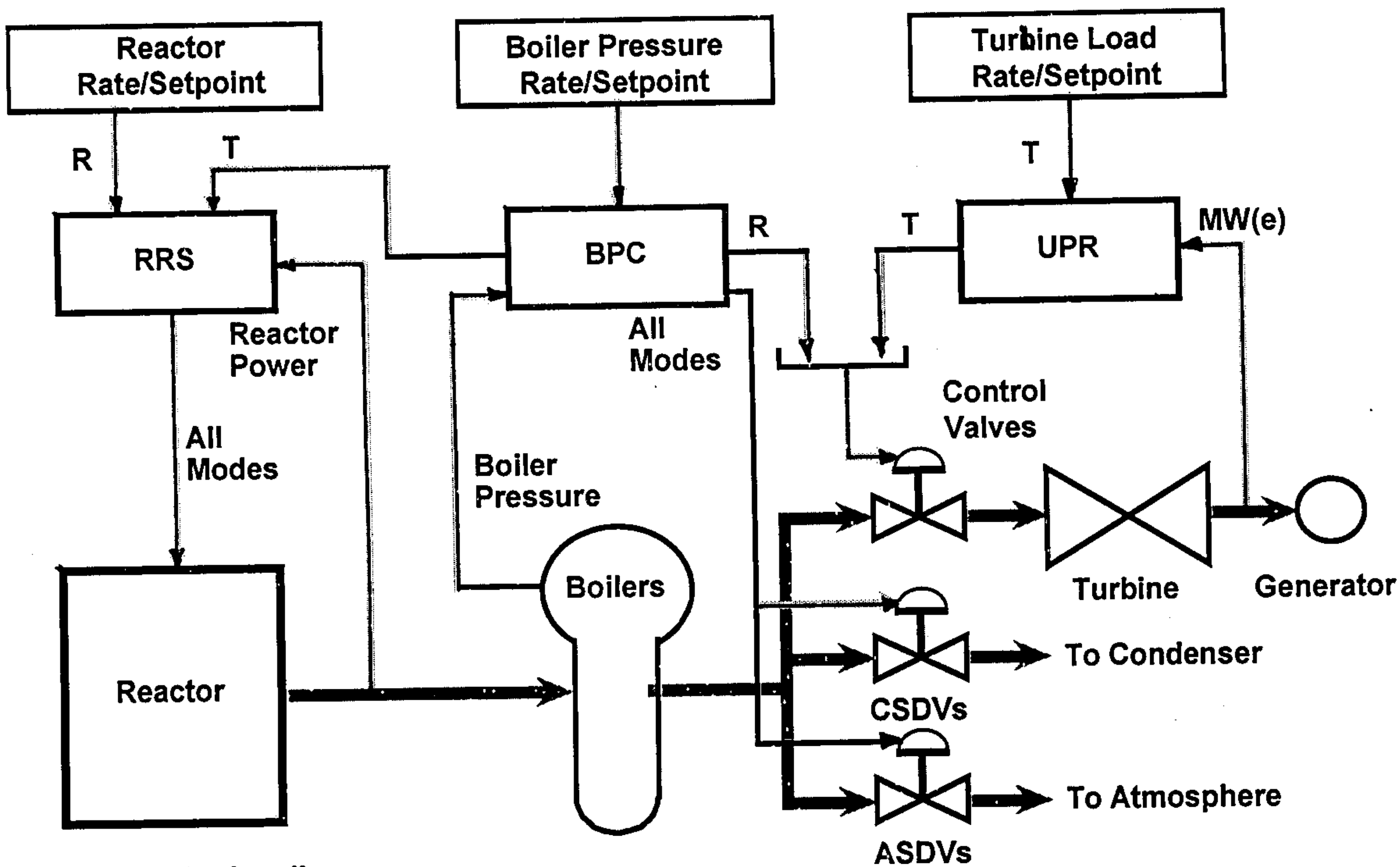


Figure 5: Simplified reactor-leads-turbine overall unit control system

3.3 Changing between Reactor Leading and Turbine Leading modes



R - Reactor Leading
T - Turbine Leading

4.1 CANDU OVERALL UNIT CONTROL

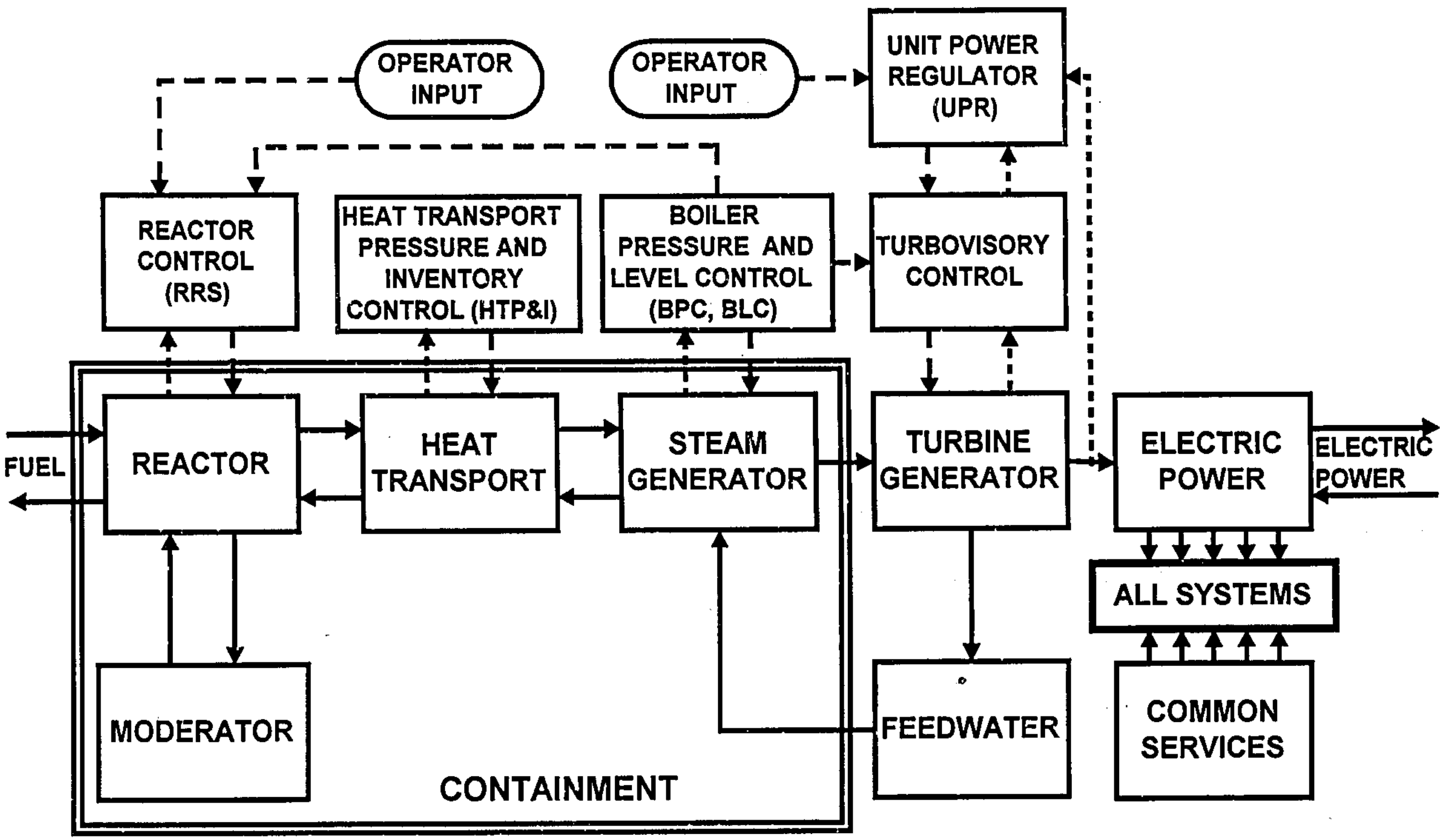


Figure 6: Simplified CANDU Unit Control Block Diagram.

4.2 Computerized Station Control Systems

Because of the complex interdependence of control systems in a CANDU unit, all major control functions are performed by Digital Control Computers (DCC). The main programs with the parameters measured and the different variables controlled and manipulated, are summarized in Table 1.

Table 1: Main CANDU Control Programs.

Program Name	Measured Parameter (s)	Variable(s) Controlled	Variable(s) Manipulated
1. Unit Power Regulator (UPR)	• Electrical output	• Electrical output • Steam flow	• Steam flow
2. Reactor Regulating System (RRS)	• Reactor bulk power	• Neutron flux	• Zone water level • Rod position
3. Heat Transport Pressure and Inventory Control (HTP&I)	• HTS Pressure	• D_2O pressure • Pressurizer level	• Pressurizer steam bleed & heaters • D_2O feed & bleed
4. Boiler Pressure Control (BPC)	• Boiler pressure • Reactor power • Steam flow	• Boiler pressure	• Reactor setpoint • Steam flow
5. Boiler Level Control (BLC)	• Boiler level • Reactor power • Feedwater flow • Steam flow	• Level (inventory)	• Feedwater flow

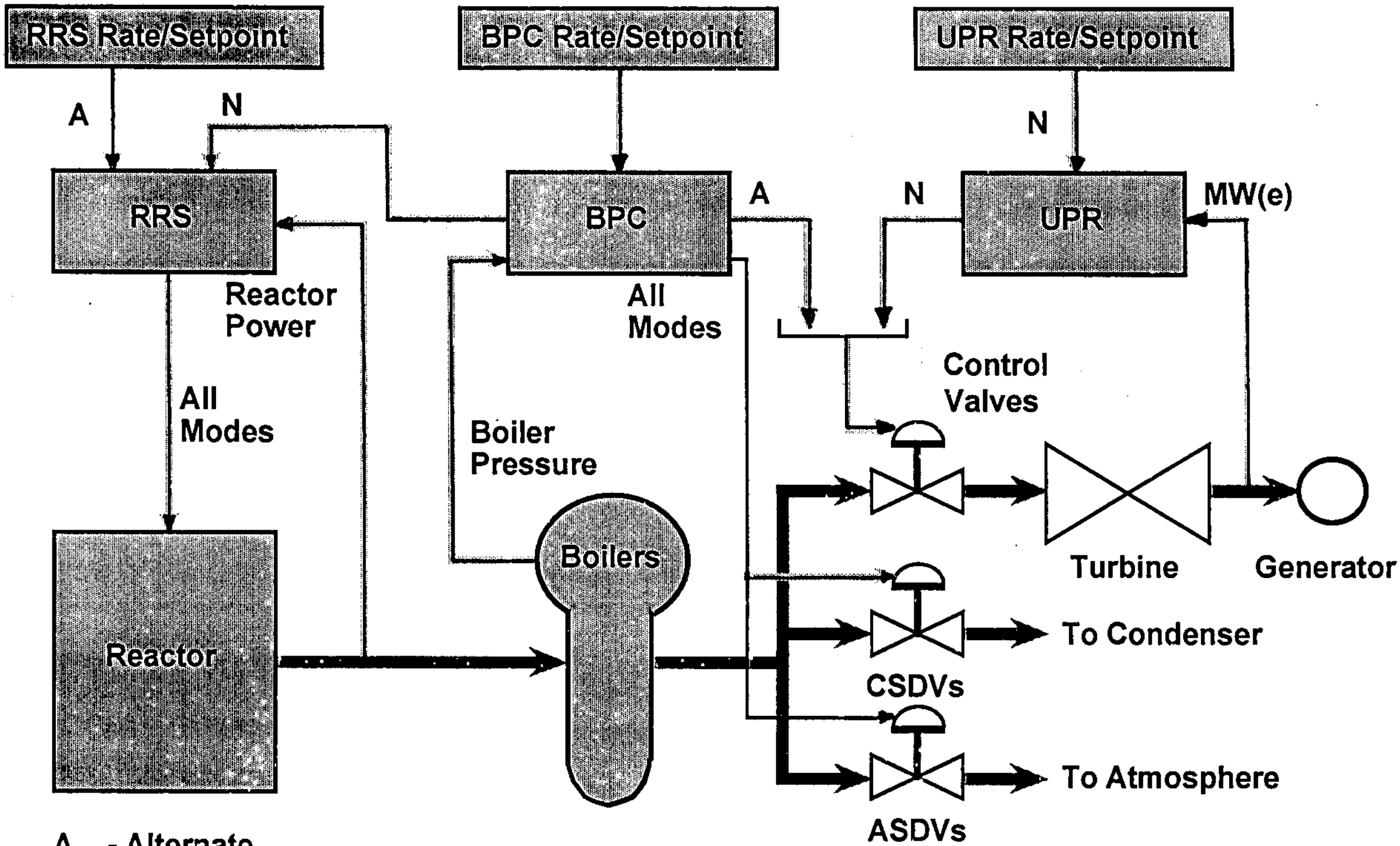
4.3 CANDU 9 Operating Characteristics

- The gross output of the generator is 925 MW and the station service power is 55 MW, yielding a net unit electrical output of 870 MW.
- The unit is capable of sustained operation at any net electrical output of up to 100 percent of rated full power output.
- The overall plant control is normally of the reactor-following-turbine type.
- The reactor and turbine are controlled by computer from zero to 100 percent of full power.
- For reactor power increases, the nuclear steam system portion of the plant is capable of maneuvering at the following rates:

Power Range	Maximum Rate
0 - 25 percent of full power	4 percent of actual power per second
25 - 80 percent of full power	1 percent of full power per second
80 - 100 percent of full power	0.15 percent of full power per second

- The overall plant power maneuvering rate is a function of turbine design, and is typically limited to 5 - 10% of full power per minute.

4.4 CANDU OVERALL UNIT CONTROL SUMMARY



A - Alternate
N - Normal

CHAPTER 2: OVERALL UNIT CONTROL

MODULE D: CANDU UNIT POWER REGULATOR

MODULE OBJECTIVES:

At the end of this module, you will be able to describe:

1. the main functions and modes of operation of the Unit Power Regulator (UPR) program
2. the main interrelations between the UPR program and the turbine hardware control
3. the control signal inputs to and outputs from UPR;
4. how the 'speeder gear' or 'turbine load set motor' is controlled;
5. the monitoring functions of UPR.

1.0 INTRODUCTION

- At all CANDU stations control of Unit Power under NORMAL conditions follows the "Turbine-Leading-Reactor" mode
- Control is via the digital computer program Unit Power Regulator (UPR)

1.1 PRIMARY PROGRAM FUNCTIONS

- Control turbine load changes
  - Accepts operator specified setpoint and rate of change via keyboard entry and ramps the turbine target load up or down to achieve the requested electrical output.
  - UPR calculates maximum maneuver rate based on measured turbine stress and limits the rate of setpoint change to this value
- Monitor turbine & generator variables
  - generates messages to alert the operator of problems
  - inhibits turbine loading as required

1.2 MODES OF OPERATION

- Monitor Mode
  - no UPR turbine control
  - monitoring routines running and providing messages to the operator
- Control Mode
  - continuous UPR turbine control
  - monitoring routines running and providing inhibits on turbine loading as well as messages

2.0 UPR CONTROL

- UPR controls the turbine when the unit is in NORMAL mode.
- The turbine loading routine in UPR is active only after turbine synchronization.
- Turbine monitoring functions of UPR are active before and after synchronization.
- When UPR is controlling the turbine it drives the load limiter to a value 80 MW higher than the turbine load setpoint, to prevent a sudden large load increase on a frequency upset.

2.1 UPR CONTROL MODE PRE-REQUISITES:

UPR will terminate CONTROL and enter MONITOR mode if any prerequisite is absent.

- "Turbine Control" (EHC) operating mode pushbutton in "Remote Operation"
- "Turbine Load Control" handswitch in "Local"
- UPR, BPC & RRS running in the same DCC
- BPC permissive allows UPR to control turbine
- Turbine synchronized
- Rational load measurement: (either generator MW or turbine 1st stage pressure)
- Difference between Generator Measured Load & Load Reference less than 40 MW i.e. $|LR-L| < 40MW$

2.2 TERMINOLOGY:

- TARGET LOAD (TL): the UPR Generator Load Setpoint as entered via key-board (K/B)
- LOAD REFERENCE (LR): a UPR calculated value that is incremented towards the setpoint on each program pass. The increment depends on the loading rate.
- ACTUAL LOAD (L): the measured generator gross load.

2.3 INTERRELATIONS BETWEEN UPR PROGRAM AND HARDWARE CONTROL

- the Electro-Hydraulic Controller (EHC) controls the position of the "speeder gear" or "turbine load set motor", which is the demanded opening or setpoint for the governor valves
- by determining the opening of the governor valves, the "speeder gear" dictates the amount of steam flow to the turbine, and hence, for a given steam pressure, the energy received by the turbine
  - when not synchronized, the "speeder gear" determines the speed of the turbine (hence the name)
  - when synchronized the "speeder gear" determines the power output of the turbine

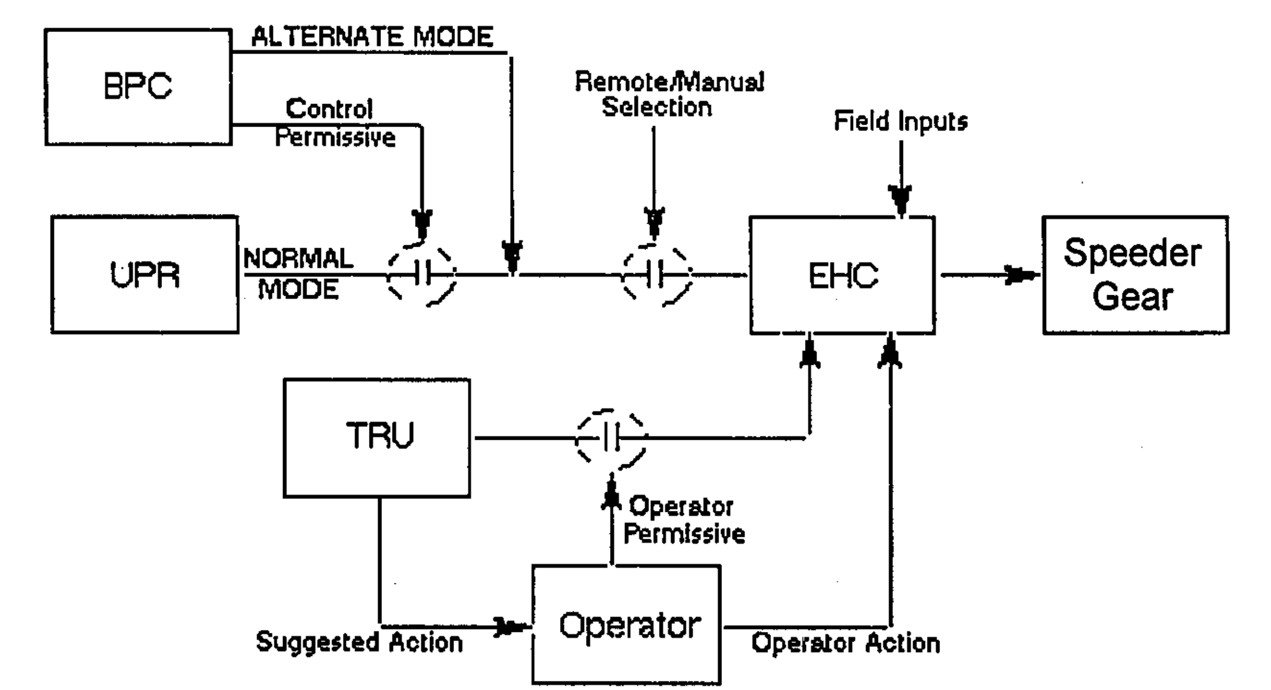


Figure 1: Block diagram of UPR control interactions

2.4 UPR CONTROL INPUTS AND OUTPUTS

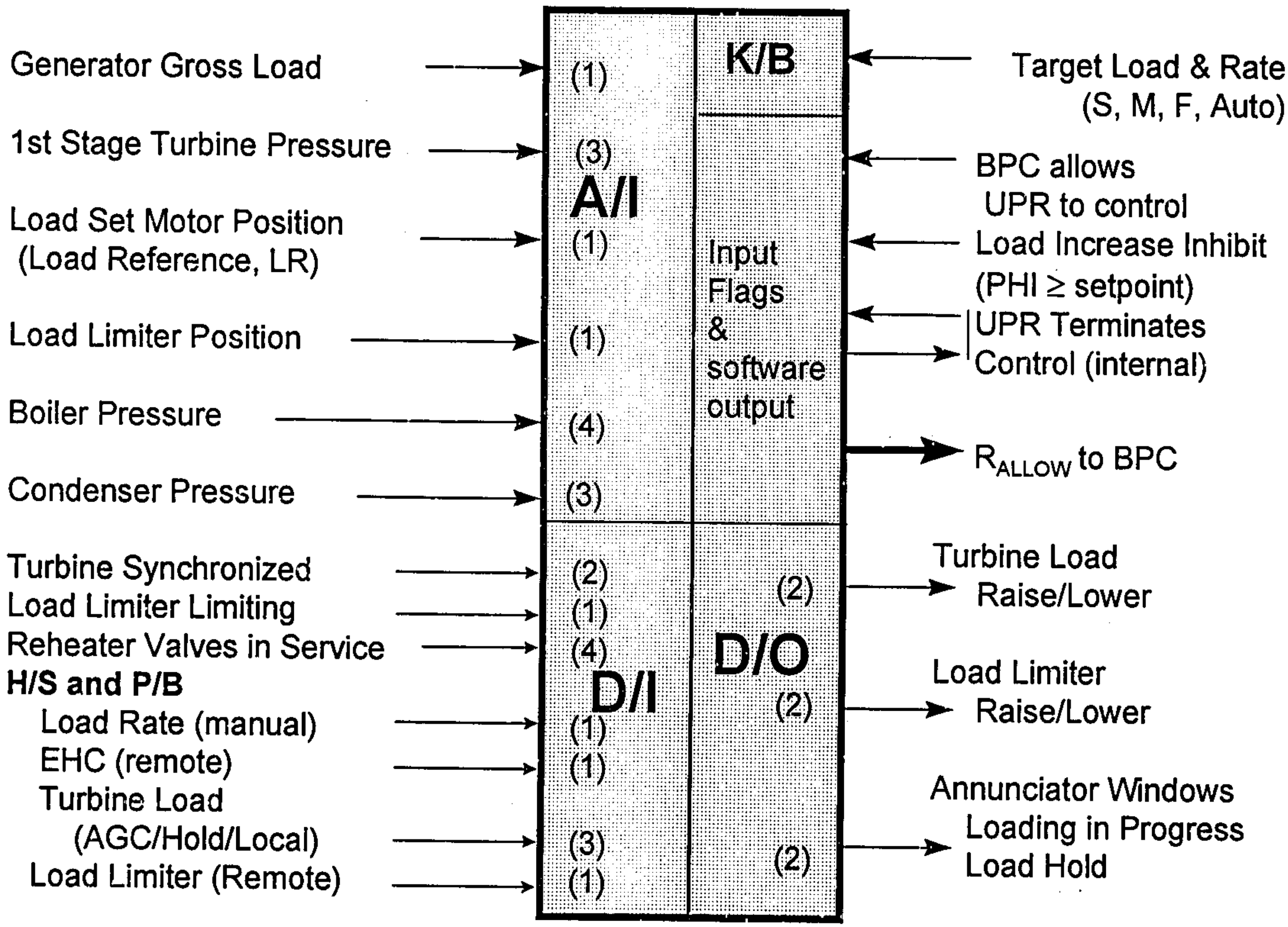


Figure 2: Input and Output Signals to and from UPR

2.5 TURBINE LOAD SET MOTOR CONTROL - SIMPLIFIED

- Load Reference (LR) is the setpoint this UPR pass (2 seconds)
- Error |LR-L| > 40 MW for 2 UPR passes terminates control
- Error determines Timed D/O pulse duration per FS figure # 2
- 2 D/Os to raise and 2 D/Os to lower LSM position

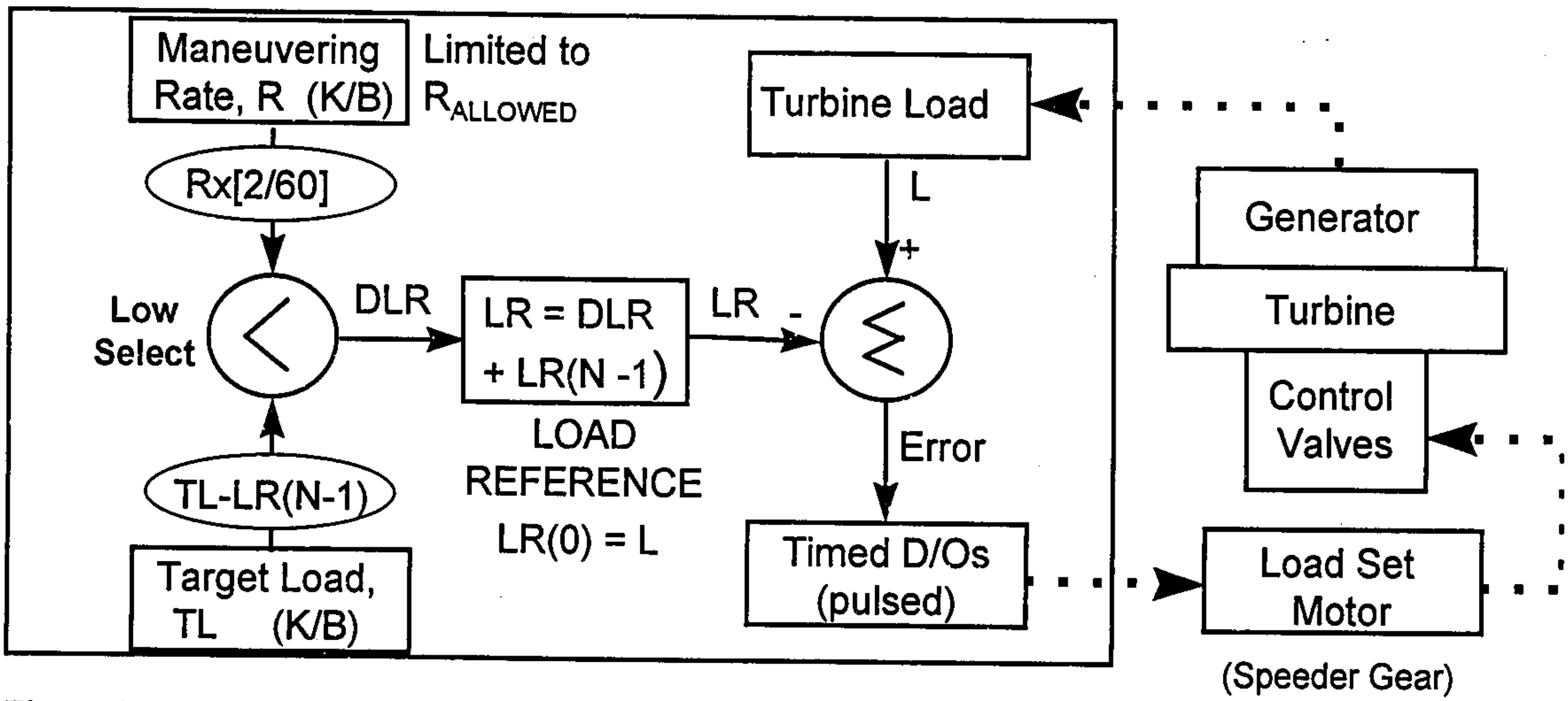


Figure 3: Simplified block diagram of Turbine Load Set Motor (Speeder Gear) Control

2.6 LOAD SET MOTOR CONTROL- PULSE DURATION

- When the contact is closed the Load Set Motor ramps the turbine at 17.6 MW/s (1056 MW/min)
  - The maximum ramp (approx ± 2.2%/s)* occurs when the contact stays closed for 2 seconds of the 2 second program cycle. * 17.6 MW/s ,2.2%/s = 800 MW for 100%
  - Slower ramps are achieved by closing the contact for a pulse time less than 2 seconds

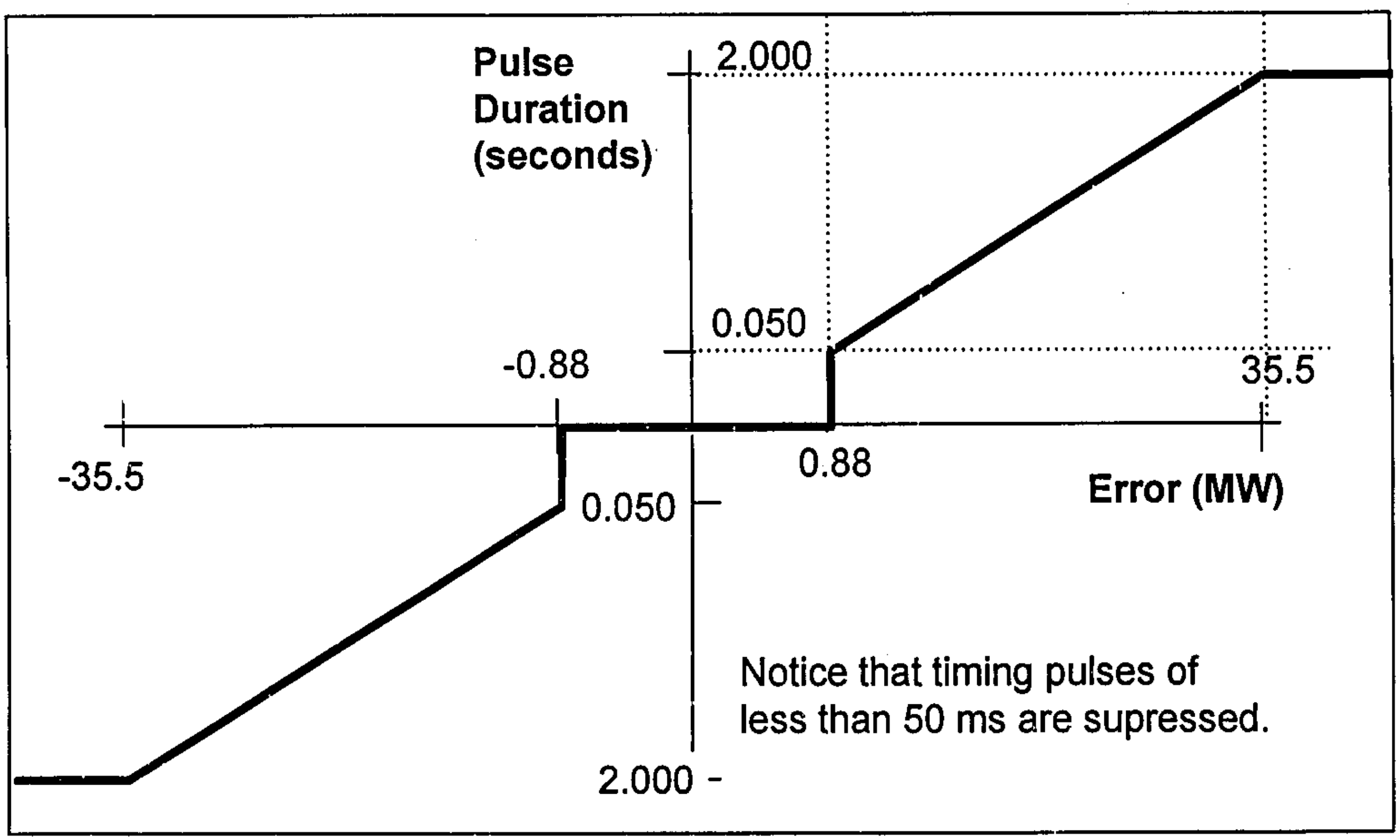


Figure 4: Load set motor control pulse duration as a function of Turbine-Generator MW Error

2.7 LOAD LIMITER CONTROL - SIMPLIFIED

- Load Limiter in 'Remote', UPR pulses the load limiter to 80MW above LR
- Error determines Timed D/O pulse duration (FS figure # 1)

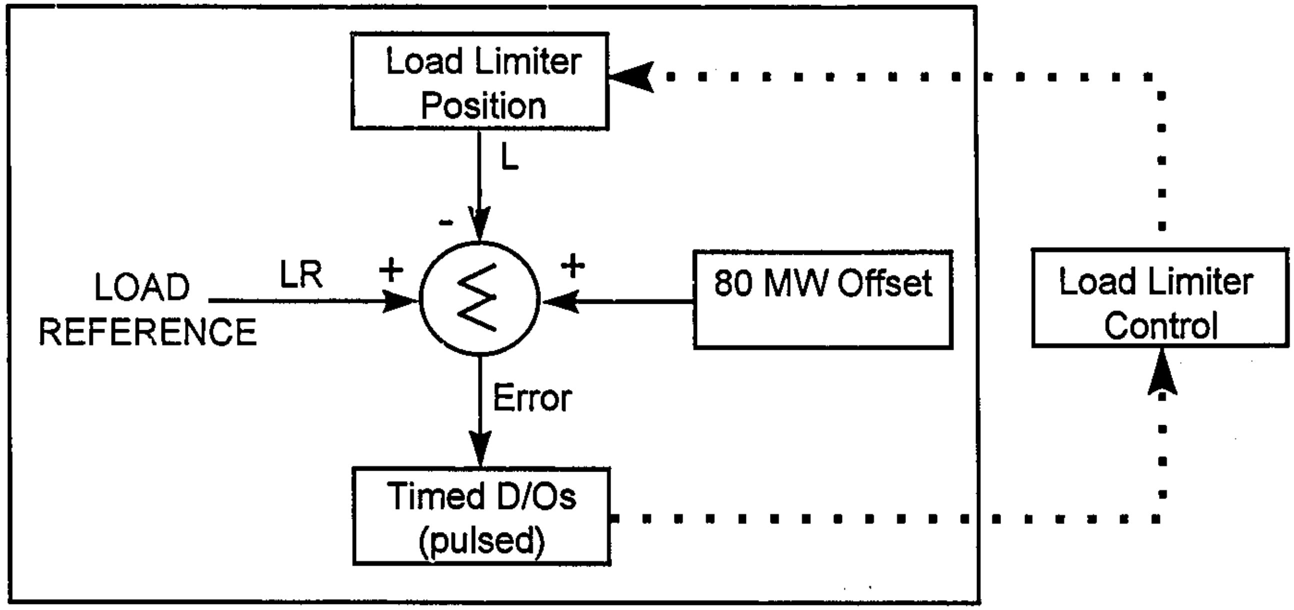


Figure 5: Simplified Load Limiter control diagram

2.8 LOAD LIMITER CONTROL - PULSE DURATION

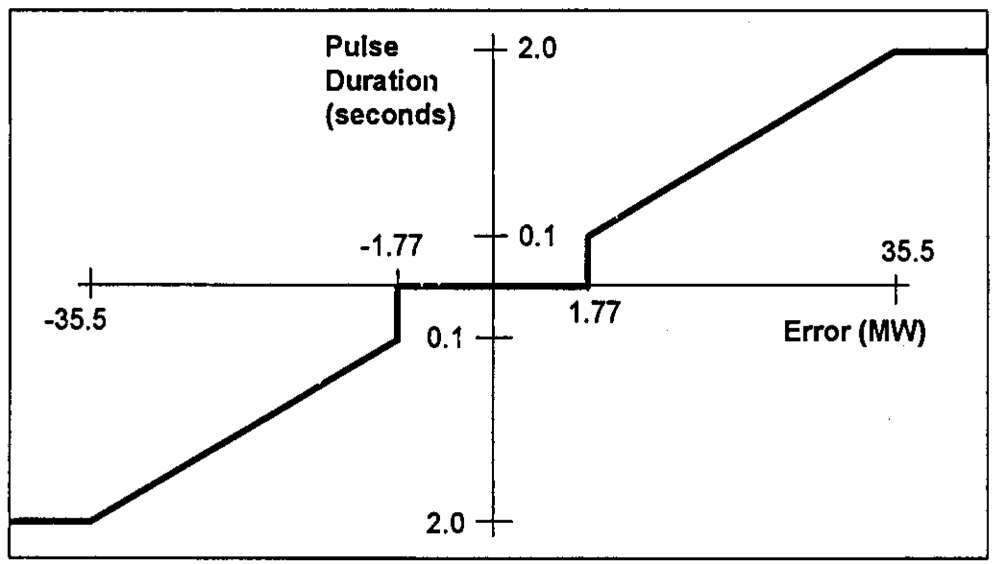


Figure 6: Load Limiter control pulse duration as a function of Turbine-Generator MW Error

3.0 UPR MONITORING ROUTINES (14)

- Turbine Control Status (2s)
- Vibration and Differential Expansion (10s)
- Turbine Stress Monitor (60s)
- Turbine Maneuvering Rate Calculation (60s)
- Actual Turbine Loading/Unloading Rate (10s)
- EHC Manual/Standby Rate and Stress (60s)
- Generator Auxiliary System (60s)
- Measured and Inferred Generator Load (2s)
- Moisture Separator/Reheater Monitor (60s)
- Boiler Pressure (2s)
- Condenser Vacuum (10s) Position (2s)
- Reload after Motoring (10s)
- L.P. Turbine Temp (10s)
- Governor CV

3.1 TURBINE CONTROL STATUS (2s)

- Checks EHC PB in 'Remote Operation' (alarm)
- Checks Load Limiter PB in 'Remote" (alarm)
- Checks if Load Limiter Limiting (alarm & inhibits loading)

3.2 TURBINE MANEUVERING RATE CALCULATION (60s)

- Uses stress ratios to calculate R_{ALLOW} [*Stress is taken as positive for heating (loading) and negative for cooling (unloading)*].
- Rate limit applies only to load increase
- Most restrictive stress of HP rotor surface, LP rotor surface and LP wheel is used
- Defaults to 12 MW/minute if Turbine Stress Monitor unavailable
- Allowable maneuvering rate calculated as per Figure #

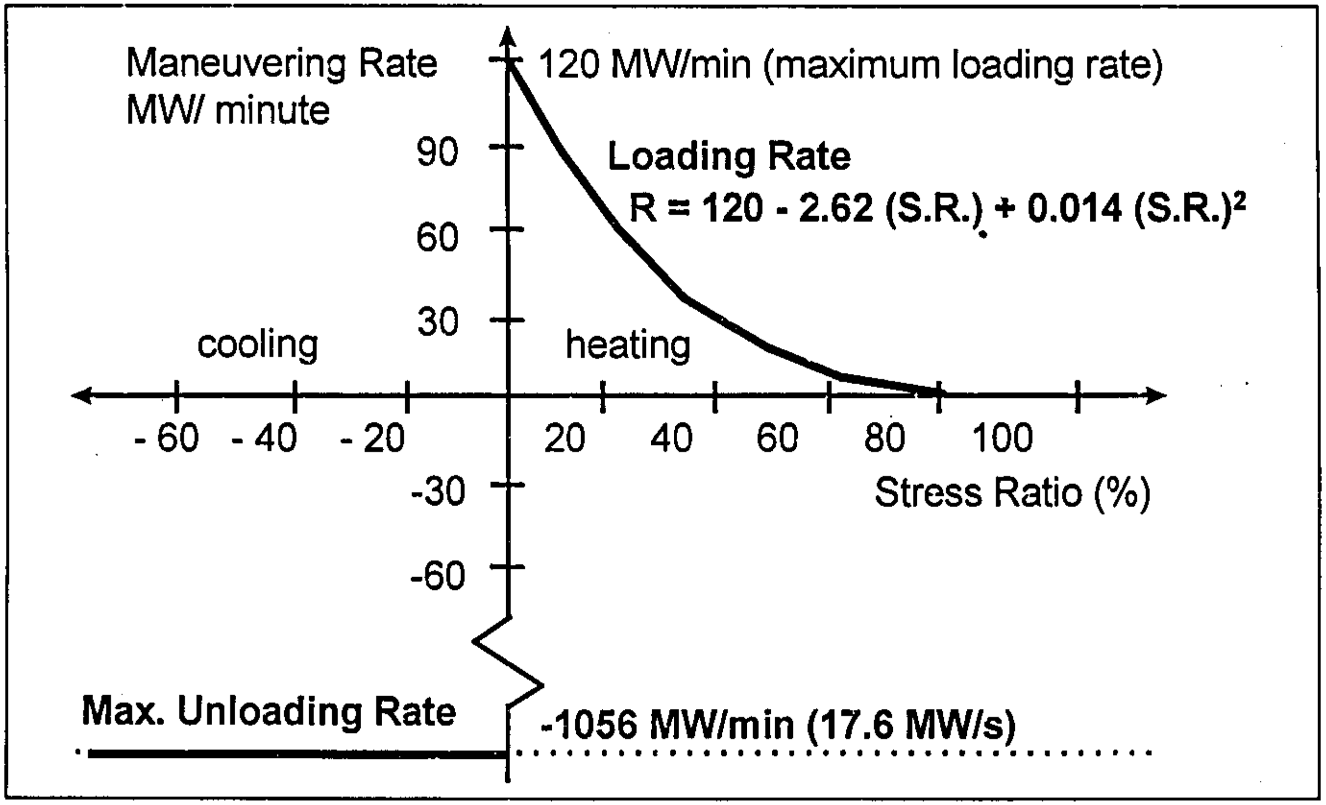


Figure 7: Turbine Maneuvering Rate as a function of the Turbine Stress Ratio

4.0 COMPUTER OR PROGRAM FAILURES

- UPR program fails in Master DCC
    - if in NORMAL mode: stops changing and remains at load - neither UPR or BPC controls turbine
    - if in ALTERNATE mode: no effect as BPC controls turbine (note that for NORMAL mode to be valid RRS, BPC and UPR must be operating in the same DCC)
    - fail indications: execute lamp on solid, acknowledge lamp off, control program window, CRT & printer alarm
    - procedure actions summary: attempt DCC panel program HS restart, transfer to healthy standby DCC, inform computer maintenance group
- UPR program fails in Standby Computer and/or Standby Computer fails
    - no effect on turbine control
    - same failure indications as given earlier
    - procedure summary: attempt DCC panel program HS restart, inform computer maintenance group
- UPR program failure in both DCCs
    - if in NORMAL mode: stops changing and remains at load - neither UPR or BPC controls turbine, but BPC continues to adjust RP setpoint
    - if in ALTERNATE mode: BPC continues to control turbine
    - No monitoring by UPR
    - same failure indications as given before
    - procedure actions summary: as given before, plus possibly restart of standby DCC

CHAPTER 2: OVERALL UNIT CONTROL

MODULE E: SIMULATOR FAMILIARIZATION

MODULE OBJECTIVES:

At the end of this module, you will be able to describe:

1. the main features of the Classroom "Advanced Reactor" Demonstrator (CARDs) for BWR, PWR and CANDU NPPs;
2. the main features of the IAEA sponsored "Advanced Reactor" Simulations for BWR and PWR NPPs;
3. the main features of the AECL sponsored CANDU NPP simulator;
4. how to operate each of the above simulators;
5. how to perform unit power maneuvers on each of the above simulators.

0 INTRODUCTION TO DESKTOP SIMULATORS

- full scope replica training simulators are used extensively in the nuclear power plant operations to ensure that all control room operators and supervisors are trained in normal, abnormal and emergency events: these simulators are physically large, operate in real time, and expensive to manufacture and to maintain
- simulations of all important systems are used in the analysis, design and licensing of nuclear power plants; depending on the complexity of the particular simulation a wide variety of equipment types are used, but real time operation is not a usual requirement
- simulations are being used increasingly in educational settings to provide a better learning environment, and with the advances in computer technology personal computers are now able to execute complete power plant simulations that a decade ago required a room full of expensive computer equipment
- this course will use three types of power plant simulations using personal computers:
  a) the Classroom "Advanced Reactor" Demonstrator (CARDs) for BWR, PWR and CANDU NPPs, produced by CAE (Canadian Aviation Electronics)
  b) the "Advanced Reactor" Simulation that was produced by Micro-Simulation Technology (MST) of the USA under contract with the IAEA; it contains five different plant simulations, but only the ones for BWR and PWR will be used in this course
  c) the CANDU 9 simulator produced by Cassiopeia Technologies Inc. (CTI) of Canada

.0 CARDs

- CARDs consists of three simulations based on the following type of nuclear power plants:
  - ⇒ 670 MW BWR
  - ⇒ 1250 MW PWR
  - ⇒ 680 MW CANDU 6
- The CARDs package serves principally as a demonstrator of the different normal and transient operation of the three main types of commercially used nuclear power plants and does not have the operator interface necessary for the functions of a training simulator. Although the models used in the CARDs are subsets of CAE's advanced full scope simulators, the package has been reduced to only the necessary software required to demonstrate the general behavior of the plant as seen from the Nuclear Steam Supply System (NSSS) with all the boundaries to this system emulated to provide the overall dynamic response.
- The demonstrators feature CAE's advanced reactor models that is a three-dimensional, multi-nodal, reactor kinetics model founded on the fundamental theory of time-dependent neutron diffusion; and advanced thermal-hydraulic model that includes multi-phase non-equilibrium, non-homogeneous, drift-flux representation.
- The demonstrators consist of the complete neutronic and thermal-hydraulic model of the reactor and reactor coolant systems. The interfaces with the neighboring systems are emulated to provide a realistic feedback. The demonstrators employ a user-friendly, graphic-based man machine interface to manipulate the inputs, insert malfunctions, and display the behavior of the systems in a dynamic color-coded manner.
- The CARDs package uses the windows environment with standard pull-down menus to perform the demonstrator functions.
- The schematic for each CARD has been shown in Module 2B

.1 CARDs pull-down menus

- there are 3 schematics for each CARD:
  - ⇒ Overview: displaying the reactor and reactor coolant system
  - ⇒ Slices: displaying the reactor core flux and temperature distribution
  - ⇒ Malfunctions: displaying the malfunctions available.
- plot function:
  - ⇒ New plot: opens the plotter
  - ⇒ Plot parameters: opens a preset choice of parameters to choose from
- Options:
  - ⇒ Run/Freeze: Runs or freezes the simulation
  - ⇒ Reset: the user can reset to either Full Power Steady State (FPSS) or a previously recorded snapshot
  - ⇒ Snapshot: Allows the user to snap a certain condition for later use.
  - ⇒ Readout: option to display or mask the readouts
  - ⇒ Power Reduction: enables the user to reduce the power to a new level.

.2 BWR CARD

- The BWR CARD simulates the core neutronic behavior of the vessel core and the thermal-hydraulic behavior of the vessel from the feedwater inlet to the main steam lines.
- Feedwater, main steam and emergency core injection flows are emulated to provide adequate dynamics during transients. The logic is also functionally emulated.
- Power changes in BWR CARD are made through varying the core flow by varying the valve position at the discharge of the recirculation pump; increasing the flow results in increasing the power, and vice versa, decreasing the flow decreases the power.
- The following malfunctions are made available through the malfunction page:
  - ⇒ Main steam line break
  - ⇒ Loss of coolant accident with and without emergency core cooling
  - ⇒ Loss of feedwater
  - ⇒ Reactor scram
  - ⇒ Dual pump trip
  - ⇒ One pump trip
- The malfunctions can be initiated by pointing and clicking on the specific malfunction window.

.3 PWR CARD

- The PWR CARD simulates the core neutronic behavior of the PWR core using 8 axial and 49 radial control volumes and the thermal-hydraulic behavior of the Reactor Cooling System (RCS) and the steam generators from the feedwater inlet to the main steam lines.
- Feedwater, main steam, and emergency core cooling flows are emulated to provide adequate dynamics during transients. The logic is also functionally emulated.
- Power changes are made through varying the reactor power setpoint using a rate defined by the user.
- The following malfunctions are made available through the malfunction page:
  - ⇒ Steam Generator malfunctions (Main steam line break and tube rupture)
  - ⇒ Pressurizer PORV stuck open
  - ⇒ Loss of Feed water
  - ⇒ Reactor Trip
  - ⇒ RCS Pump trip (individual or all)
  - ⇒ RCS Loss of coolant accident (Cold or hot leg)
  - ⇒ Control Rod drop
  - ⇒ Miscellaneous (inhibition of reactor trip or safety injection)
- The malfunctions can be initiated by pointing and clicking on the specific malfunction window.

4 CANDU CARD

- The CANDU CARD simulates the core neutronic behavior of the CANDU core using 6 axial and 49 radial control volumes and the thermal-hydraulic behavior of the Primary Heat Transport (PHT) and all the major surrounding systems such as feedwater, main steam, emergency core cooling, and the corresponding logic associated with these models.
- The major DCC programs are emulated to provide functionality of the DCC control. The programs emulated are:
  - ⇒ Heat Transport Control (HTC)
  - ⇒ Boiler Level Control (BLC)
  - ⇒ Boiler Pressure Control (BPC)
  - ⇒ Reactor Regulating System (RRS)
  - ⇒ Moderator Temperature Control (MTC)
- Power changes are made by varying the reactor power setpoint in RRS using a rate defined by the user.
- The following malfunctions are made available through the malfunction page:
  - ⇒ Steam Generator malfunctions (Main steam line break and tube rupture)
  - ⇒ Pressurizer steam break
  - ⇒ Loss of Feed water
  - ⇒ Reactor Trip
  - ⇒ PHT Pump trip (individual or all)
  - ⇒ PHT Loss of coolant accident (Inlet or outlet header breaks)
  - ⇒ Liquid Relief Valve (LRV) failures
  - ⇒ Miscellaneous (inhibition of reactor trip or safety injection)

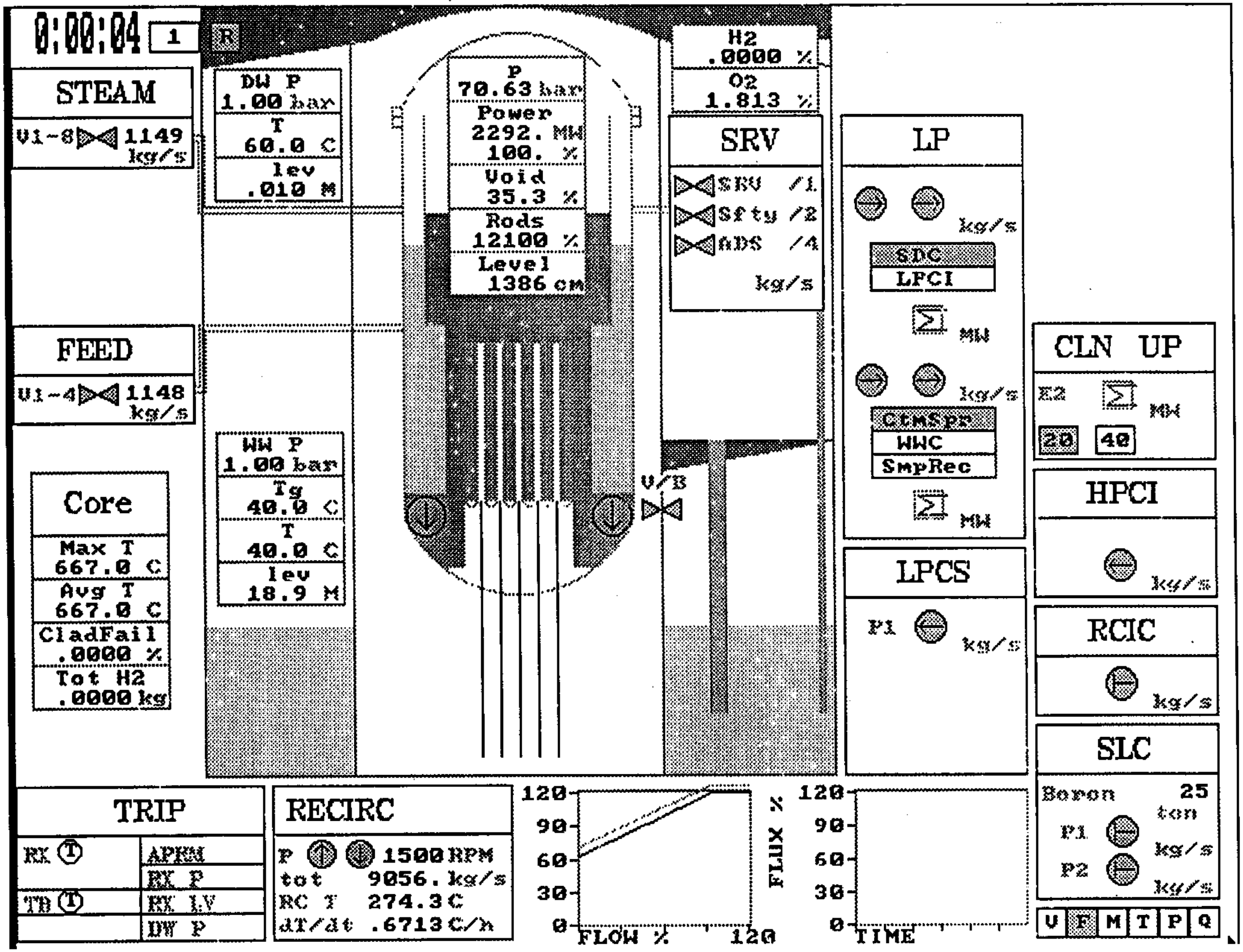
.0 IAEA "ADVANCED REACTOR" SIMULATIONS

- The ARS suite of five simulations represent NPPs in the 600-700 MW range; the following types of plants are modeled:
  - ⇒ Pressurized Water Reactor (PWR)
    - PWRs with Inverted U-Bend Steam Generators
    - PWRs with Horizontal Steam Generators
    - Advanced PWR with Passive Safety Systems
  - ⇒ Boiling Water Reactor (BWR)
  - ⇒ Pressurized Heavy Water Reactor (PHWR) - this simulation is incorrect as released in November 1997 and should not be used
- Each of the simulations includes the complete Nuclear Steam Supply System (NSSS), containment, control systems, and safety systems that affect normal and abnormal plant operation.
- The simulations generally replicate the behavior/performance of the actual plants under all design basis (normal and abnormal) conditions.
- The mathematical models are much simpler than for full scope simulators (and CARDs), and only the main systems and equipment are represented.
- A point kinetics model for reactor power calculation and has added a containment model. By including fuel and containment condition simulation in addition to the original NSSS model, ARS is a complete, if very much simplified, nuclear plant simulator.
- For PWRs, a non-equilibrium pressurizer model handles its normal controls by the spray, heater and relief valves. It also allows sudden changes and extreme conditions such as "water solid" in the pressurizer and two-phase in the reactor core. The loop flow model accommodates individual pump trip and possible reversed flows.

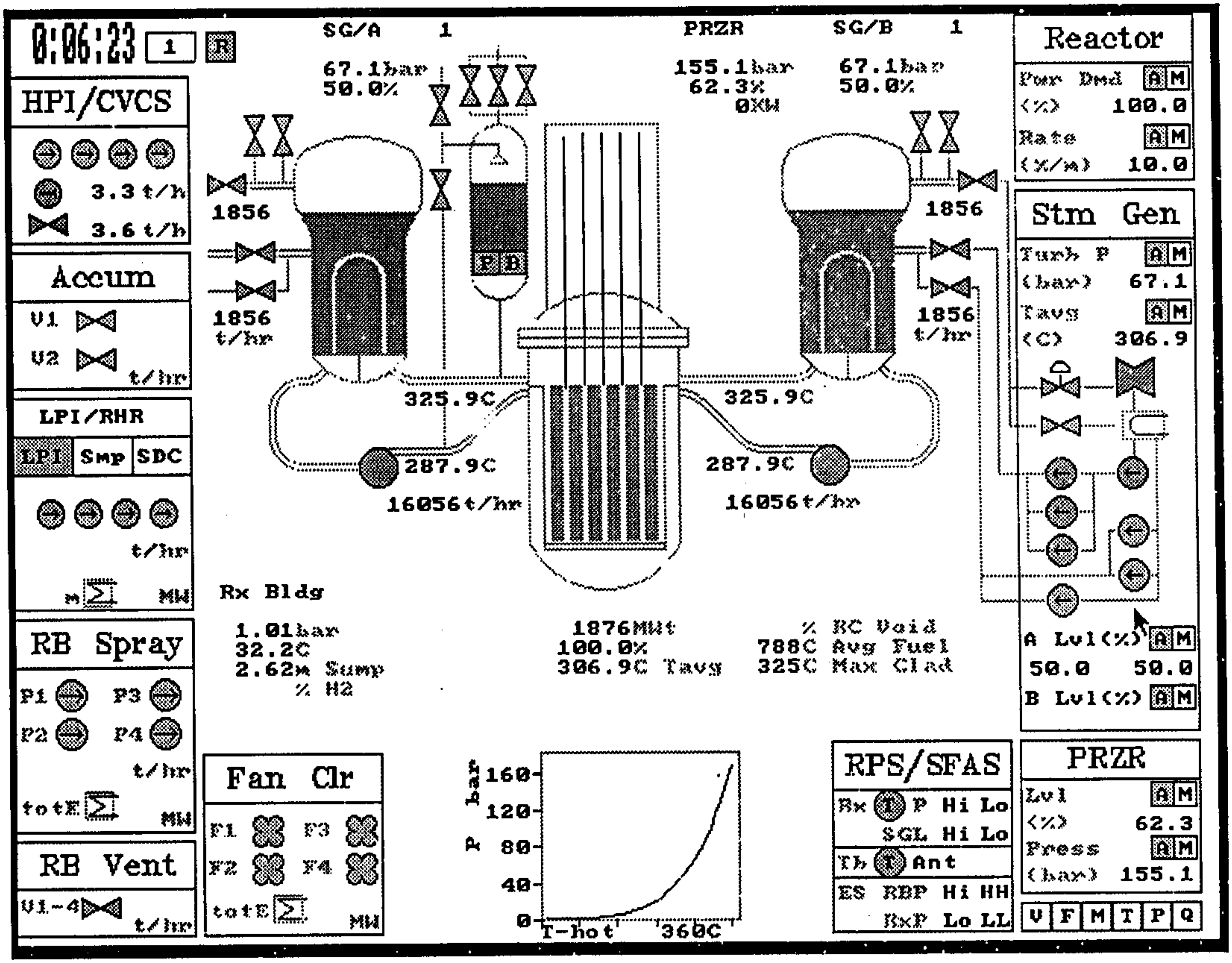
0 IAEA "ADVANCED REACTOR" SIMULATIONS (continued)

- The steam generators are modeled as homogeneous equilibrium two-phase volumes. Heat transfer from the primary to the secondary is treated rigorously during both forced and natural circulation.
- The fluid discharge rate from a break uses typical critical flow models. A mechanistic model of the coolant flow covering both forced and natural circulation provides temperature distribution in the primary coolant.
- The containment conditions are calculated based on a homogeneous equilibrium model with participation of non-condensable air and hydrogen. During a severe accident with the core being exposed to steam for extended period of time, the core may become overheated. Zirconium in the cladding may react with steam and hydrogen will be generated. Simulation of clad failure and hydrogen content in the containment is included.
- The mass and energy balance equations with correlations in momentum and heat transfer are solved for all control volumes simultaneously. Transient progress is handled by using Euler integration over every time step increment. Key plant parameters are then displayed graphically on a mimic.

1 BWR SIMULATION DISPLAY



.2 PWR SIMULATION DISPLAY



.0 CTI's CANDU SIMULATOR

- The Simulator is based on the CANDU 9 (925 MWe) unit; it includes simplified dynamic models of all the main process, control and safety systems, such as reactor, heat transport, steam generator, turbine, feedwater, electric generator, output and plant distribution systems.
- Depending on the configuration, 20-40 graphic display pages are used to mimic the actual control panel instrumentation, including the plant display system. These "pages" are used to monitor and control the plant's responses, providing for extensive interactions with the simulator and enabling the user to monitor many more parameters than is possible with the simulators using only one display for the power plant. In the Phase 2 release used in this course, the following display pages are available:

1. Plant Overview
2. Shutdown Rods
3. Reactivity Control
4. Liquid Zones Control
5. Zonal Flux Trends
6. Flux Mapping
7. PHT Main Circuit
8. PHT Feed & Bleed
9. PHT Inventory Control
10. PHT Pressure Control
11. Bleed Condenser Control
12. Steam Generator Feed Pumps
13. Steam Generator Level Control Steam
14. Generator Level Trends
15. Steam Generator Level Manual Ctrl
16. Extraction Steam
17. Turbine Generator
18. RRS / DPR
19. UPR
20. Electrical GRP1 Class IV
21. Electrical GRP1 Class III
22. Electrical GRP1 Class I/II A
23. Electrical GRP1 Class I/II B
24. Electrical GRP1 Class I/II C
25. Elect GRP1 Class IV Loads 1
26. Elect GRP1 Class IV Loads 2
27. Elect GRP1 Class III Loads 1
28. Elect GRP1 Class III Loads 2

1 SIMULATOR DISPLAY COMMON FEATURES

Revision 4 of the CANDU 9 Compact Simulator is made up of 28 interactive display screens or pages. All of these screens have the same information at the top and bottom of the displays, as follows:

- top of the screen contains 21 plant alarms and annunciations; these indicate important status changes in plant parameters that require operator actions; each of these alarms will be discussed as part of the system that is generating it and/or is involved in the corrective action;
- top right hand corner shows the simulator status:
  - ⇒ the window under 'Labview' (this is the proprietory software that generates the screen displays) has a counter that is incrementing when Labview is running; if Labview is frozen (i.e. the displays cannot be changed) the counter will not be incrementing;
  - ⇒ the window displaying 'CASSIM' (this is the proprietory software that computes the simulation responses) will be green and the counter under it will not be incrementing when the simulator is frozen (i.e. the model programs are not executing), and will turn red and the counter will increment when the simulator is running;
- to stop (freeze) Labview click once on the 'STOP' sign at the top left hand corner; to restart 'Labview' click on the ⇒ symbol at the top left hand corner;
- to start the simulation click on 'Run' at the bottom right hand corner; to the stop simulation click on 'Freeze' at the bottom right hand corner;

2 SIMULATOR DISPLAY COMMON FEATURES (continued)

the bottom of the screen shows the values of the following major plant parameters:

⇒ Reactor Neutron Power (%)

⇒ Reactor Thermal Power (%)

⇒ Generator Output (%)

⇒ Main Steam Header Pressure (kPa)

⇒ Steam Generator Level (m)

⇒ OUC Mode ('Normal' or 'Alternate')

- the bottom left hand corner allows the initiation of two major plant events:

⇒ 'Reactor Trip'

⇒ 'Turbine Trip'

these correspond to hardwired push buttons in the actual control room;

- the box above the Trip buttons shows the display currently selected (i.e. 'Plant Overview'); by clicking and holding on the arrow in this box the titles of the other displays will be shown, and a new one can be selected by highlighting it;
- the remaining buttons in the bottom right hand corner allow control of the simulation one iteration at a time ('Iterate'); the selection of initialization points ('IC'); insertion of malfunctions ('Malf'); and calling up the 'Help' screen.

3 PLANT OVERVIEW PAGE

Shows a 'line diagram' of the main plant systems and parameters. No inputs are associated with this display. The systems and parameters displayed are as follows (starting at the bottom left hand corner):

MODERATOR system is not simulated

REACTOR is a 14 zone model, each zone being represented by a point kinetic model with six groups of delayed neutrons, and coupling coefficients that account for the interaction of the flux between adjacent zones; decay heat model uses a three group approximation; reactivity calculations include reactivity control and safety devices, Xenon, voiding in channels and power level changes. The parameters displayed are:

⇒ Average Zone Level (% full)

⇒ Neutron Power (% full power)

⇒ Neutron Power Rate (%/ second)

Heat Transport main loop, pressure and inventory control systems are shown as a single loop on the Plant Overview display, additional details will be shown on subsequent displays. The parameters displayed are:

⇒ Reactor Outlet Header (ROH) and Reactor Inlet Header (RIH) average Temperature (°C) and Pressure (kPa)

⇒ Pressurizer Level (m) and Pressure (kPa); D_2O Storage Tank level (m)

3 PLANT OVERVIEW PAGE (continued)

- The four Steam Generators are individually modeled, but only the level measurements are shown separately, for the flows, pressures and temperatures average values are shown. The parameters displayed are:

  ⇒ Boiler 1, 2, 3, 4 Level (m)

  ⇒ Steam Flow (kg/sec)

  ⇒ Steam Pressure (kPa)

  ⇒ Steam Temperature (°C)

  ⇒ Moisture Separator and Reheater (MSR) Drains Flow (kg/sec)

  ⇒ Status of control valves is indicated by their colour: green is closed, red is open; the following valves are shown for the Steam System:

  Main Steam Stop Valves (MSV) status only
  Condenser Steam Discharge Valves (CSDV) status and % open
  Atmospheric Steam Discharge Valves (ASDV) status and % open

- Generator output (MW) is calculated from the steam flow to the turbine
- Condenser and Condensate Extraction Pump (CEP) are not simulated
- Simulation of the feedwater system is very much simplified; the parameters displayed on the Plant Overview screen are:

  ⇒ Total Feedwater flow to the steam generators (kg/sec)

  ⇒ Average Feedwater temperature after High Pressure Heater (HPHX)

  ⇒ Status of Boiler Feed Pumps (BFP) is indicated as red if any pumps are 'ON' or green if all the pumps are 'OFF'

3 PLANT OVERVIEW PAGE (continued)

Six trend displays show the following parameters:

⇒ Reactor Neutron Power and Reactor Thermal Power (0-100%)

⇒ Turbine Power (0-100%)

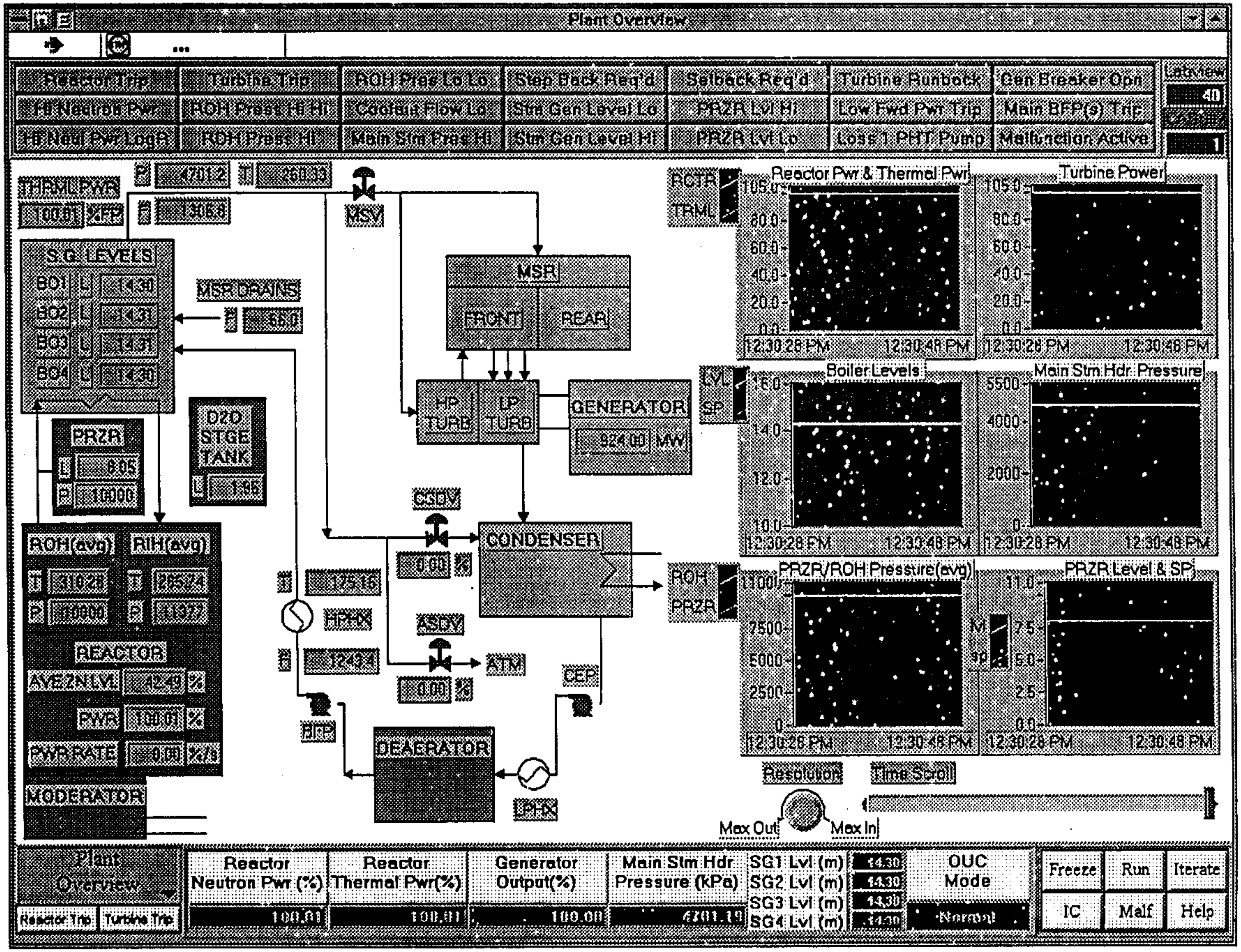
⇒ Boiler Levels - actual and setpoint (m)

⇒ Main Steam Header Pressure (kPa)

⇒ Pressurizer and Reactor Outlet Header (average) Pressure (kPa)

⇒ Pressurizer Level - actual and setpoint (m)

- Note that while the simulator is in the 'Run' mode, all parameters are being continually computed and all the displays are available for viewing and inputting changes.


Plant Overview
Reactor Trip
Turbine Trip
ROH Pres Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Hi Neutron Pwr
ROH Press Hi Hi
Coolant Flow Lo
Stm Gen Level Lo
PRZR Lvl Hi
Low Fwd Pwr Trip
Main BFP(s) Trip
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
PRZR Lvl Lo
Loss 1 PHT Pump
Malfunction Active
THRML PWR
%FP
S.G. LEVELS
MSR DRAINS
MSV
MSR
FRONT
REAR
HP TURB
LP TURB
GENERATOR
MW
PRZR
D2O STGE TANK
CSDV
CONDENSER
ASDV
ATM
CEP
HPHX
BFP
DEAERATOR
LPHX
ROH(avg)
RIH(avg)
REACTOR
AVE ZN LVL
PWR
PWR RATE
MODERATOR
Reactor Pwr & Thermal Pwr
Turbine Power
Boiler Levels
Main Stm Hdr Pressure
PRZR/ROH Pressure(avg)
PRZR Level & SP
Resolution
Time Scroll
Max Out
Max In
Plant Overview
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Main Stm Hdr Pressure (kPa)
SG1 Lvl (m)
SG2 Lvl (m)
SG3 Lvl (m)
SG4 Lvl (m)
OUC Mode
Normal
Freeze
Run
Iterate
IC
Malf
Help

TURBINE GENERATOR PAGE

Shows the main parameters and controls associated with the Turbine and the generator. The parameters displayed are:

Boiler 1, 2, 3, 4 Level (m)

status of Main Steam Safety Valves (MSSV)

status, opening and flow through the Atmospheric Steam Discharge Valves (ASDV) and the Condenser Steam Discharge Valves (CSDV)

Steam Flow to the Turbine (kg/sec)

Governor Control Valve Position (% open)

Generator Output (MW)

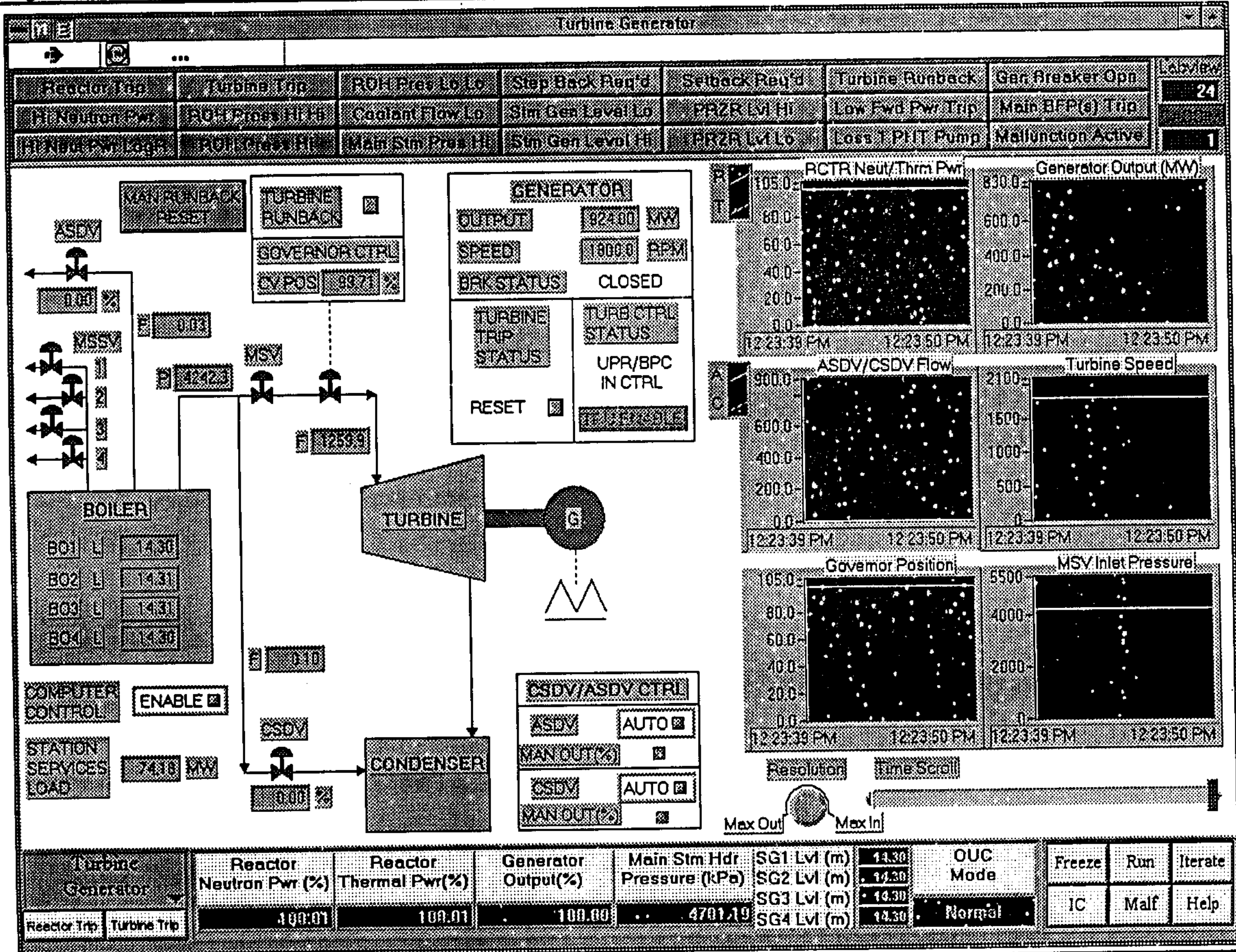
Turbine/Generator Speed of Rotation (rpm)

Generator Breaker Trip Status

- Turbine Trip Status
- Turbine Control Status
- All the trend displays have been covered elsewhere or are self explanatory

The following pop-up menus are provided:

- TURBINE RUNBACK - sets Target (%) and Rate (%/sec) of runback when 'Accept' is selected
- TURBINE TRIP STATUS - Trip or Reset
- ASDV and CSDV AUTO/MANUAL Control - AUTO Select, following which the Manual Position of the valve may be set


Turbine Generator
Reactor Trip
Turbine Trip
ROH Pres Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Hi Neutron Pwr
ROH Press Hi Hi
Coolant Flow Lo
Stm Gen Level Lo
PRZR Lvl Hi
Low Fwd Pwr Trip
Main BFP(s) Trip
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
PRZR Lvl Lo
Loss 1 PHT Pump
Malfunction Active
MAN RUNBACK RESET
TURBINE RUNBACK
GOVERNOR CTRL
CV POS 99.71 %
GENERATOR
OUTPUT 824.00 MW
SPEED 1800.0 RPM
BRK STATUS CLOSED
TURBINE TRIP STATUS
TURB CTRL STATUS
UPR/BPC IN CTRL
RESET
ASDV
MSSV
MSV
BOILER
TURBINE
CONDENSER
COMPUTER CONTROL
ENABLE
STATION SERVICES LOAD 74.18 MW
CSDV
CSDV/ASDV CTRL
ASDV AUTO
MAN OUT(%)
CSDV AUTO
MAN OUT(%)
RCTR Neut/Thrm Pwr
Generator Output (MW)
ASDV/CSDV Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
12:23:39 PM
12:23:50 PM
Resolution
Time Scroll
Max Out
Max In
Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 100.01
Reactor Thermal Pwr(%) 100.01
Generator Output(%) 100.00
Main Stm Hdr Pressure (kPa) 4701.19
SG1 Lvl (m) 14.30
SG2 Lvl (m) 14.30
SG3 Lvl (m) 14.30
SG4 Lvl (m) 14.30
OUC Mode Normal
Freeze
Run
Iterate
IC
Malf
Help

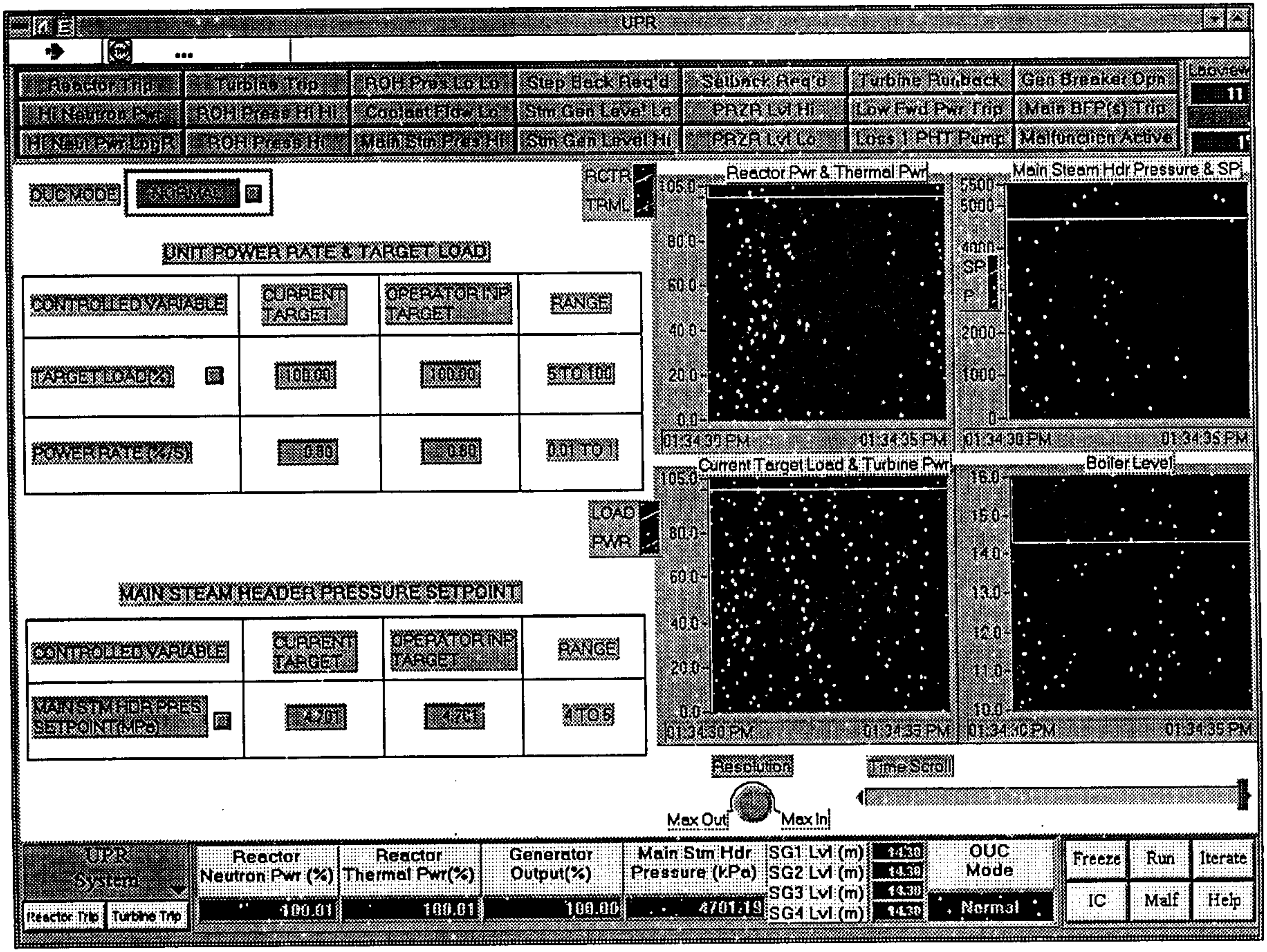
5 UPR PAGE

This screen permits control of station load setpoint and its rate of change while under Unit Power Regulator (UPR) control, i.e. 'normal' mode. Control of the Main Steam Header Pressure is also through this screen, but this is not usually changed under normal operating conditions.

- OUC (overall Unit Control) MODE can be changed from NORMAL to ALTERNATE.
- TARGET LOAD - on selection Station Load (%) and Rate of Change (%/sec) can be specified; change becomes effective when 'Accept' is selected.
  - ⇒ The OPERATOR INP TARGET is the desired setpoint inserted by the operator; the CURRENT TARGET will be changed at a POWER RATE specified by the operator.
  - ⇒ Note that the RANGE is only an advisory comment, numbers outside the indicated range of values may be input on the Simulator.
- MAIN STEAM HEADER PRESSURE SETPOINT (MPa) - alters the setpoint, which is rarely done during power operation. Caution must be exercised when using this feature on the Simulator, since the requested change takes place in a step fashion as soon as the change is made; changes should be made in increments of 0.1 MPa.

.6 POWER MANEUVER:

- To reduce unit power in the 'normal' mode do the following:
  - ⇒ select UPR display
  - ⇒ select 'TARGET LOAD (%)' pop-up menu
  - ⇒ in pop-up menu type in or use the arrows to select the 'target'
  - ⇒ 'Accept' and 'Return'


UPR
Reactor Trip
Turbine Trip
ROH Pres Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Hi Neutron Pwr
ROH Press Hi Hi
Coolant Flow Lo
Stm Gen Level Lo
PRZR Lvl Hi
Low Fwd Pwr Trip
Main BFP(s) Trip
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
PRZR Lvl Lo
Loss 1 PHT Pump
Malfunction Active
OUC MODE
NORMAL
UNIT POWER RATE & TARGET LOAD
CONTROLLED VARIABLE
CURRENT TARGET
OPERATOR INP TARGET
RANGE
TARGET LOAD(%)
100.00
100.00
5 TO 100
POWER RATE (%/S)
0.80
0.80
0.01 TO 1
MAIN STEAM HEADER PRESSURE SETPOINT
CONTROLLED VARIABLE
CURRENT TARGET
OPERATOR INP TARGET
RANGE
MAIN STM HDR PRES SETPOINT(MPa)
4.701
4.701
4 TO 5
Reactor Pwr & Thermal Pwr
RCTR
TRML
Main Steam Hdr Pressure & SP
SP
P
Current Target Load & Turbine Pwr
LOAD
PWR
Boiler Level
01:34:30 PM
01:34:35 PM
Resolution
Time Scroll
Max Out
Max In
UPR System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
100.01
Reactor Thermal Pwr(%)
100.01
Generator Output(%)
100.00
Main Stm Hdr Pressure (kPa)
4701.19
SG1 Lvl (m)
SG2 Lvl (m)
SG3 Lvl (m)
SG4 Lvl (m)
OUC Mode
Normal
Freeze
Run
Iterate
IC
Malf
Help

SIMULATOR EXERCISE 1.1: PWR Response to Power Reduction using CARDs

- **initialize CARDs PWR Simulation at 100%FP, reduce power in steps at 5%/min (trip the reactor for the 0% state) and record the following values:**

Parameter	Unit	100%	90%	75%	50%	25%	0%	Comments
Reactor Power	%							
Reactor Coolant Inlet Temp	°F							
Reactor Coolant Outlet Temp	°F							
Reactor Coolant Average Temp	°F							
RC Hotleg Flow	lb/sec							
RC Coldleg Flow	lb/sec							
RC Coldleg Temperature	°F							
RC Hotleg Temperature	°F							
Pressurizer Level	ft							
Pressurizer Pressure	psi							
Pressurizer Temperature	°F							
Boiler Level	ft							
Boiler Pressure	psi							
Boiler Temperature	°F							
Steam Flow	lb/sec							
Feedwater Flow	lb/sec							

Under "Comments" please note type of parameter change as a function of reactor power 0% → 100%FP: constant, linear increase or decrease, non-linear increase or decrease

SIMULATOR EXERCISE 1.2: BWR Response to Power Reduction using CARDs

- initialize CARDs BWR Simulation at 100%FP, reduce recirculation flow control valve opening in steps at a rate of 5%/min and record the values in the Table
- trip the reactor and record the values

		Recirc Control Valve Opening					Rx	
Parameter	Unit	95%	85%	75%	65%	55%	Trip	Comments
Reactor Power	%							
Reactor Coolant Core Inlet Flow	lb/sec							
Reactor Coolant Core Inlet Temp	°F							
Reactor Vessel Level	ft							
Reactor Vessel Pressure	psi							
Reactor Vessel Steam Temp	°F							
RC Recirc Flow	lb/sec							
RC Recirc Pump Inlet Pressure	psi							
RC Recirc Pump Outlet Pressure	psi							
Feedwater Flow	lb/sec							
Steam Flow	lb/sec							

Under “Comments” please note type of parameter change as a function of reactor power 0% → 100%FP: constant, linear increase or decrease, non-linear increase or decrease

SIMULATOR EXERCISE 1.3: CANDU Response to Power Maneuver

- initialize CARDs CANDU Simulation at 100%FP, reduce power to 90% at 1%/min (trip the reactor for the 0% state) and record the following values:

Parameter	Unit	100%	90%	Reactor Trip	Comments
Reactor Power	%				
ROH Pressure	MPa				
ROH Temperature	°C				
RIH Pressure	MPa				
RIH Temperature	°C				
Pressurizer Level	m				
HT Pump Flow	Mg/s				
Boiler Pressure	MPa				
Boiler Temperature	°C				
Boiler Level	m				
Steam Flow	kg/s				
Feedwater Flow	kg/s				

Under "Comments" please note type of parameter change as a function of reactor power 0% → 100%FP: constant, linear increase or decrease, non-linear increase or decrease

SIMULATOR EXERCISE 1.4: PWR Response to Power Reduction using ARS

- Initialize ARS PWR Simulation at 100%FP, reduce power in 25% steps at 5%/min, plot the following parameters, and compare them to the trends in Exercise 1.1 (Note different units!)
- trip the reactor for 0% power
- plot the parameters using the simulator's plotting capability

Parameter	Unit	File	Label	100%	75%	50%	25%	0%	Comments
Reactor Power	%	E	PWR						
Reactor Coolant Average Temp	°C	A	TAVG						
Reactor Coolant Loop A Flow	t/hr	A	WRCA						
RC Coldleg Temperature	°C	A	TCA						
RC Hotleg Temperature	°C	A	THA						
Pressurizer Pressure	bar	A	P						
Pressurizer Level	%	B	LVPZ						
Boiler Pressure	bar	A	PSGA						
Boiler Level	%	C	NSGA						
Boiler Temperature	°C								
Steam Flow	t/hr	A	WSTA						
Feedwater Flow	t/hr	A	WFWA						

Under "Comments" please note type of parameter change as a function of reactor power 0% → 100%FP: constant, linear increase or decrease, non-linear increase or decrease

SIMULATOR EXERCISE 1.5: BWR Response to Power Reduction using ARS

- initialize ARS BWR Simulation at 100%FP, reduce recirculation pump speed to the values shown and record the values in the Table
- trip the reactor and record the values
- plot the parameters using the simulator's plotting capability

				Recirculation Pump Speed				Rx	
Parameter	Unit	File	Label	1500	1125	750	375	Trip	Comments
Reactor Power	%	A	PWTH						
Reactor Vessel Pressure	psi	B	P						
Reactor Vessel Void	%	A	VOID						
Reactor Vessel Level	cm	B	LEV						
Recirculation Flow (total)	kg/sec	A	FLOW						
Reactor Coolant Temp	°C	C	TAVG						
Steam Flow	kg/sec	D	WSTM						
Feedwater Flow	kg/sec	D	WMFW						

Under "Comments" please note type of parameter change as a function of reactor power 0% → 100%FP: constant, linear increase or decrease, non-linear increase or decrease

SIMULATOR EXERCISE 1.6: CANDU Response to Power Maneuver

- Initialize CTI CANDU Simulation at 100%FP, reduce power using UPR in 25% steps at 0.5%/sec (trip the reactor for the 0% state) and record the following values:

Parameter	Unit	100%	75%	50%	25%	0%	Comments
Reactor Power	%						
ROH Pressure	MPa						
ROH Temperature	°C						
RIH Pressure	MPa						
RIH Temperature	°C						
Pressurizer Level	m						
HT Pump Flow	Mg/s						
Boiler Pressure	MPa						
Boiler Temperature	°C						
Boiler Level	m						
Steam Flow	kg/s						
Feedwater Flow	kg/s						
Turbine-Generator Power	%						

Under "Comments" please note type of parameter change as a function of reactor power 0% → 100%FP: constant, linear increase or decrease, non-linear increase or decrease

CHAPTER 2: OVERALL UNIT CONTROL

MODULE F: BOILER PRESSURE CONTROL

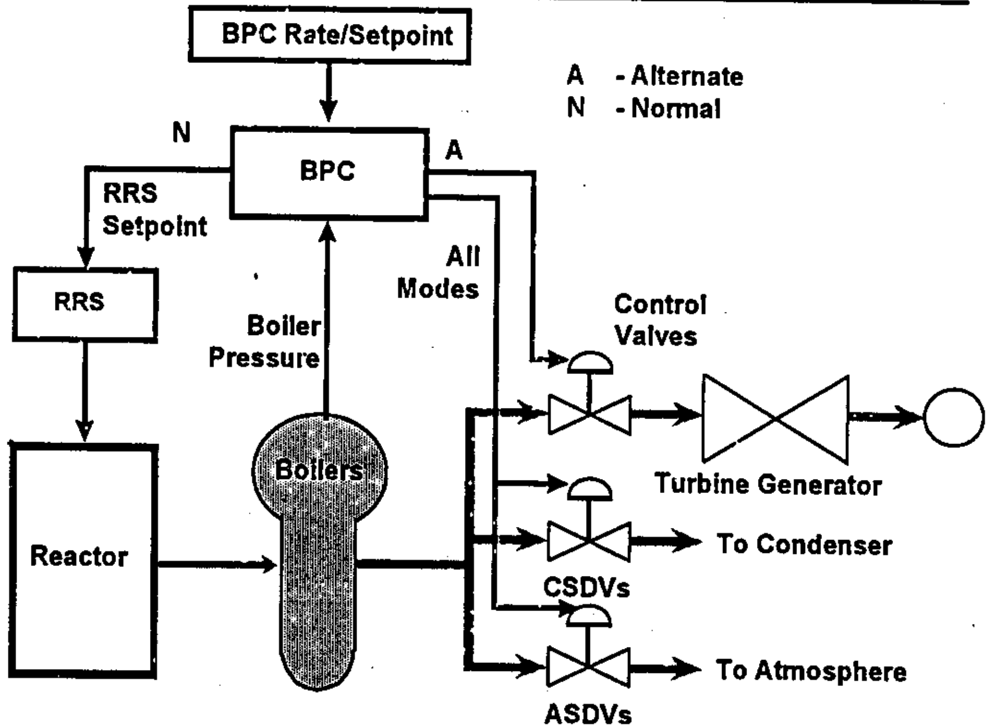
MODULE OBJECTIVES:

At the end of this module, you will be able to describe:

1. The main functions of BPC
2. The Control Modes and Operating States of BPC
3. The main input signals to and output signals from BPC
4. How BPC Controls Reactor Power
5. How BPC Controls Turbine Load
6. How BPC Controls the ASDVs and CSDVs
7. Cases of BPC failures

1.0 INTRODUCTION

- BPC maintains boiler pressure at the setpoint either by adjusting reactor power, or by manipulating steam loads, i.e BPC ensures that there is a heat flow balance between Reactor Power and Steam Load Requirements
- At Power, BPC Controls Boiler Pressure to a Fixed Setpoint
  - in NORMAL mode: BPC directs Reactor Power to follow changes in steam flow. (Reactor Lagging)
  - in ALTERNATE mode: BPC causes adjustments to steam flow (usually steam to turbine) to match a fixed reactor power output. (Reactor Leading)



- During Warm-Up and Cooldown of the Heat Transport System BPC ramps the Boiler Pressure Setpoint at the specified rate
- In all modes of operation, BPC controls the steam discharge valves to limit high steam pressure.
  - CSDVs - designed to accept 75% reactor power, permit operation of steam supply system in the event of a grid or turbine/generator fault
  - ASDVs - capacity of 11.2% reactor power, augment CSDV capacity on turbine trip to avoid opening of boiler safety relief valves.
  - provide means of pressure control when poor condenser pressure makes CSDVs unavailable.
- In case of low steam pressure the Turbine Hardware unloader (turbine protection, independent of BPC) runs back the governor.

2.0 BPC OPERATING STATES

2.1 Normal

- primary boiler pressure control is by BPC calculating Reactor Power Setpoint to minimize boiler pressure error
- secondary control via SDVs during transients (offsets: ASDV 70 kPa, CSDV 100 kPa)
- Speeder Control (Load Set Motor or LSM sets setpoint for steam flow to turbine)
  - usually UPR - LSM adjusts setting by up to ± 17.6 MW/s
  - operator can control the speeder by selecting it to manual
- Entry to NORMAL is by keyboard entry via RRS to "SET MODE NORMAL", or entering UPR rate/setpoint and the total SDV demand is < 1%FP
- Exit from NORMAL mode:
  - BPC forces RRS out of NORMAL (and takes control of the speeder), because:
    - turbine first stage pressures are irrational so BPC cannot calculate a reactor power setpoint, or
    - calculated reactor power setpoint > P_{HI} and boiler pressure is low by more than 100 kPa
  - RRS conditions for NORMAL fail, such as:
    - operator initiates a reactor power maneuver via the keyboard, or depresses the HOLD POWER push button, or
    - automatic action, including reactor trip, stepback, setback in progress, turbine trip or Power/Load Unbalance (PLU) condition, reactor power setpoint from BPC is outside the range 0.25% to 100.5% FP (or set values of P_{HI} and P_{LO}), the BPC program fails in the controlling computer, the RRS program fails in the controlling computer

2.2 Alternate

- primary boiler pressure control is by BPC adjusting the steam flow to the Turbine via the speeder (must have EHC on REMOTE)
- secondary control via SDVs if speeder on manual and during transients (offsets: ASDV 70 kPa, CSDV 100 kPa)
- Speeder Control is normally by BPC, but could also be by operator if selected to manual
- Reactor Power Setpoint is specified by the operator doing a keyboard entry to RRS
- Entry to ALTERNATE mode is:
    - via keyboard entry to PLANT CONTROL MODE SELECT, or
    - keyboard entry of Reactor Power Setpoint to RRS, or
    - on exit from NORMAL mode, or
    - if BPC and RRS are not running in the same computer
- Exit from ALTERNATE mode is by:
    - keyboard entry via RRS to "SET MODE NORMAL", or
    - entering UPR rate/setpoint and the total SDV demand is < 1%FP

3.0 HEAT BALANCE CONDITIONS

- Heat Input > Heat Output: Pressure Increasing
    - Unit Unloading (NORMAL: TURBINE LEADING)
    - Unit Loading (ALTERNATE: TURBINE LAGGING)
    - Warm-Up of the Boilers and Heat Transport System
    - Turbine Trip or Generation Rejection
- Heat Input Matches Heat Output - steady operation
- Heat Input < Heat Output: Pressure Decreasing
    - Unit Loading (NORMAL mode)
    - Unit Unloading (ALTERNATE mode)
    - Cooldown of the Boilers and HTS
    - Reactor Trip, Stepback or Setback

3.1 Why is Boiler Pressure key to unit power regulation?

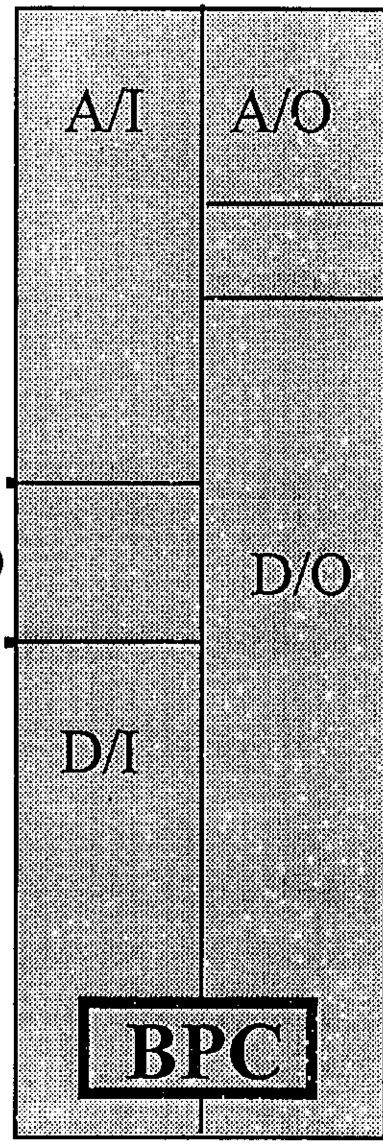
- The boiler links the secondary side to the primary side. A reactor power/steam load mismatch shows up in the boiler.
- The Boiler Steam & Water Mixture is saturated, so for any specific enthalpy there is a unique pressure and temperature.
- Pressure Measurements are <u>fast</u> while Temperature Measurements are <u>slow</u>
- Sensitivity of pressure measurement: at the usual Full Power Operating Pressure a ΔT of ½ °C corresponds to $\Delta P \cong 35$ kPa.

4.0 BPC INPUTS AND OUTPUTS

- BOILER STEAM PRESSURE (4)
- TURBINE 1ST STAGE P (3)
- CONDENSER PRESSURE (3)
- REACTOR POWER (28)
- ASDV POSITION (4)
- CSDV POSITION (6)
- LOAD LIMITER POS^n (1)
- LOAD SET MOTOR POS^n (1)
- GOVERNOR VALVE POS^n (1)

- KEYBOARD INPUTS
- NORMAL/ALTERNATE (from DPR)
- LOADING RATE LIMIT (from UPR)

- WARMUP/HOLD/COOLDOWN
- CSDV H/S & ASDV H/S 'AUTO'
- CSDV TRIPPED
- HP TURBINE P "IN"
- PROCESS INTERUPT
- TURBINE TRIP
- POWER/LOAD UNBALANCE
- PLU CONFIRMATION
- EHC IN REMOTE
- VACUUM UNLD^r RESET P/B



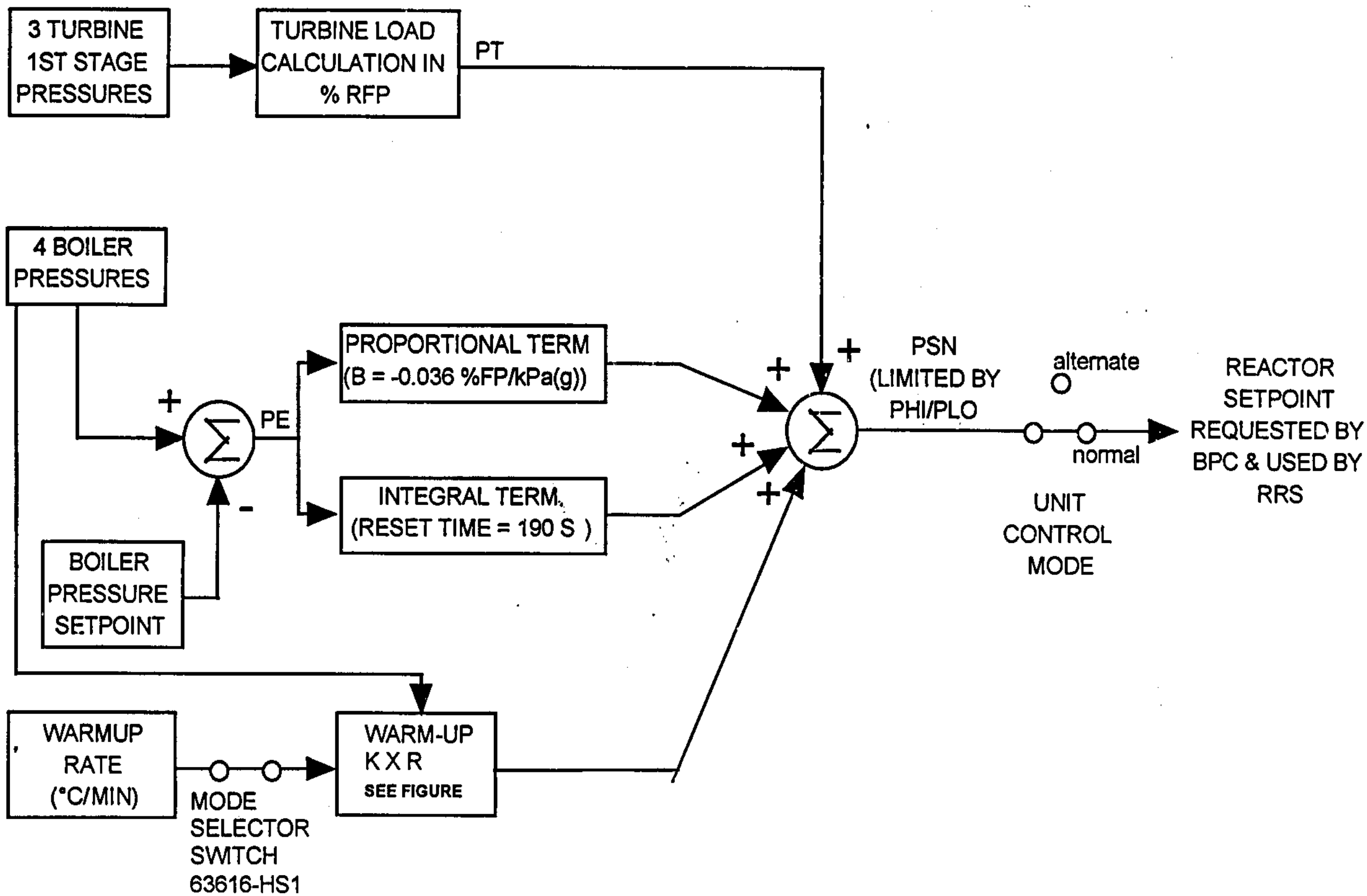
- CSDV LIFT (3)
- ASDV LIFT (4)

- PSN TO DPR

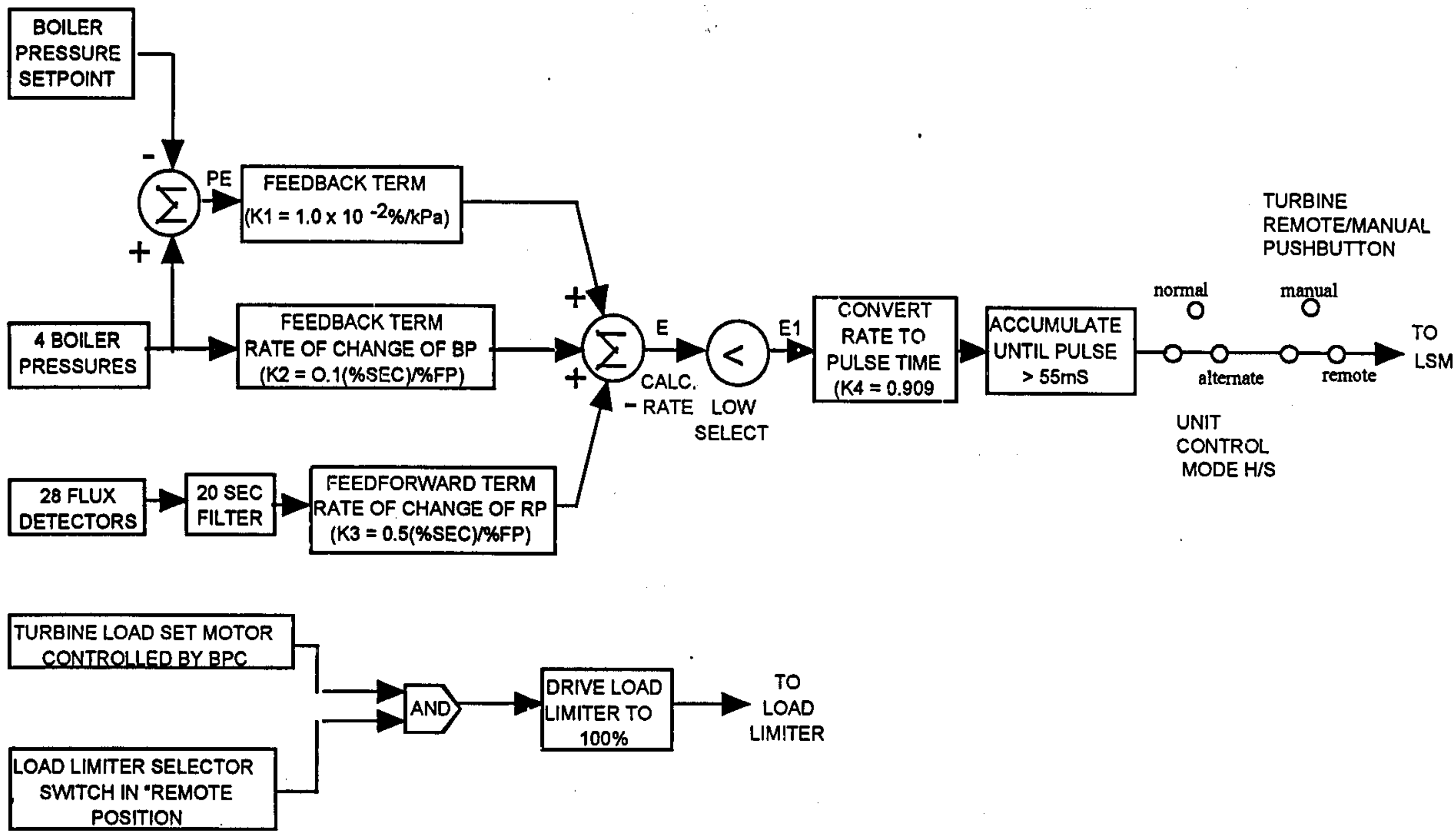
- LOAD LIMIT INC./DEC.
- RAISE/LOWER LSM (2.2%/S)
- CONDENSER P RESET P/B
- CONDENSER SPRAY VALVE
- CSDV UNLOADING HI P (WINDOW ANN.)
- RUNBACK IN PROGRESS (WINDOW ANN.)

- INPUT/OUTPUT SOFTWARE FLAGS NOT SHOWN

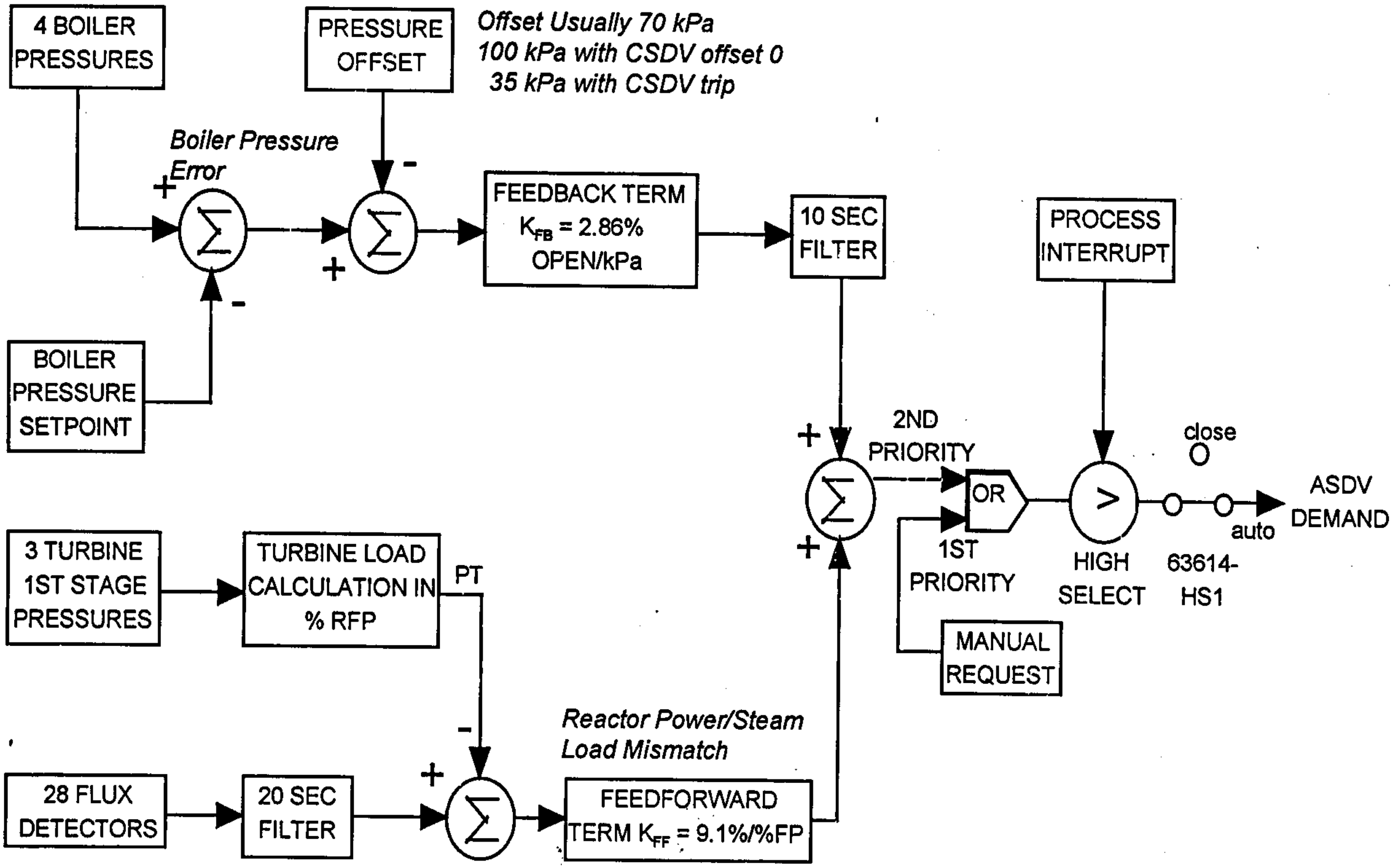
4.1 BPC COMPUTATION OF REACTOR POWER SETPOINT



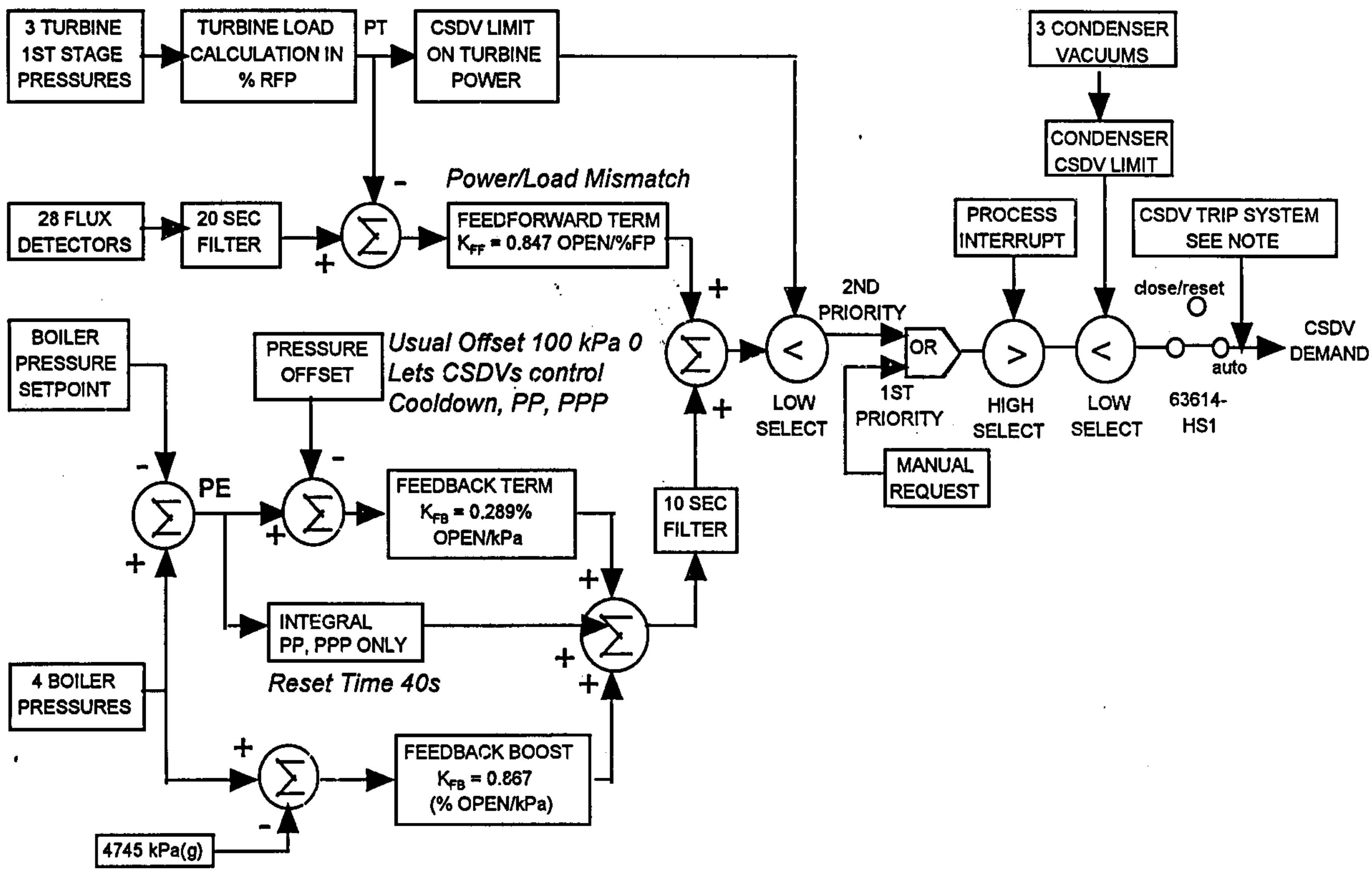
4.2 BPC COMPUTATION OF TURBINE LOAD SETPOINT



4.3 BPC COMPUTATION OF ASDV CONTROL SIGNAL



4.4 BPC COMPUTATION OF CSDV CONTROL SIGNAL



4.5 BPC FAILURE MODES

- 3 or 4 bad boiler pressure signals:
  - 4 good - take 2nd highest
  - 3 good - take median
  - 2 good - take higher
  - 1 or 0 good - fail BPC.
- loss of CSDV control: BPC looks at feedback from the 3 CSDV A/Os
  - if one is in error by >10% from demanded valve position, it checks the actual position of the two valves
  - if both CSDVs controlled by that A/O are more than 20% in error, BPC sets a BADFEEDBACK flag
  - BPC repeats above for next A/O
  - if the BADFEEDBACK flag is present for 2 or more pairs of valves, BPC is failed.
- BPC Failure in Master - the following automatic actions take place
  - goes to ALTERNATE mode
  - RRS continues to control in Master
  - BPC Controls in Alternate
  - UPR Monitors
- BPC Failure in Slave - No Effect
  - but will be a problem on transfer of control

4.5 BPC FAILURE MODES (continued)

- BPC Fails in both DCCX & DCCY
  - There is no Pressure Regulation
  - Protection comes from:
    - setback
    - steam safeties
    - turbine unloading on low pressure
  - Automatic switch to ALTERNATE
    - Load Set Motor freezes (subject to hardware unloading)
  - SDVs close and cannot be controlled from the keyboard.