Advanced CANDU Reactor (ACR-700) Simulator

User Manual

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FOREWORD

Given the renewed worldwide interest in nuclear technology, there has been a growing demand for qualified nuclear professionals, which in turn has resulted in the creation of new nuclear science and technology education programs and in the growth of existing ones. Of course, this increase in the number of students pursuing nuclear degrees, has also contributed to a large need for qualified faculty and for comprehensive and up-to-date curricula. The International Atomic Energy Agency (IAEA) has established a programme in PC-based Nuclear Power Plant (NPP) simulators to assist Member States in their education and training endeavors. The objective of this programme is to provide, for a variety of nuclear reactor types, insight and practice in their operational characteristics and their response to perturbations and accident situations. To achieve this, the IAEA arranges for the supply or development of simulation programs and their associated training materials, sponsors training courses and workshops, and distributes documentation and computer programs.

The simulators operate on personal computers and are provided for a broad audience of technical and non-technical personnel as an introductory educational tool. The preferred audience, however, are faculty members interested in developing nuclear engineering courses with the support of these very effective hands-on educational tools. It is important to remember, however, that the application of these PC-based simulators is limited to providing general response characteristics of selected types of power reactor systems and that they are not intended to be used for plant-specific purposes such as design, safety evaluation, licensing or operator training.

The IAEA simulator collection currently includes the following simulators:

- A WWER-1000 simulator provided to the IAEA by the Moscow Engineering and Physics Institute in Russia.

- The IAEA generic Pressurized Water Reactor (PWR) simulator has been developed by Micro-Simulation Technology of USA using the PCTRAN software. This simulator is a 600 MWe generic two-loop PWR with inverted U-bend steam generators and dry containment system that could be a Westinghouse, Framatome or KWU design.

- The IAEA advanced PWR simulator has been developed by Cassiopeia Technologies Inc. (CTI) of Canada, and is largely based on a 600 MWe PWR design with passive safety systems, similar to the Westinghouse AP-600.

- The IAEA generic Boiling Water Reactor (BWR) simulator has also been developed by CTI and represents a typical 1300 MWe BWR with internal recirculation pumps and fine motion control rod drives. This simulator underwent a major enhancement effort in 2008 when a containment model based on the ABWR was added.

- The IAEA Pressurized Heavy Water Reactor (PHWR) simulator is also a CTI product and is largely based on the 900 MWe CANDU-9 system.

- The IAEA advanced PHWR simulator by CTI from Canada, which represents the ACR-700 system.

- The IAEA advanced BWR, which largely represents the GE ESBWR passive BWR design and was also created by CTI.

This activity was initiated under the leadership of Mr. R. B. Lyon. Subsequently, Mr. J. C. Cleveland. Ms. S. Bilbao y León and later Mr. M. J. Harper and Mr. S.D. Jo from the Division of Nuclear Power became the IAEA responsible officers.

More information about the IAEA simulators and the associated training is available at http://www.iaea.org/NuclearPower/Technology/Training/Simulators/

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1. INTRODUCTION

1.1. Purpose

This publication consists of course material for workshops on the *advanced* heavy water reactor, known as the Advanced CANDU Reactor (ACR-700 MWe) simulator. Participants in the workshops are provided with instruction and practice in using the simulator, thus gaining insight and understanding of the design and operational characteristics of ACR-700 nuclear power plant systems in normal and accident situations.

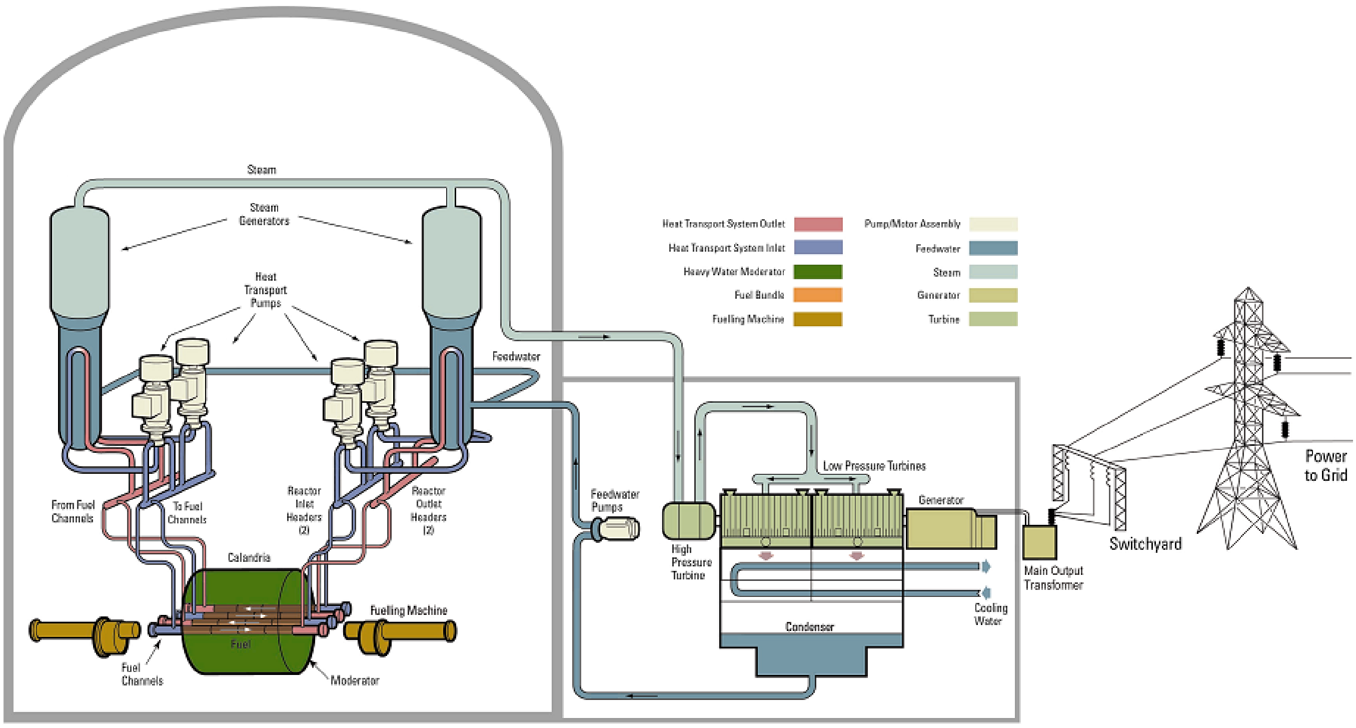
This manual is written with the assumption that the readers already have some knowledge of the CANDU and ACR. Therefore no attempt has been made to provide detailed descriptions of each individual ACR subsystem, which should be available in publications from Atomic Energy of Canada Ltd. (AECL), the reactor developer for CANDU and ACR. However, a brief system overview is presented, and details are provided where necessary to describe the functionality and the interactive features of the individual simulator screens, which relate to the specific ACR subsystems.

The user manual covers basic NPP plant operations, like plant load maneuvering, and trips and recovery — e.g. turbine trip and reactor trip. In addition, it covers plant responses to malfunction events. Some malfunction events lead to reactor trip or turbine trip. Other serious malfunctions (e.g. LOCA) lead to accident situations, causing actuation of the passive core cooling safety system.

It should be mentioned that the equipment and processes modeled in the simulator represent realistic ACR characteristics. However, for the purpose of the educational simulator, there are necessary simplifications and assumptions made in the models, which may not reflect any specific reactor vendor's design or performance.

Most importantly, the responses manifested by the simulator, under accident situations, should not be used for safety analysis purposes, despite the fact that they are realistic for the purpose of educational training. As such, it is appropriate to consider that those simulator model responses perhaps only provide first order estimates of the plant transients under accident scenarios.

1.2. Background and Highlights of Differences – CANDU vs ACR



Atomic Energy of Canada Limited (AECL) has developed the ACR-700[1] (Advanced CANDU Reactor-700) as the next generation CANDU with goals of reduced capital cost, shorter construction schedule, high capacity factor, low operating cost, increased operating life, simple component, replacement, and enhanced safety features.

The ACR design is based on the use of modular horizontal fuel channels surrounded by a heavy water moderator, the same feature as in all CANDU[2] reactors.

The major differences in ACR as compared with CANDU are:

- the use of slightly enriched uranium fuel (2.1 % wt U-235 in 42 pins of the fuel bundle), and
- light water (as opposed to heavy water D_2O) as the coolant, which circulates in the fuel channels.

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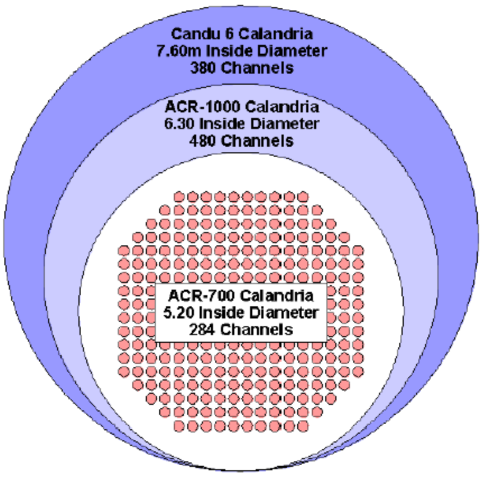
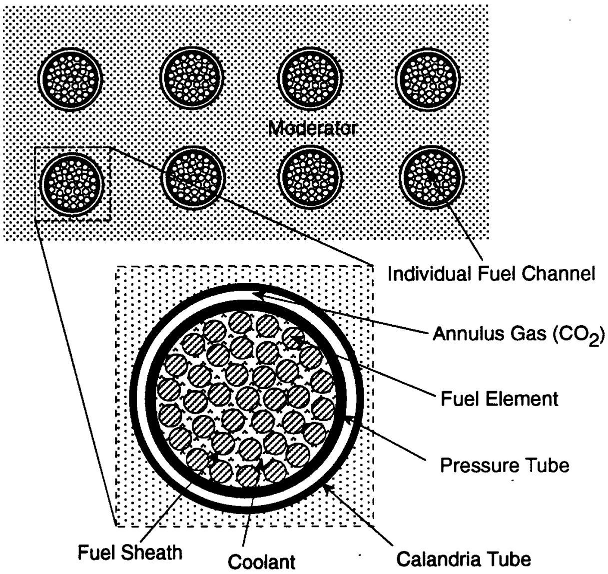
[1] ACR-700™ (Advanced CANDU Reactor) is a trademark of Atomic Energy Canada Limited (AECL).

[2] CANDU™ (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL).

This results in a more compact reactor design (Calandria inside diameter 31.6 % less than that for CANDU 6) and a reduction of heavy water inventory (72% less D_2O mass inventory when compared with CANDU 6).



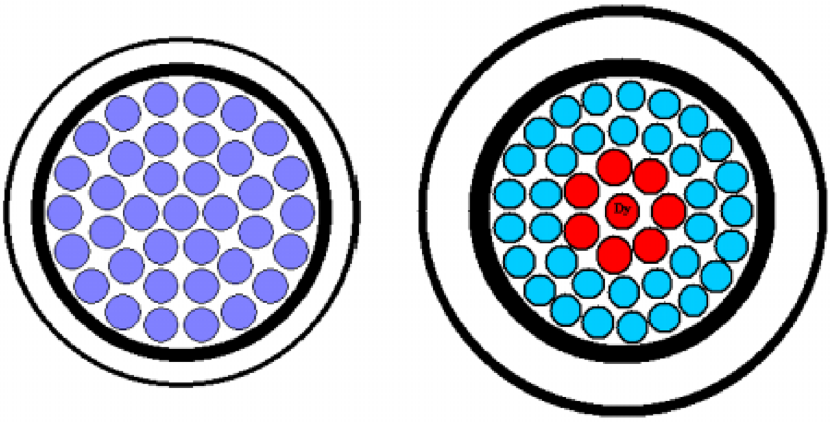


The design also features higher pressures and temperatures of reactor coolant and main steam (coolant outlet temperature 326 deg. C, and reactor inlet header (RIH) pressure 13 MPa), thus providing an improved thermal efficiency than the existing CANDU plants.

In particular, the use of the CANFLEX[3] fuel bundle, with lower linear rating and higher critical heat flux, permits increased operating and safety margins of the reactor (average channel power increased from 5.3 MW (CANDU 6) to 6.8 MW(ACR-700)). The details of the CANFLEX fuel bundles in an ACR-700 core are illustrated below:

- Reactor core configuration with 284 fuel channels
- 12 CANFLEX fuel bundles per channel.
- 43 fuel elements in one CANFLEX fuel bundle.
- The bundle has two elements size: centre pin and inner ring of seven elements with a diameter of 13.5 mm.
- The outer two rings consist of 35 elements with 11.5 mm diameter.
- 2.1 % wt U-235 in 42 pins. The center pin contains burnable poison (U, Dy)O2 pellet with 7.5% wt Dysprosium in natural Uranium.

[3] CANFLEX® is a registered trademark of AECL and the Korea Atomic Energy Research Institute (KAERI)



CANDU BUNDLE
(37-Elements)
CANDU - 6 Fuel Channel

CANFLEX BUNDLE
(43-Elements)
ACR Fuel Channel

Passive safety features draw from those of the existing CANDU plants (e.g., the two independent shutdown systems), and other passive features are added to strengthen the safety of the plant (e.g. a gravity supply of emergency feedwater to the steam generators).

The passive safety features for ACR-700 can be summarized as follows:

- Two independent SD systems located in low pressure and temperature moderator.

- Low pressure and low temperature moderator surrounding fuel channels provide additional passive heat sink, in the unlikely event that both the primary coolant and emergency cooling systems were unavailable.

- Water filled shield tank surrounding Calandria would contain and maintain a collapsed core in a cooled state, should moderator cooling be impaired.

- Emergency Core Cooling (ECC) uses a burst disc system, which functions automatically when primary system pressure drops below a prescribed level.

- Gravity supply of Emergency feedwater to SG's.

- Steel lined, pre-stressed concrete containment structure forms a safe pressure retaining envelope boundary in the unlikely event of an accident.

- Heat removal of containment atmosphere after an accident is provided by local air coolers.

- Hydrogen control is provided by passive autocatalytic recombiners.

The detailed differences between CANDU 6 and ACR-700 are highlighted in the following tables:

Comparison of CANDU 6 and ACR-700 Unit Data

DATA	CANDU 6	ACR-700
Reactor		
Type	PTR	PTR
Thermal Output to Steam Generators [MWth]	2064	2034
Coolant	Pressurized Heavy Water	Pressurized Light water
Moderator	Heavy Water	Heavy water
Calandria diameter [m]	7.6	5.2
Fuel channel	Horizontal Zr 2.5wt% Nb alloy with modified 403 SS end fittings	Horizontal Zr 2.5wt% Nb alloy with modified 403 SS end fittings
Number of fuel channels	380	292
Lattice pitch [mm]	286	220
Reflector thickness [mm]	655	480
Fuel		
Fuel	Sintered pellets of Natural UO_2	Sintered pellets of slightly enriched UO_2 & Natural UO_2 in central element
Enrichment level	0.71 wt% ^{235}U	Average 2.1 wt% U-235 in 42 elements, central element NU with Dysprosium
Fuel burn-up [MWd/te U]	7,500	20,500
Fuel bundle assembly	37 element	43 element CANFLEX
Length of bundle [mm]	495.3	495.3
Outside diameter (maximum) [mm]	102.7	103
Bundle weight [kg]	24.1 (includes 19.2 kg U)	22.7 (includes 18 kg U)
Bundles per fuel channel	12	12
Heavy Water		
Moderator Systems [Mg D_2O]	265	126
Heat Transport Systems [Mg D_2O]	192	0
Reserve [Mg D_2O]	9	4
Total [Mg D_2O]	466	130

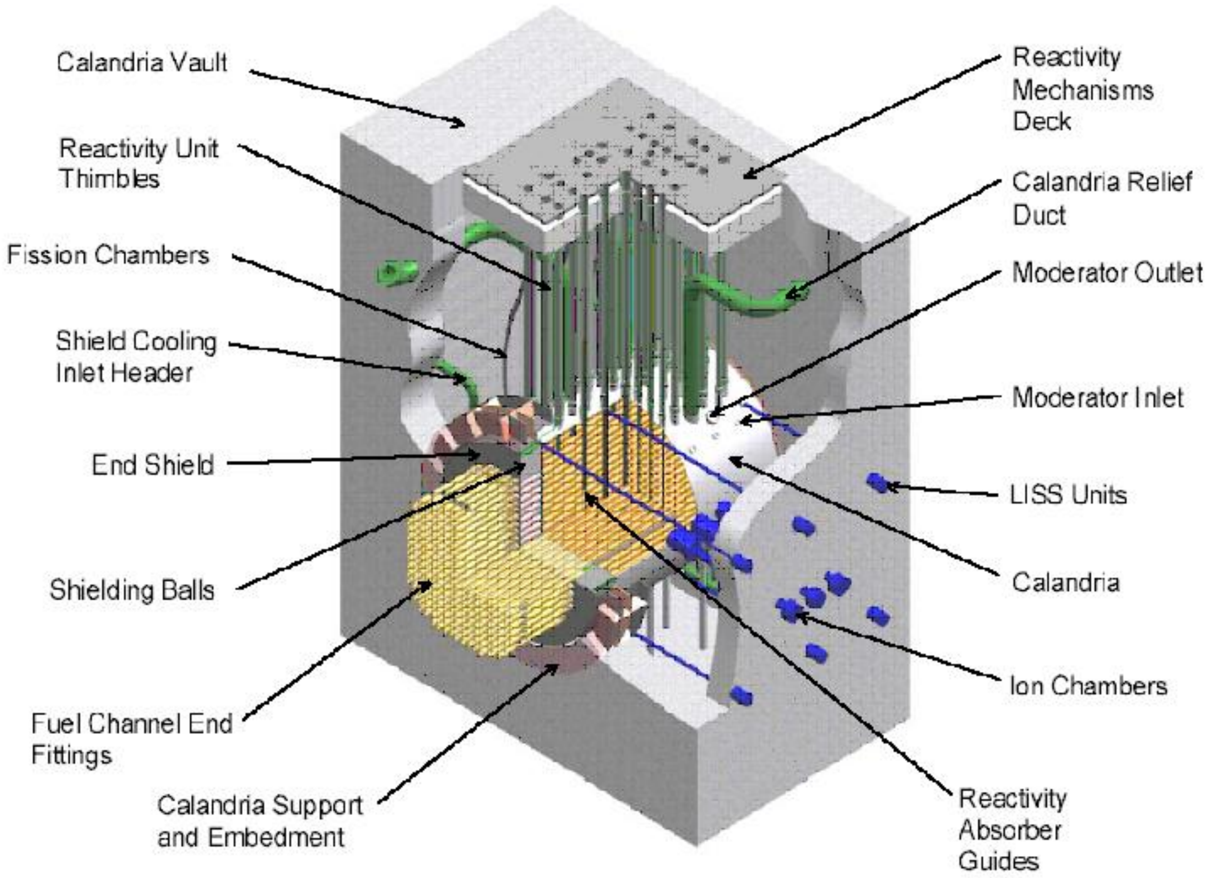
DATA	CANDU 6	ACR-700
Heat Transport System		
Reactor outlet header pressure [MPa (g)]	9.9	12.1
Reactor outlet header temperature [ºC]	310	326
Reactor inlet header pressure [MPa (g)]	11.2	13.3
Reactor inlet header temperature [ºC]	266	280
Reactor core coolant flow (total) [Mg/s]	7.7	7.13
Single channel flow (maximum) [kg/s]	28	26
Steam Generators		
Number	4	2
Type	Vertical U tube with integral preheater	Vertical U tube with integral preheater
Steam temperature (nominal) [ºC]	260	281
Steam quality	0.9975	0.999
Steam pressure [MPa (g)]	4.6	6.4
Heat Transport Pumps		
Number	4	4
Pump type	Vertical, centrifugal, single suction, double discharge	Vertical, centrifugal, single suction, double discharge
Motor type	AC, vertical, squirrel cage induction	AC, vertical, squirrel cage induction
Rated flow [L/s]	2228	2350
Rated head [m]	215	235
Motor rating [MWe]	6.7	6.4
Containment		
Type	Pre-stressed concrete with epoxy liner	Pre-stressed concrete with steel liner
Inside diameter [m]	41.5	39.5
Height (Top of base slab to Inside of Dome) [m]	51.9	59
Designed for: LOCA Pressure [kPa(g)]	124	250
MSLB Pressure [kPa(g)]	200	450
Turbine Generator		
Steam Turbine Type	Hitachi impulse type, tandem compound double exhaust flow, reheat condensing turbine with a last stage blade length of 132 cm (52 inches)	Impulse type, tandem compound double exhaust flow, reheat condensing turbine with a last stage blade length of 132 cm (52 inches)
Steam Turbine Composition	One double flow high-pressure cylinder, two external moisture separators/reheaters and two double flow low pressure cylinders	One single flow, high-pressure cylinder, two external moisture separators/reheaters and two double flow, low pressure cylinders
Net heat to turbine [MWth]	2062	2030
Gross/Net electrical output* (nominal) [MWe]	728/666	753/703*
Gross Turbine Generator Efficiency	35.30%	37.0%
Steam temperature at main stop valve [ºC]	258	279
Steam pressure at main stop valve [MPa (g)]	4.41	6.2
Final feedwater temperature [ºC]	187.0	215
Condenser Vacuum [kPa (a)]	4.9*	4.9*

CANDU 6 data quoted is based on the Qinshan Phase III CANDU 6 design (50 Hz).

* Gross electrical output is dependent on cooling water temperature, the turbine-generator and condenser design, and the grid frequency (60 Hz).

2. BRIEF ACR-700 SYSTEMS OVERVIEW

2.1 Reactor Configuration



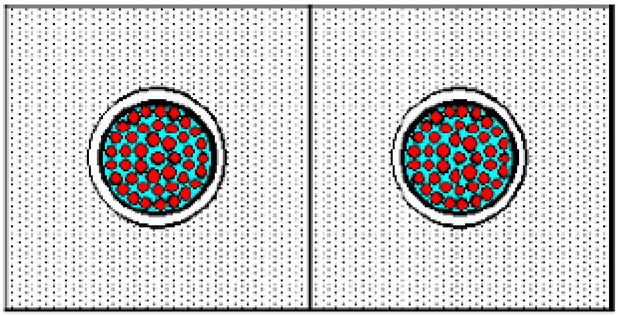
The ACR reactor consists of a set of 292 horizontally aligned fuel channels arranged in a square pitch. The fuel channels contain the fuel and the high-pressure light water coolant. They are mounted in a calandria vessel containing the heavy water moderator. Individual calandria tubes surround each individual fuel channel.

The calandria vessel is enclosed by end shields, which support each end of the calandria. They are filled with shielding balls and water to provide shielding. The fuel channels are located by adjustable restraints on the two endshields and are connected by individual feeder pipes to the Heat Transport System.

The calandria vessel is enclosed in a concrete vault (calandria vault) filled with light water for shielding. The calandria vault is closed at the top by the reactivity mechanisms deck.

- The core configuration parameters and lattice pitch (LP) are shown below:

Core Configuration	CAs and SORs out
Total Fission Power	2114.4 MW
Core Thermal Power	2037 MW (th)
Fuelling Scheme	2-bundle shift
Radial Form Factor	0.95
Peak Channel Power	7.3 MW
Peak Bundle Power	850 kW
Core-average Exit Irradiation	3.5 n/kb
Core-average Discharge Bundle Burnup	20.5 MWd/kgU
Channel Average Exit Burnup	24 MWd/kgU
Maximum Fuel Element Burnup	28 MWd/kgU
Core-average Dwell Time	100 FPD



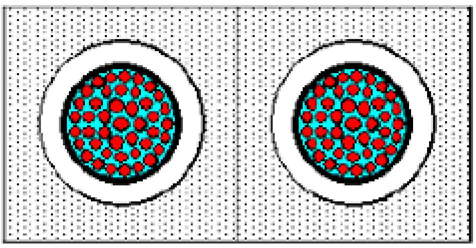
NU CANDU Lattice

LP = 286 mm.

PT_{OR} = 56 mm.

CT_{OR} = 66 mm.

V_M/V_F = 16.4



ACR-700 Lattice

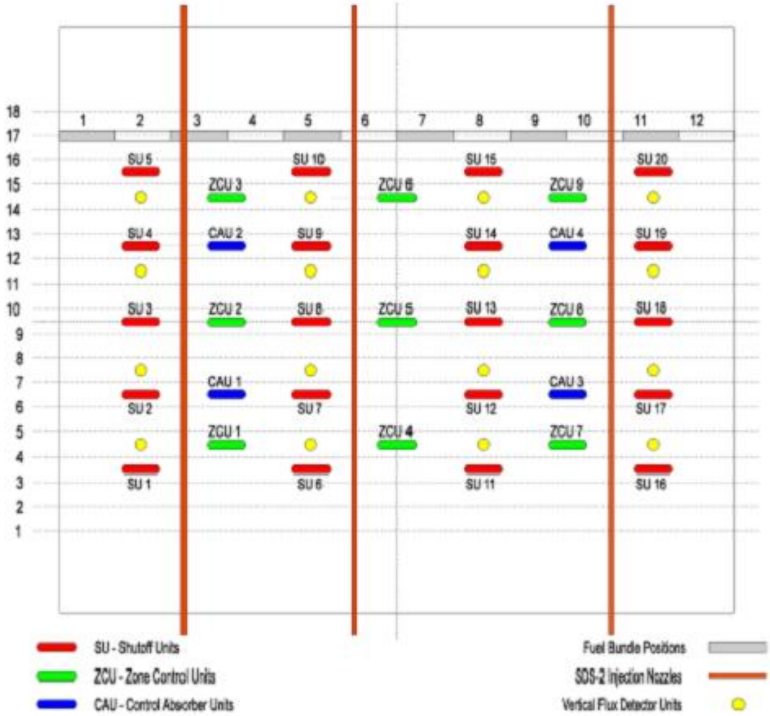
LP = 220 mm.

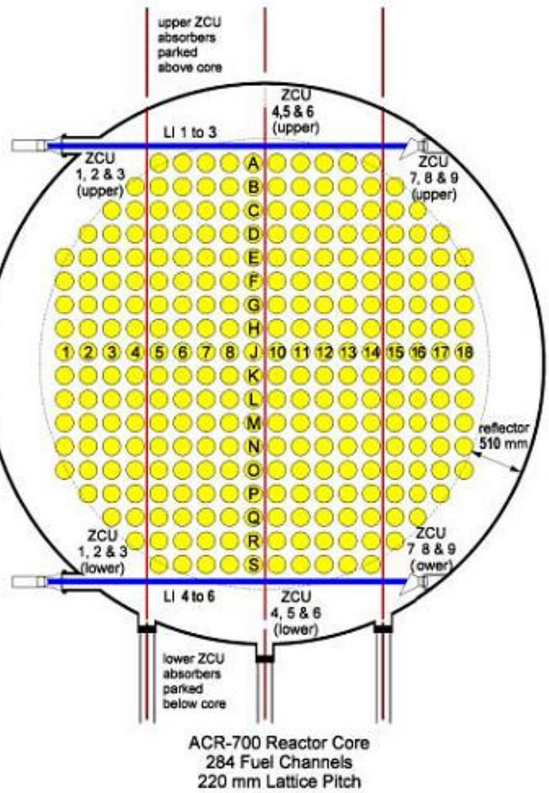
PT_{OR} = 58 mm.

CT_{OR} = 78 mm.

V_M/V_F = 7.1

2.2 Reactivity Control Units





Reactivity control units include neutron flux measuring devices, reactivity control devices, and safety shutdown systems. Flux detectors are provided in and around the core to measure neutron flux, and reactivity control devices are located in the core to control the nuclear reaction.

In-core flux detectors are used to measure the neutron flux in different zones of the core. These are supplemented by fission chamber and ion chamber assemblies mounted in housings on the calandria shell. The signals from the in-core flux detectors are used to adjust the absorber insertion in the zone control assemblies. By varying the absorber position in these assemblies the local neutron absorption in each zone of the reactor changes, thereby controlling the local neutron-flux level. *In contrast with CANDU 6 which uses liquid zones (in-core light water columns with varying liquid level) as the zone control assemblies, ACR uses control rods as the zone control assemblies.*

Control absorber elements penetrate the core vertically. These are normally parked out of the reactor core and are inserted to control the neutron flux level at times when a greater rate or amount of reactivity control is required than can be provided by the zone control assemblies.

Slow or long-term reactivity variations are controlled by the addition of a neutron-absorbing liquid to the moderator. Control is achieved by varying the concentration of this "neutron absorbent material" in the moderator. For example, the liquid "neutron absorbent material" is used to compensate for the excess reactivity that exists with a full core of fresh fuel at first startup of the reactor. In this regard, two moderator poison addition systems are provided:

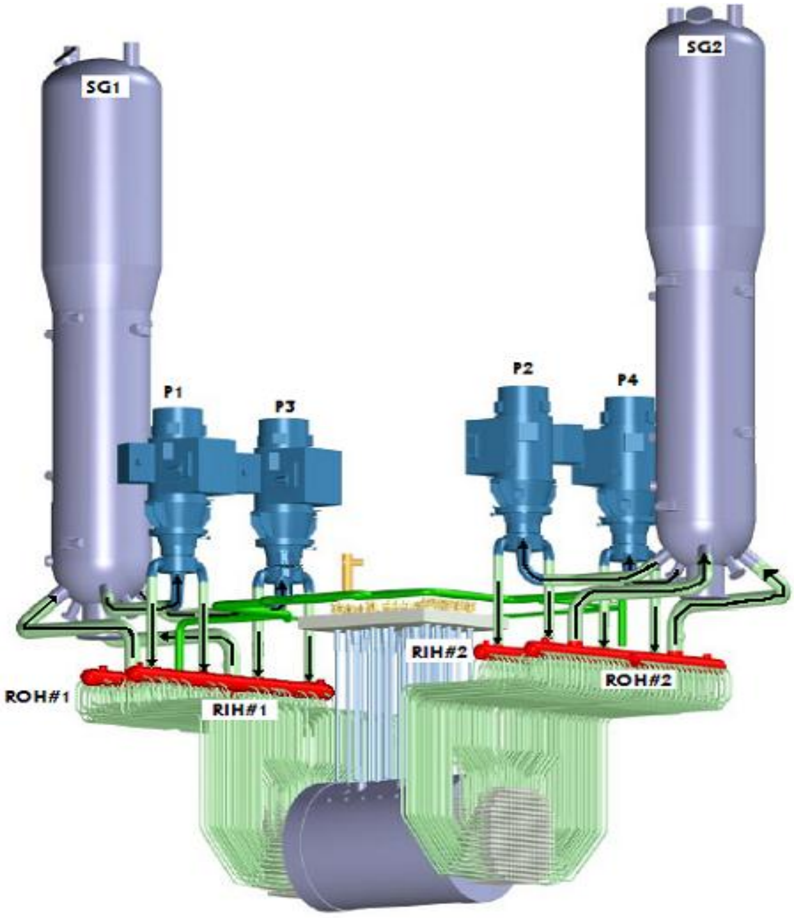
(a) Boron addition, initiated manually, is used as a source of long-term negative reactivity when the reactor has excess fuel reactivity.
(b) Gadolinium addition, normally initiated manually, is used as a source of short-term negative reactivity, to compensate for a lack of xenon (gadolinium burns out at a rate similar to xenon production rate.) Under special conditions (positive flux rate and large power error) Reactor Regulating System (RRS) will add gadolinium automatically.

Two independent reactor safety shutdown systems are provided. Each shutdown system, acting alone, is designed to shut the reactor down and maintain it in a safe shutdown condition. The safety shutdown systems are independent of the reactor regulating system and are also independent of each other. The first shutdown system, SDS1, consists of shutoff units (absorber element, guide assembly, and drive mechanisms), which drop neutron absorbing elements into the core by gravity on receipt of a shutdown signal from the safety system. The second shutdown system, SDS2, uses injection of a strong neutron absorbing solution into the moderator. The automatic shutdown systems respond to both neutronic and process signals.

2.3 Heat Transport System

The heat transport system (HTS) circulates pressurized light water coolant through the reactor fuel channels to remove heat produced by nuclear fission in the core. The fission heat is carried by the reactor coolant to the steam generators, to produce steam on the secondary side that subsequently drives the turbine generator. The heat transport system is complemented by auxiliary systems, which support its operation and maintain parameters within operation ranges to suit the various system functions.

The 3D isometric view of the Heat Transport System is shown below:



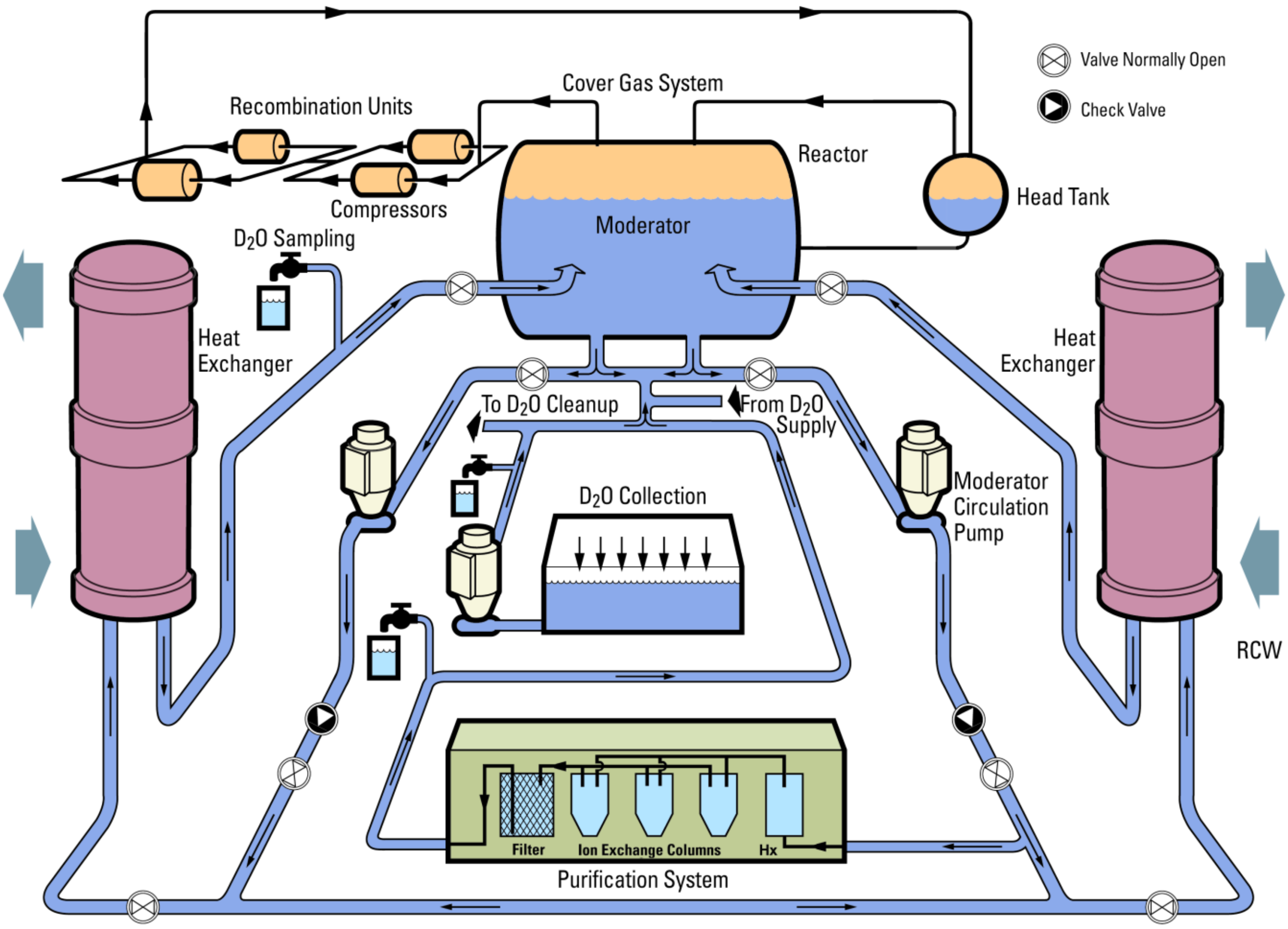
The pressure and volume of the coolant in the HTS are controlled by the pressure and inventory control system. The long term cooling (LTC) system is used to remove decay heat following a reactor shutdown and to cool the HTS to a temperature suitable for maintenance of the heat transport and auxiliary system components.

The HTS and its auxiliary systems are similar to the equivalent systems in the CANDU 6 design. However, the overall design of these systems has been improved based on operation feedback from existing CANDU plants and has been simplified with the use of light water in a single-loop configuration.

The major components of the heat transport system are the reactor fuel channels, two steam generators, four electrically driven heat transport pumps, two reactor inlet headers, two reactor outlet headers, and the interconnecting piping. Light water coolant is fed to the fuel channels from the inlet headers at each end of the reactor and is returned to the outlet headers at the opposite end off the reactor.

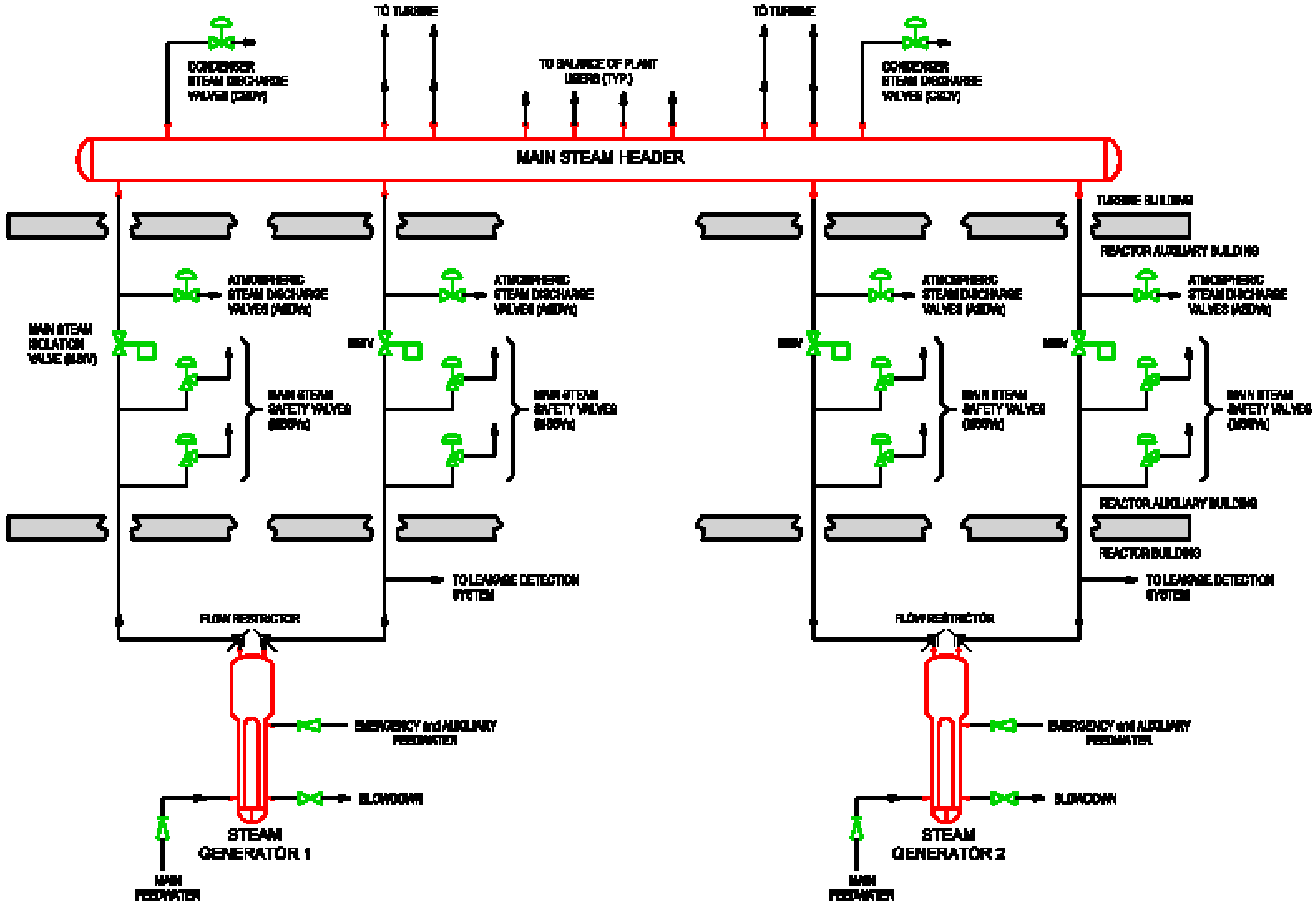
The principal function of the heat transport system main circuit is to provide reliable cooling of the reactor fuel under all operating conditions, for the life of the plant and with minimal maintenance. The heat transport system also provides a barrier to the release of radioactive fission products during normal operation to ensure that radiation doses to plant staff remain within acceptable limits. It is designed to retain its integrity under normal and abnormal operating conditions.

2.4 Moderator System



Neutrons produced by nuclear fission are moderated by the heavy water in the calandria. The heavy water moderator is circulated by the moderator pumps through the calandria at a relatively low temperature and low pressure and cooled by the moderator heat exchangers. The moderator heat exchangers remove the nuclear heat generated in the moderator and the heat transferred to the moderator from the fuel channels. Helium is used as a cover gas over the heavy water in the calandria. Chemistry control of the moderator water is maintained by the moderator purification system. The moderator system also acts as a back-up heat sink under certain postulated accident conditions.

2.5 Steam and Feedwater System

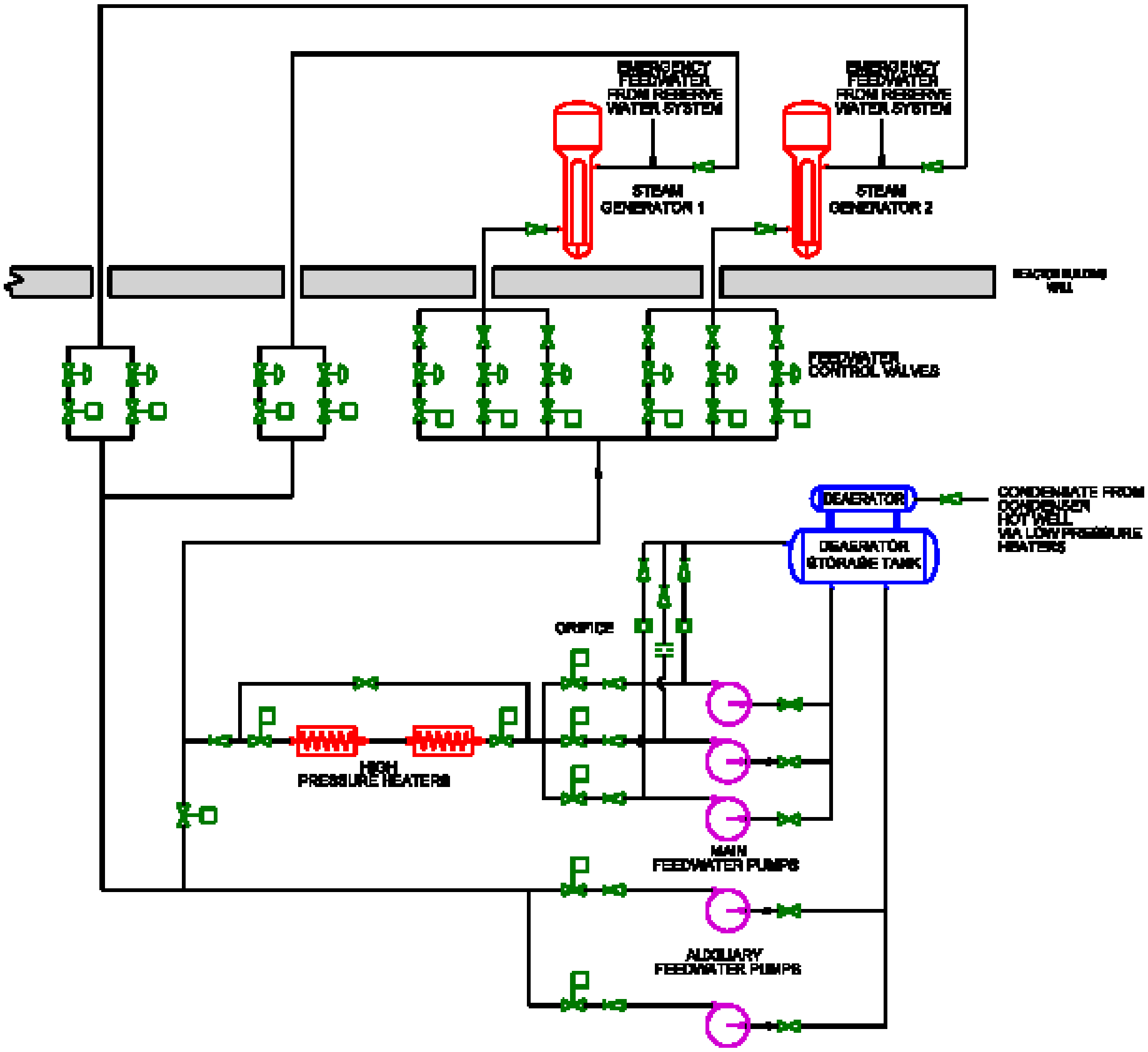


Main Steam System Flow Diagram

The main steam lines supply steam from the two steam generators in the reactor building to the turbine through the steam balance header, in the turbine building, at a constant pressure.

The system controls the steam generator pressure using the condenser steam discharge valves (CSDVs) and the atmospheric steam discharge valves (ASDVs). Main steam safety valves (MSSVs) are provided for overpressure protection of the steam generator secondary side. The feedwater system takes hot, pressurized feedwater from the feedwater train in the turbine building and discharges the feedwater into the preheater section of the steam generators.
Main steam isolation valves (MSIVs) are provided to isolate the main steam supply to the turbine in the event of steam generator tube leak, after reactor shutdown when the long term cooling system is placed in service and the heat transport system is depressurized.

The feedwater system controls the feedwater flow to maintain the required steam generator level. The flow diagram for the system is shown below.



Feedwater System Flow Diagram

2.6 Balance of Plant

The balance of plant (BOP) consists of the turbine building, steam turbine, generator and condenser, the feedwater heating system with associated auxiliary, and electrical equipment. The BOP also includes the water treatment facilities, auxiliary steam facilities, pumphouses and/or cooling towers, main switchyard, and associated equipment to provide all conventional services to the ACR-700 two-unit plant.

Two steam generators are provided in the heat transport system. They discharge steam into a common header located in the turbine building that supplies the required steam to the turbine generator and the auxiliary steam systems. The power generating equipment consists of the following:

- A turbine generator set with a nominal gross output of 753 MW(e). This consists of a tandem compound, reheat condensing type steam-driven single shaft turbine, composed of one high pressure and two low pressure cylinders, with a thermal cycle involving two stage moisture separator/reheater vessels located between the high pressure turbine exhaust and the low pressure turbine inlets. The generator is cooled with water and hydrogen and provided with a static excitation system.
- A condenser with tubes at right angles to the turbine axis.
- A regenerative feedwater heating system with low pressure stages, deaerating feedwater heater and high pressure stages.
- Other auxiliaries associated with the turbine generator set.

2.7 Safety Systems

The ACR safety design has the following inherent and engineered safety characteristics:

- On-power refuelling assures that very little reactivity needs to be held up in movable control devices or in neutron absorbent material dissolved in the moderator (no chemicals are added to the reactor coolant for reactivity control). Thus any malfunctions in the reactor control system produce only modest reactivity changes.
- The control and shutdown devices are in the low pressure moderator and are not subject to large hydraulic forces.
- The equilibrium core has a significantly negative power reactivity coefficient, which provides inherent protection against transients with inadvertent increase of reactor power.
- The void reactivity coefficient is small and negative, and offers a good balance of inherent nuclear protection between loss-of-coolant accidents (LOCA) and accidents with fast cooldown of the heat transport system.
- Natural coolant circulation can remove decay heat from the fuel if Class IV electrical power to the heat transport pump motors is lost.
- Two independent shutdown systems are provided. Each system can shut down the reactor for the entire spectrum of design basis events.
- Emergency core cooling (ECC) is provided by an emergency coolant injection (ECI) system, which injects water into the heat transport system after a LOCA. A long term cooling (LTC) system provides adequate decay heat removal from the reactor core in the recovery/recirculation phase after a LOCA.
- For a loss of the main feedwater pumps and/or Class IV electrical power, the auxiliary feedwater pumps with power supplied from the Class III power systems provide effective cooling with the reactor shut down. The auxiliary feedwater supply is also backed up by

passive emergency feedwater with gravity water supply from the reserve water tank to the steam generators.
- A separate secondary control area is provided as a backup to the main control room for certain emergency conditions.
- Distributed control systems control the plant routinely, freeing the operator from mundane tasks thus reducing the likelihood of operator error. The safety system responses are automated to the extent that no operator action is needed for a minimum of eight hours following most design basis accidents.

The safety systems are those systems designed to quickly shut down the reactor, remove decay heat, and limit the radioactivity release subsequent to the failure of normally operating process systems. These are:
- the shutdown system number 1 (SDS1),
- shutdown system number 2 (SDS2),
- emergency core cooling (ECC) system, and
- containment system.

The safety support systems are those that provide services needed for proper operation of the safety systems (e.g., electrical power, cooling water, instrument air).

(a) Shutdown System No. 1 (SDS1)

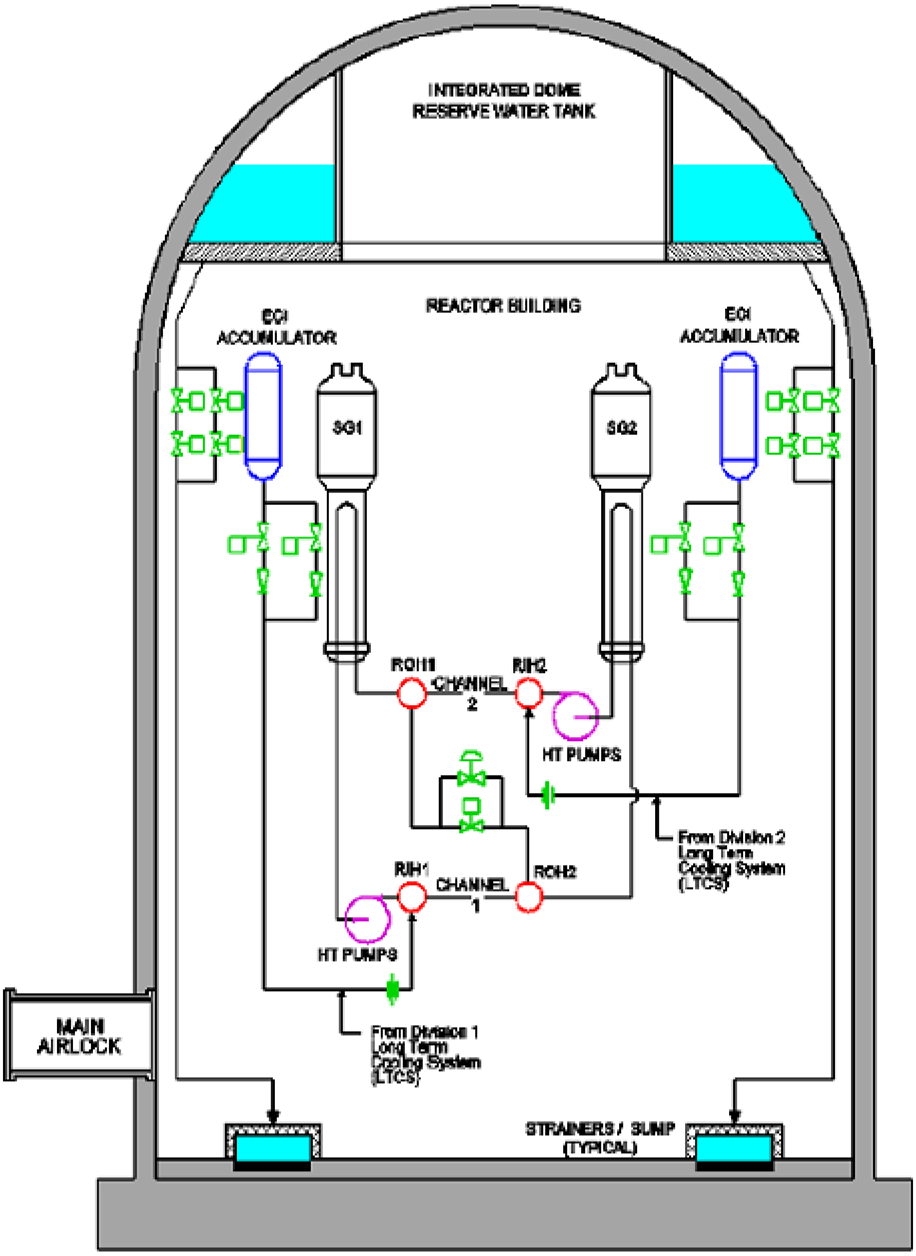
SDS1 quickly terminates reactor power operation and brings the reactor into a safe shutdown condition by inserting shutoff rods into the reactor core. Reactor operation is terminated when a certain neutronic or process parameter enters an unacceptable range. The measurement of each parameter is triplicated and the system is initiated when any two out of the three trip channels are tripped by any parameter or combination of parameters.

(b) Shutdown System No. 2 (SDS2)

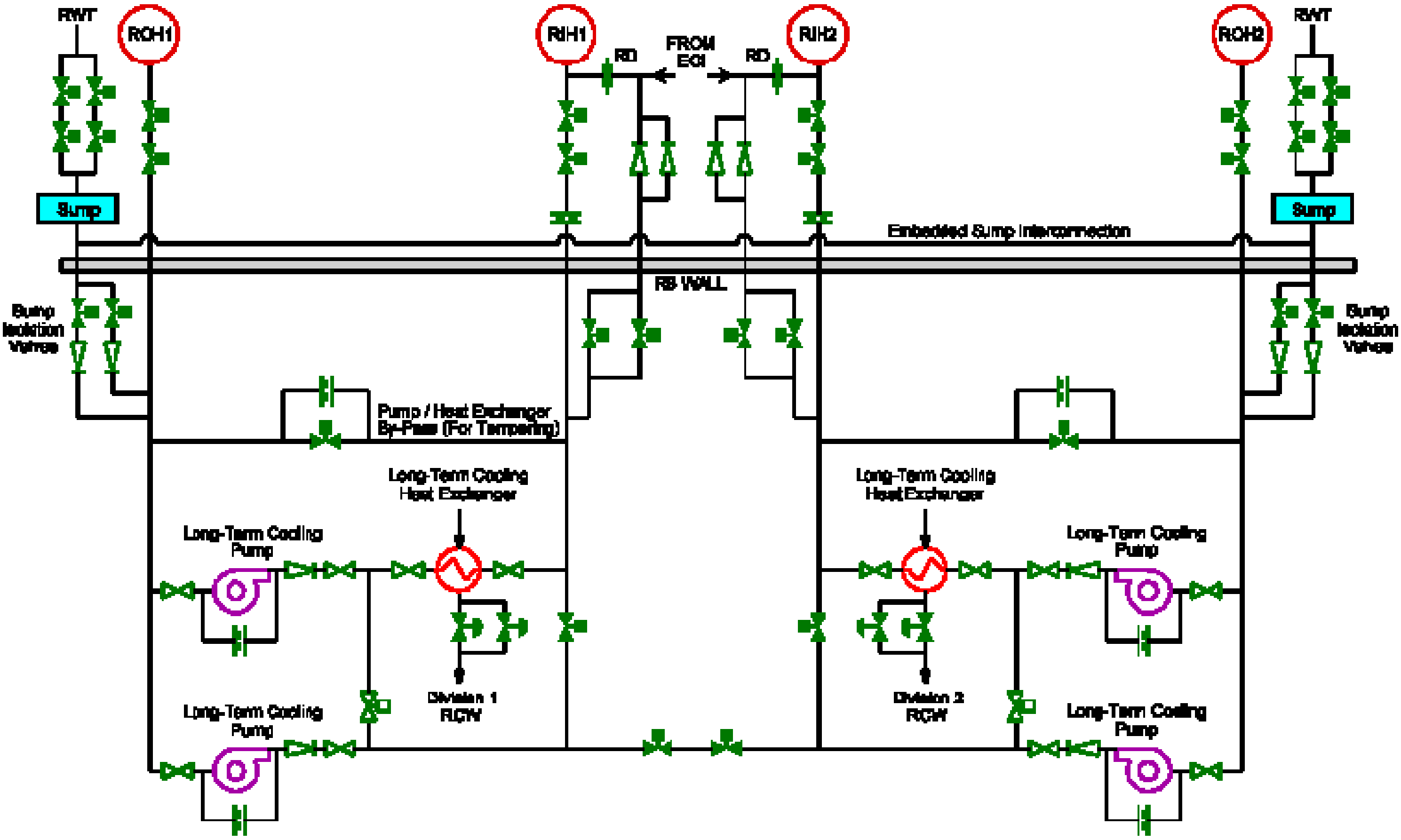
SDS2 provides a second independent method of quickly terminating reactor power operation by injecting a strong neutron absorbing solution (gadolinium nitrate) into the moderator when any two out of three trip channels are tripped by any parameter.

(c) Emergency Core Cooling (ECC) System

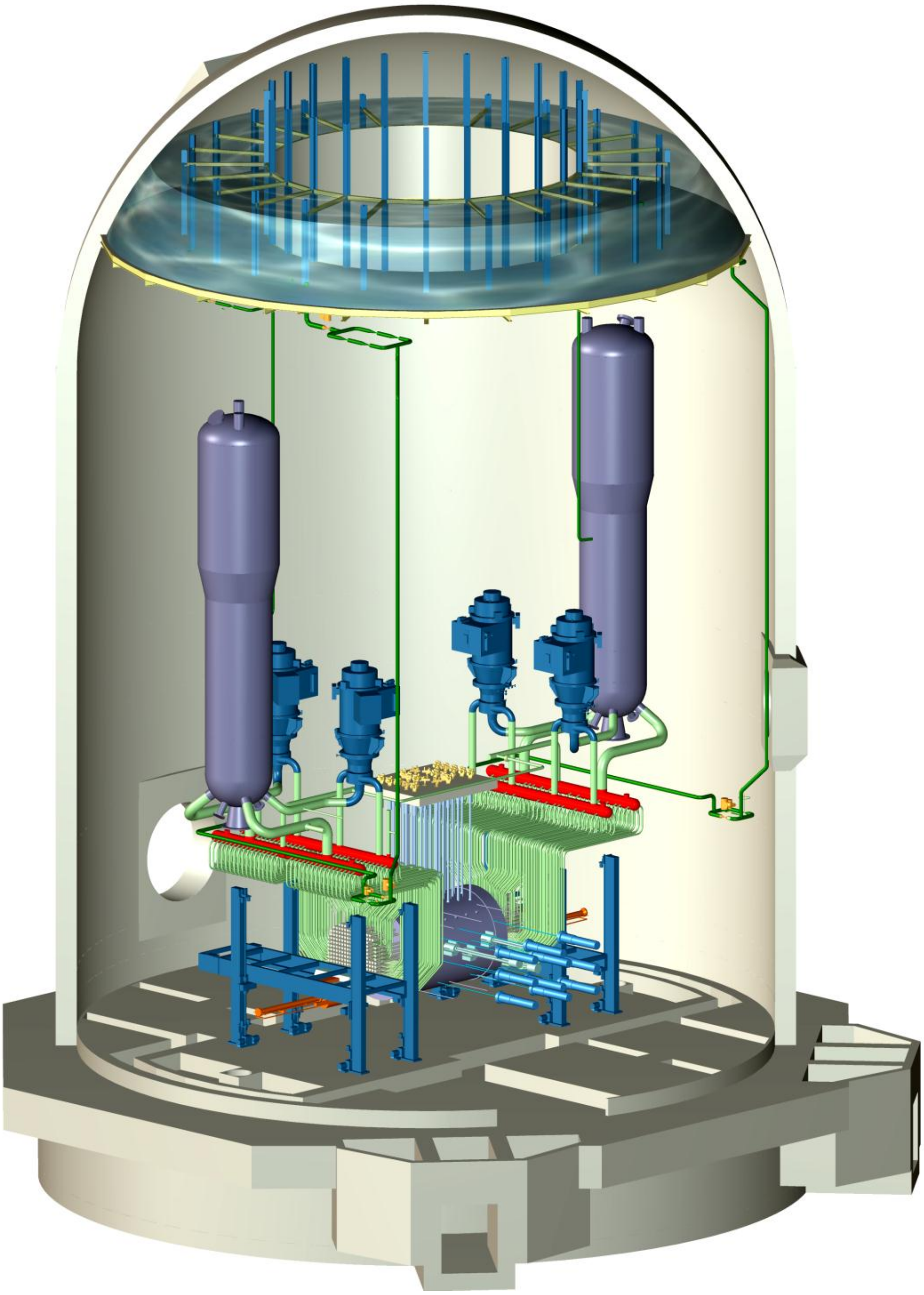
The ECC system is designed to supply water (emergency coolant) to the reactor core to cool the reactor fuel in the event of a loss-of-coolant accident (LOCA). The design bases events are LOCA events where ECC is required to fill and maintain the heat transport circuit inventory.



Emergency Coolant Injection System



Schematic Diagram of the LTC System



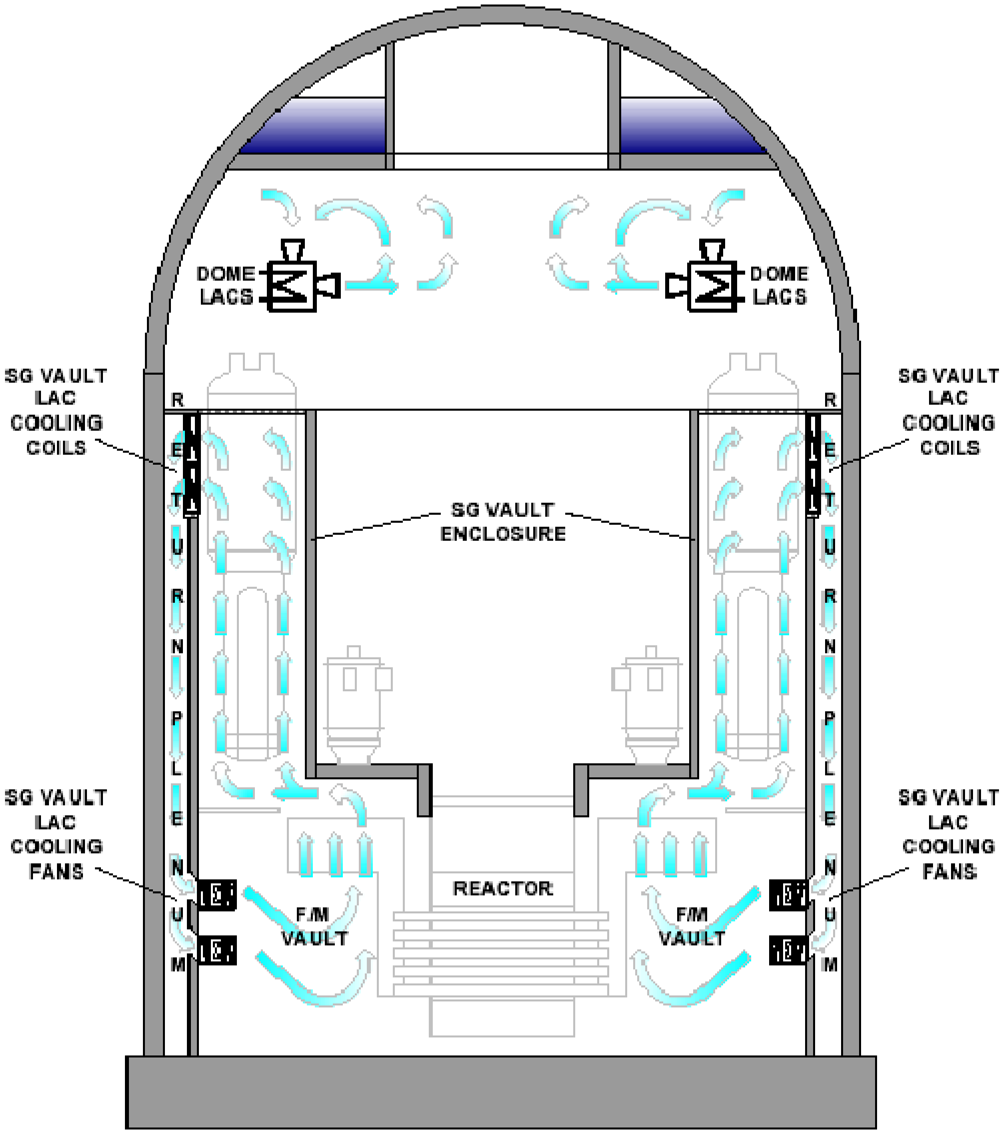
(d) Containment System

The containment system is a safety system, with the function of limiting releases of radioactive material from within containment generally; it prevents releases in excess of the site dose limits. The containment system consists of the reactor building and liner, electrical and process penetrations and other appurtenances, which together form the containment envelope. In addition, the following subsystems act to ensure the continuity of the containment envelope or to reduce the contained pressures and energies following an accident:

(a) Main and auxiliary airlocks for the passage of personnel, equipment and fuel.
(b) Containment isolation.
(c) Equipment for hydrogen control (e.g. air coolers for mixing, passive auto-catalytic recombiners for H2 reduction).

The basic function of the containment system is to form a continuous, pressure-retaining envelope around the reactor core and the heat transport system. Following an accident, the containment system limits release of resultant radioactive material to the external environment. The containment system includes the steel-lined, pre-stressed concrete reactor building containment structure, main and auxiliary airlocks, building air coolers for pressure reduction, and a containment isolation system consisting of valves or dampers in the ventilation ducts and certain process lines penetrating the containment envelope. This containment design ensures a low leakage rate while at the same time providing a pressure retaining boundary for LOCAs. The containment system automatically closes all penetrations open to the reactor building atmosphere when an increase in containment pressure or radioactivity level is detected.

Measurements of containment pressure and radioactivity are triplicated and the system is actuated using two-out-of-three logic. Heat removal from the containment atmosphere after an accident is provided by local air coolers suitably distributed in various compartments inside the reactor building. Hydrogen control is provided in the reactor building by passive autocatalytic recombiners that limit hydrogen content to below the acceptable limits within any significant enclosed compartment of the containment following an accident.



Containment Cooling System

3. 700 MW(E) ADVANCED CANDU REACTOR NPP SIMULATOR

The purpose of the 700 MW(e) advanced CANDU reactor (ACR-700) NPP simulator is educational — to provide a training tool for university professors and engineers involved in teaching topics in nuclear energy. As well, nuclear engineers, scientists and trainers in the nuclear industry may find this simulator useful in broadening their understanding of ACR transients and power plant dynamics.

The simulator can be executed on a personal computer (PC), to operate essentially in real time, and to have a dynamic response with sufficient fidelity to provide ACR plant responses during normal operations and accident situations. It also has a user-machine interface that mimics the actual control panel instrumentation, including the plant display system, and more importantly, allows user's interactions with the simulator during the operation of the simulated ACR plant.

The minimum hardware configuration for the simulator consists of a Pentium PC or equivalent (minimum 1.7 GHz CPU speed), minimum of 512 Mbytes RAM , at least 30 Gbytes hard drive, 32 MB display adaptor RAM, hi-resolution video card (capable of 1024 × 768 resolution), 15 inch or larger high resolution SVGA colour monitor, keyboard and mouse. The operating system can be Windows 2000, or Windows XP.

The requirement of having a single PC to execute the models and display the main plant parameters in real time on a high-resolution monitor implies that the models has to be as simple as possible, while having realistic dynamic response. The emphasis in developing the simulation models was on giving the desired level of realism to the user. This means being able to display all plant parameters that are critical to operating the unit, including the ones that characterize the main process, control and protective systems. The current configuration of the Simulator is able to respond to the operating conditions normally encountered in power plant operations, as well as to many malfunctions, as summarized in Table I.

The simulation uses a modular modeling approach: basic models for each type of device and process to be represented as algorithms and are developed in FORTRAN. These basic models are a combination of first order differential equations, logical and algebraic relations. The appropriate parameters and input-output relationships are assigned to each model as demanded by a particular system application.

The interaction between the user and the simulator is via a combination of monitor displays, mouse and keyboard. Parameter monitoring and operator controls, implemented via the plant display system at the generating station, are represented in a virtually identical manner on the simulator. Control panel instruments and control devices, such as push-buttons and hand-switches, are shown as stylized pictures, and are operated via special pop-up menus and dialog boxes in response to user inputs.

This manual assumes that the user is familiar with the main characteristics of water cooled thermal reactor nuclear power plants, as well as understanding the unique features of the CANDU.

TABLE I. SUMMARY OF SIMULATOR FEATURES

SYSTEM	SIMULATION SCOPE	DISPLAY PAGES	OPERATOR CONTROLS	MALFUNCTIONS
REACTOR	• Neutron flux levels over a range of 0.001 to 110% full power, 6 delayed neutron groups • Decay heat (3 groups) • All reactivity control devices – zone control rods (ZCU), absorber rods (MCA); gadolinium control. • Xenon/Iodine poison • Spatial kinetic simulated for 18 reactor zones, enabling display of flux tilt. • Reactor Regulating System (RRS) • Reactor Shutdown System - SDS1	• ACR Reactor Power control • ACR Control Rods & SD rods • ACR Trip parameters	• reactor power setpoint and rate of change (input to control computer) • manual control of reactivity devices - control rods (ZCU), absorber rods (MCA) and gadolinium addition/removal • reactor trip • reactor setback • reactor stepback	• reactor setback and stepback fail • one bank of MCA rods drop into the reactor core • all MCA rods "stuck" to manual
REACTOR COOLANT (light water)	• Main circuit coolant loop with four pumps, two steam generators, six equivalent "lumped" reactor coolant channels. • Fuel and coolant heat transfer simulated for 18 reactor zones • Pressure and inventory control which includes pressurizer, bleed condenser, feed & bleed control, and pressure relief, coolant makeup. • Operating range is from zero power hot to full power	• ACR Reactor Coolant System • ACR Coolant Inventory & Pressurizer • ACR Inventory Control • ACR Pressure Control	• coolant heat transport system (HTS) pumps • coolant makeup pumps • pressurizer pressure control: heaters; spray; pressure control valve; relief valve • pressurizer level control by regulating coolant feed & bleed flow via control valves. • isolation valves for coolant feed and bleed	• pressurizer pressure relief valve fails open • coolant feed valve fails open • coolant bleed valve fails open • pressurizer heaters #2 to # 6 turned "ON" by malfunction • reactor inlet header break • loss of one HTS pump • Loss of two HTS pumps in one loop

SYSTEM	SIMULATION SCOPE	DISPLAY PAGES	OPERATOR CONTROLS	MALFUNCTIONS
STEAM & FEEDWATER	• SG dynamics, including shrink and swell effects • Steam supply to turbine and reheater • Turbine by-pass to condenser • Extraction steam to feed heating • Steam generator pressure control • Steam generator level control • SG feed system	• ACR feedwater & extraction steam	• Feed pump on/off operation • SG level controller mode: Auto or manual • level control setpoint changes during Auto operation • level control valve opening during manual operation • extraction steam valves opening	• all level control isolation valves fail closed • one level control valve fails open • one level control valve fails closed • main feedwater pump trips • all main steam safety relief valves (MSSV) open • steam header break • steam flow transmitter failure
TURBINE-GENERATOR	• Simple turbine model • Mechanical power and generator output are proportional to steam flow • Speeder gear and governor valve allow synchronized and non-synchronized operation • Turbine steam bypass	• ACR turbine generator	• turbine trip • turbine run-back • turbine run-up and synchronization • condenser steam discharge valves (CSDV) • atmospheric steam discharge valve (ASDV)	• turbine spurious trip • condenser steam discharge valves (CSDV) failed closed
OVERALL UNIT	• Fully dynamic interaction between all simulated systems • Overall unit power control with reactor leading mode; or turbine leading mode • Unit annunciation & time trends • Computer control of all major system functions	• ACR Plant Overview • ACR Control Loops • ACR MW Demand SP & SGPC	• reactor power setpoint and rate entry in reactor-lead mode. • turbine load setpoint (MW) and loading rate entry in turbine-lead mode.	
SAFETY SYSTEM	• Emergency Core Cooling System (ECC) • Simple Model for containment.	• ACR Passive Core Cooling		• reactor inlet header break

Note: For simplicity, the following systems are not modeled in the ACR-700 simulator:

(a) Moderator System.
(b) Condenser and Condensate System.
(c) Shutdown System #2 (SDS2).

3.1 Simulator startup

- Select program icon ‘ACR Simulator’ for execution
- Click anywhere on ‘ACR simulator” screen
- Click ‘OK’ to ‘load full power IC?’
- The simulator will display the ‘ACR plant overview’ screen with all parameters initialized to 100% full power
- At the bottom right hand corner click on ‘Run’ to start the simulator

3.2 Simulator initialization

If at any time it is necessary to return the simulator to one of the stored initialization points, do the following:

- ‘Freeze’ the simulator
- Click on ‘IC’
- Click on ‘Load IC’
- Click on ‘FP_100.IC’ for 100% full power initial state
- Click ‘OK’ to ‘Load C:\ACR\FP_100.IC’
- Click ‘YES’ to 'Load C:\ACR\FP_100.IC’
- Click ‘Return’
- Start the simulator operating by selecting ‘Run’.

3.3 List of ACR simulator display screens

(1) ACR Plant Overview
(2) ACR Control Loops
(3) ACR Control & Shutdown Rods
(4) ACR Reactor Power Control
(5) ACR Trip Parameters
(6) ACR Reactor Coolant System
(7) ACR Coolant Inventory & Pressurizer
(8) ACR Coolant Inventory Control
(9) ACR Coolant Pressure Control
(10) ACR Turbine Generator
(11) ACR Feedwater & Extraction steam
(12) ACR MW demand SP & SGPC
(13) ACR Passive core cooling
(14) ACR Trends

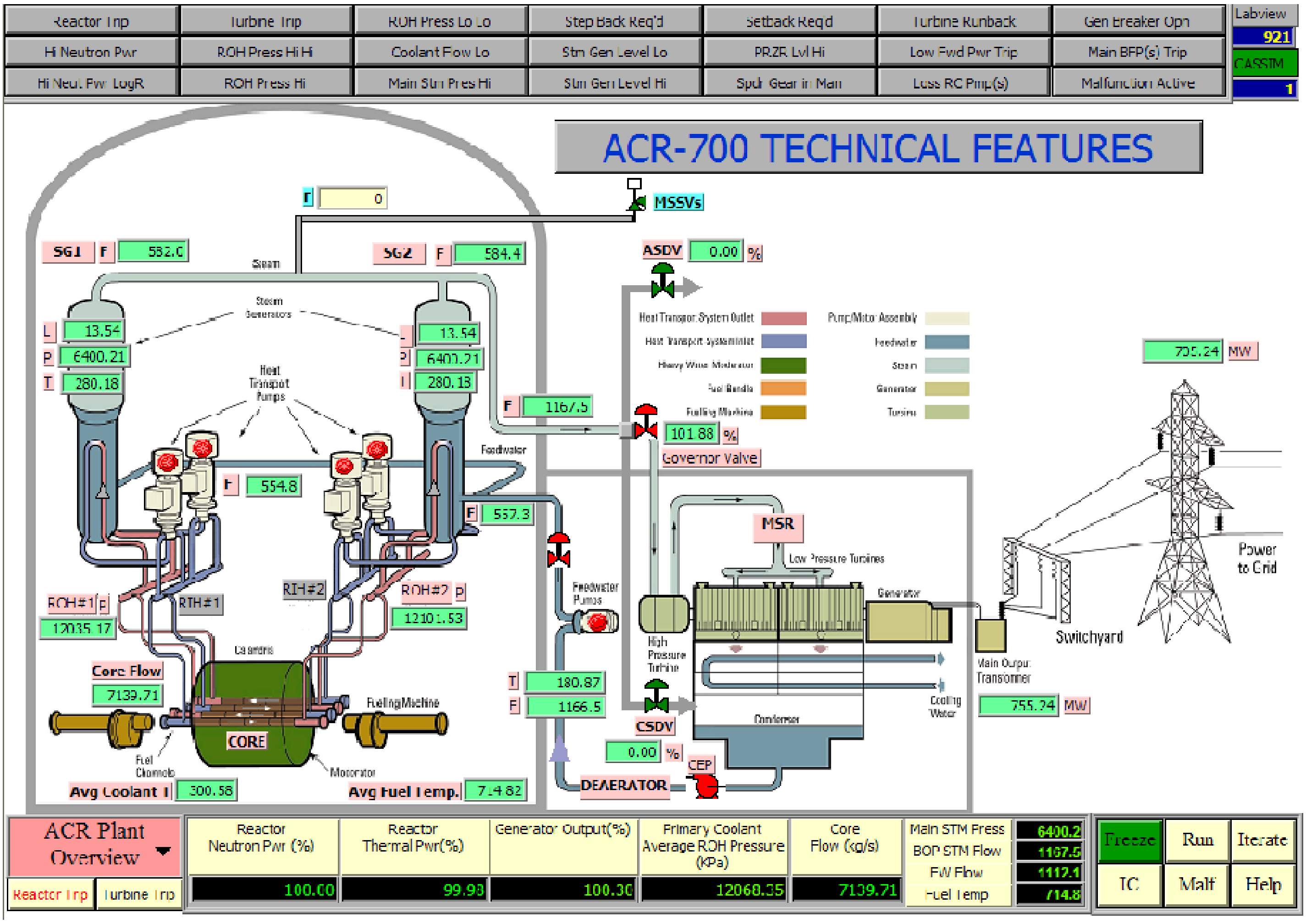
3.4. Simulator display common features

The ACR simulator is made up of 14 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;
- Top right hand corner shows the simulator status:
  - ⇒ the window under ‘labview’ (this is the proprietary 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 proprietary 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 ‘Stop’ the simulation click on ‘Freeze’ at the bottom right hand corner;
- The bottom of the screen shows the values of the following major plant parameters:
  - ⇒ Reactor neutron power (%)
  - ⇒ Reactor thermal power (%)
  - ⇒ Generator output (%)
  - ⇒ Primary coolant average reactor outlet header (ROH) pressure (kPa)
  - ⇒ Core flow (kg/sec)
  - ⇒ Main steam pressure (KPa)
  - ⇒ Balance of Plant (BOP) steam flow (Kg/sec)
  - ⇒ Feedwater Flow (Kg/sec)
  - ⇒ Average fuel temperature (Deg. C)
- 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. ‘ACR 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, if the on-line help program is provided.

3.5 ACR plant overview



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):

- REACTOR is a 3-D spatial kinetic model with six groups of delayed neutrons. The decay heat model uses a three-group approximation. The reactivity calculations include reactivity feedback effects, reactivity control and safety devices: shutoff rods (SOR), zone control units (ZCU), absorber rods (MCA), Xenon/Iodine, fuel temperature, moderator temperature, coolant temperature, Gadolinium, fresh fuel reactivity. The parameters displayed are:
  - ⇒ Neutron power (% full power)
  - ⇒ Reactor thermal power (% full power)
- Reactor coolant main loop with four heat transport system (HTS) pumps, two steam generators, two Reactor Inlet Headers (RIH#1, RIH#2); two Reactor Outlet Headers (ROH#1, ROH#2); coolant channels in core. Pressure and inventory control systems are shown on subsequent displays. The parameters displayed are:
  - ⇒ Reactor pressure (KPa) at ROH#1, ROH#2.

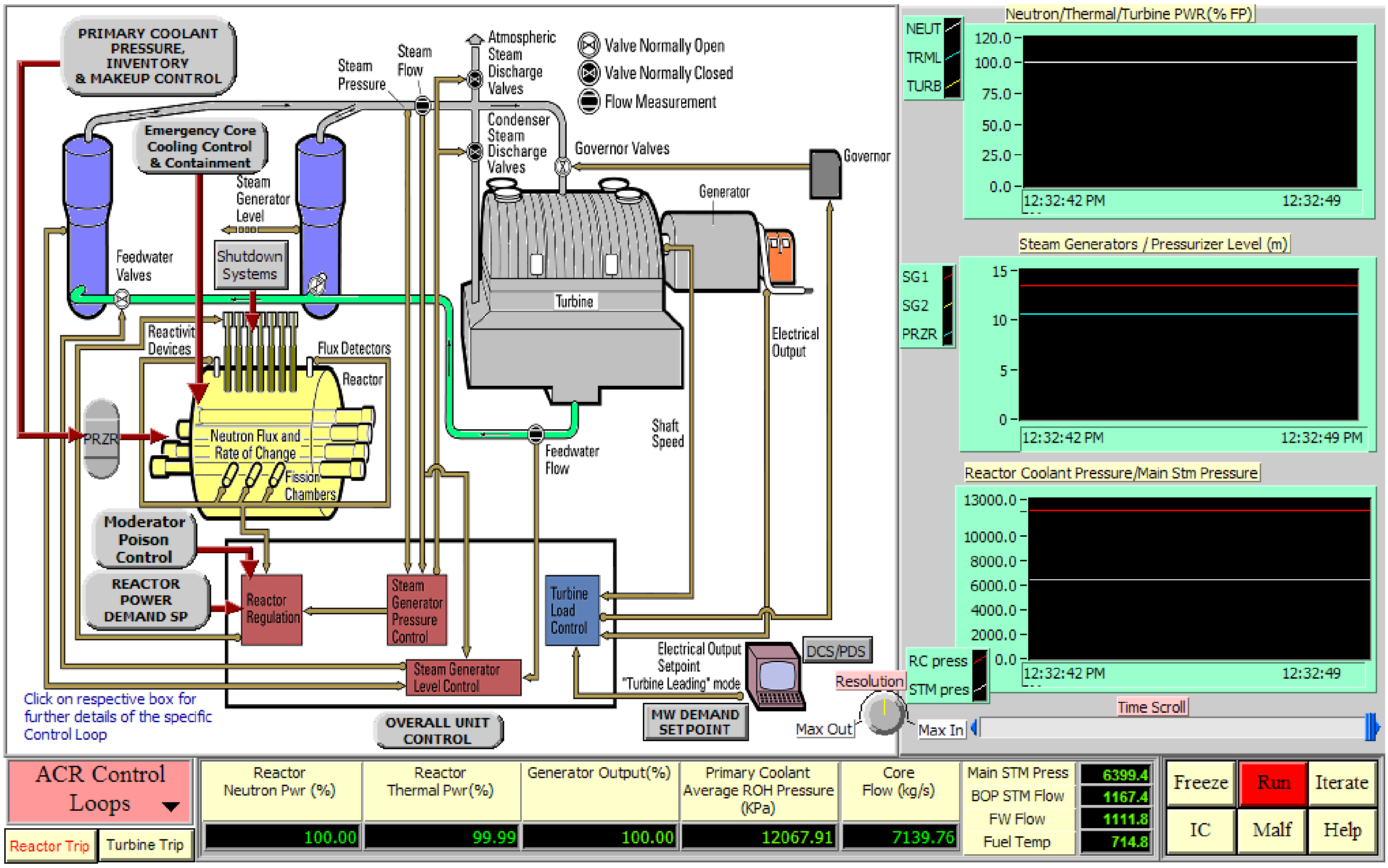
- ⇒ Reactor core flow (kg/sec)
- ⇒ Average reactor coolant temperature (°C)
- ⇒ Average fuel temperature (°C)
- ⇒ Status of the four coolant heat transport pumps (HTS P#1, 2, 3, 4)

- The two steam generators are individually modeled, along with balance of plant systems. The parameters displayed are:
  - ⇒ SG 1, 2 level (m)
  - ⇒ SG 1, 2 steam flow (kg/sec)
  - ⇒ SG 1, 2 steam pressure (kPa)
  - ⇒ SG 1, 2 steam temperature (°C)
  - ⇒ Total steam flow (kg/sec) from the steam generators
  - ⇒ Steam flow to main steam relief valves (MSSV's). Opening status of the main steam relief valves (MSSV's). The MSSV's are represented by one valve symbol - that is, in the event that any MSSV opens, the valve symbol colour will be red; green when all MSSV's are closed.
  - ⇒ Steam flow to Atmospheric Discharge Valves (ASDV), and Condenser Steam Discharge Valves (CSDV). The ASDV and CSDV are respectively represented by one valve symbol - that is, in the event that any valve opens, the valve symbol colour will be red; green when all valves are closed
- Steam Turbine is a simple model. The parameters displayed are:
  - ⇒ Status of turbine control valves is indicated by their colour: green is closed, red is open
  - ⇒ Moisture separator and reheater (MSR) drains flow (kg/sec)
  - ⇒ Main steam turbine stop valves (MSV) status
  - ⇒ Condenser steam bypass (dump) valves 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 simplified; the parameters displayed on the plant overview screen are:
  - ⇒ Total feedwater flow to the steam generators (kg/sec)
  - ⇒ Average feedwater temperature after the high pressure heaters (HPHX)
  - ⇒ Status of boiler (SG) feed pumps (BFP) is indicated as red if any pumps are 'ON' or green if all the pumps are 'OFF'

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.

Note: to facilitate simulator users a better understanding of the ACR-700 technical features, a "hot" button is provided for users to navigate the five unique features of ACR-700: (a) Horizontal Fuel Channel (b) Fuel Bundle Design (c) D_2O moderation (d) On-Power Refueling (e) Passive Safety.

3.6 ACR control loops



The plant power control function of a ACR type NPP is performed by two, separate control modes — one for the turbine generator, called 'turbine leading'; and the other one for the reactor, called 'reactor leading'. These two distinct modes of overall plant control can be switched between each other and are well coordinated for plant startup, shutdown, power operations of all kinds, and for plant upset conditions.

In the 'turbine leading' control mode, generator power is controlled according to the power demanded by means of a remote reference value (e.g. operator input), and/or by a value derived from the actual generator frequency deviation from the grid. Using this deviation from setpoint, the reactor power is adjusted using the main steam pressure error – i.e. deviation from normal main steam pressure setpoint. The latter point requires the following explanation: ACR-700 NPP is operating with a *constant* main steam pressure of 6,400 KPa. Any mismatch of energy flow from the SG to the turbine will result in changes in the main steam pressure. For example, suppose at full power, the turbine control valve is adjusted to 90 % opening, from its normal 100 % opening. The steam generators are generating 100 % full power steam flow, but the turbine only allows 90 % steam flow to pass. As a result, the extra steam flow capacity in the SGs will increase the main steam pressure to a value higher than the current setpoint of 6,400 KPa. The control system, sensing that current main steam pressure is *higher* than the setpoint, will signal the reactor regulating system (RRS) in *lowering* its reactor power setpoint accordingly. As a consequence, the primary coolant heat transfer to the SGs is reduced in a manner that would allow the SG main steam pressure to return to its setpoint.

This mode of control is typically used for baseload operation with constant or scheduled load; as well as load following operation with a frequency control function. It is important to note that steam generator pressure is maintained constant during this control mode operation.

In the ‘reactor leading’ control mode, the reactor power control is determined by operator input, and/or plant upset conditions (e.g. turbine trip), which in turn will set a new reactor power setpoint. The water-steam system, consisting of the turbine with its bypass system, and the steam generators, will adjust turbine load (MW) and/or other steam loads such as steam dump to atmosphere or condenser, to match with any reactor power changes whilst maintaining the steam generator pressure constant.

In support of these two control modes and plant safety functions, the ACR has the following control loops as illustrated by the ‘ACR control loops screen’ in the simulator:

(1) *Reactor power demand SP*
Reactor power demand setpoint (SP) is determined by operator input and/or by the automatic limitation functions such as the reactor stepback, which requires a step change in power reduction, or reactor setback, which requires power reduction at a fixed rate. The automatic limitation functions are triggered by specific reactor/coolant process conditions, which exceed alarm setpoints. The Reactor power demand setpoint (SP) provides input to the computer control program “demand power routine” (DPR).

(2) *Reactor Regulating System (RRS)*
The reactor regulating system (RRS) is composed of input sensors (fission chambers, in-core flux detectors, and process measurements), reactivity control devices (zone control units, control absorbers), hardware interlocks, and display devices. The power measurement and calibration routine uses measurements from a variety of sensors (self-powered in-core flux detectors, fission chambers, process instrumentation) to arrive at calibrated estimates of bulk and zonal reactor power.

The demand power routine (DPR) computes the desired reactor power setpoint and compares it with the measured bulk power to generate a bulk power error signal that is used to operate the reactivity devices.

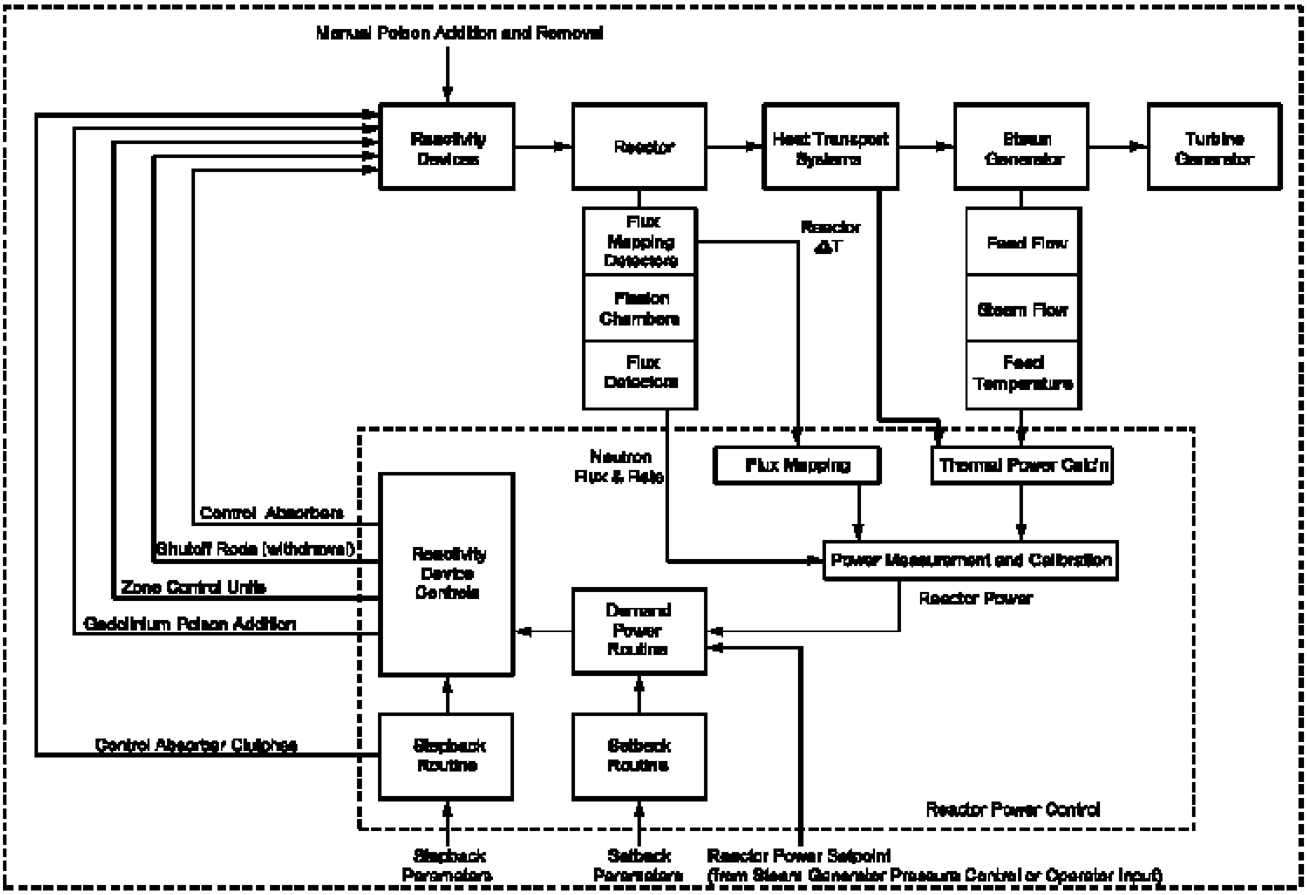
The primary reactivity control devices are the 18 zone control absorber (ZCU) elements (configured as nine units each containing two absorber elements). The zone control absorber element insertions are varied in unison for bulk power control, or differentially for tilt control.

In the “Turbine Leads” mode of operation, the reactor power setpoint is calculated by the steam generator pressure control program (SGPC). In the “Reactor Leads” mode of operation, the reactor power setpoint is set by the operator, or, in the case of abnormal plant conditions requiring power reductions, is automatically calculated by the RRS program.

In addition to controlling reactor power to a specified setpoint, the reactor regulating system monitors a number of important plant variables, and reduces the reactor power when any of these variables exceed specified limits. This power reduction may be fast (stepback), or slow (setback), depending on the possible consequences of the variable lying outside its normal operating range. The signal processing logic associated with

RRS, implemented in the distributed control system (DCS), is redundant and fail-safe in software and hardware.

A general block diagram of the reactor regulating system is shown in figure below:



Reactor Regulating System Block Diagram

(3) *Moderator Poison Control*
The reactivity depth of the zone control units (ZCU) in conjunction with the eight control absorbers (MCA) is sufficient to shut down the reactor even in the fresh fuel state when the fuel temperature reactivity feedback is at its maximum.

The normal method of maintaining the reactor adequately subcritical is by the manual addition of poison to the bulk moderator. Addition of moderator poison is also a possible but unlikely means for the operator to reduce reactor power to low levels. The poison solution is pre-mixed in the respective tank and can be added under gravity to the moderator circulating pump suction line by opening a single valve from the poison tank, via controls in the main control room.

The moderator poison system has a very large reactivity depth and is capable of reducing reactor power, and keeping it adequately subcritical under any conditions. Plant operating procedures define the level of moderator poison required to achieve the guaranteed shutdown state (GSS) in various circumstances.

Two moderator poison addition systems are provided:

(a) Boron addition, initiated manually, is used as a source of long-term negative reactivity when the reactor has excess fuel reactivity. *Note: Boron addition and removal is not modeled in the simulator.*
(b) Gadolinium addition, normally initiated manually, is used as a source of short-term negative reactivity, to compensate for a lack of xenon (gadolinium burns out at a rate similar to xenon production rate.) Under special conditions (positive flux rate and large power error) RRS will add gadolinium automatically.

Removal of poison from the moderator is via the moderator ion exchange purification facility. Procedural controls ensure that no poison removal takes place during a deliberate shutdown state using moderator poison. Successful poison addition requires the moderator circulating pumps to be operating.

(4) *Primary Coolant Pressure Control*
Reactor coolant pressure control in the ACR is performed by the pressurizer pressure control system. This provides the capability of maintaining or restoring pressure at the design value following normal operational transients that would cause pressure changes. It is done by the control of heaters and a spray in the pressurizer. The system also provides steam relief capability by controlling the power relief valves.

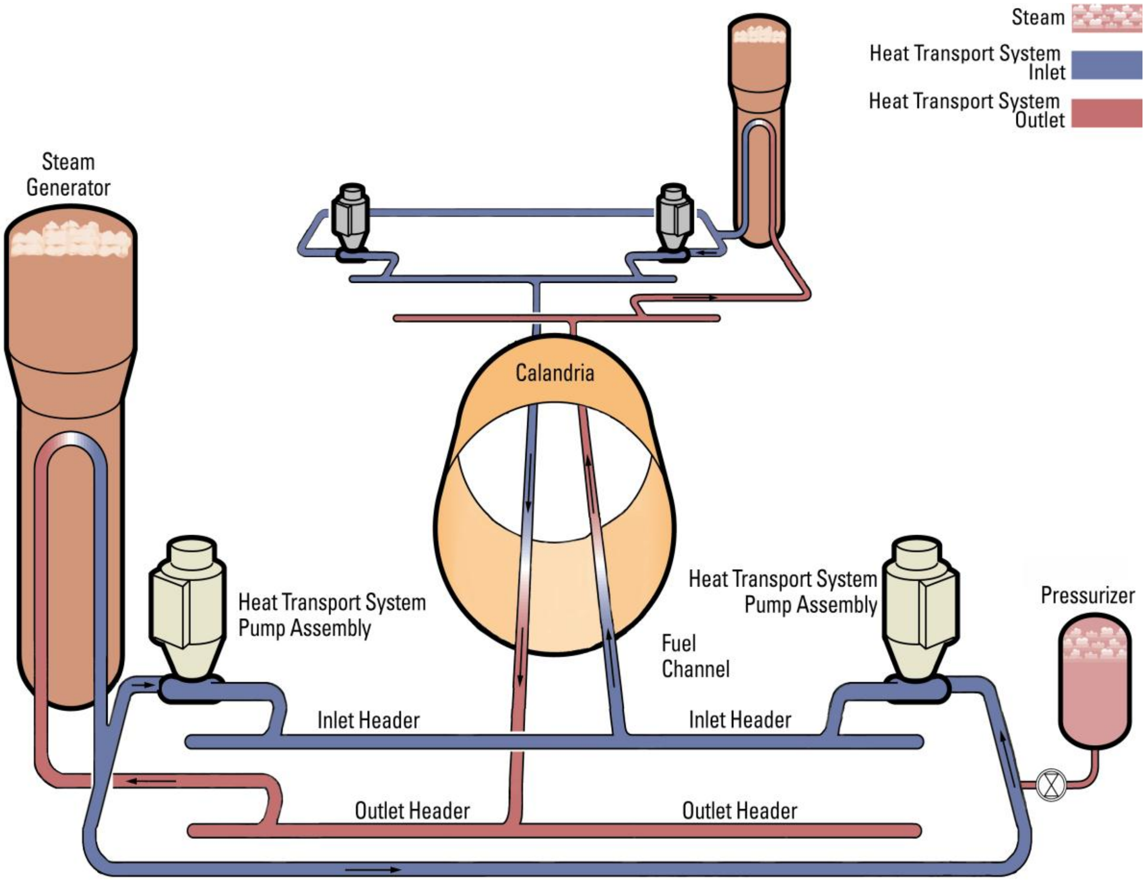
Under normal operating conditions, the pressurizer is the principal component in the pressure control of the HTS. It is a pressure vessel that is partly full of liquid water with the remainder being saturated vapour in equilibrium with the liquid. The Pressurizer is connected to the Reactor Outlet Header (ROH) of the HTS via a motorized valve. When this motorized valve is fully open, the control is in "Normal" mode. Should any pressure changes occur in the reactor coolant outlet header in the HTS, there will be in-surge or out-surge of coolant into or out of the Pressurizer through this motorized valve, depending on the differential pressure between the Pressurizer and ROH.

Hence it is necessary to provide the capability of maintaining or restoring pressurizer at the design pressure value following normal operational transients which would cause Pressurizer pressure changes. This "Normal" mode of pressure control is handled by the pressurizer pressure control system which controls a number of immersion heaters, as well as a spray system in the pressurizer. The system also provides steam relief capability by controlling the steam relief valves to the Bleed Condenser.

At low reactor powers the pressurizer may be isolated from the HTS by closing the motorized valve that is normally open to connect HTS to the pressurizer. In this case, it is known as the "Solid" mode of pressure control. In "solid" mode, the pressure of the HTS measured at the reactor outlet headers is controlled by adjusting the feed and bleed flows in and out of the HTS.

(5) *Primary Coolant Inventory & Makeup control*

The primary coolant inventory & makeup control is performed by the pressurizer level control system. It provides the capability of establishing, maintaining and restoring the pressurizer water level to the target value which is a function of the average coolant temperature (affecting coolant swell and shrink). It maintains the coolant level in the pressurizer within prescribed limits by adjusting the flow of the coolant feed and coolant bleed system, thus controlling the reactor coolant water inventory.



(6) *MW demand setpoint demand*
Megawatts (MW) demand setpoint is determined by operator input. This input will used as reference target for raising or lowering the turbine load under “turbine-lead” mode.

(7) *Steam Generator Pressure Control (SGPC)*
Steam generator pressure is maintained at an equilibrium, constant value determined by the heat balance between the heat input to the steam generator and the turbine steam consumption. If during power maneuvers, or plant upset, there is a mismatch between reactor thermal power and the turbine power, steam generator pressure will vary and deviate from the pressure setpoint. Under “turbine leading” control mode, control signals will be sent to the reactor power control system to reduce or increase reactor neutron power, in order that steam generator pressure will return to its setpoint. Likewise, under “reactor leading” control mode, control signals will be sent to the turbine governor control system to reduce, or raise turbine load, in order that steam generator pressure will return to its setpoint.

In the event of a sudden turbine load reduction, such as abnormal load rejection, or turbine trip, where the above described control system is not fast enough to alleviate steam pressure changes due to such transients, an automatic steam bypass (dump) system is provided to dump the steam to the condenser and/or to atmosphere, if the steam generator pressure exceeds a predetermined setpoint.

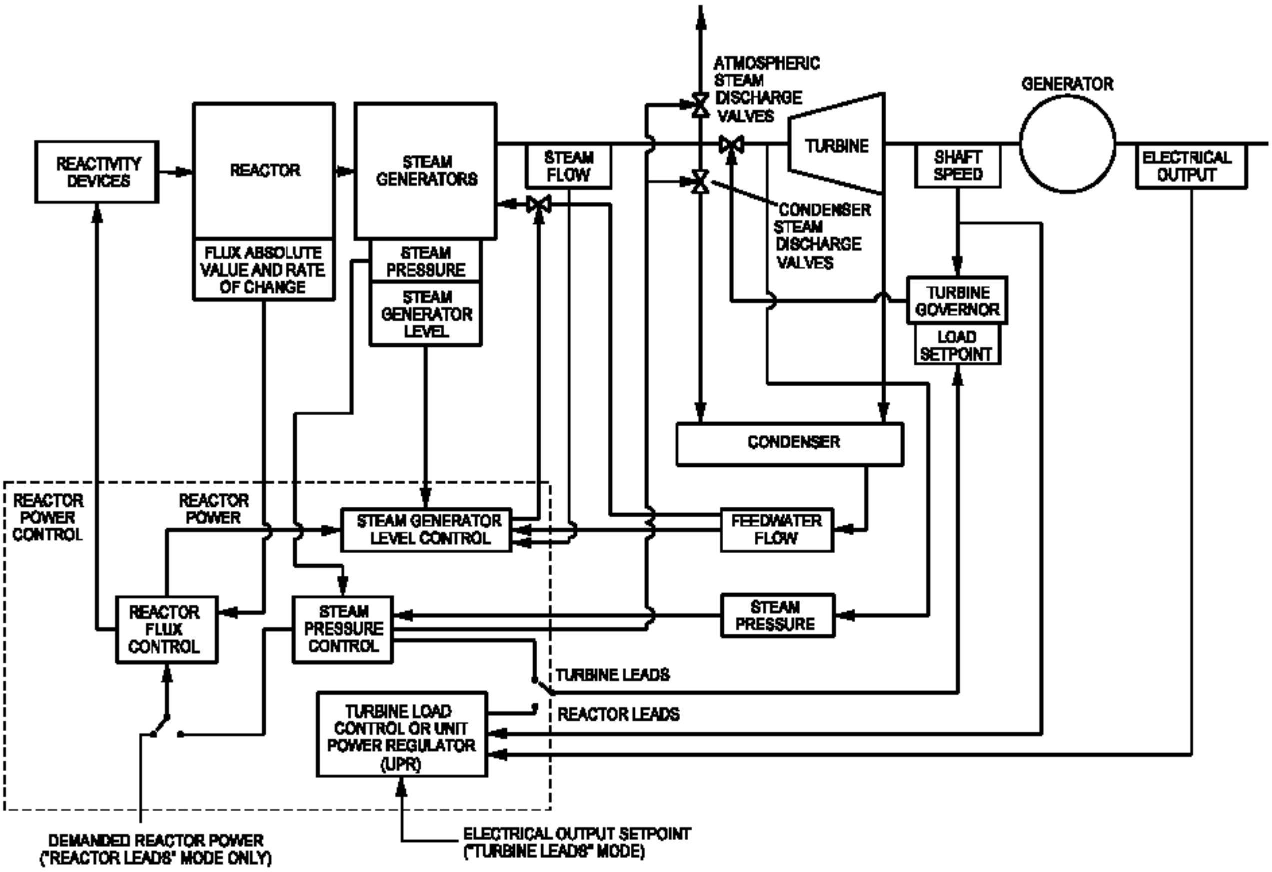
(8) *Steam generator level control*
The steam generator level control system maintains a programmed water level that is a function of turbine load. The control is a three-element controller that regulates the feedwater valve by matching feedwater flow (1^{st} element) to steam flow (2^{nd} element) from the steam generator, while maintaining the generator level (3^{rd} element) to its setpoint.

(9) *Turbine Load Control*
In the “Turbine Leads” mode, the turbine load control can be done by the operator entering the target load known as Mega Watts (MW) Demand Setpoint, and loading/unloading rate. This communicates its actions to the turbine generator governor controller through the steam generator pressure control (SGPC) program. The turbine generator governor controller will regulate the steam flow through the turbine to meet turbine load target by controlling the opening of the turbine governor valve.

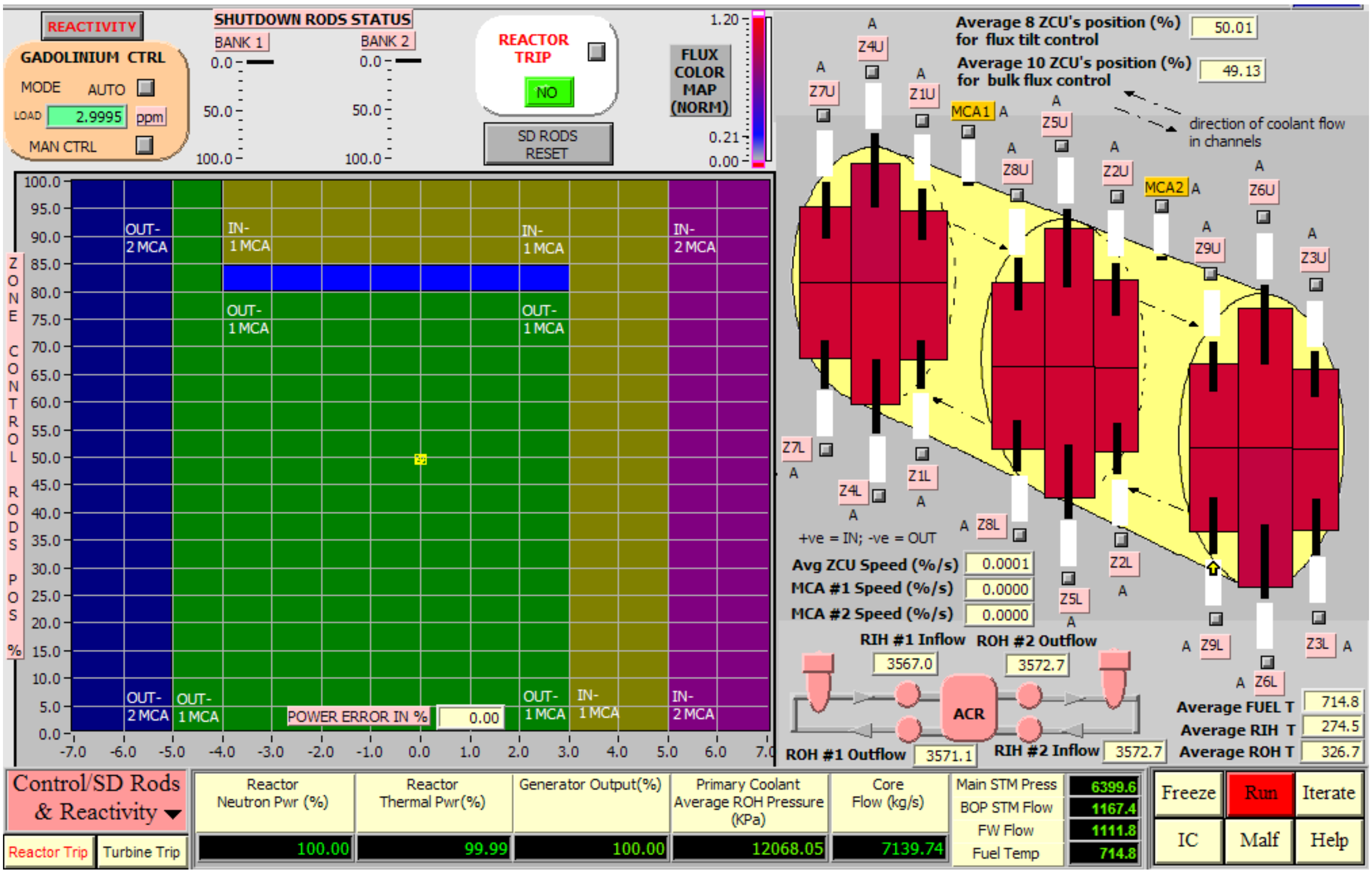
(10) *Emergency Core Cooling and Containment Control*
See details in section (2.7) (c) and (d).

(11) *Reactor Shutdown Systems Control*
See details in section (2.7) (a) and (b).

The control loops described above can be summarized by the following control block diagram:



3.7 ACR Control Rods and Shutdown rods & Reactivity



This screen shows the status of the shutdown system #1 (SDS1), as well as the respective positions, and average speed of the 18 zone control units (ZCU). Similarly, the positions and the speeds of the two banks of absorber rods (MCA) are also displayed. The reactivity contributions from the reactor feedback effects, and each reactivity control device: shutoff rods, zone control units (ZCU), absorber rods (MCA), and gadolinium can be observed from the pop-up window by pressing the "Reactivity" button on the top left of the screen.

All the reactivity devices considered for regulation as well as shutdown purposes, are installed from above the Calandria, known as the Reactivity Mechanism Deck. Reactivity control is provided for the following effects:

(1) Long-term bulk reactivity is mainly controlled by on-power refuelling. This is the only method for adding absolute positive reactivity to the core.
(2) Small, frequent reactivity changes, for both global and spatial neutron power, are controlled by the zone control unit (ZCU) system.
(3) Negative reactivity to supplement the zone control units (ZCU), particularly for fast power reductions and to override the negative fuel temperature effect for large power level decreases, is provided by the insertion of mechanical control absorbers (MCA) from their

normal "poised position" (above the core), to part way or all the way to their fully inserted position at core.

(4) Excess reactivity due to fresh fuel and decay of xenon following a long shutdown, are compensated by adding poison (boron or Gadolinium) to the moderator.
Note: only Gadolinium poison is modeled in this simulator.

(5) Reactivity variations due to on-power refuelling during equilibrium operation: since the reactor is fuelled continuously and on-power at a rate which keeps the reactor critical, the control requirements for refuelling are within the range of the zone controller response. Soluble poison concentration is normally near zero. For a standard 2-bundle shift fuelling scheme, the reactivity increase due to refuelling in an average channel is less than 0.2 mk. This reactivity change is sufficiently controlled by the zone controllers.
Note: on-power refuelling is not modeled, hence the reactivity variations due to on-power refuelling will not be observed in this simulator.

(6) Rapid shutdown of the reactor is by dropping solid control absorbers (shutdown rods) into the core, and/or by the fast injection of large amounts of liquid poison into the moderator.
Note: only SDS1 is modeled in this simulator.

Specific details regarding the respective reactivity devices are provided below:

(a) Zone Control Units (ZCU)

The zone control system consists of nine vertical assemblies with two independently moveable segments in each assembly, hence 18 ZCUs. Reactivity is adjusted by varying the lengths of the absorbers inserted into the core, based on a signal from the station computer. The zone controller system is designed so that, during normal operation, the average zone control absorber element remains in the range 20% to 80% of full insertion.

The zone control system is designed to perform two main functions:

a) Bulk control - i.e., control of power output. The zone control system will provide short-term fine control of reactivity to maintain reactor power at demanded level during normal operation. The bulk flux control is mainly carried out by the 10 zone control rods located near the center of the reactor vessel, namely, Z2U, Z2L, Z4U, Z4L, Z5U, Z5L, Z6U, Z6L, Z8U, Z8L.
b) Spatial control - i.e., control of flux and power shapes. The zone control system will maintain the desired global flux and power distributions by counteracting any power distortion or oscillation brought on by a space dependent reactivity perturbation. In practice, the perturbations can be caused by:
(1) fuel burnup and refuelling of channels,
(2) power level changes,
(3) changes in the heat transport system conditions,
(4) xenon oscillations,
(5) movement of absorber elements, and
(7) small variations in moderator poison concentration.
The spatial control is mainly carried out by the 8 zone control units (ZCU) located near the four corners of the reactor vessel, namely, Z1U, Z1L, Z3U, Z3L, Z7U, Z7L, Z9U, Z9L.

(b) Control Absorber Units (MCA)

Eight control absorber units (MCAs) are provided for rapid controlled power reductions and to compensate for the fuel temperature reactivity effect for shutdown under fresh fuel conditions. For the simulator, the eight control absorbers are modeled as two banks of absorber rods.

The control absorber elements are physically similar to the shutdown SORs. Normally, the control absorbers are positioned outside the core. Their arrangement is shown in the Figure shown in section 2.2. Since the reactivity increase following a power reduction is significant and usually rapid, the zone controllers alone are incapable of counteracting the increase in all cases. In particular, the reactivity increase is the highest following a hot shutdown (when fuel temperature drops to coolant temperature), and for fresh fuel. In this case, MCAs are used to compensate for the reactivity increase. The control absorbers are normally inserted in banks (of two absorber elements each) but can also be inserted individually. The percentage insertion depends on the degree of reactor power reduction. The optimum speed of insertion is determined primarily from control considerations. In summary, the maximum rate of positive reactivity insertion due to any set of reactivity devices of the reactor regulating system ranges between 0.05 mk/s for MCAs and 0.2 mk/s for the ZCU.

(c) Shutdown Systems

The ACR-700 reactor is equipped with two physically independent shutdown systems. These systems are designed to be both functionally different and geometrically separate. The functional difference is achieved by the use of 24 shutoff units for SDS1 and six liquid injection nozzles for SDS2. The 24 shutoff rods are inserted vertically by gravity drop. Their locations are shown in the Figure in section 2.2. The six poison injection nozzles are positioned horizontally, as shown in same Figure (indicated on the figure as LI1 through 6). A concentrated solution of gadolinium in D2O is injected under pressure into the moderator space between the calandria tubes. The in-core instrumentation feeding flux signals to the shutdown systems is also separated in a geometrical sense. Vertical flux detector units and fission chambers on side ‘D’ are used for SDS1 while horizontal flux detector units and ion chamber units on side ‘B’ are used for SDS2. Other instrumentation monitoring the core conditions also feed into SDS1 and SDS2. *Note: SDS2 is not modeled in this simulator*.

The display parameters shown on this screen are as follows:

- The positions of each of the two SDS SHUTDOWN ROD banks are shown relative to their normal (fully withdrawn) position. In this ACR Simulator, the reactivity worth for each SDS SHUTDOWN ROD bank is - 30 mk, so the total reactivity worth for the two SDS SHUTDOWN ROD banks, when fully inserted in core is - 60 mk. The trip time is 150 ms delay, plus 1.5 sec. for full SDS1 rods drop
- REACTOR TRIP status is shown as NO (green) or YES (yellow), the trip can be reset here; note that SDS RESET must also be activated before reactor shutdown system will begin withdrawing the Shutdown Rods.
- The REACTIVITY CHANGE (mk) of each device and parameter from the initial 100% full power steady state is shown. These include:
    1. SHUTDOWN RODS (SOR): -60 MK for full SOR drop.
    2. ZONE CONTROL UNITS (ZCU): 18 ZCUs; total reactivity worth is 9 MK; +4.5 MK fully withdrawn; -4.5 MK fully inserted in core. The maximum speed of ZCU movement gives + or – 0.2 mk per sec. of reactivity rate change.

Note: control buttons are provided on the "Reactivity pop-up" to allow users to increase and decrease the reactivity worth of the ZCU online. This is to facilitate user's observation of the ZCU control system response with various design values of total MK worth as an educational exercise.

3. MECHANICAL ABSORBER ROD UNITS (MCA): total of 8 absorber rods, 4 in the core center region, 4 in the core outer region. Total reactivity is -12 MK for full insertion. The speed for all the absorber rods is constant, and the full insertion travel time is 120 sec. For the simulator, the absorber rods are divided into 2 banks, with each bank's reactivity worth of – 6 MK.

4. XENON: full power steady state Xenon load –26 MK; peak Xenon load 12 hours after full power trip –63 MK.

5. FUEL TEMPERATURE reactivity feedback: -0.014 MK/deg. C (from 687 to 787 deg. C).

6. MODERATOR TEMPERATURE reactivity feedback: -0.024 MK/deg.C (from 70 to 90 deg. C).

7. COOLANT TEMPERATURE reactivity feedback: -0.01 MK/deg.C (from 290 to 310 deg. C)

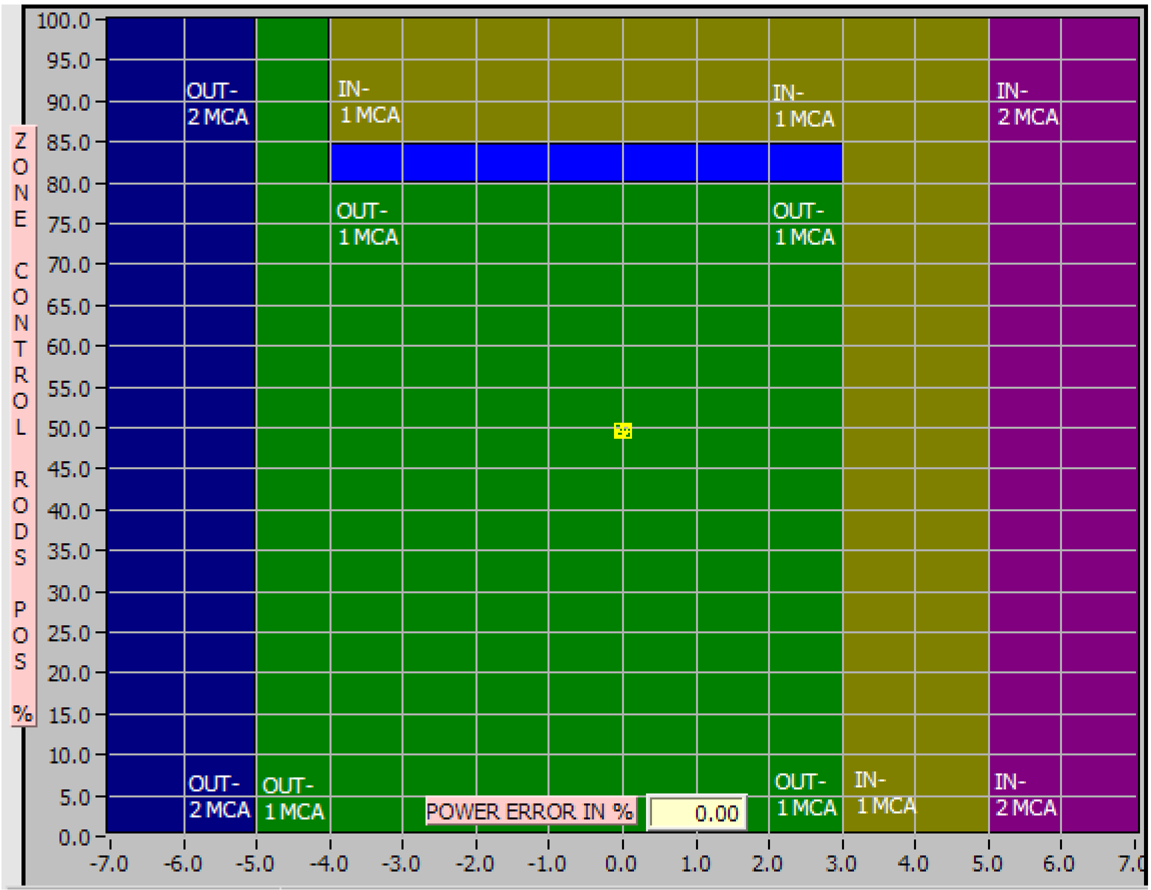
8. GADOLINIUM reactivity feedback: 1 ppm will yield – 6 MK. Addition rate ~ 0.5 MK/minute; removal rate ~ 0.1 MK/minute.

9. FRESH FUELreactivity: +44 MK.

⇒ Note that reactivity is a computed parameter, and not a measured parameter at the actual plant. It is displayed on the simulator as a means of understanding how the reactor is being controlled, using reactivity as the parameter.

⇒ Note also that when the reactor is critical, the Total Reactivity must be zero.

This screen also shows the movement of the zone control units (ZCU) and mechanical absorbers rods (MCA) as a function of the Reactor Power Error (%) (see definition below). The relationship is depicted by the movement of a yellow cursor shown on a graphical X-Y plot. The Plot has Y-axis as Average Zone Control Rods Position (%) and X-axis as Reactor Power Error, and is known as Reactivity Limit Control Diagram.



Reactivity Limit Control Diagram

As mentioned above, the mechanical absorber rods are divided into two banks. The drive logic for the absorber banks is as follows:

- If the absorber banks control is set in AUTO, the absorber banks will move according to the power error versus zone control rods position as per the above Reactivity Limit Control Diagram.
- In the GREEN color region: designating Reactor Power Error as PE (%) & Average Zone Control Rods Position (%) as ZCP, the green color region is defined by:
  (a) 3 % ≥ PE ≥ -4% ; 80 % ≥ ZCP ≥ 0 % and
  (b) -4 % > PE ≥ -5% ; 100 % ≥ ZCP ≥ 0 %

  In this region, the absorber bank 2 will be driven OUT first (if it is in core), and absorber bank 1 will start to drive OUT when bank 2 is completely driven out.
- In the LIGHT BROWN color region - it is defined by:
  (a) 5 % ≥ PE ≥ 3% ; 100 % ≥ ZCP ≥ 0 % and
  (b) 3 % ≥ PE ≥ -4% ; 100 % ≥ ZCP ≥ 85 %

  In this region, the absorber bank 1 will be driven IN first, and bank 2 will start to drive IN when bank 1 is completely driven in core.
- In the DARK BLUE region - it is defined by:
  -7 % ≥ PE ≥ -5% ; 100 % ≥ ZCP ≥ 0 %

  In this region, both banks of absorber rods will be driven OUT simultaneously.
- In the MAGENTA color region – it is defined by :

7 % ≥ PE ≥ 5% ; 100 % ≥ ZCP ≥ 0 %

In this region, both banks of absorber rods will be driven IN simultaneously.

- In the LIGHT BLUE region – it is defined by :
  3 % ≥ PE ≥ -4% ; 85 % ≥ ZCP ≥ 80 %

  This is a transitional region between the GREEN region and the LIGHT BROWN region, where the absorber rods which are driving OUT in GREEN region, will reverse direction (driving IN) in LIGHT BROWN region, or vice versa. Hence, this region serves as a deadband for which the absorber rods may not move, until clear demarcation in entering the GREEN region or LIGHT BROWN region is established by the relationship of Power Error (%) versus zone control rods position (%).

NOTE: the 18 zone control units (ZCUs) are normally controlled by the Reactor Regulating System (RRS) in "auto" mode. The control of ZCU can be switched to "manual" mode where each ZCU can be controlled individually with the control button for "IN", "STOP", "OUT". Likewise, the two banks of "absorber" rods are normally controlled by RRS in "auto" mode. The control of individual bank of "absorber" rods can be switched to "manual" mode where each bank can be controlled individually with the control button for "IN", "STOP", "OUT".

The screen also displayed the following parameters related to the reactivity control devices:

1. Average 8 ZCUs position (%) responsible for flux tilt control;
2. Average 10 ZCUs position (%) responsible for bulk flux control;
3. Average of all 18 ZCU speeds in % per sec.
4. Absorber rods MCA bank #1 speed in % per sec.
5. Absorber rods MCA bank #2 speed in % per sec.

As well included on this screen is the Gadolinium (Gd) control system, which can be used for relatively short term core reactivity control. If the control system is in AUTO mode, and the Power error (%) > 5 %, and neutron log rate > 0 % /sec., Gd will be added automatically, resulting in a negative reactivity rate of - 0.5 MK per minute, with a delay of 30 seconds. Gd in core will be slowly burnt out at a time constant of 9 hours at nominal core conditions. However, if needed, Gd can be removed MANUALLY, resulting in a positive reactivity rate of 0.1 MK/minute.

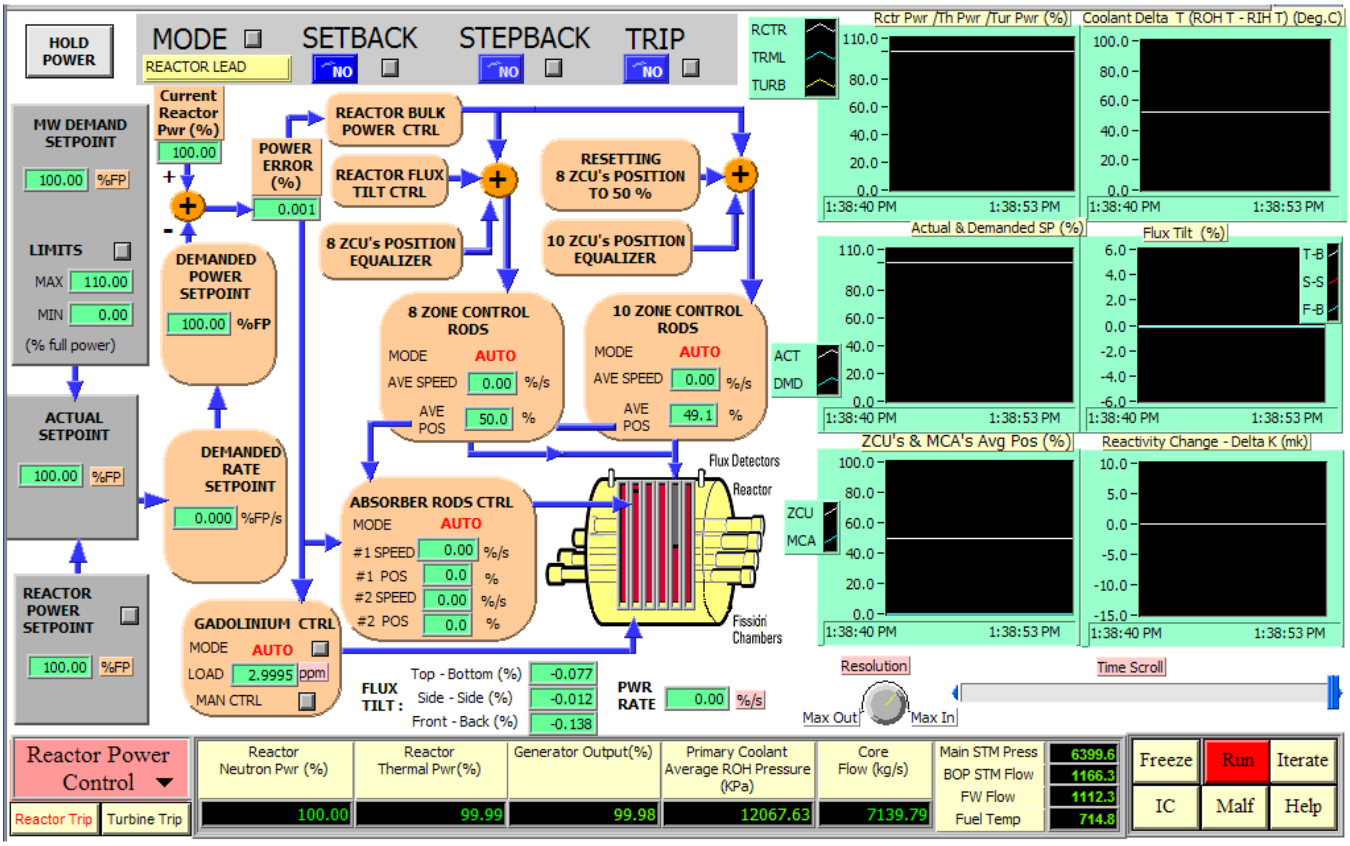
The screen also shows the reactor core normalized flux intensity map in color.

- The flux intensity scale is from 0 (grey color) - 1.2 (red color).
- The core flux mapping is represented in a simplified manner by 18 cells, with each cell representing a section of the core coinciding with the location of one zone control unit (ZCU). Each cell's flux intensity is represented by a color map.
- Axially, each cell is also aligned with a section of a lumped reactor channel (fuel and coolant) being modeled in the simulator. For the simulator, there are 6 lumped reactor channels modeled:
  Lumped Channel #1: represented by cell Z1U, Z2U, Z3U
  Lumped Channel #2: represented by cell Z4U, Z5U, Z6U
  Lumped Channel #3: represented by cell Z7U, Z8U, Z9U
  Lumped Channel #4: represented by cell Z1L, Z2L, Z3L
  Lumped Channel #5: represented by cell Z4L, Z5L, Z6L
  Lumped Channel #6: represented by cell Z7L, Z8L, Z9L

The coolant flows in adjacent channels are in opposite directions, namely, coolant at channel 1, 3, 5 flows in one direction to one Reactor Outlet Header; coolant at channel 2, 4, 6 flows in opposite direction to the other Reactor Outlet Header.

- In conjunction with the flux map of the core, the flow path of the reactor coolant through the core is also shown below the flux map. “Cold” reactor coolant from the U tubes steam generators outlets enters the reactor at the respective Reactor Inlet Headers entry points- RIH#1, RIH#2. The reactor coolant from the inlet headers then travels through the respective reactor core coolant channels.
- The reactor coolant carries the heat energy from the fuel pellets as it travels through core channels, and mixes with other coolant streams before leaving the reactor at the two “hot” Reactor Outlet Headers –ROH#1, ROH#2. The parameters displayed are:
  1. RIH#1, #2, coolant inlet flow rates in Kg/sec;
  2. ROH#1, #2 coolant outlet flow rates in Kg/sec.
  3. Average fuel temperature- deg. C
  4. Average coolant temperature at RIHs
  5. Average coolant temperature at ROHs.

3.8. ACR reactor power control



This screen permits control of reactor power setpoint and its rate of change while under Reactor Regulating System (RRS) control, i.e. in 'REACTOR LEADING' mode. Several of the parameters key to RRS operation are displayed on this page.

- The status of reactor control is indicated by the four blocks marked MODE, SETBACK, STEPBACK AND TRIP. They are normally blue but will turn red when in the abnormal state.
  - ⇒ MODE will indicate whether the reactor is under TURBINE LEADING to REACTOR LEADING control, this status can also be changed here.
  - ⇒ SETBACK status is indicated by YES or NO; setback is initiated automatically under the prescribed conditions by RRS, but at times the operator needs to initiate a manual setback, which is done from this page on the simulator: the target value (%) and rate (%/sec) need to be input.
  - ⇒ STEPBACK status is indicated by YES or NO; stepback is initiated automatically under the prescribed conditions by RRS, but at times the operator needs to initiate a manual stepback, which is done from this page on the simulator: the target value (%) needs to be input.
  - ⇒ TRIP status is indicated by YES or NO; reactor trip is initiated by the shutdown system, if the condition clears, it can be reset from here. Note however, that the "tripped" shutdown system must also be reset before RRS will pull out the shutdown rods, this must be done on the "shutdown rods" page

- Key components of RRS control algorithm are also shown on this screen.

  ⇒ REACTOR POWER SETPOINT target and rate are specified by the user on the simulator in terms of %FP and %FP/sec, i.e. as linear measurements, instead of the logarithmic values used in practice. The requested rate of change should be no greater than 0.8 % of full power per second in order to avoid a reactor LOG RATE trip. This is readily achieved in the 'at-power' range (above 15%FP), but only very small rates should be used at low reactor power levels (below 1%FP), such as encountered after a reactor trip.

  ⇒ The MW DEMAND SETPOINT is set equal to the MW SETPOINT under "TURBINE LEADING" control; the upper and lower limits on this setpoint can be specified here.

  ⇒ The ACTUAL SETPOINT is set equal to the accepted "REACTOR POWER SETPOINT' *TARGET* under RRS control in "REACTOR LEADING" mode.

  ⇒ HOLD POWER 'On' will select 'REACTOR LEADING' mode and stops any requested changes in DEMANDED POWER SETPOINT.

  ⇒ DEMANDED RATE SETPOINT is set equal to the accepted "REACTOR POWER SETPOINT' *RATE*, limited by the maximum rate of 0.8 % of full power per second.

  ⇒ DEMANDED POWER SETPOINT is the incremental power target, which is set equal to current reactor power (%) + rate (% / s) * program cycle time (sec). In this way, the DEMANDED POWER STEPOINT is "ramping" towards the REACTOR POWER SETPOINT target, at the accepted rate of change.

  ⇒ From the DEMANDED POWER SETPOINT, CURRENT REACTOR POWER, TARGET RATE, CURRENT RATE OF CHANGE OF REACTOR, the POWER ERROR can be determined as follows:

$$Perr = KB * \frac{(Nflux - PDEM)}{Nflux} + KRATE * (\frac{Nflux - NfluxP}{Nflux * DT} - RD)$$

  Where Perr = Reactor Power Error (%)

  Nflux = Current Bulk Reactor Power (%)

  NfluxP = Current Bulk Reactor Power in previous RRS program cycle (%)

  PDEM = Demanded Power Setpoint (%)

  RD = Reactor power rate demanded (%/sec)

  DT = Time (sec.) between successive execution of the RRS program

  KB = Gain constant for difference between current power versus demanded power.

  KRATE = Gain constant for the rate difference between the current reactor power rate versus the demanded rate.

  ⇒ The FLUX TILT component of the reactor consisting of multi-cells, representing respective section of the core, can be determined as follows:

$$FluxTC_i = KT * (Nflux_i - Nflux)$$

where $FluxTC_i$ = Flux tilt component for i^{th} cell in the reactor core (%)

Nflux = average reactor flux in core (%)

$Nflux_i$ = reactor flux at the i^{th} cell of the reactor core (%)

⇒ Having defined the Power Error, and Flux Tilt, the control algorithm for controlling the zone control units (ZCU) can now be described. A digital control algorithm commonly known as "velocity" control algorithm is used to compute the *speed* of the respective zone control units (ZCU), according to its assigned control function, namely: for bulk flux control, or for flux tilt control. As described above, the bulk flux control is mainly carried out by the 10 zone control rods located near the center of the reactor vessel, namely, Z2U, Z2L, Z4U, Z4L, Z5U, Z5L, Z6U, Z6L, Z8U, Z8L. The spatial control is mainly carried out by the 8 zone control units (ZCU) located near the four corners of the reactor vessel, namely, Z1U, Z1L, Z3U, Z3L, Z7U, Z7L, Z9U, Z9L.

⇒ The speed of a ZCU_j in the 10 zone "bulk flux" control group is determined by evaluating the individual control functions as given below:

ZCU_j speed = Reactor Bulk Power Control + Resetting Control for the 8 ZCUs (responsible for flux tilt control) to 50 % position + Equalizing Control for the 10 ZCUs (responsible for bulk flux control) to achieve uniform position in core.

⇒ The speed of a ZCU_i in the 8 zone "flux tilt" control group is determined by evaluating the individual control functions as given below:

ZCU_i speed = Reactor Bulk Power Control + Flux Tilt Control + Equalizing Control for the 8 ZCUs (responsible for flux tilt control) to achieve uniform position in core.

⇒ In control equation format, the speed of the respective ZCU is given by :

$$\textit{For the } 8 \textit{ Zones,}$$
$$ZCU\_SPD_i = G1 * Perr + G2 * FluxTC + KL * (ZCUP_i - AVG8ZCUP)$$
$$\textit{For the } 10 \textit{ Zones,}$$
$$ZCU\_SPD_j = G3 * Perr + G4 * (Avg8ZCUP - 0.5) + G5 * (Avg10ZCUP - ZCUP_j)$$

Where ZCU_SPD_i = ZCU_i speed in %/s , i = 1,…8 in the 8 zones group.
G1 = gain constant for power error.
G2 = gain constant for flux tilt error.
KL = gain constant for the position equalizer for the 8 zones group.
$ZCUP_i$ = ZCU_i (in the 8 zones group) position (%).
AVG8ZCUP = average ZCU position in the 8 zones group.

ZCU_SPD_j = ZCU_j speed in %/s, j = 1, … 10 in the 10 zones group
G3 = gain constant for power error.
G4 = gain constant for resetting control of the 8 zones group.
G5 = gain constant for the position equalizer for the 10 zones group.
$ZCUP_j$ = ZCU_j (in the 10 zones group) position (%).
AVG10ZCUP = average ZCU position in the 10 zones group.

The ZCU control functions responsible for bulk flux control and flux tilt control respectively are illustrated by “arrows” in the following diagram extracted from the simulator screen.

⇒ The auto/manual mode (changeable by user), ZCU average speed, and the average position of the ZCUs are displayed respectively for the 10 zones group responsible for bulk power control, and the 8 zones group responsible for flux tilt control.

⇒ The auto/manual mode (changeable by user), absorber rods speed, and the average position are displayed on this screen, respectively for the Bank #1, and Bank #2.

⇒ As well, the auto/manual mode for Gadolinium control (changeable by user), and the current Gd load (ppm) in core are displayed. The Manual control button for Gd addition and removal is also provided.

$$FluxTC_i = KT*(Nflux_i - Nflux)$$

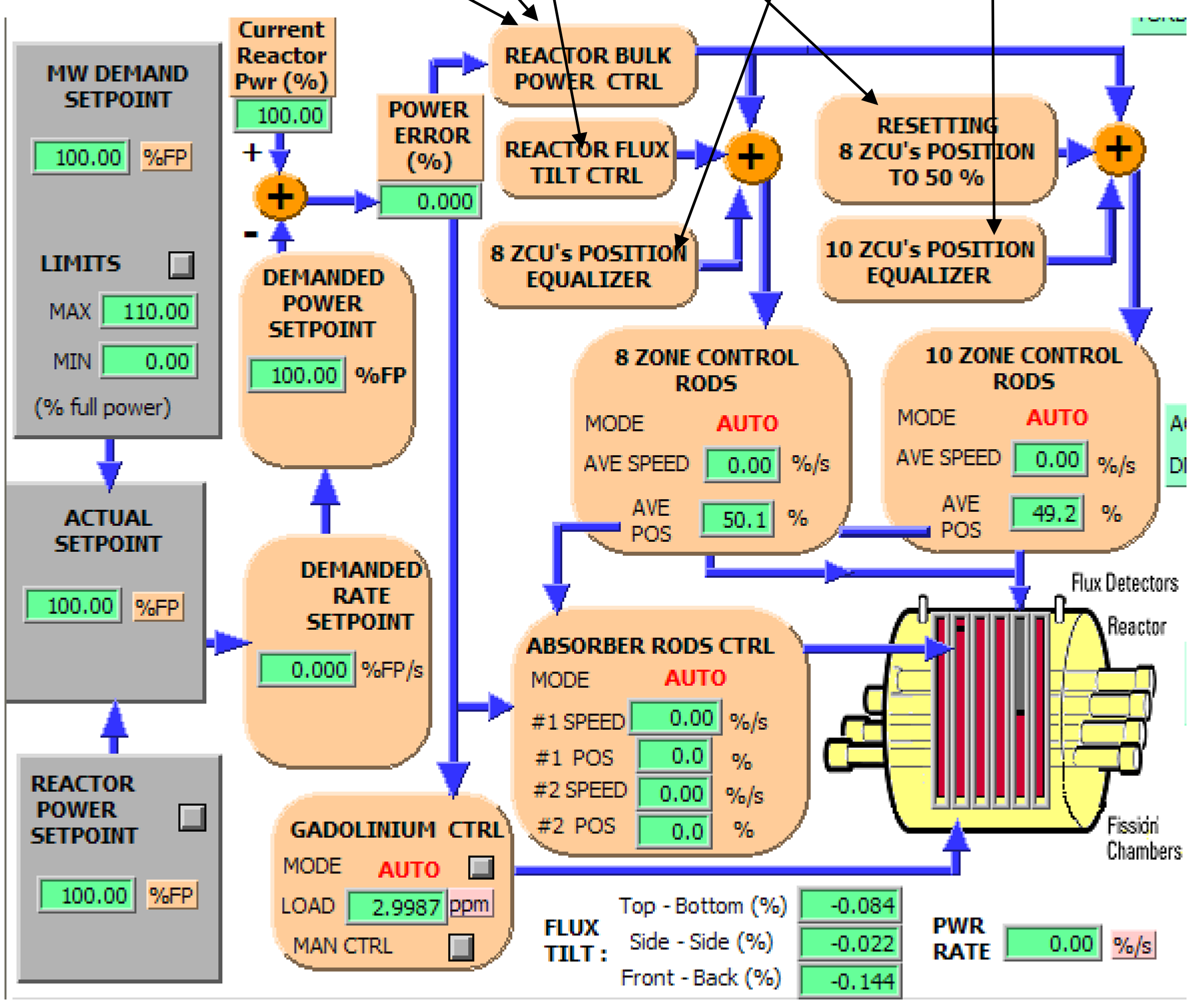
$$Perr = KB*\frac{(Nflux - PDEM)}{Nflux} + KRATE*\left(\frac{Nflux - NfluxP}{Nflux*DT} - RD\right)$$

For the 8 Zones,

$$ZCU\_SPD_i = G1*Perr + G2*FluxTC_i + KL*(ZCUP_i - AVG8ZCUP)$$

For the 10 Zones,

$$ZCU\_SPD_j = G3*Perr + G4*(Avg8ZCUP - 0.5) + G5*(Avg10ZCUP - ZCUP_j)$$

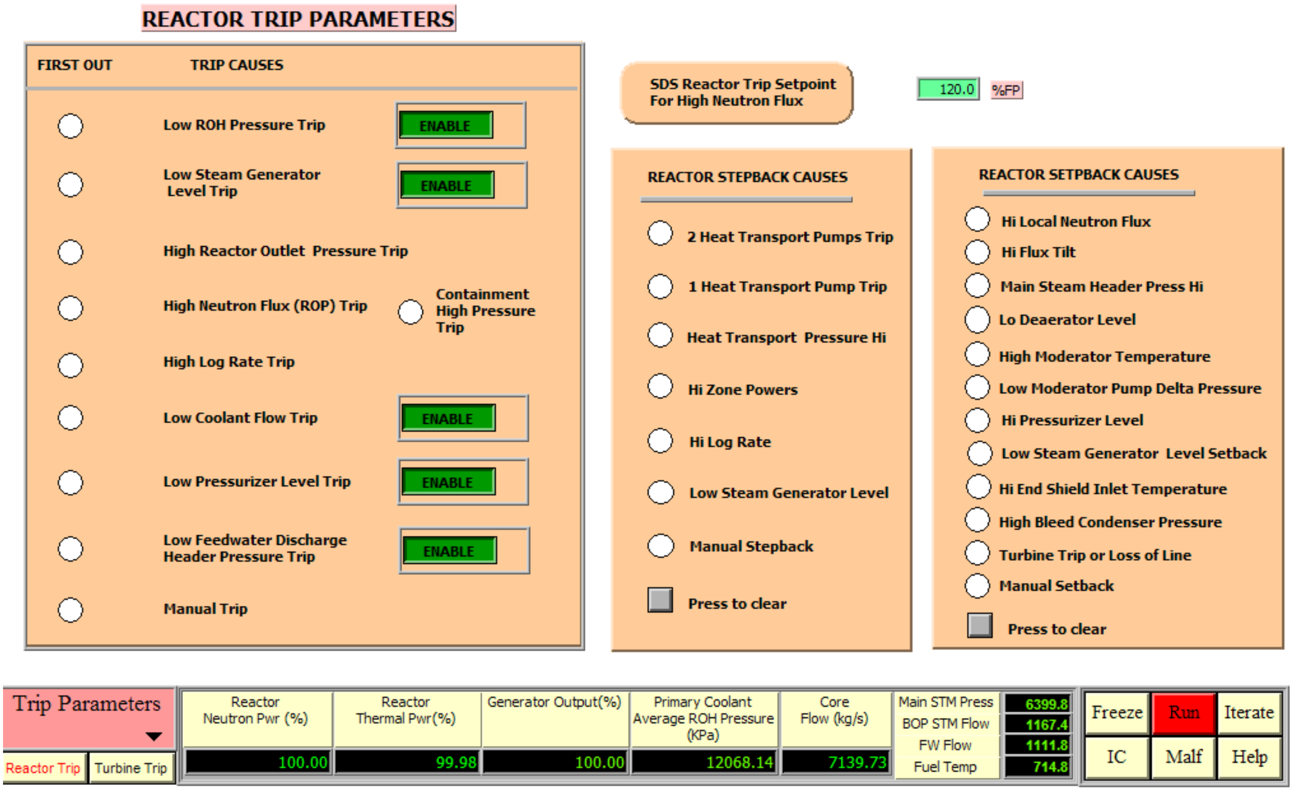


⇒ The rate of change in reactor power is displayed, as result of the control rods movement.

⇒ The following time trends are displayed:

- Reactor power, thermal power and turbine power (%)
- Coolant ΔT (ROH temperature – RIH temperature) (Deg. C)
- Actual and demanded SP (%)
- Flux tilt error (%) (top-bottom, side-side, front-back)
- ZCU's and MCA average position in core (%)
- Core reactivity change (ΔK) - mk

3.9. ACR trip parameters



This screen displays the parameters that cause REACTOR TRIP, REACTOR STEPBACK, and REACTOR SETBACK.

⇒ Reactor stepback is the reduction of reactor power in large step, in response to certain process parameters exceeding alarm limits, as a measure in support of reactor safety.

⇒ Reactor setback is the ramping of reactor power at fixed rate, to the setback target, in response to certain process parameters exceeding alarm limits, as a measure in support of reactor safety.

The TRIP PARAMETERS[4] for REACTOR TRIP are:

Trip Parameter	**Design Setpoint**
ROP (% FP)	123
High Log Rate (%/sec)	10
HTS high pressure	12.75 MPa(g) or 12.45 MPa(g) with 3s delay
HTS low pressure	10.9 MPa(g) ≥ 95% FP, 8.35 MPa(g) at 0% FP, linear in between
Pressurizer low level (m)	6.5 ≥ 95% FP, 0 ≤ 25% FP, linear in between
HTS low coolant flow (kg/s)	85% nominal flow (instrumented channel)
Steam generator feedline low pressure)	5.4 Mpa(g)
Steam generator low level (m)	9.9 m – subject to change.
Containment high pressure	TBD

[4] Note: The trip parameters indicated here could be different than those in current ACR-700 design, as these parameters are subject to changes, as a result of latest safety and design review by AECL.

The causes for REACTOR STEPBACK[5] are:

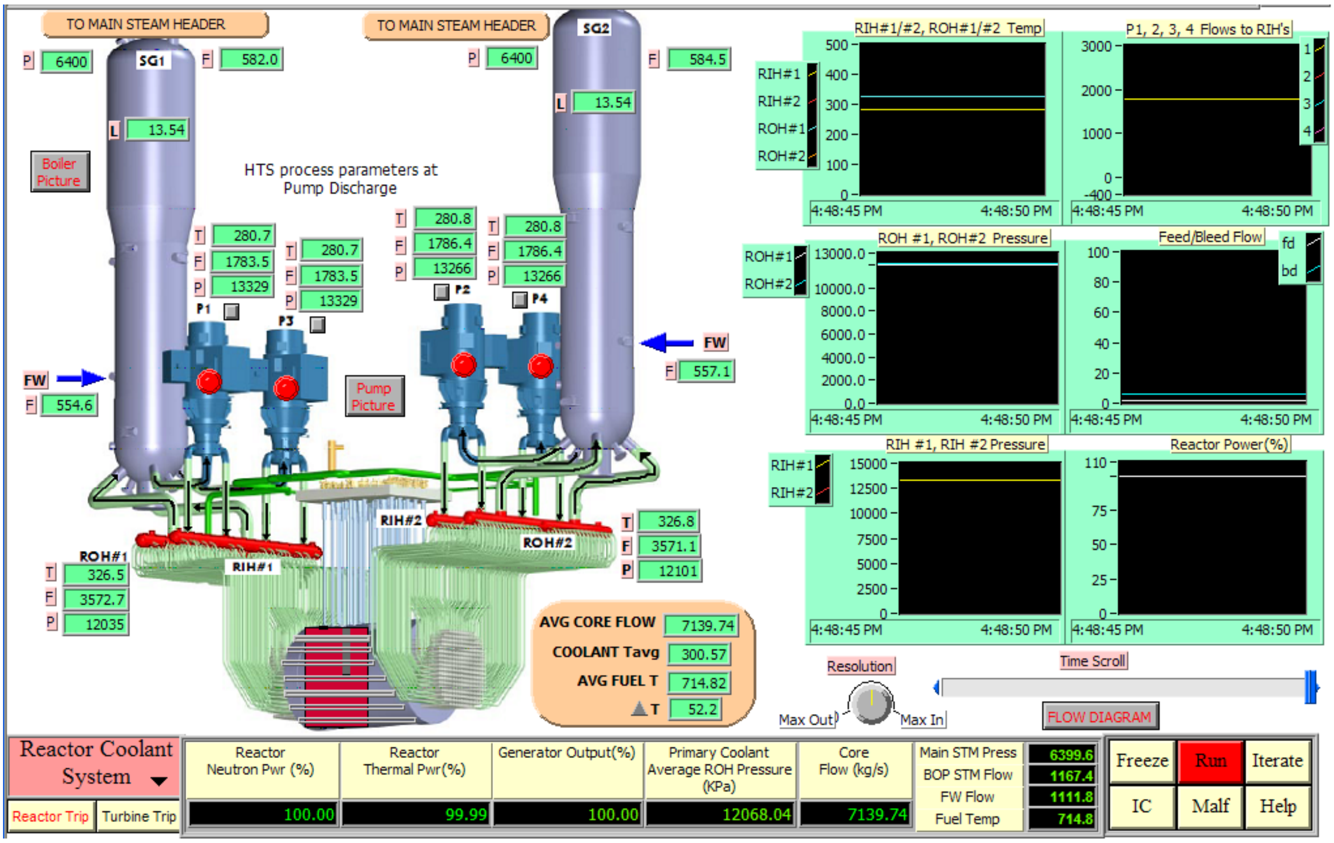
⇒ Two reactor coolant pumps trip (target 2 % FP).
⇒ One reactor coolant pump trip (target 2 % FP).
⇒ Heat transport pressure high (initiated at P > 12.55 MPa; target 2 % FP).
⇒ Hi zone flux (initiated if zone flux is > 115 % of nominal zone flux at full power; target 2 %).
⇒ High log rate (initiated when d(lnP)/dt) > 7 %/s; target 2 % FP).
⇒ Low steam generator level (not implemented as Stepback parameter, but implemented as Setback parameter – see below).
⇒ Manual stepback (initiated by operator; target set by operator).

The causes for REACTOR SETBACK are:

⇒ Hi zonal flux (> 120 %) — setback at 0.1%/s to end point 60 %.
⇒ Hi flux tilt (> 20 %) — setback at 0.1 %/s to end point 20 %.
⇒ Main steam header pressure Hi — setback if > 6800 KPa, at 0.5%/s to end point 10 %.
⇒ Low deaerator level — setback if < 2.075 m, at 0.8%/s to end point 2 %.
⇒ High moderator temperature — setback (moderator temp is not modeled).
⇒ Low moderator pump delta P — setback (moderator pump delta P is not modeled).
⇒ Hi pressurizer level — setback if > 12 M, at 0.1%/s to end point 2 %.
⇒ Low steam generator level — setback if < 10.11M at FP, at 0.8%/s to end point 2%. Note - the low steam generator level setback setpoint is a function of reactor power: SP = 8.8 +1.31*Reactor_Power (normalized)
⇒ Hi end shield inlet temperature — setback (end shield inlet temp is not modeled).
⇒ Hi bleed condenser pressure — setback (not implemented yet - TBD).
⇒ Turbine trip or loss of line — setback, at 0.5%/s to 75%.
⇒ Manual setback (initiated by operator; target set by operator).

[5] Note: The Stepback and Setback Parameters indicated here could be different than those in current ACR-700 design, as these parameters are subject to changes, as a result of latest safety and design review by AECL.

3.10. ACR reactor coolant system



This screen shows a layout of the Heat Transport System (HTS): two steam generators, four heat transport pumps, reactor inlet header (RIH) #1, #2, reactor outlet header (ROH) #1, #2, reactor vessel with coolant feeder piping.

The primary coolant is circulated through four heat transport pumps into the core through the through two reactor inlet headers, known as RIH #1, and RIH #2 respectiively. After entering the RIH#1, #2, the coolant then travels through the fuel channels in the core, and exits the core at two reactor outlet headers, known as ROH #1, #2. The two ROHs are connected to two steam generators respectively.

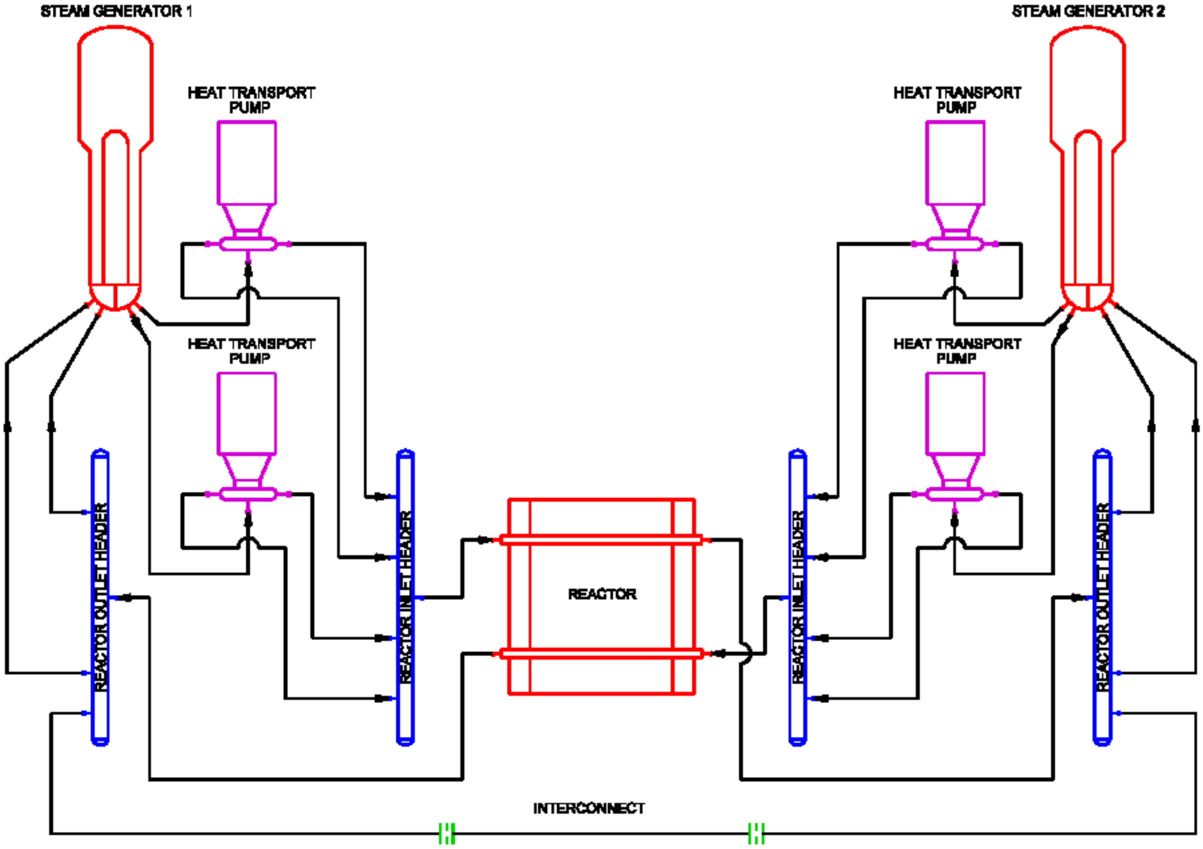
The heated coolant then flows down through the two steam generators where the heat is transferred to the secondary system. The primary coolant is then taken from the bottom of each of the steam generator into the heat transport pumps (two for each steam generator) to repeat the cycle.

The system components and parameters shown on the screen are:

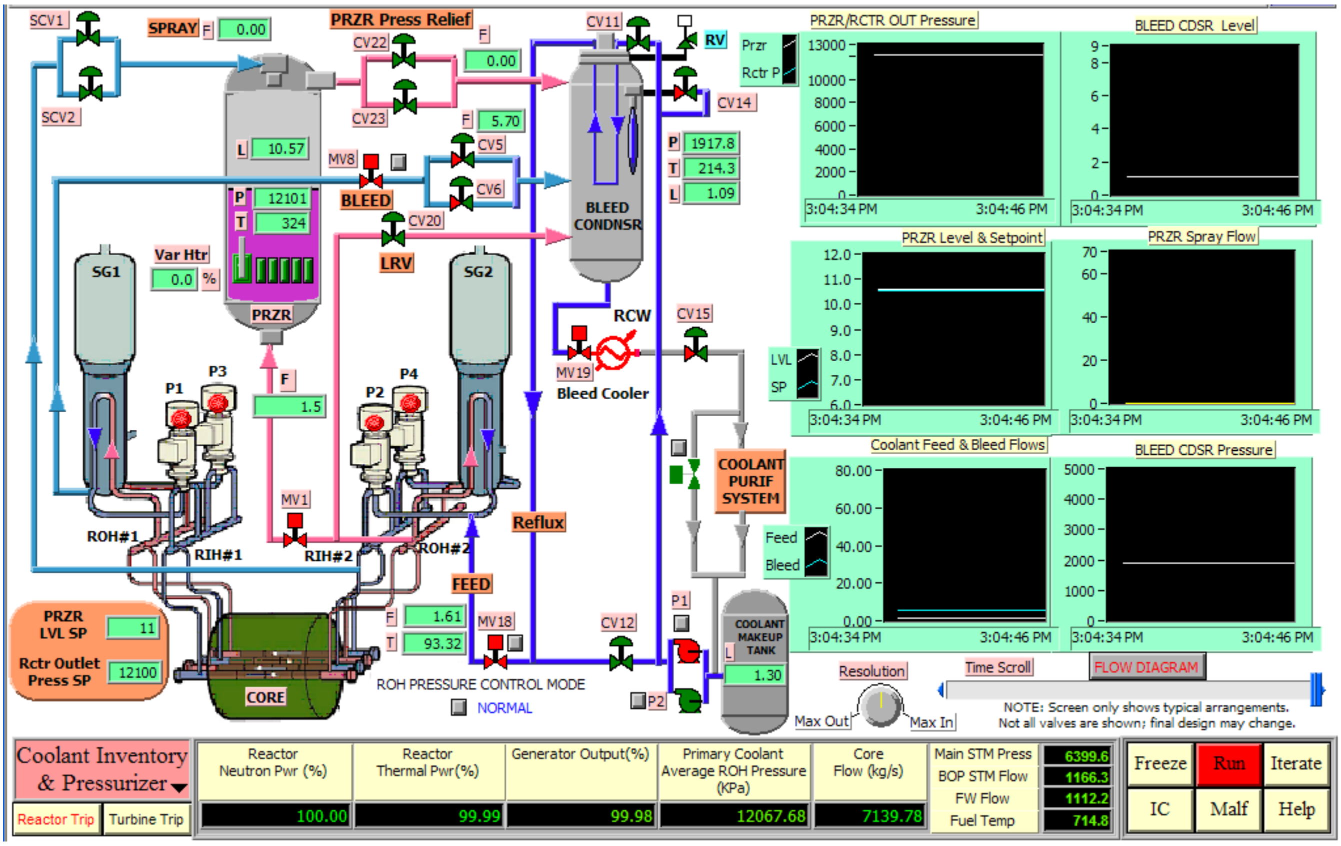
- Average fuel temperature (°C); average coolant temperature (°C); average core flow (kg/s)); Δ T across the core = coolant outlet temperature - coolant inlet temperature.
- Heat transport pump’s discharge flow (kg/s); discharge pressure (KPa); discharge temperature (°C)

- Heat transport pump pop-up control which allows ‘START’, ‘STOP’ and ‘RESET’ operations
- Pressure (kPa), flow (kg/s) and temperature (°C) at the reactor outlet header (ROH) #1, #2.
- For each steam generator (SG) — feedwater flow (kg/s); feedwater level in drum (m); steam drum pressure (KPa); main steam flow from SG to main steam header (kg/s).
- The following time trends are displayed:
  - ⇒ RIH #1, #2; ROH #1, #2 temperatures (°C)
  - ⇒ HTS pumps (P1, P2, P3, P4) discharge flows (Kg/s) to RIHs
  - ⇒ ROH #1, #2 pressures
  - ⇒ The coolant feed flow (kg/s); the coolant bleed flow (kg/s)
  - ⇒ RIH #1, #2 pressures (KPa)
  - ⇒ Reactor power (%)

Also shown on the screen is a pop-up window for the Flow Diagram of the Heat Transport System, as shown below:



3.11. ACR coolant inventory and pressurizer

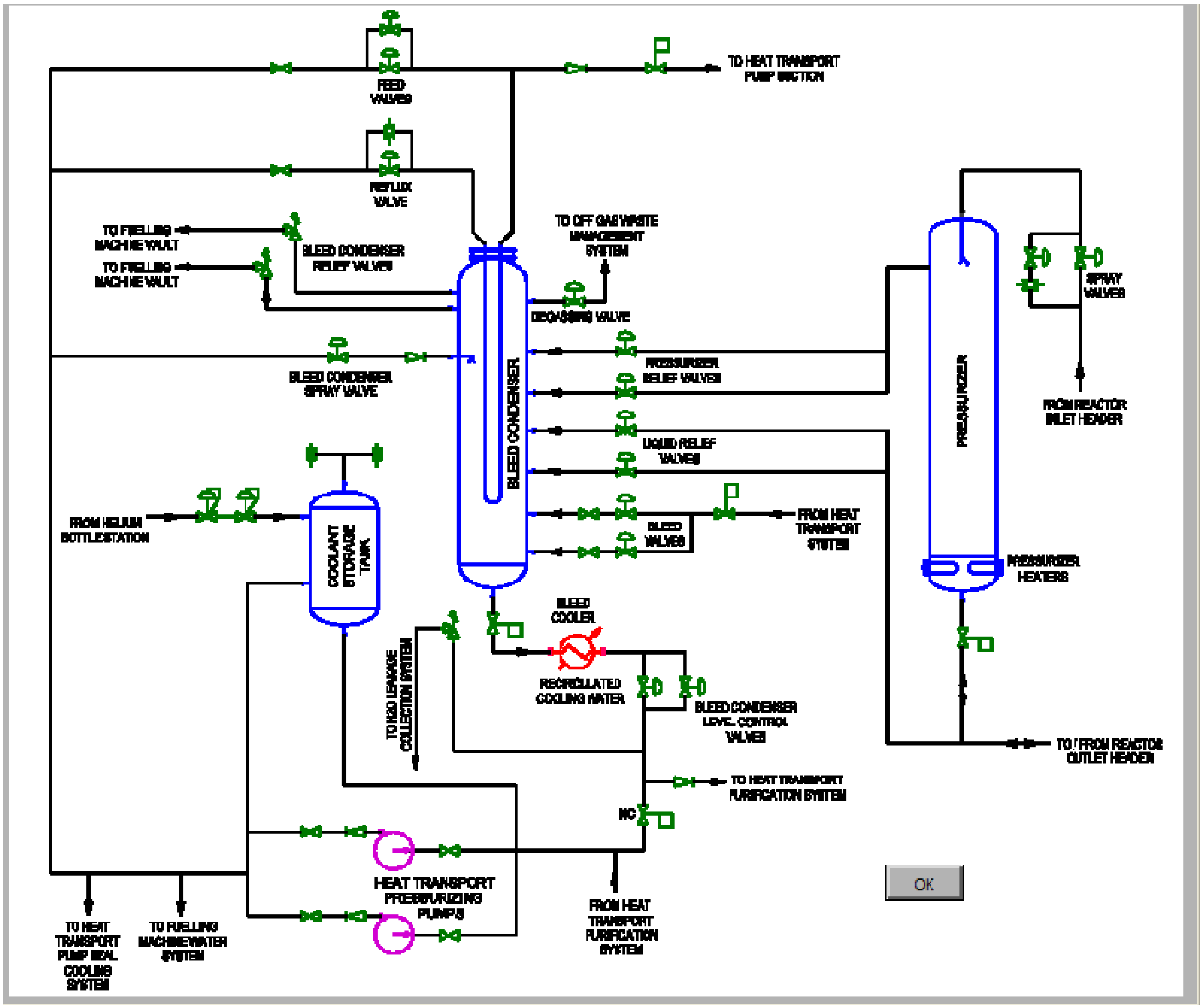


This screen shows the coolant inventory and pressure control system, including the pressurizer, pressurizer pressure relief, coolant feed and bleed circuits, bleed condenser, bleed cooler, coolant purification system and coolant makeup storage tank.

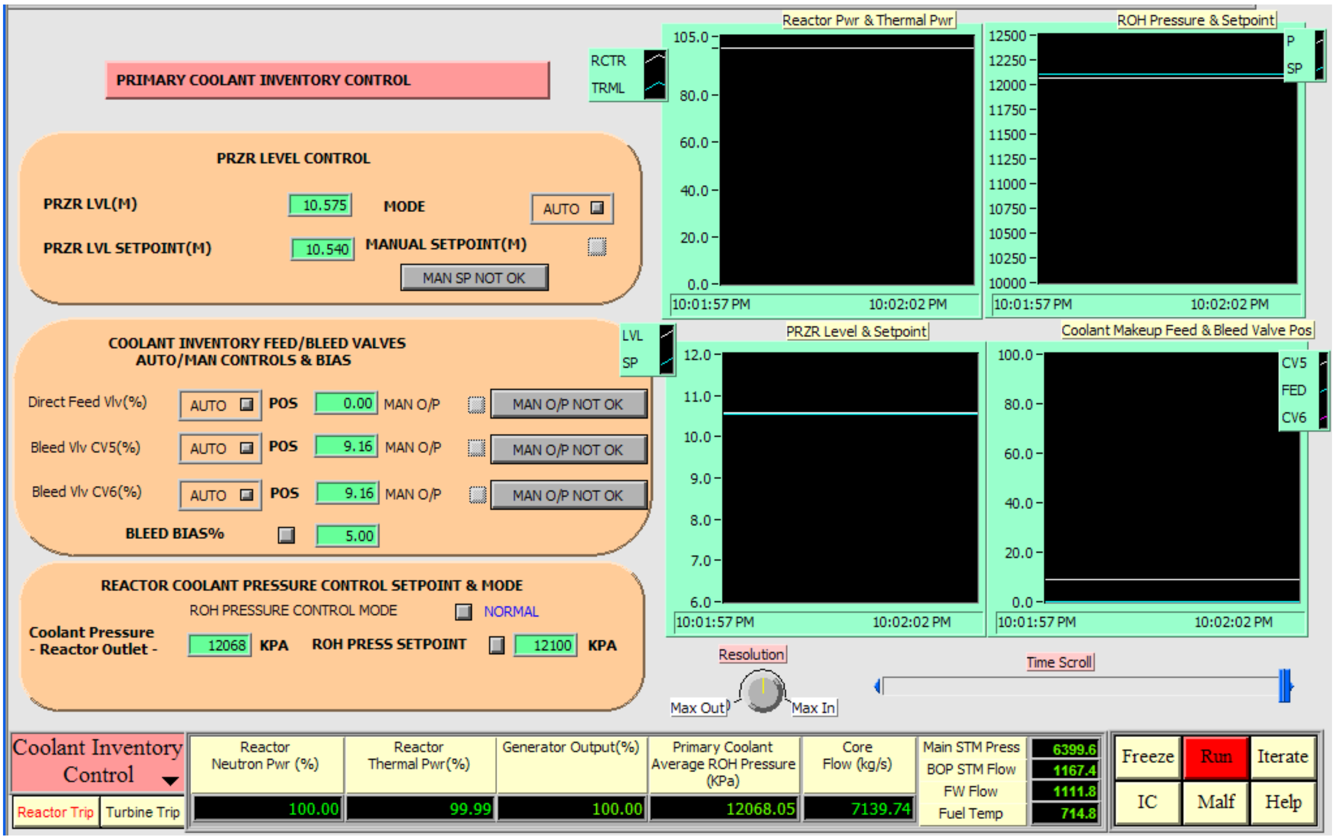
- Starting with the coolant makeup storage tank at the bottom left hand corner, its level is displayed in meters. The tank supplies the flow and suction pressure for the feed pumps P1 and P2: normally one pump is running, the pop-up menu allows START, STOP and RESET operations.
- The flow (kg/sec) and temperature (°C) of the coolant feed flow are displayed. The feed flow then passes through the feed isolation valve MV18 before entering Steam Generator #2, at the suction point of the coolant heat transport pumps P2, P4. Note that some of the coolant at the feed circuit is heated up through the bleed condenser reflux line via CV11. The heated feed coolant then mixes with the coolant from CV12 discharge, before the mixture passes through MV18.
- Coolant flow from the reactor outlet header (ROH) #2 is normally to and from the pressurizer via a short connecting pipe, a negative flow (kg/sec) indicating flow out of the pressurizer into ROH #2. Vice-versa would indicate a positive flow. Pressurizer pressure (kPa), temperature (°C) and level (m) are displayed.
- Pressurizer pressure is maintained by one variable and five on-off heaters which turn ON if the pressure falls, and by pressure relief valves CV22 and CV23 are open if the

pressure is too high. As well, coolant is drawn from connecting lines with the reactor inlet header (RIH) # 2 via control valves SCV1, SCV2 for the purpose of spraying to depressurize the pressurizer.

- There is coolant bleed flow (kg/sec) from the steam generator SG #1 outlet - "cold" coolant suction lines of heat transport pumps P1, P3. The coolant bleed flow, via the bleed control valves CV5, CV6 and isolating MV8, will help maintain coolant inventory in the main coolant circuit, if the inventory becomes too high, as sensed by high pressurizer level.
- The outflow from the bleed condenser goes through bleed cooler, then to the coolant purification system. From it, the coolant goes to the coolant makeup storage tank.
- Parameters displayed for the bleed condenser are: pressure (kPa), temperature (°C) and level (m). The bleed condenser pressure is controlled via CV14 by spraying cold coolant supplied from the feed pumps discharge line. Furthermore, the relief valve is available to relieve excessive high bleed condenser pressure.
- PRESSURIZER LEVEL SETPOINT and REACTOR OUTLET PRESSURE SETPOINT are also shown.
- A ROH PRESSURE CONTROL MODE control pop-up is provided to facilitate the heat transport coolant pressure to be controlled in two modes: NORMAL or SOLID. "SOLID" mode represents the condition that the pressurizer is isolated from the heat transport circuit, meaning that the isolating valve MV1 will be fully closed. Therefore in SOLID mode, there will be much pronounced pressure effects (increase or decrease), with changes in coolant mass inventory. This mode is usually used during plant shutdown or cold startup, when a fast coolant pressure decrease or increase is required. In NORMAL mode, as usually the case in normal plant operation, the isolating valve MV1 is fully open, thus allowing the pressurizer to assist in maintaining coolant pressure and mass inventory at setpoint.
- The following time trends are displayed:
  - ⇒ Pressurizer pressure (KPa); reactor outlet pressure –average of ROH #1, #2 pressures (KPa)
  - ⇒ Bleed condenser level (m); bleed condenser pressure (KPa)
  - ⇒ Pressurizer level (m) and setpoint (m)
  - ⇒ Pressurizer spray flow (kg/s)
  - ⇒ Coolant bleed flow (kg/s); coolant feed flow (kg/s)
- Also shown on the screen is a pop-up window for the Flow Diagram of the Feed & Bleed circuit, as shown below:


TO HEAT TRANSPORT PUMP SUCTION
FEED VALVES
REFLUX VALVE
TO FUELLING MACHINE VAULT
TO FUELLING MACHINE VAULT
BLEED CONDENSER RELIEF VALVES
TO OFF GAS WASTE MANAGEMENT SYSTEM
DEGASSING VALVE
BLEED CONDENSER SPRAY VALVE
PRESSURIZER RELIEF VALVES
LIQUID RELIEF VALVES
BLEED CONDENSER
PRESSURIZER
SPRAY VALVES
FROM REACTOR INLET HEADER
FROM HELIUM BOTTLE STATION
COOLANT STORAGE TANK
BLEED VALVES
FROM HEAT TRANSPORT SYSTEM
PRESSURIZER HEATERS
BLEED COOLER
TO H2O LEAKAGE COLLECTION SYSTEM
RECIRCULATED COOLING WATER
BLEED CONDENSER LEVEL CONTROL VALVES
TO / FROM REACTOR OUTLET HEADER
TO HEAT TRANSPORT PURIFICATION SYSTEM
NC
OK
HEAT TRANSPORT PRESSURIZING PUMPS
FROM HEAT TRANSPORT PURIFICATION SYSTEM
TO HEAT TRANSPORT PUMP SEAL COOLING SYSTEM
TO FUELLING MACHINE WATER SYSTEM

3.12. ACR coolant inventory control



The screen shows the parameters relevant to controlling the inventory in the reactor coolant loop.

⇒ Inventory control is achieved by controlling pressurizer level ⇐

- Pressurizer level is normally under computer control, with the setpoint being ramped as a function of reactor power and the expected shrink and swell resulting from the corresponding temperature changes.
- The screen provides a PRESSURIZER LEVEL CONTROL section, showing the CONTROL mode: AUTO or MANUAL; current PRESSURIZER LEVEL (m); PRESSURIZER LEVEL SETPOINT (m); Level control may be transferred to MANUAL using the mode control pop-up, and the SETPOINT can then be controlled manually using MANUAL SETPOINT pop-up.

⇒ NOTE: in order to control the pressurizer level MANUALLY, one must use the pop-up menu to switch the control mode from AUTO to MANUAL first, then the level setpoint value will be "frozen", as shown in the numeric value display. Observe the display message below the MANUAL SETPOINT button. If it says: "MAN O/P OK", that means the level setpoint can now be controlled by the "MAN" pop-up menu. If it says: "MAN O/P NOT OK", that means the MANUAL setpoint signal from the "MAN" pop-up, and the "frozen" setpoint value do not match. One must then use the "MAN" pop-up menu to enter a value equal to the "frozen" numeric value display, then the message will say "MAN O/P OK".

- The screen also provides a section for COOLANT INVENTORY FEED & BLEED VALVES AUTO/MANUAL CONTROL and BIAS. Using pop-up controls in this section, one can switch the control mode: “AUTO/MANUAL” for DIRECT FEED VALVE, BLEED VALVE CV4 and CV5. The feed and bleed valves are normally in AUTO mode, but may be placed on MANUAL and the valve opening can be controlled manually via pop-up menus.

  ⇒ NOTE: in order to control these valves MANUALLY, one must use the pop-up menu to switch the control mode from AUTO to MANUAL first, then the control signal to the control valve will be “frozen”, as shown in the numeric value display. Observe the display message above the valve control. If it says: “MAN O/P OK”, that means the control valve can now be controlled by the “MAN” pop-up menu. If it says: “MAN O/P NOT OK”, that means the MANUAL control signal from the “MAN” pop-up, and the “frozen” control signal to the control valve do not match. One must then use the “MAN” pop-up menu to enter a value equal to the “frozen” numeric value display, then the message will say “MAN O/P OK”.

  The amount of coolant feed and bleed is controlled about a bias value that is set to provide a steady flow of bleed to the purification system. The amount of flow may be adjusted by changing the value of the BIAS by using the BIAS control pop-up provided.

- The last section on this screen makes provisions for changing the reactor outlet pressure setpoint and the pressure control mode for the heat transport system. The current reactor outlet pressure is shown and the reactor outlet pressure setpoint (kPa) may be controlled manually via the control pop-up provided. As well, a ROH PRESSURE CONTROL MODE control pop-up is provided to facilitate the heat transport coolant pressure to be controlled in two modes: NORMAL or SOLID. “SOLID” mode represents the condition that the pressurizer is isolated from the heat transport circuit, meaning that the isolating valve MV1 will be fully closed. Therefore in SOLID mode, there will be much pronounced pressure effects (increase or decrease), with changes in coolant mass inventory. This mode is usually used during plant shutdown or cold startup, when a fast coolant pressure decrease or increase is required. In NORMAL mode, as usually the case in normal plant operation, the isolating valve MV1 is fully open, thus allowing the pressurizer to assist in maintaining coolant pressure and mass inventory at setpoint.

- The following time trends are displayed:

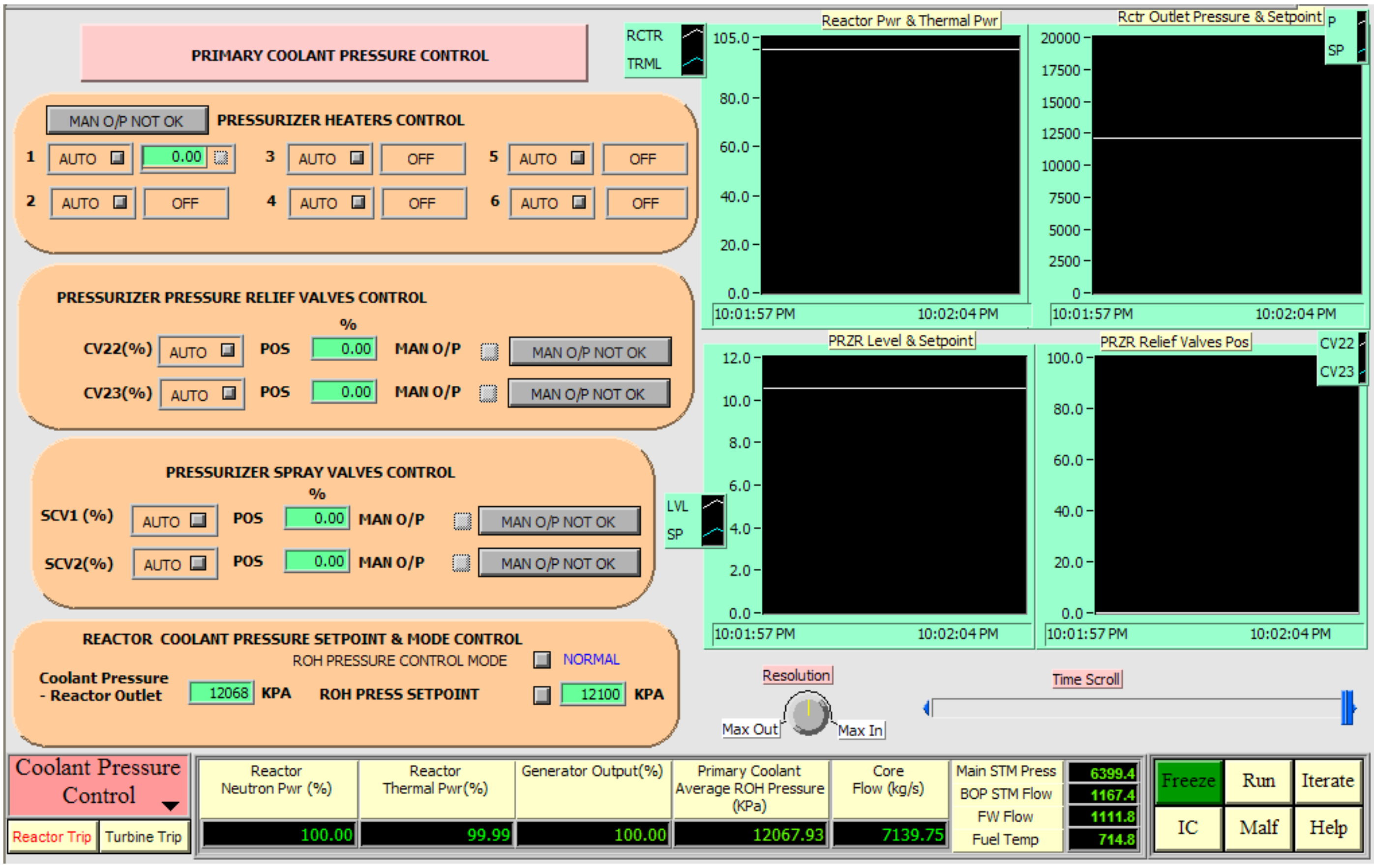
  ⇒ Reactor neutron power (%); reactor thermal power (%)

  ⇒ Reactor outlet header pressure – average of ROH #1, #2 pressure (KPa) & setpoint (KPa)

  ⇒ Pressurizer level (m) & setpoint (m)

  ⇒ Reactor coolant makeup feed valve position (%); reactor coolant bleed valve position (%)

3.13. ACR coolant pressure control



This screen is designed for reactor coolant pressure control:

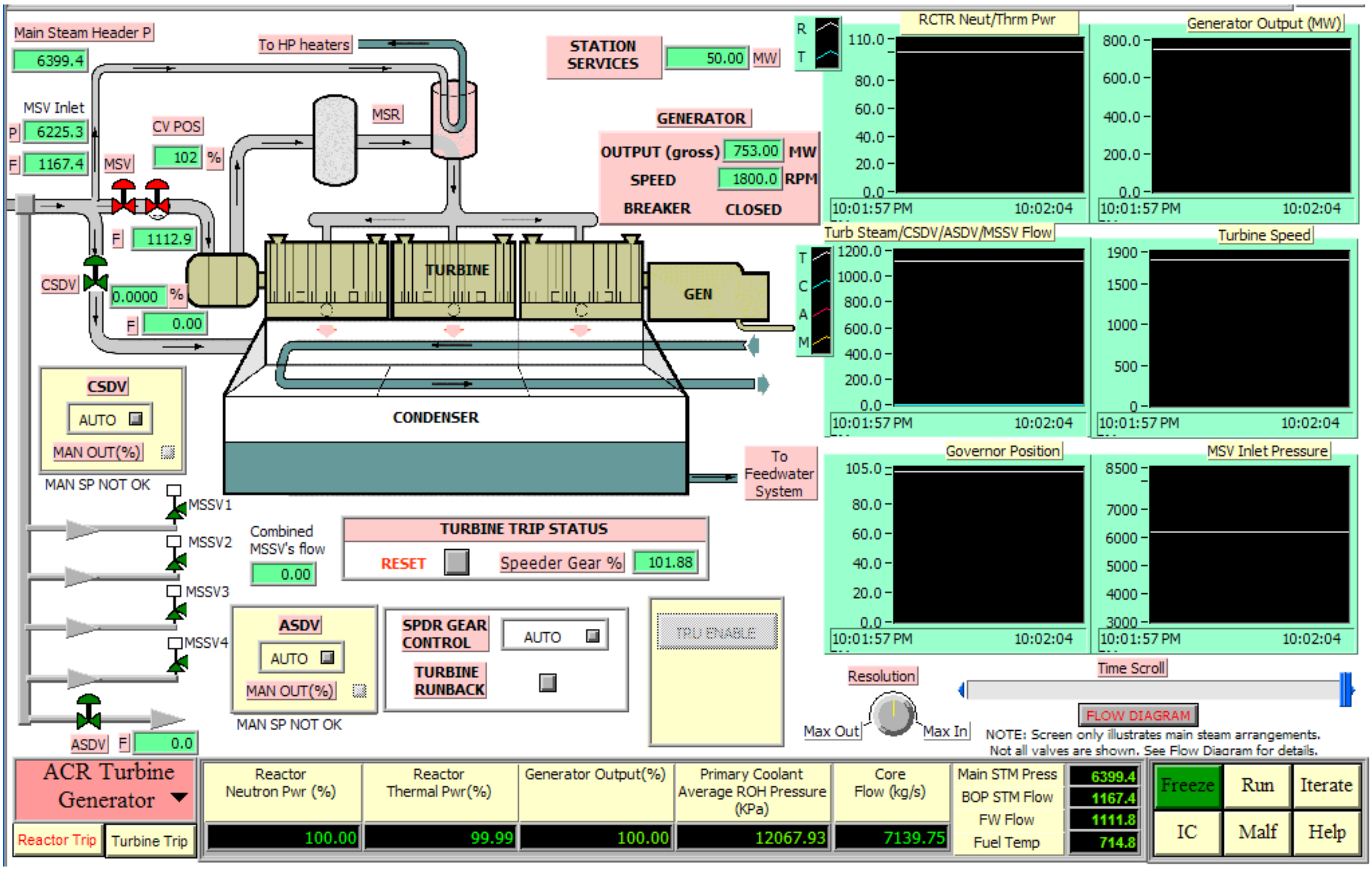
- The first section on screen provides controls for the pressurizer heaters. The six HEATERS are normally in AUTO, with the variable Heater (#1) modulating. The other five heaters are either ON or OFF, and under AUTO control. Via the pop-up menus MANUAL operation can be selected, and each heater may be selected to START, STOP or RESET.

  ⇒ NOTE: in order to control the variable Heater (#1) MANUALLY, one must use the pop-up menu to switch the control mode from AUTO to MANUAL first, then the control signal to the Heater #1 will be “frozen”, as shown in the numeric value display. Observe the display message above the Heater control. If it says: “MAN O/P OK”, that means Heater # 1 can now be controlled by the “MAN” pop-up menu. If it says: “MAN O/P NOT OK”, that means the MANUAL control signal from the “MAN” pop-up, and the “frozen” control signal to the Heater does not match. One must then use the “MAN” pop-up menu to enter a value equal to the “frozen” numeric value display, then the message will say “MAN O/P OK”.

- In the next section, PRESSURIZER RELIEF VALVES CONTROL is via CV22 and CV23. These are normally in AUTO mode, but may be placed on MANUAL and the valve opening can be controlled manually via pop-up menus.

- In the third section, the PRESSURIZER SPRAY VALVES CONTROL is via SCV1 and SCV2. These are normally in AUTO mode, but may be placed on MANUAL and the valve opening can be controlled manually via pop-up menus.
  - ⇒ NOTE: in order to control the pressurizer relief valves or pressurizer spray valves MANUALLY, one must use the pop-up menu to switch the control mode from AUTO to MANUAL first, then the control signal to the control valve will be "frozen", as shown in the numeric value display. Observe the display message above the valve control. If it says: "MAN O/P OK", that means the control valve can now be controlled by the "MAN" pop-up menu. If it says: "MAN O/P NOT OK", that means the MANUAL control signal from the "MAN" pop-up, and the "frozen" control signal to the control valve does not match. One must then use the "MAN" pop-up menu to enter a value equal to the "frozen" numeric value display, then the message will say "MAN O/P OK".
- The last section on this screen makes provisions for changing the reactor outlet pressure setpoint and the pressure control mode for the heat transport system. The current reactor outlet pressure is shown and the reactor outlet pressure setpoint (kPa) may be controlled manually via the control pop-up provided. As well, a ROH PRESSURE CONTROL MODE control pop-up is provided to facilitate the heat transport coolant pressure to be controlled in two modes: NORMAL or SOLID. "SOLID" mode represents the condition that the pressurizer is isolated from the heat transport circuit, meaning that the isolating valve MV1 will be fully closed. Therefore in SOLID mode, there will be much pronounced pressure effects (increase or decrease), with changes in coolant mass inventory. This mode is usually used during plant shutdown or cold startup, when a fast coolant pressure decrease or increase is required. In NORMAL mode, as usually the case in normal plant operation, the isolating valve MV1 is fully open, thus allowing the pressurizer to assist in maintaining coolant pressure and mass inventory at setpoint.
- The following time trends are displayed:
  - ⇒ Reactor neutron power (%); reactor thermal power (%)
  - ⇒ Reactor outlet pressure (KPa) & setpoint (KPa)
  - ⇒ Pressurizer level (m) & setpoint (m)
  - ⇒ Pressurizer relief valve position (%)

3.14. ACR turbine generator



This screen shows the main parameters and controls associated with the turbine and the generator. The parameters displayed are:

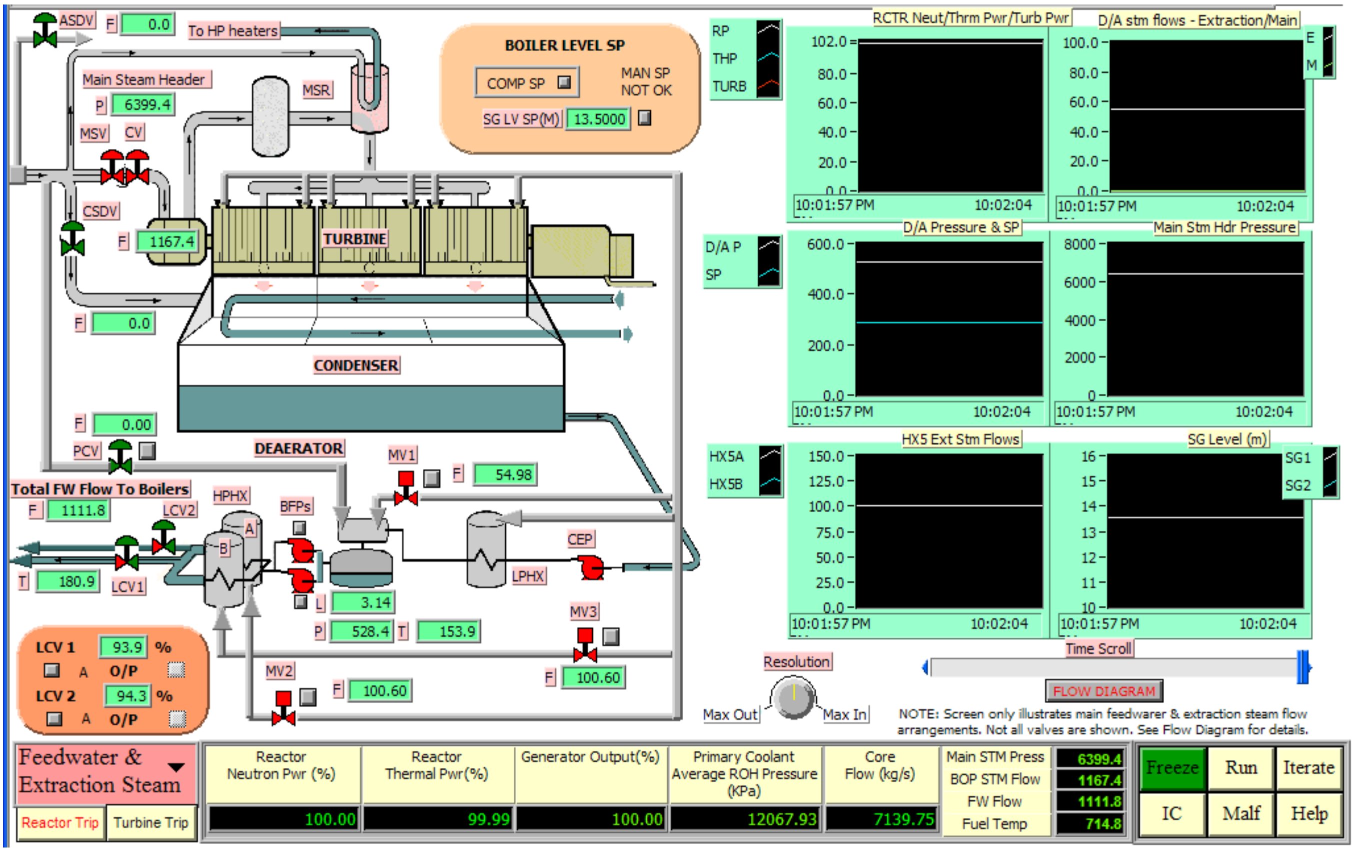
- Main steam pressure (KPa) and main steam flow (Kg/s); main steam stop valve (MSV) status
- Main steam header pressure (KPa)
- Status of main steam safety relief valves (MSSVs)
- Control status (auto/manual), opening (%) and flow (Kg/s) through the steam bypass valves – CONDENSER STEAM DISCHARGE VALVES (CSDV); ATMOSPHERIC STEAM DISCHARGE VALVES (ASDV).
- Steam flow to the turbine (kg/sec)
- Governor control valve position (CV) (% open)
- Generator output (MW); station services (MW)
- Turbine/generator speed of rotation (rpm)
- Generator breaker trip status
- Turbine trip status (tripped or reset)
- Turbine control status — auto (by computer) or manual

- The trend displays are:
  - ⇒ Reactor neutron & thermal power (%)
  - ⇒ Generator output (MW)
  - ⇒ Turbine steam flow (Kg/s); steam BYPASS flow (Kg/s)
  - ⇒ Turbine speed (RPM)
  - ⇒ Turbine governor position (%)
  - ⇒ Main steam stop valve (MSV) inlet pressure (KPa)

  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
- Steam bypass valve ‘AUTO/MANUAL’ control — AUTO select allows transfer to MANUAL control, following which the manual position of the valve may be set.
- Computer or manual control of the speeder gear.
- Turbine runup/speedup controls

3.15. ACR feedwater and extraction steam



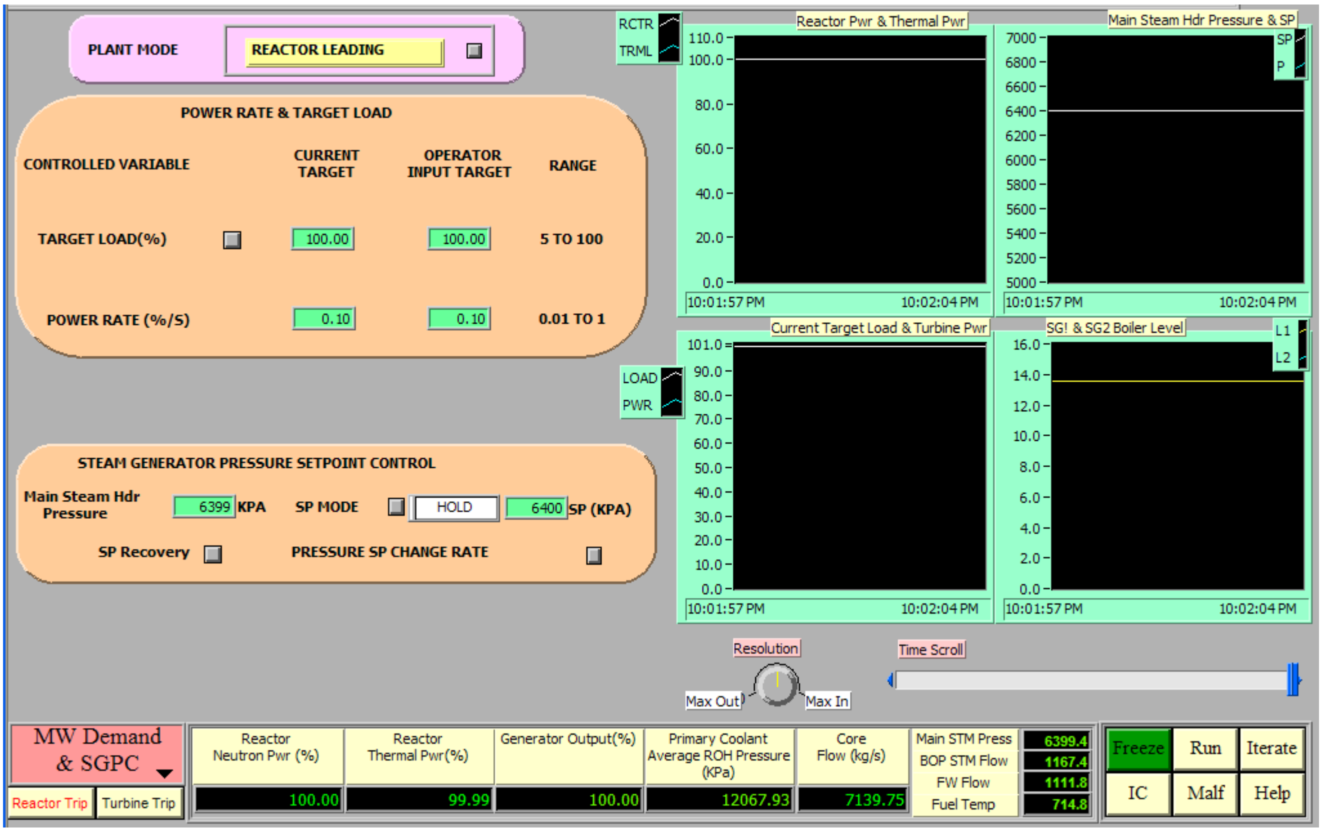
This screen shows the portion of the feedwater system that includes the condenser, low pressure heater, deaerator, the boiler feed pumps, the high pressure heaters and associated valves, with the feedwater going to the steam generator level control valves, after leaving the HP heaters.

The following display parameters and pop-up controls are provided:

- Main steam header pressure (KPa), steam flow through the turbine governor valve and the bypass valve (Kg/s).
- Deaerator level (m) and deaerator pressure (KPa); extraction steam motorized valve status and controls from turbine extraction, as well pressure controller controls for main steam extraction to deaerator. The extraction steam flows (Kg/s) are shown respectively for turbine extraction as well as for main steam extraction to the deaerator.
- Main feedwater pump and auxiliary feedwater pump status with associated pop-up menus for 'ON/OFF' controls.
- HP heater motorized valves MV2 and MV3 and pop-up menus for open and close controls for controlling extraction steam flow to the HP heaters.
- Feedwater flow rate (Kg/s) at SG level control valve (LCV1 & LCV2) outlet and feedwater temperature (°C).
- Pop-up controls for "auto/manual" for SG level control valves LCV1 & LCV2

- Pop-up controls for changing SG level setpoint control from “computer SP” to “manual SP”, or vice versa.
  - ⇒ NOTE: in order to change the SG setpoint control from “computer SP” to “manual SP”, one must use the pop-up menu to switch the control mode from COMPUTER SP to MANUAL SP first, then the “steam generator level SP” value will be “frozen”, as shown in the numeric value display. Observe the display message next to SP control status. If it says: “MAN SP OK”, that means the SG level SP can now be controlled by the “MAN SP” pop-up menu. If it says: “MAN SP NOT OK”, that means the MANUAL SP value from the “MAN SP” pop-up, and the “frozen” SP value (as displayed) do not match. One must then use the “MAN SP” pop-up menu to enter a value equal to the “frozen” numeric value display, then the message will say “MAN SP OK”.
- The following trends are displayed:
  - ⇒ Reactor neutron power (%); reactor thermal power (%); turbine power (%)
  - ⇒ Steam flow to deaerator (Kg/s)
  - ⇒ Deaerator pressure (KPa) & setpoint (KPa)
  - ⇒ Main steam header pressure (KPa)
  - ⇒ High pressure heaters HX5A, HX5B extraction steam flows (Kg/s)
  - ⇒ Steam generator level (m)

3.16. ACR MW demand setpoint (SP) and steam generator pressure control (SGPC)

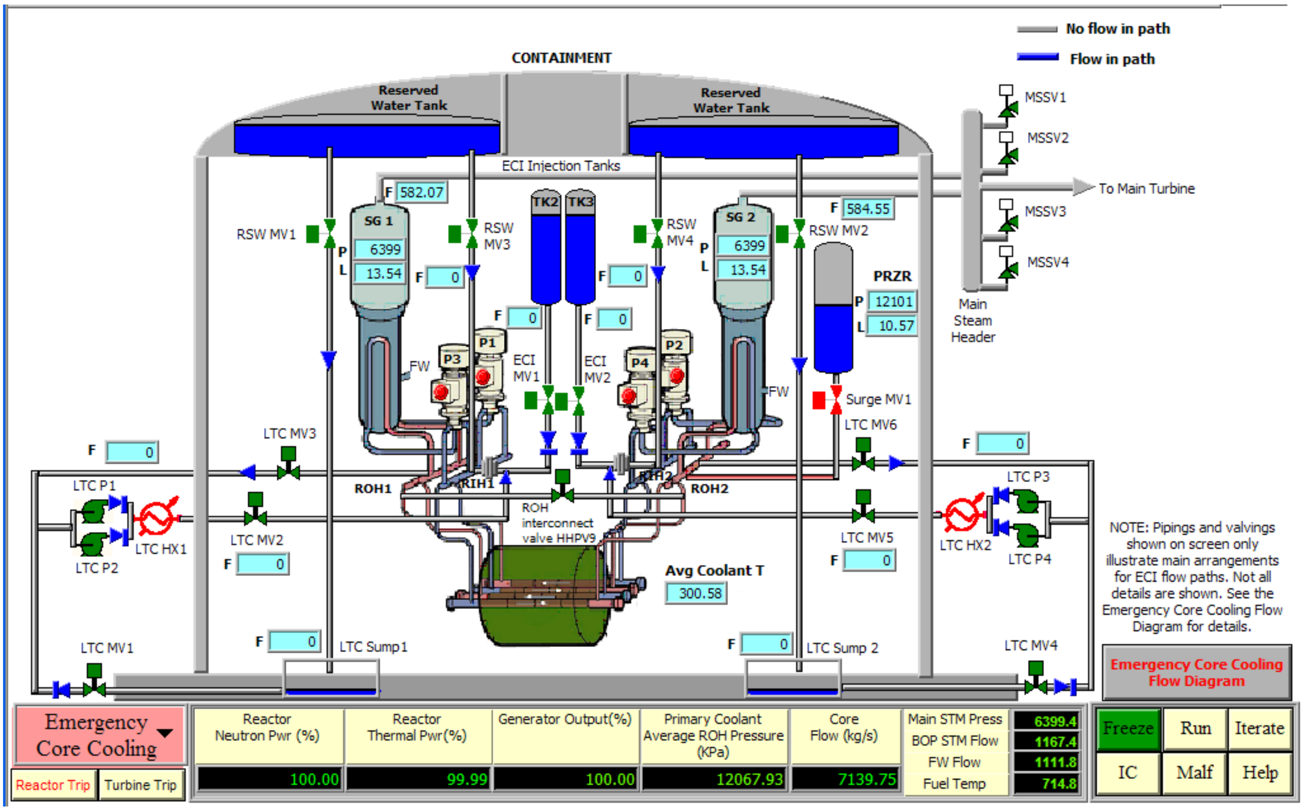


- This screen permits control of station load setpoint and its rate of change while under "TURBINE LEADING" control mode. Control of the main steam header pressure is also through this screen, but this is not usually changed under normal operating conditions.
- ACR OVERALL UNIT CONTROL MODE can be changed from "REACTOR LEADING" to "TURBINE LEADING".
- TARGET LOAD — on selection station load (%) and rate of change (%/sec) can be specified; change becomes effective when 'accept' is selected.
  - ⇒ The OPERATOR INPUT TARGET is the desired setpoint inserted by the operator; the CURRENT TARGET will be changed at a TARGET and 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.
- STEAM GENERATOR PRESSURE SETPOINT CONTROL — alters the setpoint of the steam generator pressure controller, which is rarely done during power operation. *Caution must be exercised when using this feature on the simulator*. However, this feature can be used for educational study of ACR plant responses under different secondary pressure conditions.
  - ⇒ To change SG pressure setpoint, first use the "SP Mode" pop-up to change the SP mode from "HOLD" to "INCREASE" or "DECREASE", depending on new

pressure setpoint target. After that, use the “pressure SP change rate” pop-up to enter new values for “pressure SP TARGET” (in MPa), and the “pressure SP change rate” (in MPa /minute). Observe that the SP value changes immediately, after the new SP target and rate are “accepted”. As well, the main steam header pressure shown in the display will be changed. At any time, if one wants to return the original pressure setpoint, just press the button “SP recovery” once. It can observe that the pressure SP will recover to 6400 KPa, and the main steam header pressure will follow accordingly.

- The following trends are provided:
  - ⇒ Reactor neutron power (%); reactor thermal power (%)
  - ⇒ Main steam header pressure (KPa) & setpoint (KPa)
  - ⇒ Current target load (%),and turbine power (%)
  - ⇒ Steam generator 1 & 2 level (m)

3.17. ACR passive core cooling



This screen shows the passive core cooling system in an advanced ACR. The passive emergency core cooling system requires no operator actions to mitigate design basis events like loss of coolant accident (LOCA). The system relies on natural forces such as gravity, natural circulation, and compressed gas. Only few valves are used in the system, supported by reliable power sources.

The ECC function is accomplished by two sub-systems:

- The emergency coolant injection (ECI) System, for high-pressure coolant injection after a LOCA.
- The Long Term Cooling (LTC) system for long term recirculation/recovery after a LOCA. The LTC system is also used for long term cooling of the reactor after shutdown following other accidents and transients.

Following a loss-of-coolant accident, the reactor shutdown and emergency core cooling systems acting together must, as a design target, prevent excessive fuel damage.

In the event of a major break in the heat transport system, the water escapes through the break, depressurizing the system (the *blowdown* phase). The reactor is tripped automatically. The combination of increase in pressure differential across the fuel sheath caused by the gaseous fission products and the increase in sheath temperature is a factor affecting the sheath failure threshold during blowdown. If the threshold is exceeded, the sheath can swell and could result in sheath rupture. However, during blowdown the sheath temperature increase is limited and excessive sheath failures are prevented. The need to remove residual heat in the fuel at the end of blowdown, and decay heat produced thereafter, leads to the requirement for an emergency core cooling system (ECC).

The emergency core cooling system is designed to supply emergency coolant to the reactor in two stages. During the high pressure stage, water is injected into the reactor core via the Emergency Core Injection (ECI) system on detecting a LOCA.

The system consists of two ECI water accumulators TK2, TK3; each accumulator is pressurized during normal reactor operation by compressed nitrogen gas. A floating ball seal is located in each of the ECI accumulators. At the end of injection when the water level nears the bottom of the accumulator, pressure forces the ball against a seat at the bottom of the accumulator, creating a seal and terminating injection. This provides a passive means of defense against injection of nitrogen gas into the Heat Transport System (HTS). Each of the two ECI accumulators is connected to one of the two heat transport system reactor inlet headers (RIH's) by an injection line via ECI MV1, and MV2 respectively. One-way rupture discs in the injection lines isolate the ECI system from the Heat Transport System (HTS). The one-way rupture discs withstand the high differential pressure that is normally present in the reverse direction (ECI system to HTS).

During normal operation the ECI system is poised to detect any LOCA that results in a depletion of HTS inventory to such an extent that heat removal by normal means is not assured. When the HTS pressure drops below the rupture pressure of the one-way rupture discs, the rupture discs burst, thereby enabling emergency coolant injection to the RIH. Water is injected into the heat transport system from the pressurized ECI accumulators. Valves (HHPV9) on the ECI interconnect line between the reactor outlet headers (ROH) open upon detection of a LOCA to assist in establishing a sustainable cooling flow path. To enhance the effectiveness of the high pressure injection of water into the heat transport system, the main steam safety valves (MSSV1 to 4) open on detection of a LOCA to provide a rapid cool down of the steam generators and depressurization of the heat transport system.

High pressure injection continues until the ECI accumulators are nearly empty, at which time the Long Term Cooling (LTC) system begins operation in long-term recovery mode. At this time the ECI injection valves (ECI MV1, MV2) close to ensure there is no injection of nitrogen gas into the HTS, this is backed up by floating ball seals inside the ECI accumulators.

For a LOCA, the LTC system is initiated during the operation of the ECI system. On detecting a LOCA, water is automatically introduced into the containment sumps and the LTC pumps start automatically. When the water accumulators are nearly empty, the ECI accumulator isolation valves close. The recovery stage begins by pumping water from the sumps into the HTS via the LTC heat exchangers. The LTC delivers flow to the reactor inlet headers, thereby utilizing the cooling path already established by the high pressure ECI system. The LTC system is also used for long term cooling of the reactor after shutdown following other accidents and transients.

The followings provide a qualitative description of the ECC event sequence. That is, the event sequence describes the behavior that would be expected, should a LOCA occur:

- A large break is postulated to occur in a large diameter pipe of the heat transport system (HTS), discharging coolant into containment.

- The pressure, temperature and humidity of the containment atmosphere increase.
- The HTS depressurization causes coolant voiding in the core and a decrease in reactivity.
- The reactor shuts down on a process trip (e.g., low Heat Transport System pressure, low Heat Transport System flow) depending on break size and initial reactor power.
- Containment isolation is automatically initiated on a high reactor building pressure signal. The high reactor building pressure signal also conditions ECI signal.
- The heat transport system loses inventory and depressurizes at a rate depending on the break size and location.
- Following reactor trip, the turbine runs back. The condenser steam dump valves (CSDVs) open to by-pass steam to the condenser. The atmospheric steam discharge valves (ASDVs) open and close to maintain system pressure.
- The main feedwater system feeds the steam generators from the condenser hotwell throughout the event.
- The HTS flow decreases faster in the core pass downstream of the break. If the break is large enough, the flow will reverse in that pass. For some break sizes, the flow momentarily falls very low as the break upstream of the core pass balances the pumps. Some channels may become steam-filled and others may experience stratified two-phase flow, exposing some fuel elements to steam cooling. Fuel temperatures rise. A rise in fuel temperatures increases the internal fuel element gas pressures, whereas a rise in sheath temperatures reduces the sheath strength. Increased internal fuel element gas pressure along with the decreased coolant pressure increases fuel sheath stresses. If the fuel sheath temperature becomes high enough, sheath failure can occur.
- The pressurizer discharges its inventory into the HTS. The decreasing pressurizer level causes the light water bleed valves to close, and feed valves to open up, adding light water makeup to the HTS.
- Following reactor trip, the average fuel temperature decreases as the heat generation rate decreases and the temperature profile in the fuel pin flattens out. The sheath temperature increases depending on the heat transfer from the sheath to the coolant.
- When the HTS pressure falls below a specified setpoint, the ECI signal, which is conditioned by the high reactor building pressure signal, is generated. This signal results in the following events:
  - ⇒ Emergency Coolant Injection (ECI) System is initiated by the ECI signal. The one-way rupture discs burst open at a pressure differential of 0.52 MPa. The ECC piping downstream of one-way rupture discs is pressurized to the heat transport system pressure. Thus, the ECI injection flow will begin when the pressure in the heat transport system is about 0.52 MPa less than the ECI injection pressure from the ECC accumulators. ECI injection continues until the associated ECI accumulator is nearly empty.
  - ⇒Valves on the ECI interconnect line between the reactor outlet headers open up on the ECI signal to assist in establishing a cooling flow path.
  - ⇒ Steam generator crash cooldown is initiated 30 seconds after the ECI signal through the automatic opening of the main steam safety valves (MSSV's). This assists in ECC injection by further depressurizing the HTS.
  - ⇒ On the ECI signal, water is automatically introduced into the containment sumps from the Reserve Water Tank (RWT) and the LTC pumps start automatically. The long-term cooling (LTC) pumps start automatically on a high reactor-building sump level signal. When the ECI accumulators are nearly empty, the ECI accumulator isolation valves close and the LTC stage begins by pumping water from the reactor-building sump. LTC

delivers flow to the reactor inlet headers, thereby utilizing the cooling flow path already established by the ECI system.

⇒ On the ECI signal, the RWS injection valves to the reactor inlet headers are open. When the HTS depressurize, the water from the RWT could directly be injected to the reactor inlet headers.

- Soon after ECC injection and steam generator crash cooldown begin, emergency coolant water begins to refill the core pass. As a result, fuel and sheath temperatures start to decrease.
- The ECI refills both core passes and a quasi-steady-state flow pattern is established.
- Long-term cooling is maintained by the flow of ECCS coolant through the circuit, with decay heat removal by the ECCS heat exchangers and through the break.

For details of the ECC Flow Diagram, press the button "Emergency Core Cooling Flow Diagram" shown at the bottom right corner of the screen.

4. ACR BASIC OPERATIONS & TRANSIENT RECOVERY

4.1. Plant load maneuvering — reactor lead

POWER MANEUVER: 10 % power reduction and return to full power

(1) Initialize the simulator to 100%FP.

(2) Select “ACR reactor power control” screen.

(3) Run the simulator by pressing the “run” button.

(4) Select the plant mode to be “REACTOR LEAD”.

(5) Record in Table II the following parameters in the “full power” column, before power maneuvering.

TABLE II. PLANT LOAD MANEUVERING – REACTOR LOAD

Parameter	Unit	(1) Full Power ____%	(2) 90 % just reached	(3) 90 % stabilized	(4) return to 100 % stabilized	Comments
Reactor Neutron Power	%					
Reactor Thermal Power	%					
Reactor Power SP	%					
Actual Setpoint	%					
Demanded Power Setpoint	%					
Demanded Rate Setpoint	%/sec					
Current Reactor Power	%					
Power Error	%					
Average Coolant Temperature (from ACR Reactor Coolant Screen)	°C					
Average Core Flow	Kg/s					
Average Fuel Temp	°C					

Coolant Delta T = ROH temp – RIH temp	°C					
Reactor Outlet Header #1 Pressure	KPa					
Reactor Outlet Header #2 Pressure	KPa					
Reactor Inlet Header #1 Pressure	KPa					
Reactor Inlet Header #2 Pressure	KPa					
Pressurizer Level	M					
Flux Tilt: Top – bottom Side – side Front - back	%					
ZCU - 8 zones group control rods average position	%					
ZCU - 10 zones group control rods average position	%					
MCA – Bank #1 rods average position	%					
MCA – Bank #2 rods average position	%					
Gadolinium Concentration	ppm					
Pressurizer Temperature	°C					
Coolant Feed Flow	Kg/s					
Coolant Bleed Flow	Kg/s					
Main Steam Pressure	KPa					
Total steam flow from steam generators	Kg/s					
Total Feedwater Flow	Kg/s					

(5) Reduce power using “reactor power setpoint” pop-up.

⇒ Press the “reactor power setpoint” pop-up button at the bottom left corner of the screen
⇒ Enter “reactor power SP target” = 90 %; enter “power rate” = 0.3 %/sec, and press “accept”
⇒ Observe parameter changes during transient and record comments
⇒ Freeze simulator as soon as reactor neutron power just reaches 90% and record parameter values in the column (2) for “90%” power just reached.
⇒ Unfreeze simulator and let parameters stabilize, record parameter values in the column (3) for “90%” power stabilized.

(6) Explain the responses for -

⇒ Primary coolant pressure
⇒ Coolant feed and bleed flow changes
⇒ Coolant temperature
⇒ Steam generators pressure
⇒ Feedwater flow and steam flow
⇒ ZCU and MCA rods movement
⇒ Gadolinium load changes
⇒ Flux tilt

(7) Return reactor power to 100% FP at 0.3 %/sec by using the “reactor power setpoint” pop-up

(8) When reactor power has returned to 100 % and the parameters have stabilized, unfreeze, record parameter values in the column (4) of Table II “return to 100 % stabilized”

(9) Note any major difference in parameter values between column (4) and column (1). Can you explain why the differences in parameter values, if any?

4.2. Plant load maneuvering — turbine lead

POWER MANEUVER: 10% power reduction and return to full power

- Initialize simulator to 100% full power
- Verify that all parameters are consistent with full power operation.
- Select the “ACR MW demand SP & SGPC” page

⇒ Change the scale on the “reactor PWR & thermal PWR” and “current target load & turbine PWR” graphs to be between 80 and 110 percent; the “main steam Hdr pressure & SP” to 5000 and 7000 KPa, “SG level” to 10 and 15 meters, and set “resolution” to “max out” .

⇒ Record down the following parameters in the “full power” column (1) of Table III, before power maneuvering.

⇒ Go to "reactor power control screen", and record down the following parameters in the "full power" column (1), before power maneuvering.

- Go back to "MW demand setpoint & SGPC" screen
- Reduce unit power in the 'turbine lead' mode, i.e.

⇒ Select the plant mode to be "turbine lead"

⇒ Select 'TARGET LOAD (%)' pop-up menu

⇒ In pop-up menu lower 'target' to 90.00% at a 'rate' of 0.3 %/sec

⇒ 'accept' and 'return'

- Observe the response of the displayed parameters until the transients in reactor power and steam pressure are completed without freezing the simulator and/or stopping labview.
- When the parameters have stabilized, freeze the simulator and record the parameter values in column (2) 90 % stabilized of Table III. Go to "reactor power control" screen, and record parameter values in column (2) of Table IV.

TABLE III. PLANT LOAD MANEUVERING – TURBINE LEAD (1)

Parameter	Unit	(1) Full Power ____%	(2) 90 % stabilized	(3) return to 100 % stabilized	Comments
Reactor Neutron Power	%				
Reactor Thermal Power	%				
Main Steam Header Pressure	KPa				
Main Steam Pressure Setpoint	KPa				
Current Target Load	%				
Turbine Power	%				
SG 1 Level	m				
SG2 Level	m				

TABLE IV. PLANT LOAD MANEUVERING – TURBINE LEAD (2)

Parameter	Unit	(1) Full Power ____%	(2) 90 % stabilized	(4) return to 100 % stabilized	Comments
Reactor Neutron Power	%				
Reactor Thermal Power	%				
Turbine Power	%				
Average Coolant Temperature (from ACR Reactor Coolant Screen)	°C				
Average Core Flow	Kg/s				
Average Fuel Temp	°C				
Coolant Delta T = ROH temp – RIH temp	°C				
Reactor Outlet Header #1 Pressure	KPa				
Reactor Outlet Header #2 Pressure	KPa				
Reactor Inlet Header #1 Pressure	KPa				
Reactor Inlet Header #2 Pressure	KPa				
Pressurizer Level	M				
Flux Tilt: Top – bottom Side – side Front - back	%				
ZCU - 8 zones group control rods average position	%				
ZCU - 10 zones group control rods average position	%				
MCA – Bank #1 rods average position	%				

MCA – Bank #2 rods average position	%				
Gadolinium Concentration	ppm				
Pressurizer Temperature	°C				
Coolant Feed Flow	Kg/s				
Coolant Bleed Flow	Kg/s				
Main Steam Pressure	KPa				
Total steam flow from steam generators	Kg/s				
Total Feedwater Flow	Kg/s				

- Explain the main changes.
  - ⇒ Why main steam header pressure rises first then drops back to the steam pressure setpoint value, although the steam pressure setpoint value is unchanged?
  - ⇒ Why steam generator's level drops initially and then recovers?
  - ⇒ Turbine power (%) lags target load (%), but follows it nicely. However, the reactor neutron & thermal power overshoot below 90 % power, but recover later. But their values drift up and down for sometime before they stabilize. Recall previous power maneuvering in "reactor leading" mode, the reactor neutron & thermal power decrease orderly and do not drift as much during power changes. Can you explain why this occurs in this power maneuvering in turbine lead mode? What is the difference in the way reactor power is controlled in "reactor lead" mode, versus "turbine lead" mode ?
- Continuing the above operation, raise "UNIT POWER" to 100% at a rate of 0.3%FP/sec.
- When reactor power has returned to 100 %, and the parameters have stabilized, freeze the simulator and record the parameter values in column (3) 100 % stabilized of Table III. Go to "reactor power control" screen, and record parameters in column (3) of Table IV.
- Note any major difference in parameter values between column (3) and column (1). Can you explain why the differences in parameter values, if any?

4.3. Power level reduction to 0% FP

- Initialize the simulator to 100% FP, using "reactor lead" mode, reduce reactor power in 25% steps at 0.5%/sec

- During power changes, go to the following screens and record the parameters in Table V.

  ⇒ Control rods & SD rods
  ⇒ Reactor power control
  ⇒ Reactor coolant system
  ⇒ ACR coolant inventory & pressurizer
  ⇒ Turbine generator
  ⇒ Feedwater & extraction steam
  ⇒ 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.
  ⇒ Note any alarms encountered during the power changes. In case reactor setback, or stepback occurs, the trip parameters screen will indicate the causes for such alarms.

TABLE V. POWER LEVEL REDUCTION TO 0% FP

Parameter	Unit	100%	75%	50%	25%	0%	Comments
Reactor Power	%						
Turbine-Generator Power	%						
ZCU - 8 zones group rods average position	%						
ZCU – 10 zones group rods average position	%						
MCA Bank #1 Rods average position	%						
MCA Bank #2 Rods average position	%						
Maximum Flux Tilt error during power changes	%						*
Gadolinium load	ppm						*
Maximum reactivity change (ΔK) during power changes	mK						*
ROH # 1 pressure	KPa						
ROH # 2 pressure	KPa						
ROH # 1 temp	°C						
ROH # 2 temp	°C						
RIH # 1 pressure	KPa						
RIH # 2 pressure	KPa						
RIH # 1 temp	°C						
RIH # 2 temp	°C						
Average coolant temp	°C						
Average core flow	Kg/s						
Average fuel temp	°C						
Pressurizer Level	m						
Coolant feed flow	Kg/s						
Coolant bleed flow	Kg/s						
SG1 Pressure	KPa						
SG2 Pressure	KPa						
SG1 Level	m						
SG2 Level	m						
Main Steam Flow	kg/s						
Feedwater Flow	kg/s						

*** NOTE**: it may be necessary to record these values from the relevant trend in "reactor power control" screen, or in the TRENDS screen.

4.4. Turbine trip and recovery

Turbine trip transient occurs as a result of either a load rejection or turbine malfunction. On turbine trip -

⇒ The turbine main steam stop valves and governor valves will close, immediately shutting off steam flow to the turbine.

⇒ As well, the generator breaker will trip open, causing the nominal MW power produced by the generator to drop to 0 MW almost instantly.

⇒ As a result of losing MW from the generator, there is a large mismatch between the reactor thermal power and the turbine power at the SG. This mismatch will cause a rapid increase in steam generator pressure, which will cause disturbances to the reactor coolant system.

⇒ If action is not taken to reduce the reactor neutron power immediately, the SG pressure safety relief valve will open on high SG pressure, causing depressurization of the steam generator. This again will cause disturbances in the primary systems.

To cope with the disturbances caused by the turbine trip, the plant control system is designed with the following control actions:

⇒ The reactor neutron power will be reduced quickly to 75 % by rapid insertion of control rods — this is known as reactor power "setback". The intent is to reduce the reactor power in "ramp" fashion, but still maintain the reactor power at high enough level such that Xenon level buildup as a result of the setback, will not "overcome" the positive reactivity margin available at the reactor power control system. In other words, at such reduced power level, the reactor power control system still has enough positive reactivity (from the rods) to bring the reactor back to full power, if the turbine trip can be cleared quickly.

⇒ The turbine bypass valves will open automatically when turbine trip is detected (first the CSDVs, then the ASDVs, if SG pressure is still high), trying to alleviate steam pressure build-up. After the reactor power "setback" has been completed, the turbine bypass valves will modulate their opening to pass sufficient steam flow to the condenser, in order to maintain SG pressure at the constant setpoint. In this way, the turbine bypass valves temporarily replace the turbine as the steam load, and hence eliminate the mismatch of reactor thermal power and turbine power as mentioned previously.

To observe the transients as described above, using the simulator:

⇒ First initialize the simulator to 100% full power, and run the simulator.

⇒ Go to control rods & SD rods screen; record the average position of the ZCU rods and MCA rods, Gadolinium load (ppm). Observe any flux tilt in the flux map.

⇒ Go to "reactor power control" Screen; record the flux tilt error (%). Record the reactivity feedback effects due to Xenon (mk).

⇒ Go to "turbine generator" screen; record the position of the main steam stop valve, turbine governor control valves, turbine bypass valves (CSDV, ASDV), SG safety MSSVs.

⇒ Record the coolant pressure, SG pressure, and generator output.

⇒ Press the turbine trip button on the left hand bottom corner of the screen, and confirm turbine trip.

- Record the position of the main steam stop valve, turbine governor control valves, turbine bypass valves (CSDV, ASDV), SG MSSVs.
- Record the reactor power, SG pressure, and generator output, as the transient evolves.
- What is reactor power when turbine speed settles at 5 rpm?
- What is the steam flow through the bypass valve on the turbine generator screen?
- What is the peak SG pressure during the transient?
- Go to control rods & SD rods screen; record the position of the “ZCU” rods and “MCA” rods. How much have the ZCU rods moved (average position %)? How much have the MCA rods moved (average position %)? Observe any flux tilt in the flux map.
- Go to “reactor power control” screen, record any flux tilt error (%), coolant pressure and temperature. Record the reactivity feedback effects (mk) for Xenon. What is the difference in mk for Xenon before & after the turbine trip?
- Go to turbine generator screen, reset turbine trip, select ‘TRU ENABLE’, and select “TRU speedup” to synchronize the generator and continue to load turbine.
- After turbine is in service, what happens to the steam bypass valves as the turbine power increases? Note the SG pressure reading.
- After the turbine power is equal to the reactor power, go to the “reactor power control” screen to increase reactor power to 100 % in 25 % steps at 0.5 % per sec.

Now call up the 100 % FP IC, then call up the IC: “ZCU_DOUBLE_WORTH”, meaning that the ZCU’s now have a total MK worth of 18 MK, instead of 9 MK. Repeat the TURBINE TRIP & RECOVERY exercise, using this new MK worth of ZCUs. Discuss and compare the results for these two exercises.

4.5. Reactor trip and recovery

Reactor trip (or reactor scram) is a reactor protective action initiated by the reactor safety shutdown system on detection of alarm limits exceeded by specific parameters in the reactor core, coolant and balance of plant systems. The parameters and the related reactor trip setpoints are described in Section 2.9 “ACR trip parameters”.

Most importantly, the reactor also can be tripped by the operator MANUALLY, on account of abnormal incidents, or accidents.

⇒ The reactor trip action is to drop the two banks of “shutdown” rods (known as SDS1 into the core by gravity. As well, the liquid poison shutdown system (SDS2) is also activated. Note SDS2 is not simulated.

⇒ As well, all the ZCU and MCA rods are inserted into the core at maximum speed.

⇒ The end result is to put lots of negative reactivity into the core such that the nuclear fission chain reaction in the core is stopped immediately.

This exercise demonstrates the manual reactor trip transient, and how to recover and return the reactor to full power:

- Initialize the simulator to 100% FP.
- Go to “ACR Control rods & SD rods” screen, note the “shutdown rods” position, ZCU rods position; MCA rods position.
- Go to “reactor power control” screen; record the reactivity mk contribution from the reactivity devices and the feedback effects — i.e. SD rods, ZCU rods, MCA rods, Xenon, fuel temperature, moderator temperature & Gadolinium.
- Manually trip the reactor using the pop-up control at the left bottom of the screen.
- Observe the response of the overall unit. Go to trends screen; observe the trends for reactor power, reactor coolant pressure, SG pressure, steam flow, feedwater flow, and generator power.
- Wait until generator power is zero and reactor neutron power is less than 0.1%.
- Go to “ACR Control rods and SD rods” screen, reset the reactor trip and shutdown system (SDS). Observe that the SD rods are withdrawing.
- Record the time (using the display under the trends) needed to withdraw all shutdown rods. As well, observe the position of the yellow cursor show on the Reactivity Limit Diagram, after the SD rods are fully withdrawn, and the RRS is taking full control. You may observe that initially the RRS will input a small power increase setpoint of 1.45 % FP (see ACR Reactor Power Control Screen), in order to pull out the ZCU and MCA rods. Observe the movement of the yellow cursor, and the corresponding movement of the ZCU rods and MCA rods.
- As well, go to “ACR Reactor Power Control Screen”, observe the trends for overall mk, average ZCU and MCA position. Can you explain the trends for overall mk, and the average ZCU and MCA position ?
- Raise reactor power to 60%FP, in small step, at rate of 0.3 %/sec
- Observe the response of the reactor regulating system and the reactivity changes that take place.

Now call up the 100 % FP IC, then call up the IC: “ZCU_DOUBLE_WORTH”, meaning that the ZCU’s now have a total MK worth of 18 MK, instead of 9 MK. Repeat the REACTOR TRIP & RECOVERY exercise, using this new MK worth of ZCUs. Discuss and compare the results for these two exercises.

5. ACR MALFUNCTION TRANSIENT EVENTS

Note: The ACR malfunction transient events described below are caused by malfunctions initiated in the simulator. To initiate a malfunction:

⇒ Press the “MALF” button at the bottom right of any screen.

⇒ A pop-up menu with a list of malfunctions will appear.

⇒ Select the specific malfunction to initiate, by clicking on the malfunction item itself. The malfunction item will be highlighted in “black”.

⇒ Click on “insert MF” button, if the malfunction is initiated immediately; or input a time delay (sec) in the display box, and then click “insert MF”; the malfunction will be initiated after the specified time delay has elapsed.

⇒ When malfunction occurs, the “malfunction active” alarm will be “on”.

⇒ To clear a malfunction which has been inserted, click on the malfunction item, and then click “clear MF”; or alternatively, click on “global clear”, which will clear all the malfunctions selected.

5.1. Fail closed all feedwater level control valves

This malfunction leads to loss of feedwater to the steam generators.

When this malfunction transient occurs:

⇒ The SG level drops quickly, causing low SG level.

⇒ Reactor will be setback when SG level drops < 10.11m.

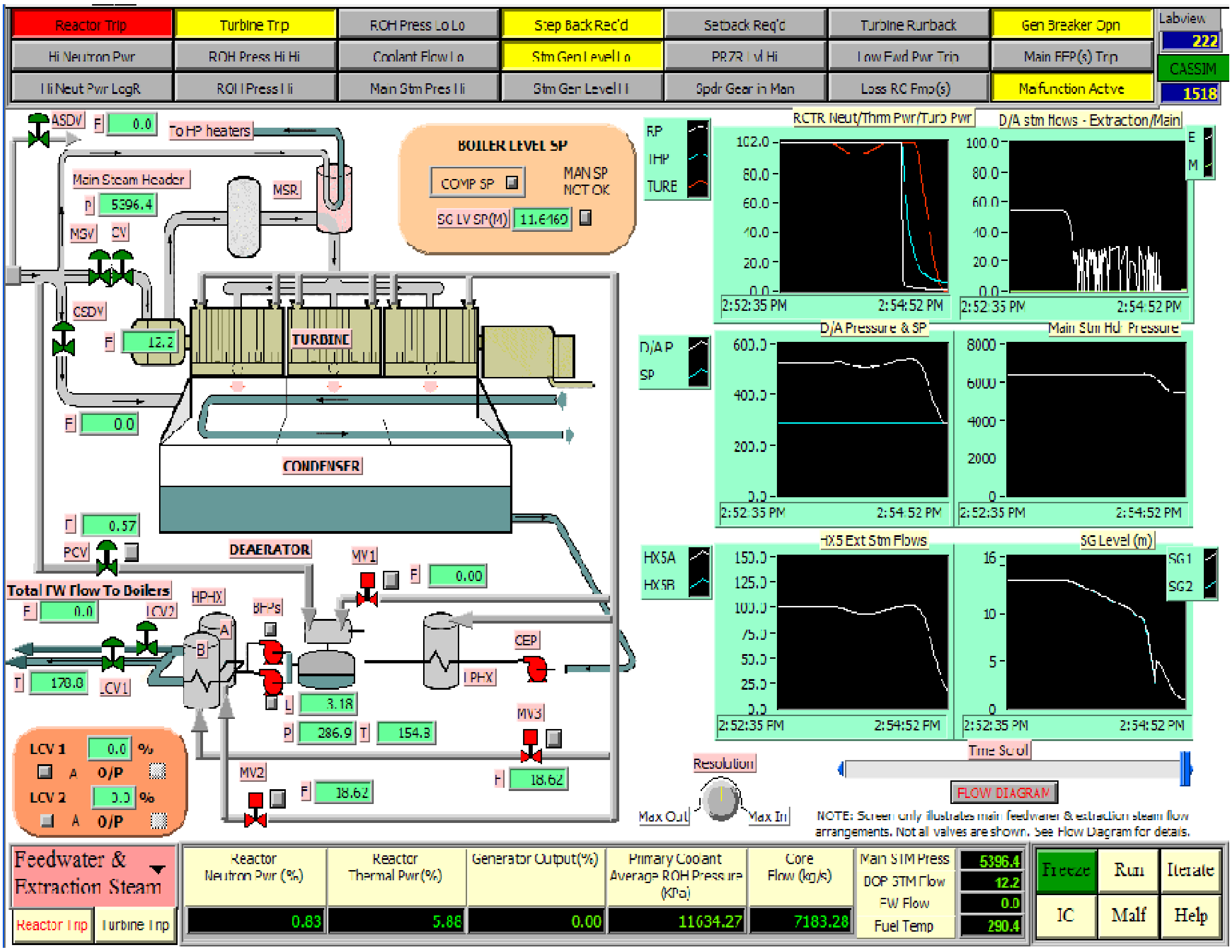
⇒ Reactor will be tripped when SG level drops < 9.9m.

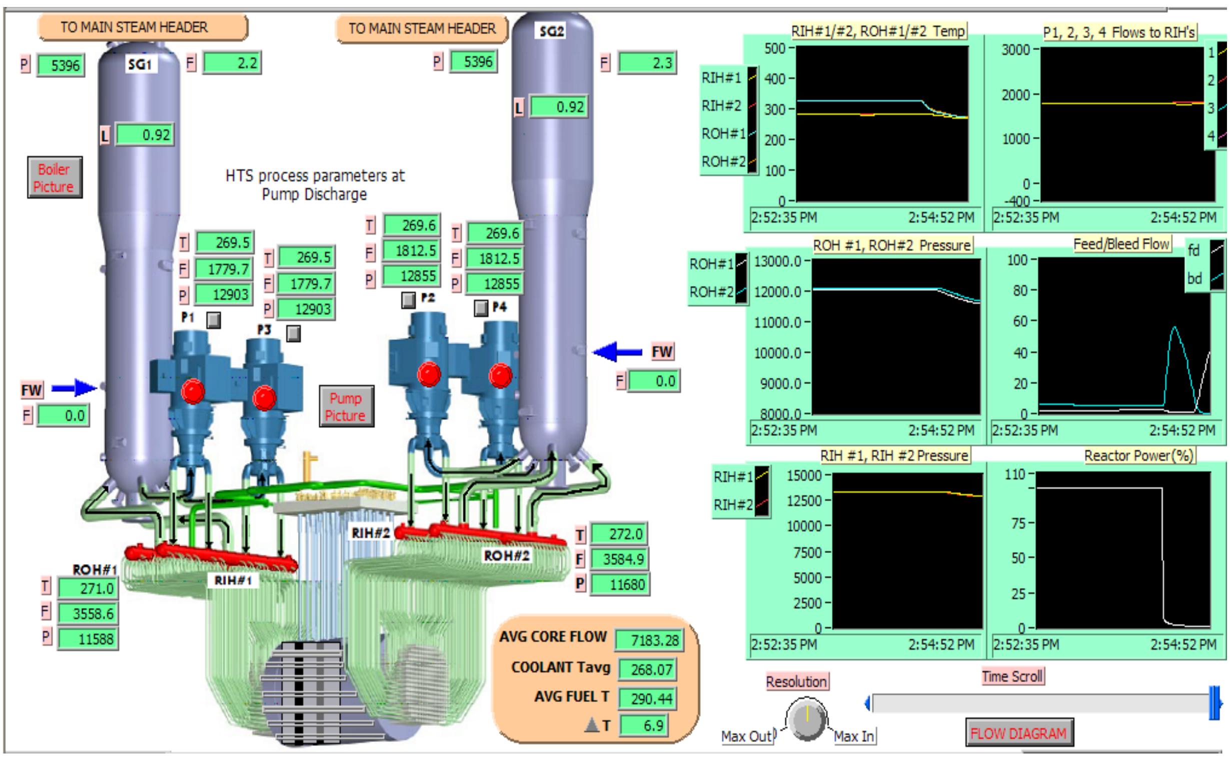
⇒ Due to loss of feedwater to the steam generators, cooling of the primary reactor coolant is reduced.

⇒ The higher temperature in the reactor coolant causes it to expand. However, as the reactor is tripped, there will be rapid reduction of reactor thermal power, causing shrinkage of reactor coolant. So the net effect is the dropping of reactor coolant pressure.

⇒ Dropping coolant pressure causes out-surge of coolant from the pressurizer, in order to alleviate coolant pressure decrease. Observe the flow direction in the surge line to pressurizer. As well, the electric heaters in the pressurizer will be turned on, until coolant pressure returns to its setpoint.

⇒ As reactor is tripped, SG pressure is dropping rapidly, causing the turbine governor to runback the turbine - that is closing the turbine governor control valves. This results in rapid reduction of MW to zero, leading to turbine generator trip, on zero forward power.





Topic for discussion:

1. Discuss the role of the steam generators as heat sink. If that heat sink is lost, like in this case, what should be the appropriate design to back up the steam generators heat sink ?

5.2. Steam generator #1 steam flow FT irrational

This malfunction causes steam flow transmitter for steam generator #1 to fail “low”. The consequence is that the steam generator level control system for SG#1 is “fooled” into thinking that the steam flow from SG #1 is rapidly decreasing, hence feedwater flow into SG #1 will be cutback immediately to match with “false” steam flow reduction, in an attempt to maintain the SG level at its setpoint value.

In reality, the steam flow from SG #1 remains at 100 % nominal flow rate. Because the feedwater flow is reduced to zero, by the control action of the SG level control system (SGLC), the consequence is a rapid drop in SG #1 level.

When this malfunction transient occurs:

⇒ Go to reactor coolant system screen, observe the steam flow from SG #1

⇒ As well, observe the feedwater flow to SG #1

⇒ Observe changes in primary coolant pressure, and the surge flow from pressurizer

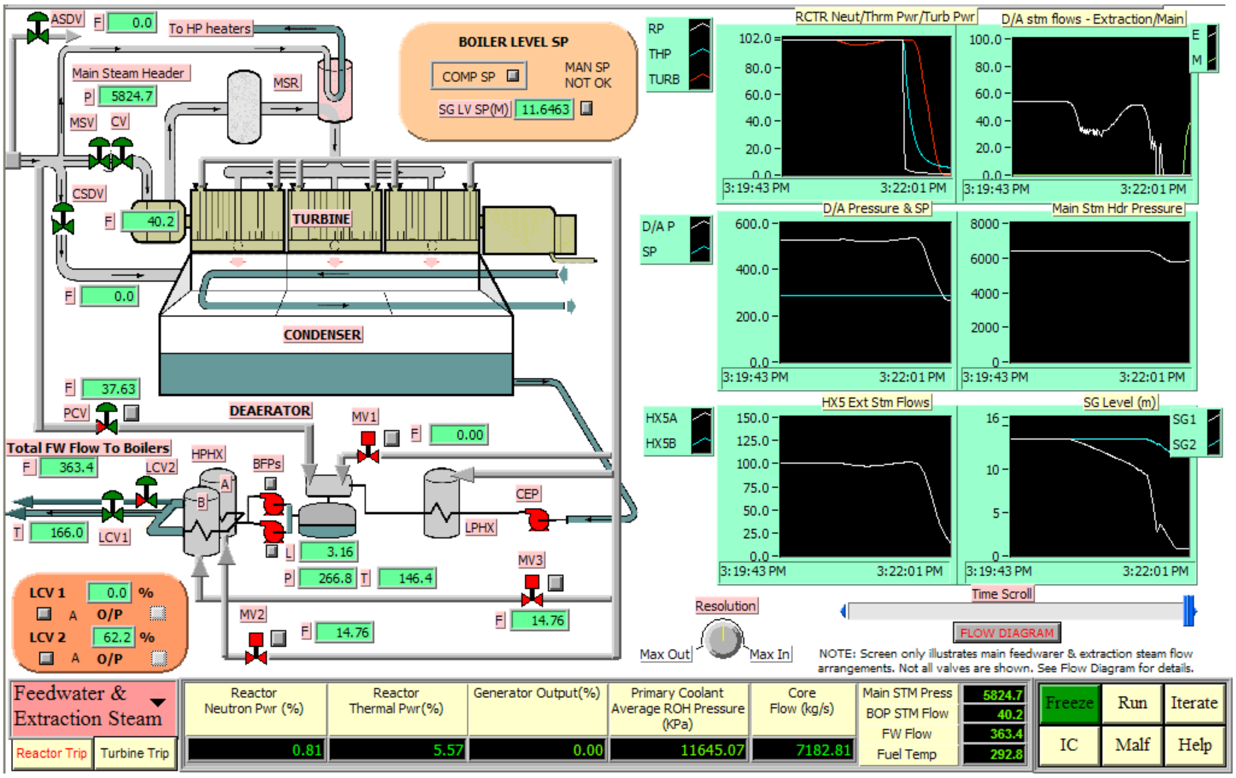
⇒ Reactor setback will occur first on low SG #1 level

⇒ Reactor trip will occur on low-low SG #1 level

⇒ Observe the coolant pressure transient, and the surge flow from pressurizer.

⇒ Observe level in SG #1

⇒ As reactor is tripped, SG pressure is dropping rapidly, causing the turbine governor to runback the turbine - that is closing the turbine governor control valves. This results in rapid reduction of MW to zero, leading to turbine generator trip, on zero forward power.



5.3. FW LCV#1 fails open

This malfunction leads to maximum feedwater flow to SG #1 with the control valve LCV #1 failed wide open. Because the feedwater flow is much more than the steam flow from SG #1, as a result, the level at SG #1 is rising steadily, leading to SG # 1 high level.

When this malfunction transient occurs:

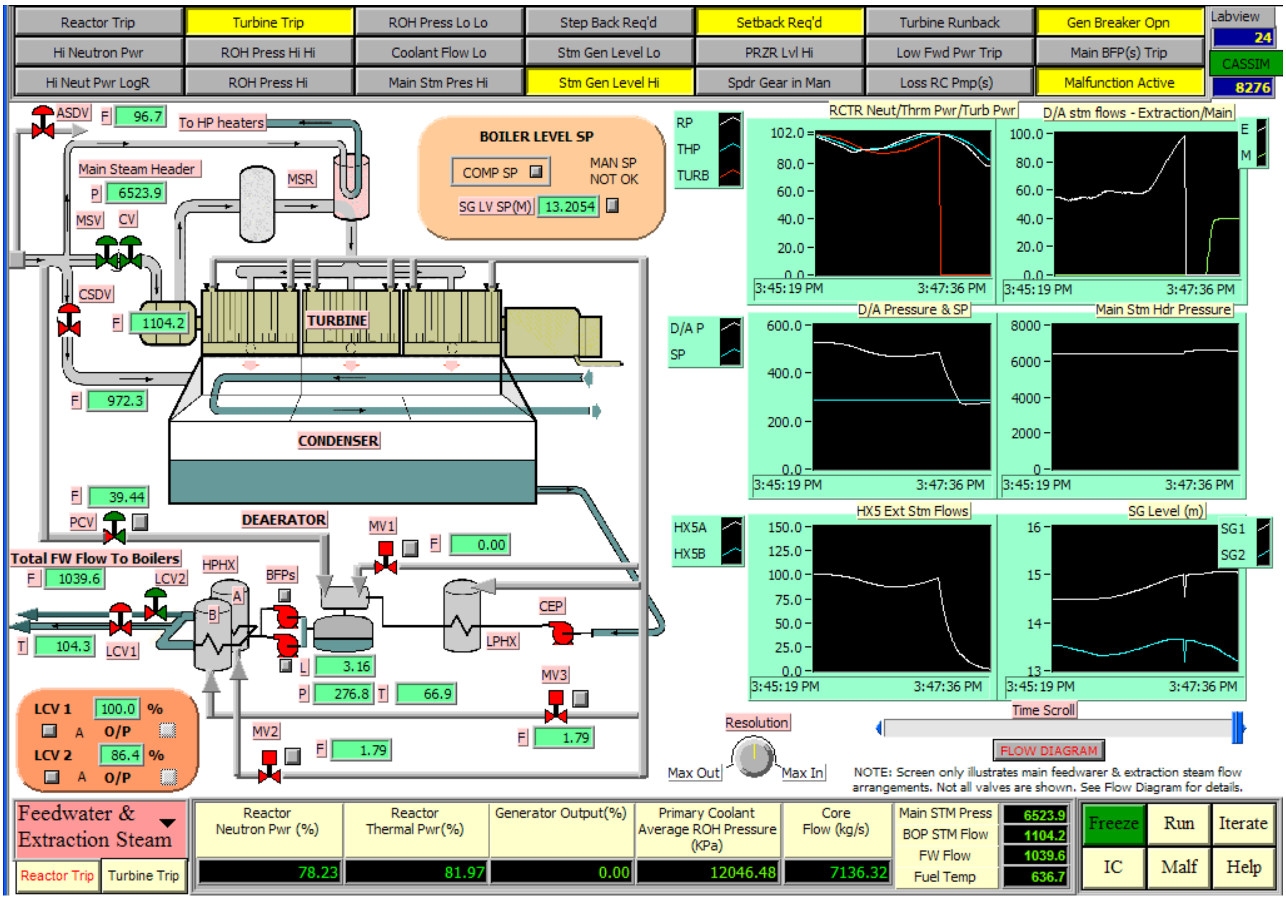
⇒ Go to "feedwater & extraction steam" screen; observe that LCV #1 is 100 % open.

⇒ Go to "reactor coolant system" screen, observe SG #1 feedwater flow, and steam flow. Note the mismatch in flow, and observe the SG #1 level.

⇒ Observe if this transient has any impact to the reactor and primary coolant systems.

⇒ As the SG level very high alarm occurs, turbine generator will be tripped.

⇒ When the turbine is tripped, there will be a Reactor Setback to 75%. The transient response will be similar to that described in Section 4.4



5.4. FW LCV#1 fails closed

This malfunction leads to loss of feedwater to SG #1. As such, the transient response is similar to that described in Section 5.2.

5.5. Main Feed Water Pump trips

This malfunction leads to loss of 50 % of normal feedwater flow to SG #1 and SG #2, due to tripping of one SG feed pump (BFP). The result is low SG level, causing reactor setback, followed by reactor trip. The transient response is similar to that described in Section 5.1.

5.6. Turbine throttle PT fails low

This malfunction causes the turbine throttle pressure transmitter to fail "low". The consequence is that the turbine governor control system is "fooled" into thinking that the main steam pressure is rapidly decreasing, hence as a regulation control action, the turbine governor will run back turbine load immediately in order to maintain main steam pressure. Because the throttle pressure transmitter has failed "low", the turbine will be run back to 0 MW. Turbine trip will follow as a consequence of generator "zero" forward power.

But in reality, the main steam pressure was never "low" in the beginning. Running back the turbine will cause immediate rise in main steam pressure. Despite the fact that the turbine Bypass valves (CSDV, ASDV) are opening to cope with the pressure rise, it takes time for the steam pressure to decrease. The peak rise in steam pressure has immediate impact on the heat transfer of the steam generators. As a result, there will be transients on coolant temperature and pressure. But turbine trip will occur very quickly, causing setback of reactor power, and the transients in the reactor and primary coolant will stabilize.

When this malfunction transient occurs:

⇒ Go to "turbine generator" screen; observe the turbine governor position.

⇒ Observe the main steam pressure transient. What is the peak steam pressure?

⇒ Explain how the turbine bypass valves operate ?

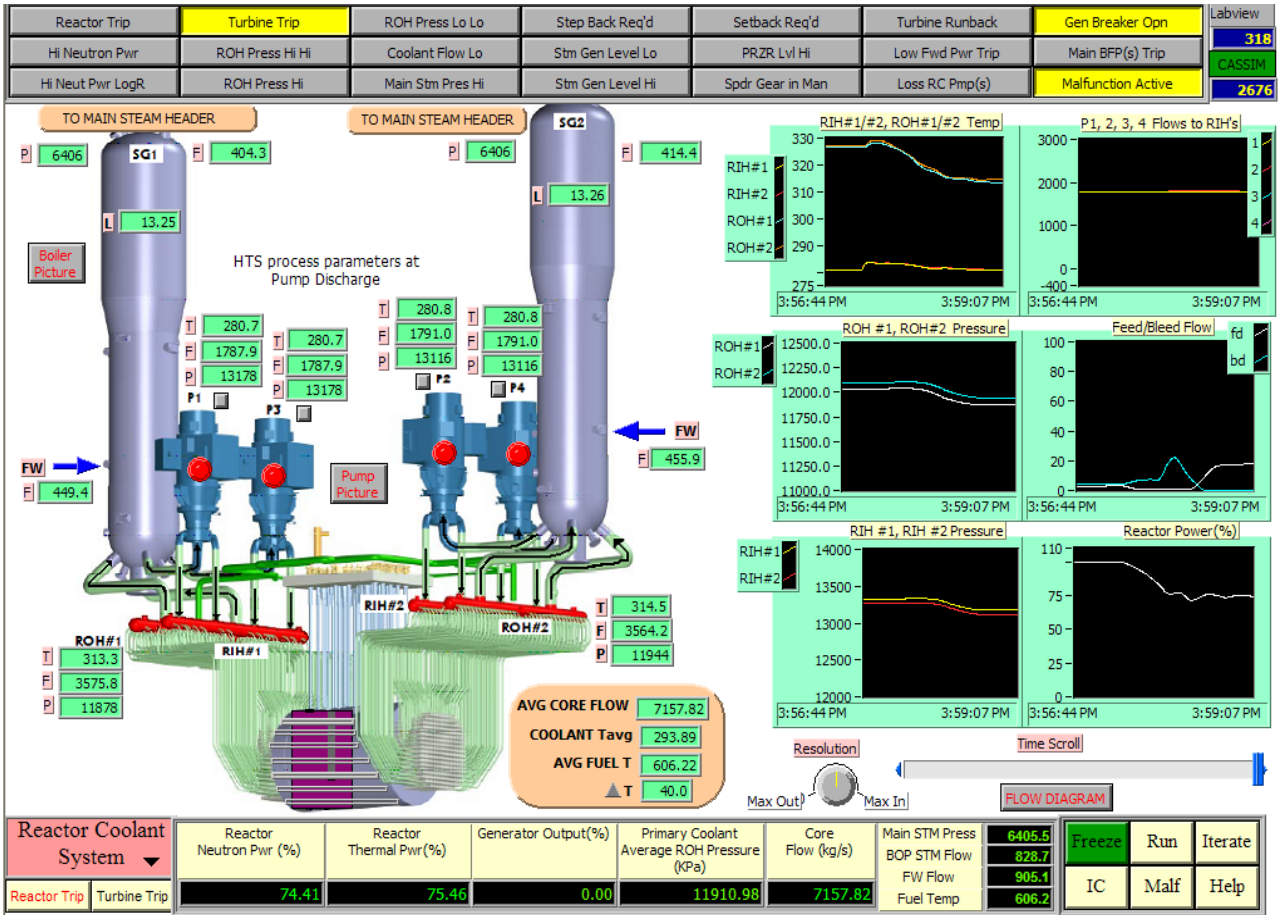
⇒ Observe turbine power is decreased very rapidly, followed by turbine trip.

⇒ Repeat this malfunction again, while the "reactor coolant system" screen is displayed.

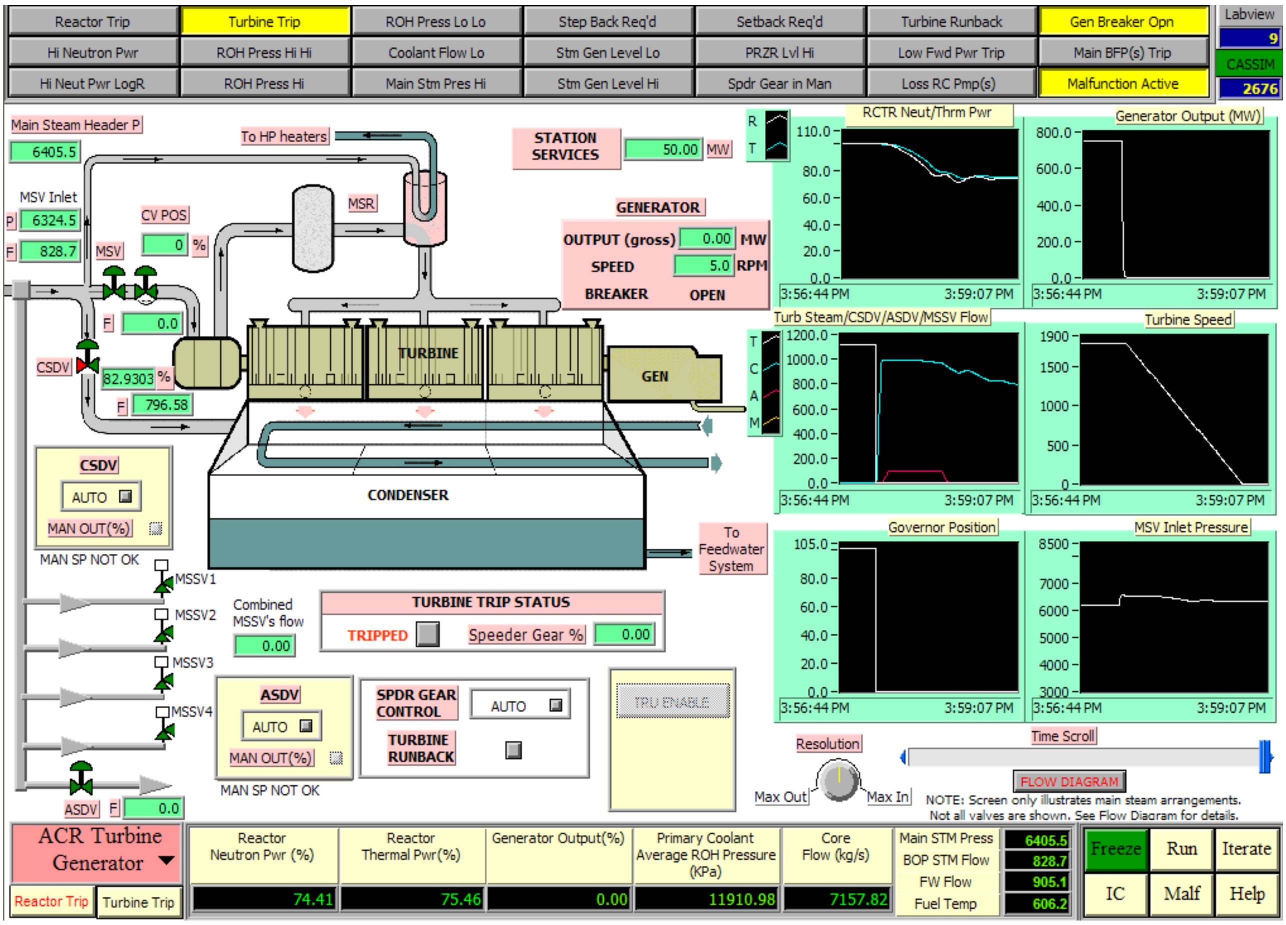
⇒ Observe the RIH, ROH temperature transients. It is necessary to change the scale of the trend accordingly in order to see the transient better.

⇒ What is the peak RIH, ROH temperature during this malfunction?

⇒ Explain why the RIH, ROH temperatures go up?



Reactor Trip
Turbine Trip
ROH Press 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
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
Labview
318
CASSIM
2676
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
Boiler Picture
HTS process parameters at Pump Discharge
Pump Picture
FW
ROH#1
ROH#2
RIH#1
RIH#2
AVG CORE FLOW 7157.82
COOLANT Tavg 293.89
AVG FUEL T 606.22
RIH#1/#2, ROH#1/#2 Temp
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
Feed/Bleed Flow
RIH #1, RIH #2 Pressure
Reactor Power(%)
Resolution
Time Scroll
Max Out
Max In
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 74.41
Reactor Thermal Pwr(%) 75.46
Generator Output(%) 0.00
Primary Coolant Average ROH Pressure (KPa) 11910.98
Core Flow (kg/s) 7157.82
Main STM Press 6405.5
BOP STM Flow 828.7
FW Flow 905.1
Fuel Temp 606.2
Freeze
Run
Iterate
IC
Malf
Help



Reactor Trip
Turbine Trip
ROH Press 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
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
Labview
9
CASSIM
2676
Main Steam Header P
6405.5
To HP heaters
STATION SERVICES 50.00 MW
MSV Inlet
CV POS
MSR
GENERATOR
OUTPUT (gross) 0.00 MW
SPEED 5.0 RPM
BREAKER OPEN
CSDV
TURBINE
GEN
CONDENSER
AUTO
MAN OUT(%)
MAN SP NOT OK
To Feedwater System
MSSV1
MSSV2
MSSV3
MSSV4
Combined MSSV's flow
TURBINE TRIP STATUS
TRIPPED
Speeder Gear %
ASDV
SPDR GEAR CONTROL
TURBINE RUNBACK
TRU ENABLE
RCTR Neut/Thrm Pwr
Generator Output (MW)
Turb Steam/CSDV/ASDV/MSSV Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
Resolution
Time Scroll
Max Out
Max In
FLOW DIAGRAM
NOTE: Screen only illustrates main steam arrangements. Not all valves are shown. See Flow Diagram for details.
ACR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 74.41
Reactor Thermal Pwr(%) 75.46
Generator Output(%) 0.00
Primary Coolant Average ROH Pressure (KPa) 11910.98
Core Flow (kg/s) 7157.82
Main STM Press 6405.5
BOP STM Flow 828.7
FW Flow 905.1
Fuel Temp 606.2
Freeze
Run
Iterate
IC
Malf
Help

5.7. All atmospheric MSSVs fail open

This malfunction will cause immediate depressurization of the steam generators. Responding to rapid dropping of main steam pressure, the turbine will be unloaded rapidly, followed by turbine trip on zero forward power. Reactor power will be setbacked to 75 % FP upon turbine trip.

On the primary side, the rapid drop in steam generator pressure causes the coolant temperature and pressure transients.

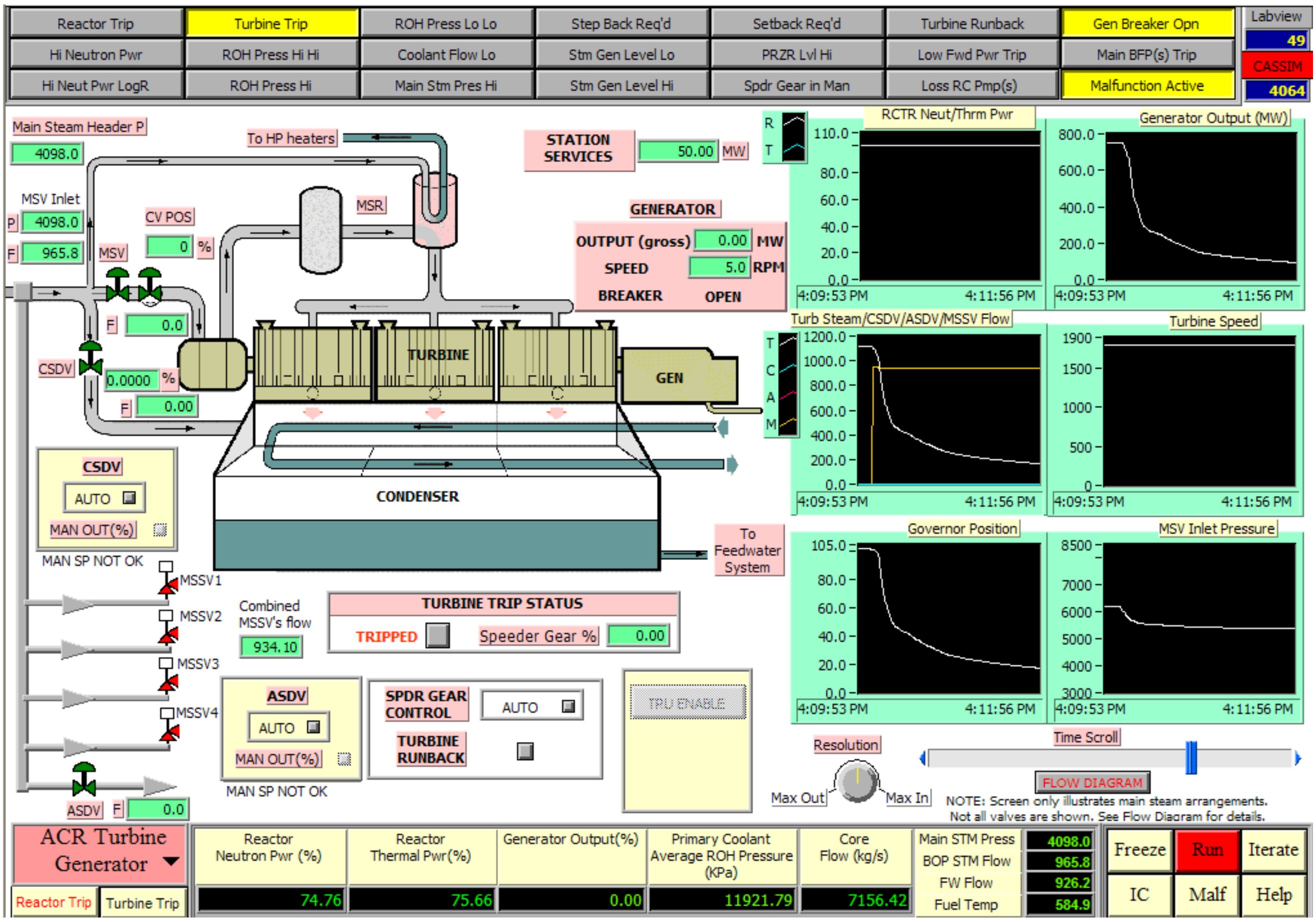
When this malfunction transient occurs:

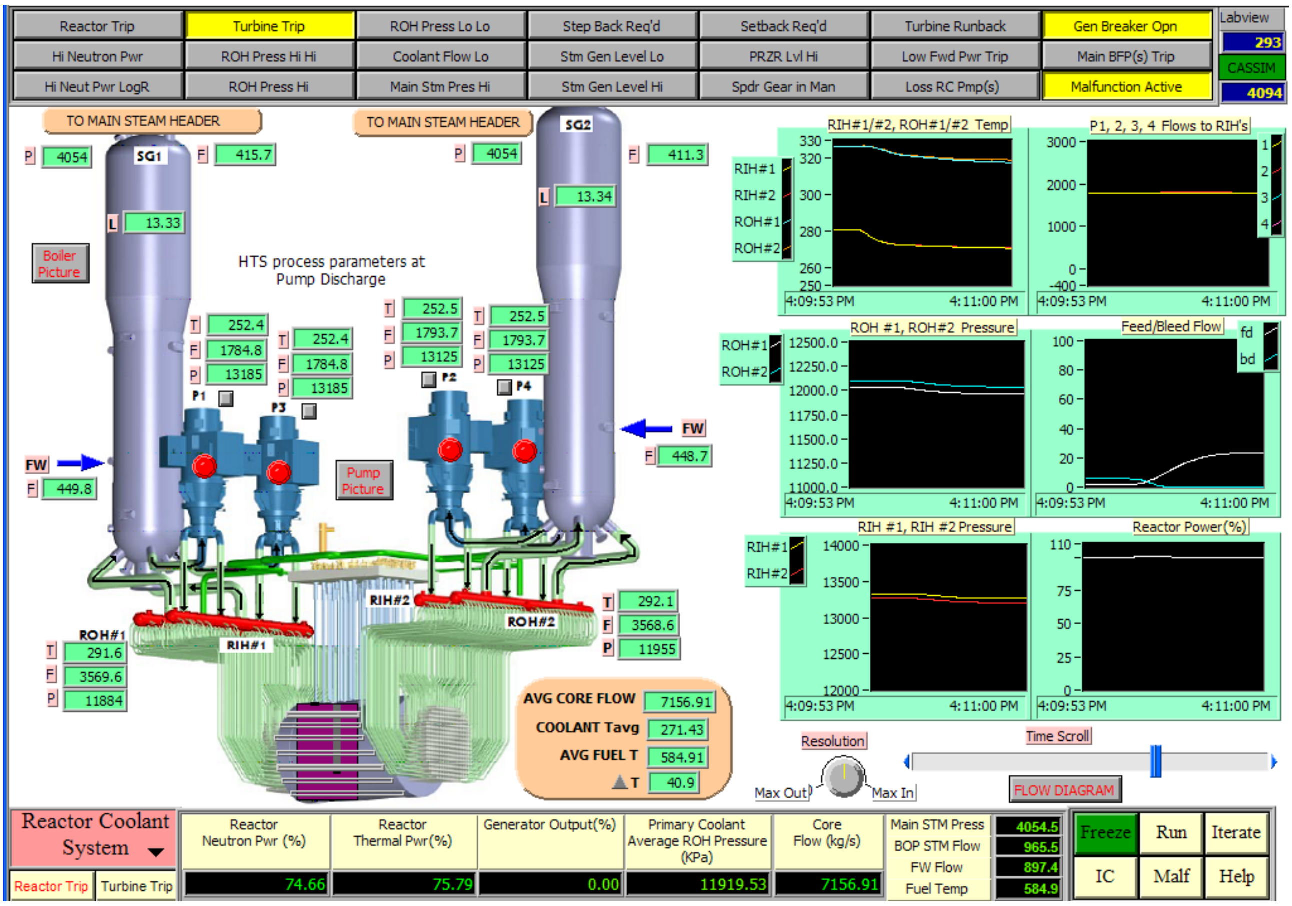
⇒ Go to "turbine generator" screen, observe the main steam safety relief valves (MSSV) opening.

⇒ Observe the turbine governor valve position, and that the turbine is unloaded rapidly. As the turbine is unloaded, observe the transient of main steam pressure. Does the turbine Bypass valve open in this transient? CSDV first or ASDV first?

⇒ Repeat this transient, but this time, go to "reactor coolant system" screen first before inserting the malfunction.

⇒ Observe reactor coolant temperature and pressure transient. Explain why ROH, RIH pressures and temperatures are decreasing ?





Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
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
293
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
4094
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
Boiler Picture
HTS process parameters at Pump Discharge
Pump Picture
FW
ROH#1
RIH#1
RIH#2
ROH#2
AVG CORE FLOW 7156.91
COOLANT Tavg 271.43
AVG FUEL T 584.91
T 40.9
RIH#1/#2, ROH#1/#2 Temp
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
Feed/Bleed Flow
RIH #1, RIH #2 Pressure
Reactor Power(%)
4:09:53 PM
4:11:00 PM
Resolution
Time Scroll
Max Out
Max In
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 74.66
Reactor Thermal Pwr(%) 75.79
Generator Output(%) 0.00
Primary Coolant Average ROH Pressure (KPa) 11919.53
Core Flow (kg/s) 7156.91
Main STM Press 4054.5
BOP STM Flow 965.5
FW Flow 897.4
Fuel Temp 584.9
Freeze
Run
Iterate
IC
Malf
Help

5.8. Turbine bypass valve CSDV fails closed

This malfunction will cause the NPP to lose its steam bypass CSDV capability, in the event of turbine trip. On turbine trip, reactor power will be setbacked back to 75 %. However, as a result of turbine Bypass CSDV valves failing closed, the SG pressure will increase rapidly, causing further reactor power setback on high main steam pressure. On the steam side, ASDVs will open rapidly. As the ASDVs capacity is not sufficient to relieve rising main steam pressure, the main steam safety relief valves (MSSVs) will open to relieve rising main steam pressure that has exceeded the MSSV's lift setpoint. The MSSVs will close on decreasing main steam pressure, as the reactor power is being setbacked by high main steam pressure, and the transient stabilizes.

When this malfunction transient occurs:

⇒ Go to "turbine generator" screen, trip the turbine using the pop-up control at the bottom left of the screen.

⇒ Observe that turbine bypass valves CSDV remain closed. Observe the response of ASDV.

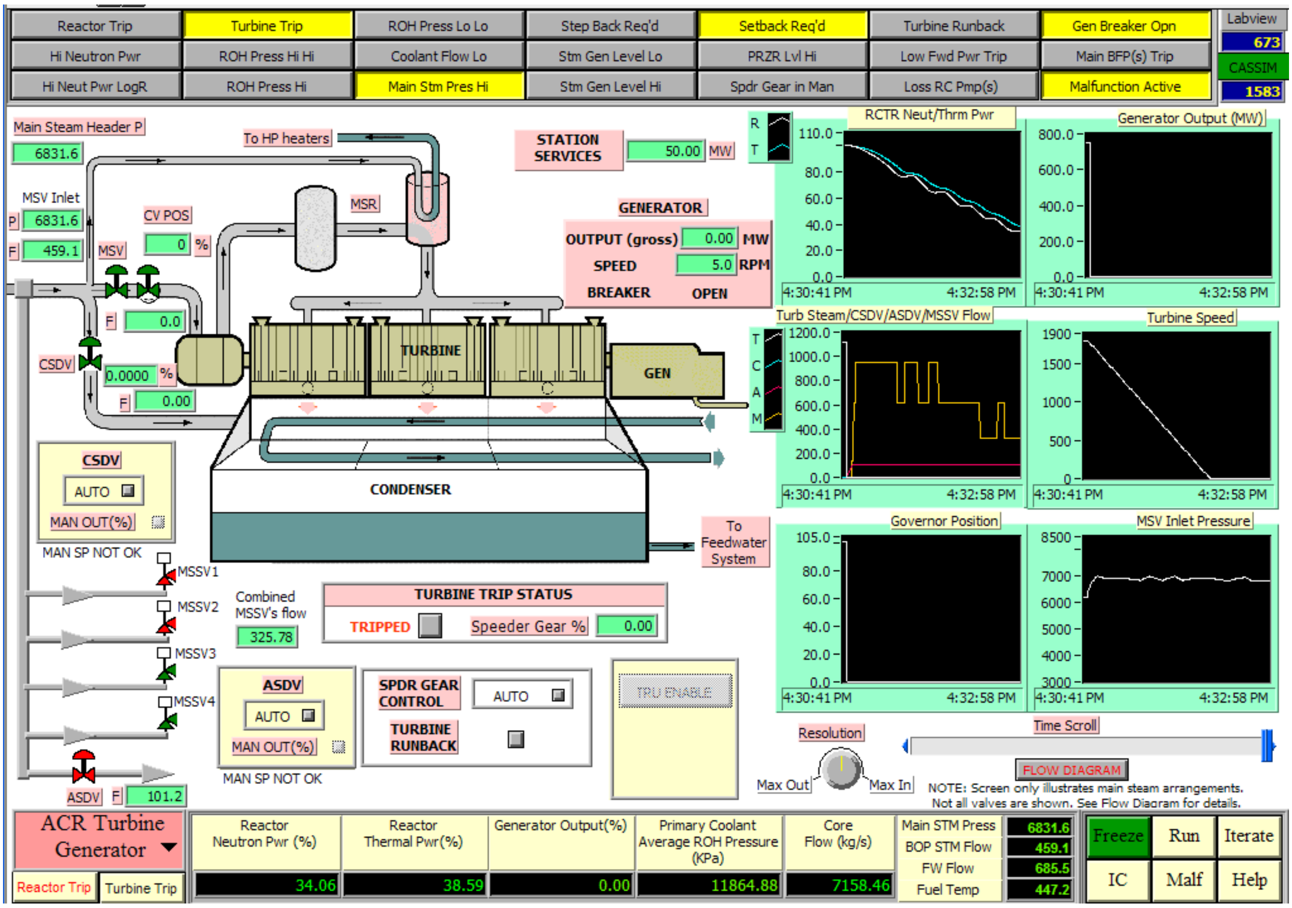
⇒ Observe that Reactor power is "setbacked". Can you explain the setback parameters in this incident?

⇒ Observe the main steam pressure transient. At what pressure does the first MSSV begin to open? What is the peak main steam pressure?

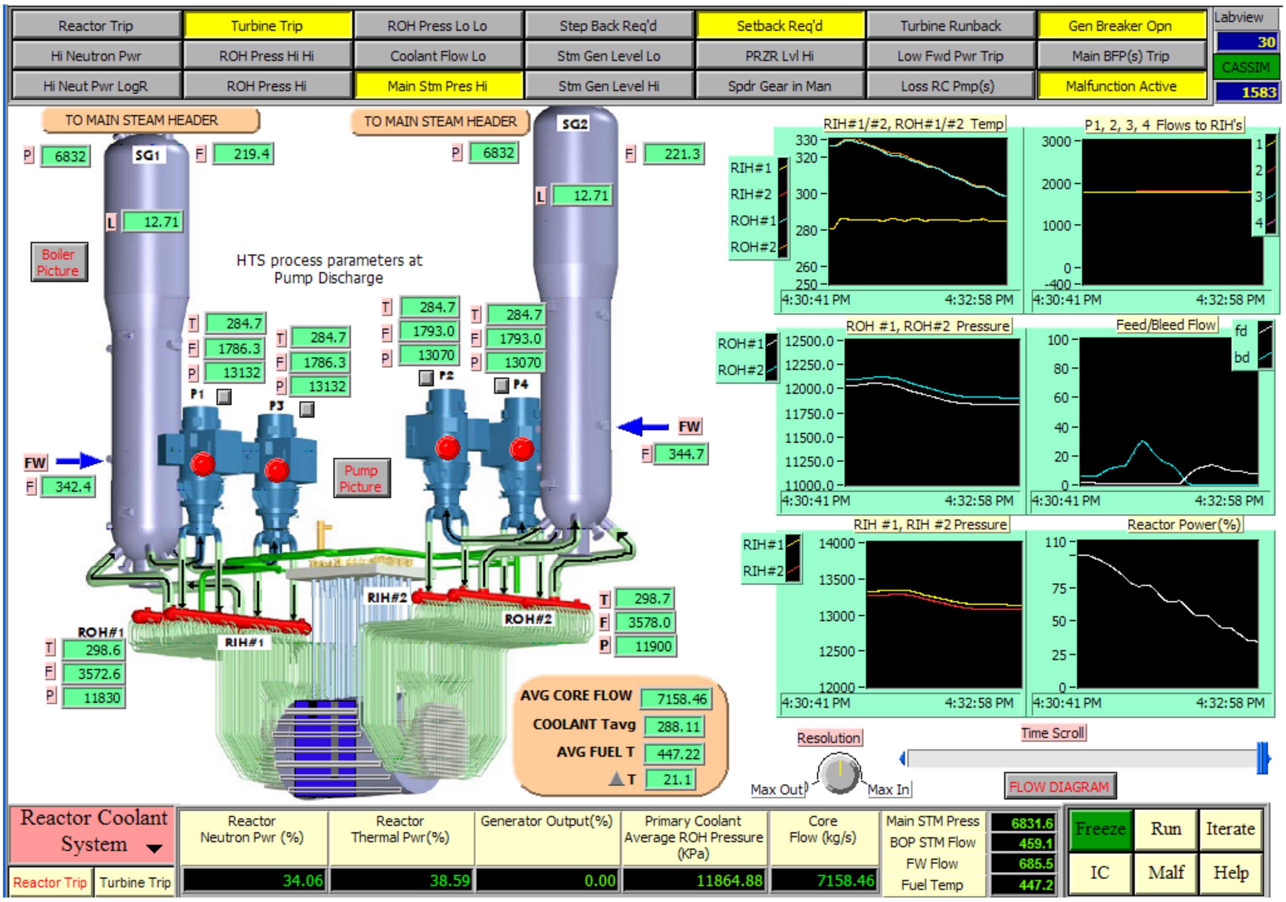
⇒ At what pressure will the MSSV start closing? Explain why MSSV closing and opening again, as seen in the trend? At what pressure will all the MSSV be closed?

⇒ Observe when Reactor Setback is finished, and at what main steam pressure? What is the reactor power at that time?

⇒ Record and explain the transients in coolant temperature and pressure.



Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
673
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
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
1583
Main Steam Header P
6831.6
To HP heaters
MSR
STATION SERVICES
50.00 MW
MSV Inlet
P 6831.6
F 459.1
CV POS
0 %
MSV
GENERATOR
OUTPUT (gross) 0.00 MW
SPEED 5.0 RPM
BREAKER OPEN
F 0.0
CSDV
0.0000 %
F 0.00
TURBINE
GEN
CONDENSER
CSDV
AUTO
MAN OUT(%)
MAN SP NOT OK
To Feedwater System
MSSV1
MSSV2
MSSV3
MSSV4
Combined MSSV's flow
325.78
TURBINE TRIP STATUS
TRIPPED
Speeder Gear % 0.00
ASDV
AUTO
MAN OUT(%)
MAN SP NOT OK
SPDR GEAR CONTROL
AUTO
TURBINE RUNBACK
TPU ENABLE
ASDV
F 101.2
RCTR Neut/Thrm Pwr
Generator Output (MW)
Turb Steam/CSDV/ASDV/MSSV Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
4:30:41 PM
4:32:58 PM
Resolution
Max Out
Max In
Time Scroll
FLOW DIAGRAM
NOTE: Screen only illustrates main steam arrangements. Not all valves are shown. See Flow Diagram for details.
ACR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
34.06
Reactor Thermal Pwr(%)
38.59
Generator Output(%)
0.00
Primary Coolant Average ROH Pressure (KPa)
11864.88
Core Flow (kg/s)
7158.46
Main STM Press 6831.6
BOP STM Flow 459.1
FW Flow 685.5
Fuel Temp 447.2
Freeze
Run
Iterate
IC
Malf
Help



Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
30
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
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
1583
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
P 6832
F 219.4
P 6832
F 221.3
L 12.71
L 12.71
Boiler Picture
HTS process parameters at Pump Discharge
T 284.7
F 1786.3
P 13132
T 284.7
F 1786.3
P 13132
T 284.7
F 1793.0
P 13070
T 284.7
F 1793.0
P 13070
P1
P2
P3
P4
Pump Picture
FW
F 342.4
FW
F 344.7
RIH#2
ROH#2
T 298.7
F 3578.0
P 11900
ROH#1
RIH#1
T 298.6
F 3572.6
P 11830
AVG CORE FLOW 7158.46
COOLANT Tavg 288.11
AVG FUEL T 447.22
ΔT 21.1
RIH#1/#2, ROH#1/#2 Temp
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
Feed/Bleed Flow
RIH #1, RIH #2 Pressure
Reactor Power(%)
4:30:41 PM
4:32:58 PM
Resolution
Max Out
Max In
Time Scroll
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
34.06
Reactor Thermal Pwr(%)
38.59
Generator Output(%)
0.00
Primary Coolant Average ROH Pressure (KPa)
11864.88
Core Flow (kg/s)
7158.46
Main STM Press 6831.6
BOP STM Flow 459.1
FW Flow 685.5
Fuel Temp 447.2
Freeze
Run
Iterate
IC
Malf
Help

5.9. Turbine spurious trip

This malfunction event is similar to the operational transient of turbine trip. See description in Section 4.4.

5.10. PRZR heaters #2 to # 6 turned "ON" by malfunction

This malfunction event causes reactor coolant pressure to increase, due to the fact that all the pressurizer on/off heaters # 2 to #6 are turned on. The rise in coolant pressure is offset by the pressurizer spray that will come into action once the coolant pressure exceeds a predetermined setpoint for spraying.

When this malfunction transient occurs:

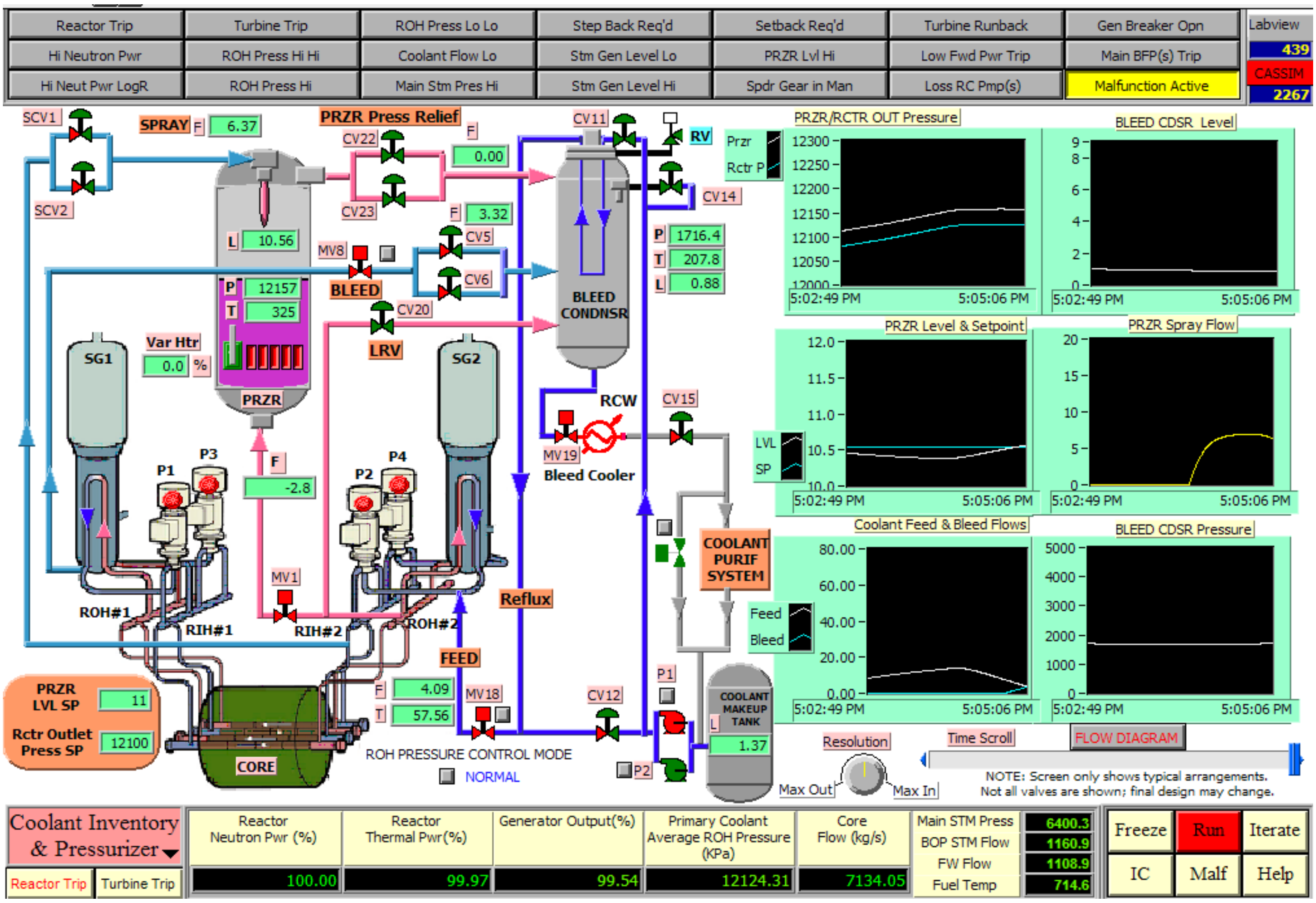
⇒ Go to “reactor coolant system” screen, observe that the pressurizer heater # 2, to # 6 are turned “on’ by malfunction.

⇒ Observe that the reactor coolant pressure increases, and then the pressurizer spray comes in, to cool the pressure down.

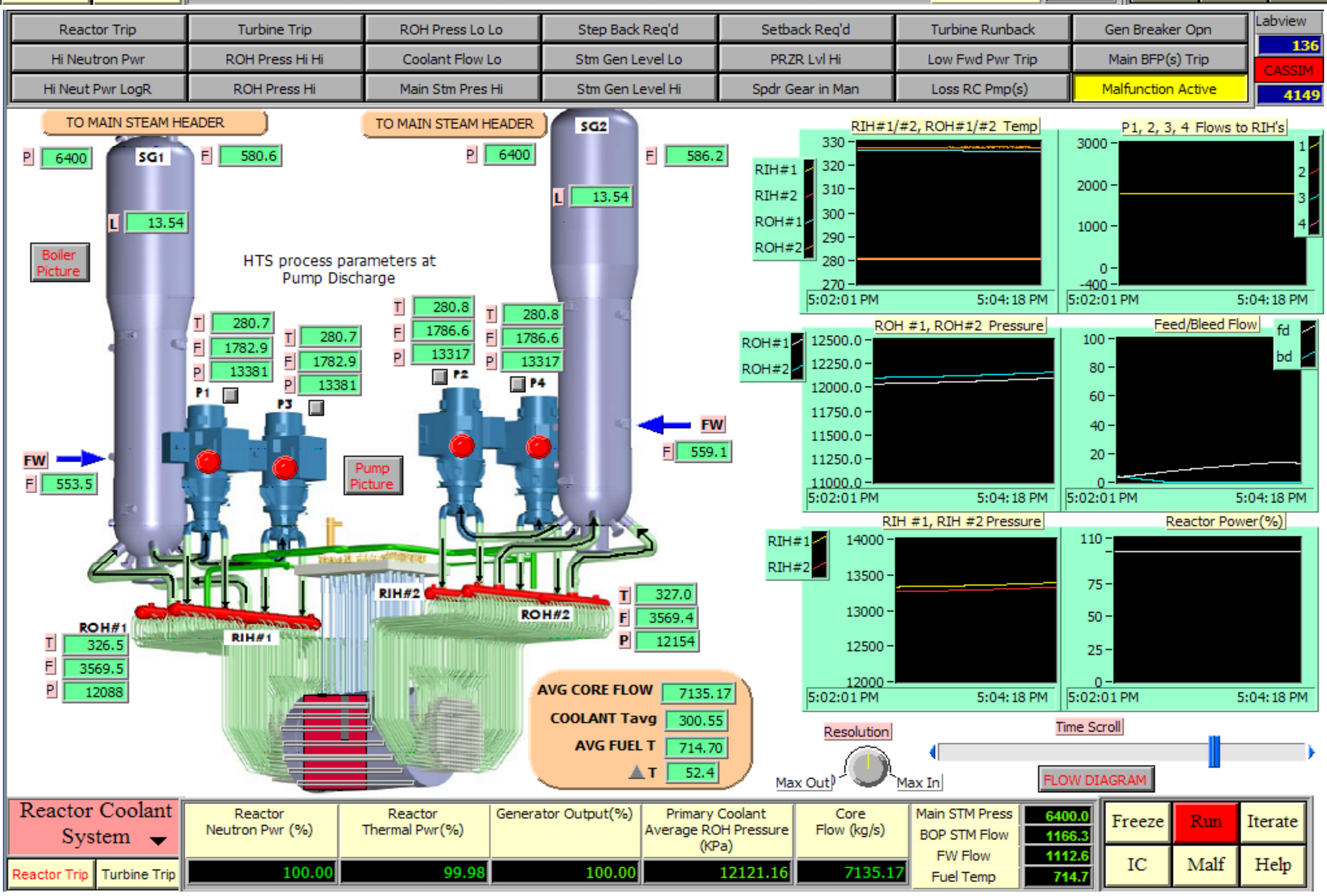
⇒ What is the net effect on reactor coolant pressure?

⇒ What happens to coolant temperature - increase or decrease? Explain the response.

⇒ Explain why the pressurizer level goes down, and then recovers?



Reactor Trip
Turbine Trip
ROH Press 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
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
Labview
439
CASSIM
2267
PRZR Press Relief
BLEED CONDSR
COOLANT PURIF SYSTEM
COOLANT MAKEUP TANK
CORE
ROH PRESSURE CONTROL MODE
NORMAL
PRZR/RCTR OUT Pressure
BLEED CDSR Level
PRZR Level & Setpoint
PRZR Spray Flow
Coolant Feed & Bleed Flows
BLEED CDSR Pressure
FLOW DIAGRAM
NOTE: Screen only shows typical arrangements.
Not all valves are shown; final design may change.
Coolant Inventory & Pressurizer
Reactor Neutron Pwr (%) 100.00
Reactor Thermal Pwr(%) 99.97
Generator Output(%) 99.54
Primary Coolant Average ROH Pressure (KPa) 12124.31
Core Flow (kg/s) 7134.05
Main STM Press 6400.3
BOP STM Flow 1160.9
FW Flow 1108.9
Fuel Temp 714.6
Freeze
Run
Iterate
IC
Malf
Help



Labview
136
CASSIM
4149
TO MAIN STEAM HEADER
HTS process parameters at Pump Discharge
Boiler Picture
Pump Picture
RIH#1/#2, ROH#1/#2 Temp
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
Feed/Bleed Flow
RIH #1, RIH #2 Pressure
Reactor Power(%)
AVG CORE FLOW 7135.17
COOLANT Tavg 300.55
AVG FUEL T 714.70
FLOW DIAGRAM
Reactor Coolant System
Reactor Neutron Pwr (%) 100.00
Reactor Thermal Pwr(%) 99.98
Generator Output(%) 100.00
Primary Coolant Average ROH Pressure (KPa) 12121.16
Core Flow (kg/s) 7135.17
Main STM Press 6400.0
BOP STM Flow 1166.3
FW Flow 1112.6
Fuel Temp 714.7

5.11. RC inventory feed valve (CV12) fails open

This malfunction causes the reactor coolant feed flow to reach the maximum. The immediate impact to the reactor coolant system is increased coolant inventory in the system. As a result, the pressurizer level will increase, leading to increase in pressurizer pressure. This is due to the fact that the vapor space in the pressurizer has been reduced by higher liquid mass in the pressurizer because of increased inventory.

The increased pressurizer pressure is offset by the spray action which comes into effect on high pressurizer pressure. But the spray will further increase the pressurizer level. The high pressurizer level will cause the inventory control system to increase the bleed flow by opening the Bleed Valve CV5. As a result, the bleed condenser level will increase. Overtime, the coolant feed flow and the coolant bleed flow will balance out, and the transient will stabilize.

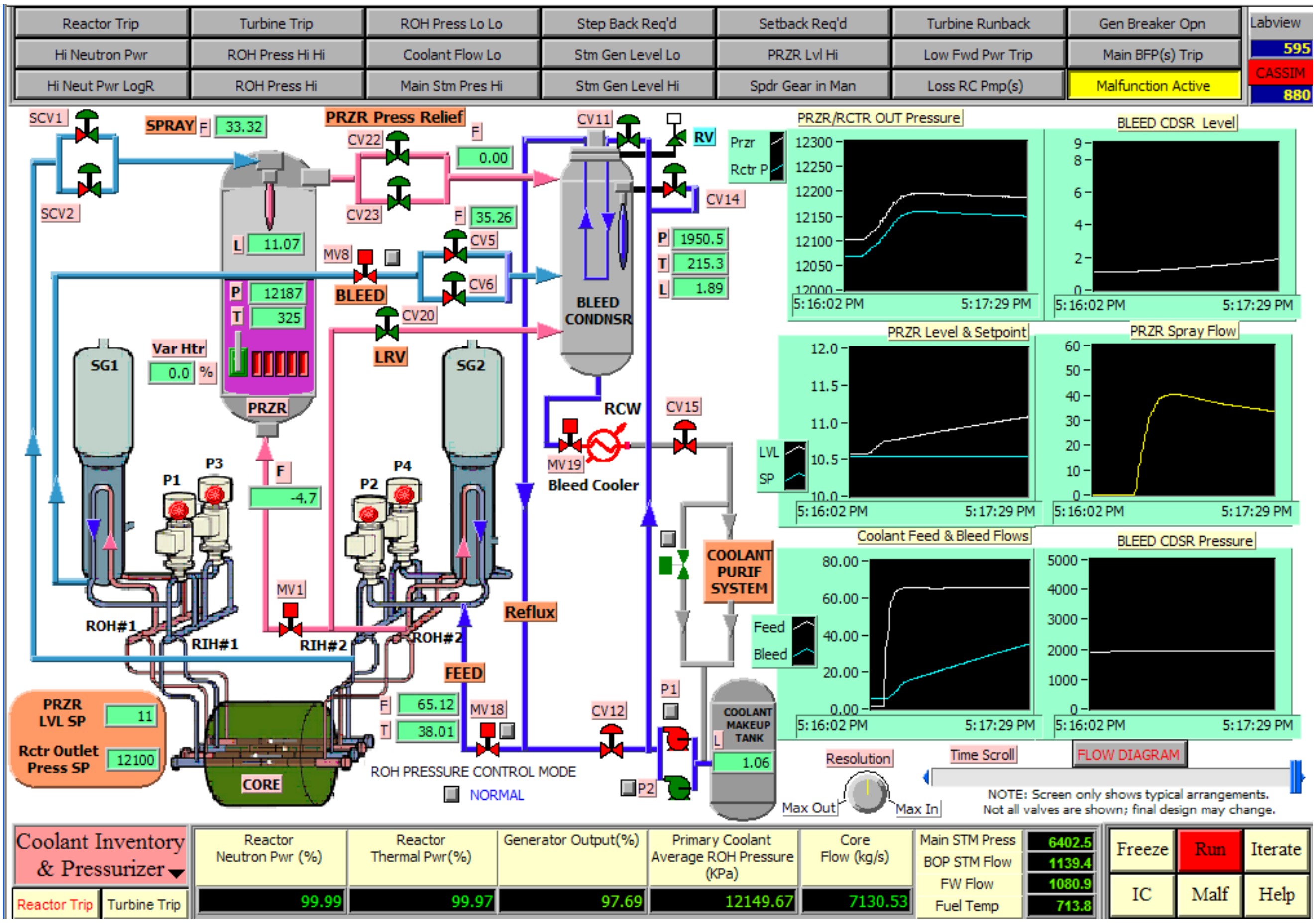
When this malfunction transient occurs:

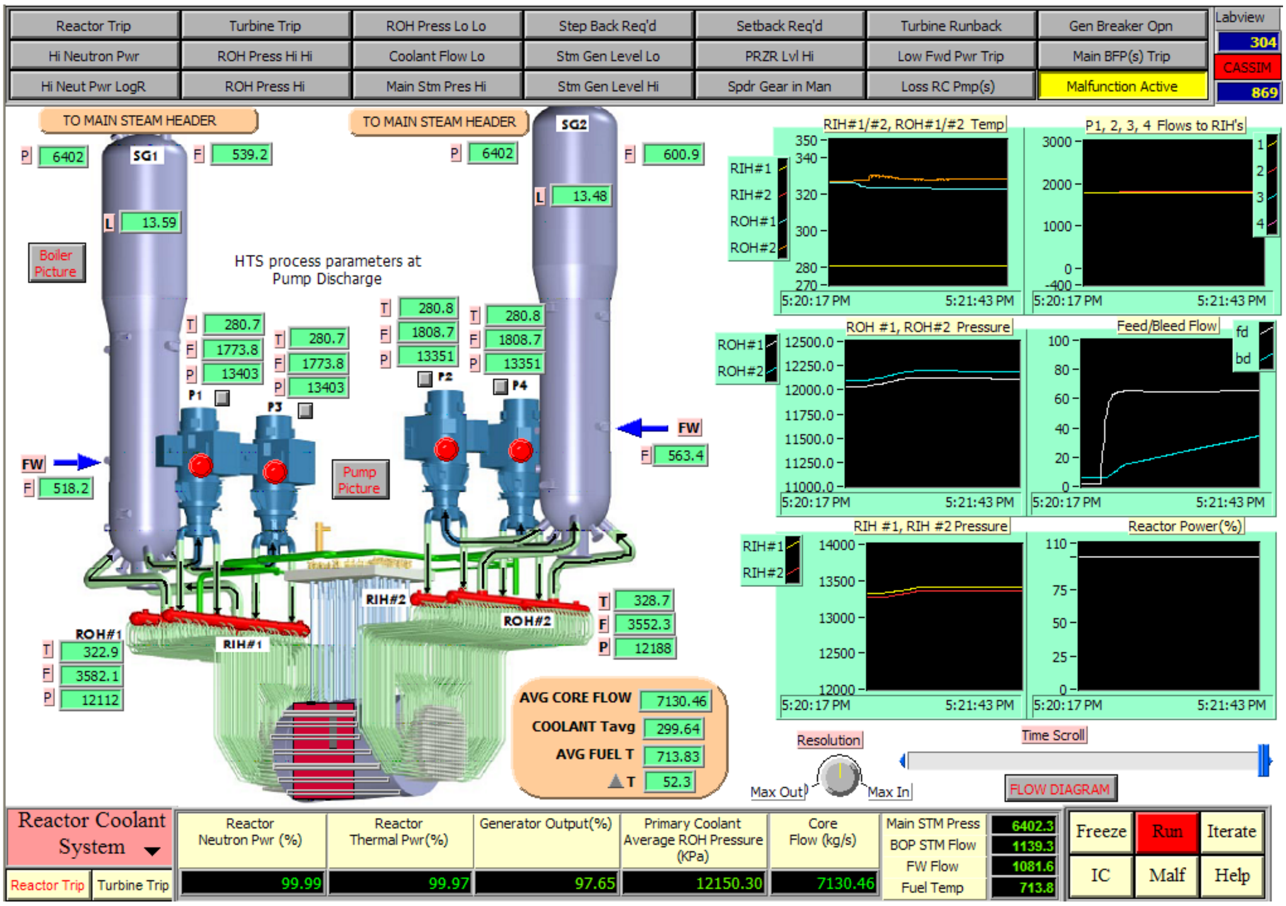
⇒ Go to the "coolant inventory and pressurizer" screen; observe that CV12 is 100 % open, and record the feed (charging) flow (kg/s).

⇒ Observe the coolant pressure transient, and that the pressurizer spray comes in.

⇒ Observe the pressurizer level and record the bleed flow (kg/s).

⇒ Observe the bleed condenser level.




Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
304
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
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
869
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
P 6402
F 539.2
L 13.59
P 6402
F 600.9
L 13.48
Boiler Picture
HTS process parameters at Pump Discharge
T 280.7
F 1773.8
P 13403
T 280.7
F 1773.8
P 13403
T 280.8
F 1808.7
P 13351
T 280.8
F 1808.7
P 13351
P1
P3
P2
P4
FW
F 518.2
FW
F 563.4
Pump Picture
ROH#1
T 322.9
F 3582.1
P 12112
RIH#1
RIH#2
ROH#2
T 328.7
F 3552.3
P 12188
AVG CORE FLOW 7130.46
COOLANT Tavg 299.64
AVG FUEL T 713.83
▲T 52.3
RIH#1/#2, ROH#1/#2 Temp
RIH#1
RIH#2
ROH#1
ROH#2
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
ROH#1
ROH#2
Feed/Bleed Flow
fd
bd
RIH #1, RIH #2 Pressure
RIH#1
RIH#2
Reactor Power(%)
5:20:17 PM
5:21:43 PM
Resolution
Max Out
Max In
Time Scroll
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 99.99
Reactor Thermal Pwr(%) 99.97
Generator Output(%) 97.65
Primary Coolant Average ROH Pressure (KPa) 12150.30
Core Flow (kg/s) 7130.46
Main STM Press 6402.3
BOP STM Flow 1139.3
FW Flow 1081.6
Fuel Temp 713.8
Freeze
Run
Iterate
IC
Malf
Help

5.12. RC inventory bleed valve (CV5) fails open

This malfunction causes the reactor coolant bleed flow to reach the maximum. As a result, the bleed condenser level will increase. The immediate impact on the reactor coolant system is decreased coolant inventory in the system.

The pressurizer level will decrease, leading to decrease in pressurizer pressure. This is due to the fact that the vapor space in the pressurizer has been increased by reduced liquid mass in the pressurizer because of decreased inventory. The decreased pressurizer pressure will turn on the pressurizer heaters. The low pressurizer level will cause the inventory control system to increase the feed flow by opening the feed valve CV12. Over time, the coolant bleed flow and the coolant feed flow will balance out, and the transient will stabilize.

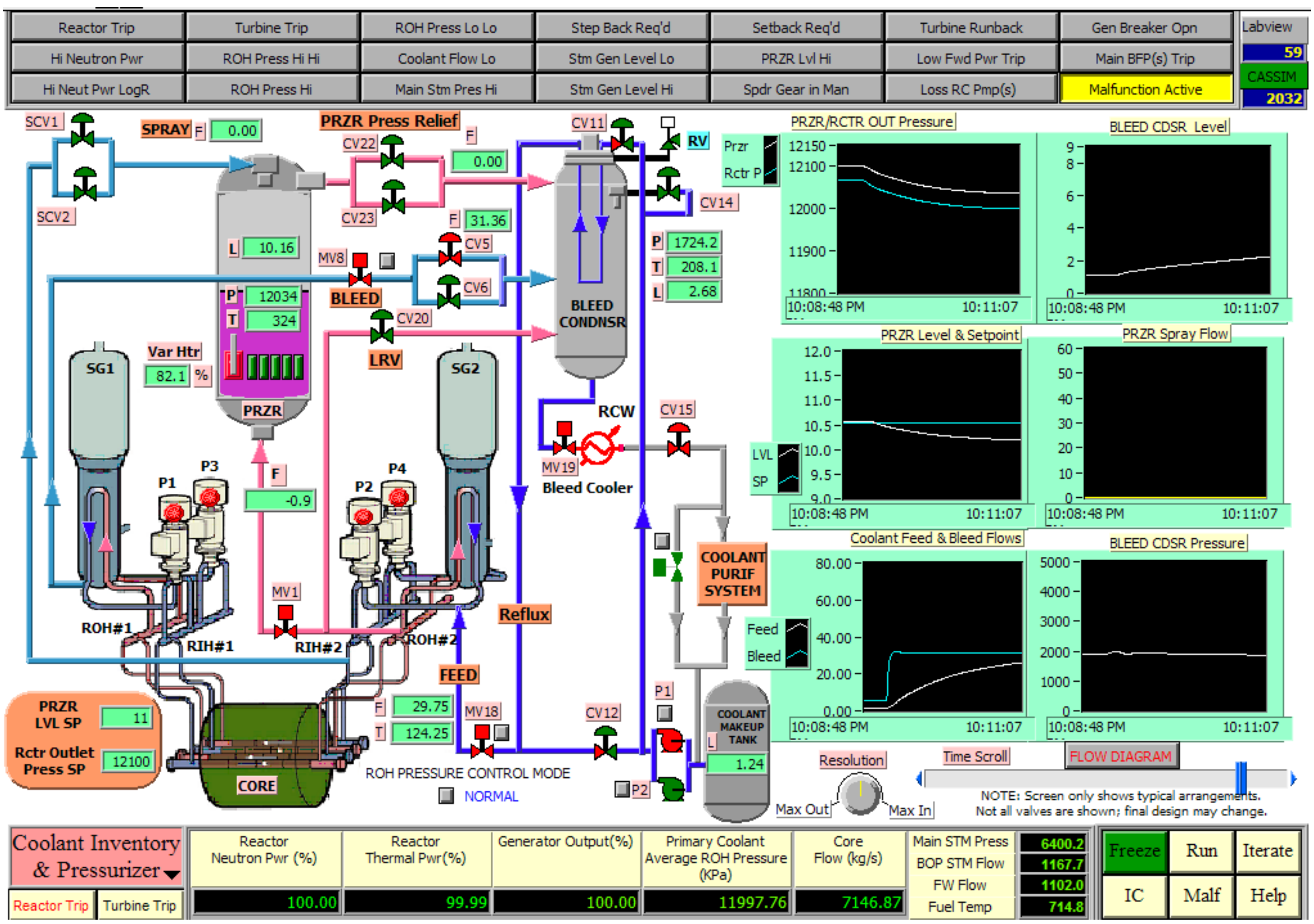
When this malfunction transient occurs:

⇒ Go to the "ACR coolant inventory and pressurizer" screen, observe that CV5 is 100 % open, and record the bleed flow (kg/s).

⇒ Observe the coolant pressure transient, and that the pressurizer heaters turn on.

⇒ Observe the pressurizer level and record the feed flow (kg/s).

⇒ Observe the bleed condenser level.



5.13. PRZR pressure relief valve (CV22) fails open

This malfunction transient causes depressurization of the pressurizer; with steam vapor going to the bleed condenser through the failed opened pressure relief valve CV22.

As the pressure is decreasing in the pressurizer, the electric heaters will be turned on. As well, pressurizer level will rise with decreasing pressure. The rising pressurizer level will cause the bleed flow to increase, trying to reduce coolant inventory in the pressurizer. Although the electric heaters are turned on, they cannot cope with the pressure loss caused by the failed CV22 venting to the bleed condenser. As a result, the coolant pressure keeps dropping during this transient, leading to reactor trip by low reactor outlet header pressure.

When this malfunction transient occurs:

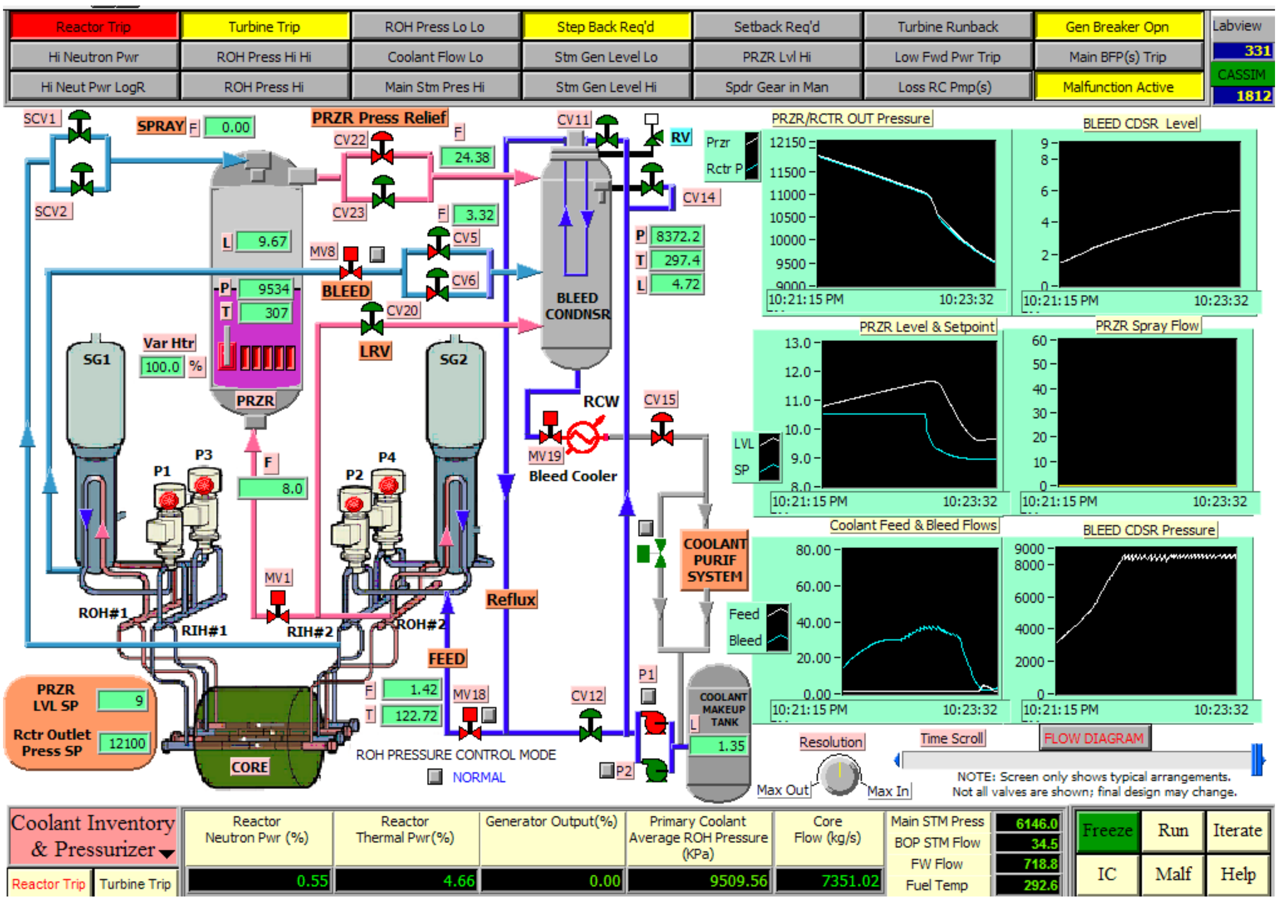
⇒ Go to "coolant inventory and pressurizer" screen. Observe that CV22 fails open.

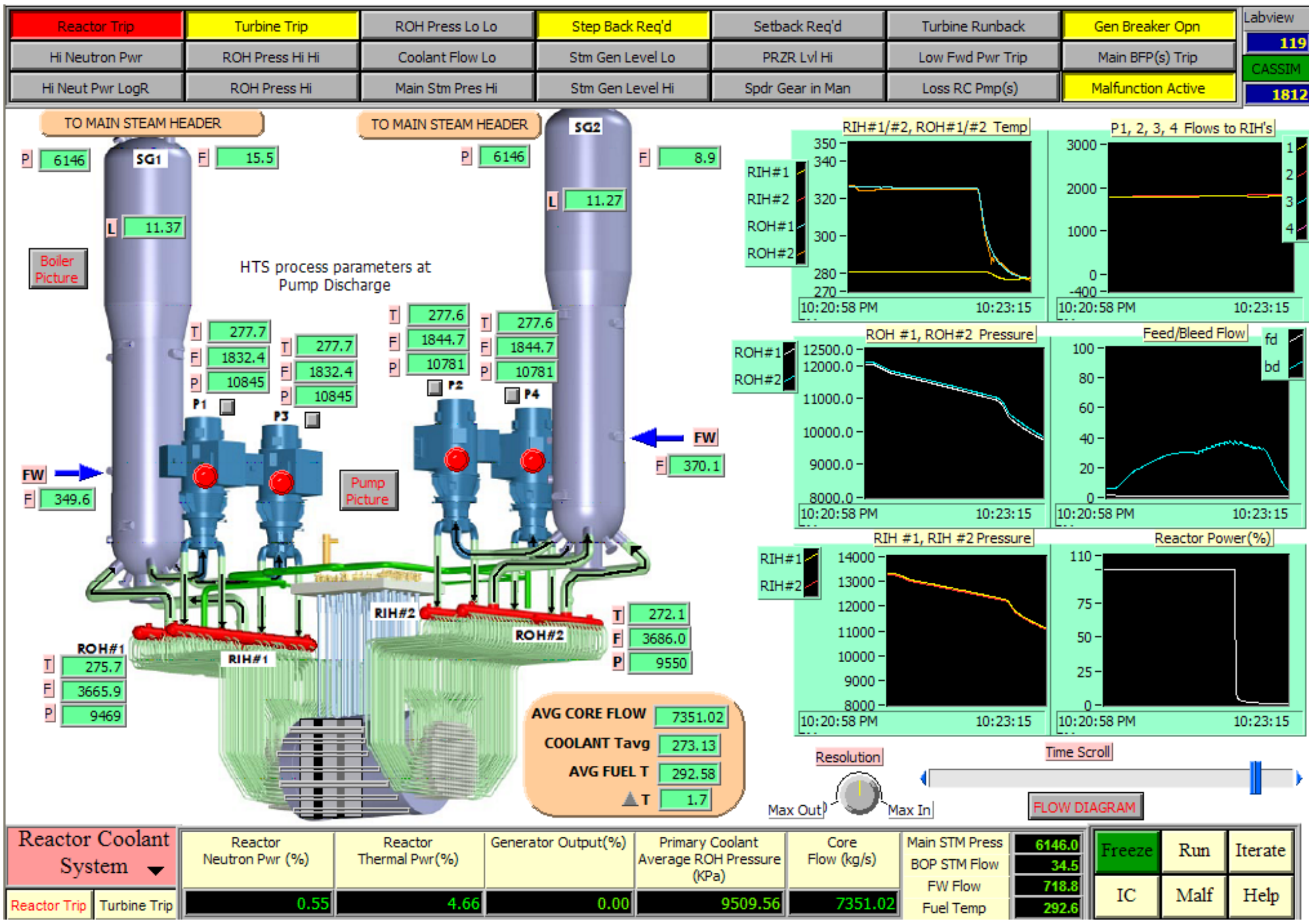
⇒ Observe the pressurizer pressure transient and level transient. Note that the electric heaters will turn on.

⇒ Record the bleed flow to the letdown condenser.

⇒ Continue to monitor coolant pressure, record when reactor trip occurs.

⇒ Monitor flow through CV22, as the pressurizer pressure continues to decrease. At what pressure will the flow from CV22 stop? Why?





Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
119
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
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
1812
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
P 6146
F 15.5
L 11.37
P 6146
F 8.9
L 11.27
Boiler Picture
HTS process parameters at Pump Discharge
T 277.7
F 1832.4
P 10845
T 277.7
F 1832.4
P 10845
T 277.6
F 1844.7
P 10781
T 277.6
F 1844.7
P 10781
P1
P2
P3
P4
FW
F 349.6
FW
F 370.1
Pump Picture
ROH#1
T 275.7
F 3665.9
P 9469
RIH#1
RIH#2
ROH#2
T 272.1
F 3686.0
P 9550
AVG CORE FLOW 7351.02
COOLANT Tavg 273.13
AVG FUEL T 292.58
T 1.7
RIH#1/#2, ROH#1/#2 Temp
RIH#1
RIH#2
ROH#1
ROH#2
350
340
320
300
280
270
10:20:58 PM
10:23:15
P1, 2, 3, 4 Flows to RIH's
3000
2000
1000
0
-400
1
2
3
4
10:20:58 PM
10:23:15
ROH #1, ROH#2 Pressure
ROH#1
ROH#2
12500.0
12000.0
11000.0
10000.0
9000.0
8000.0
10:20:58 PM
10:23:15
Feed/Bleed Flow
fd
bd
100
80
60
40
20
0
10:20:58 PM
10:23:15
RIH #1, RIH #2 Pressure
RIH#1
RIH#2
14000
13000
12000
11000
10000
9000
8000
10:20:58 PM
10:23:15
Reactor Power(%)
110
75
50
25
0
10:20:58 PM
10:23:15
Resolution
Max Out
Max In
Time Scroll
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
0.55
Reactor Thermal Pwr(%)
4.66
Generator Output(%)
0.00
Primary Coolant Average ROH Pressure (KPa)
9509.56
Core Flow (kg/s)
7351.02
Main STM Press
6146.0
BOP STM Flow
34.5
FW Flow
718.8
Fuel Temp
292.6
Freeze
Run
Iterate
IC
Malf
Help

5.14. One bank of MCA rods drops

This malfunction event will drop one bank of absorber (MCA) rods into core, imparting large negative reactivity into the core. This leads to large reduction of reactor power. The reactor regulating system (RRS) will immediately withdraw the ZCU rods for reactivity compensation.

However, because there is only limited reactivity available for the ZCU rods, even if they are fully withdrawn, their combined reactivity is insufficient to compensate for the negative reactivity imparted from dropping the bank of MCA rods into core.

As a result, the reactor power is decreasing; coolant pressure is decreasing. As well, the main steam pressure is decreasing, leading to turbine runback, and a subsequent turbine trip on zero forward power. The transient will evolve with the reactor power slowly decreasing to zero, due to Xenon buildup.

When this malfunction transient occurs:

⇒ Go to “control rods & SD rods” screen, observe that one bank of MCA Rods has been dropped into the core.

⇒ Record the overall reactivity change and reactor power, immediately after the malfunction is initiated.

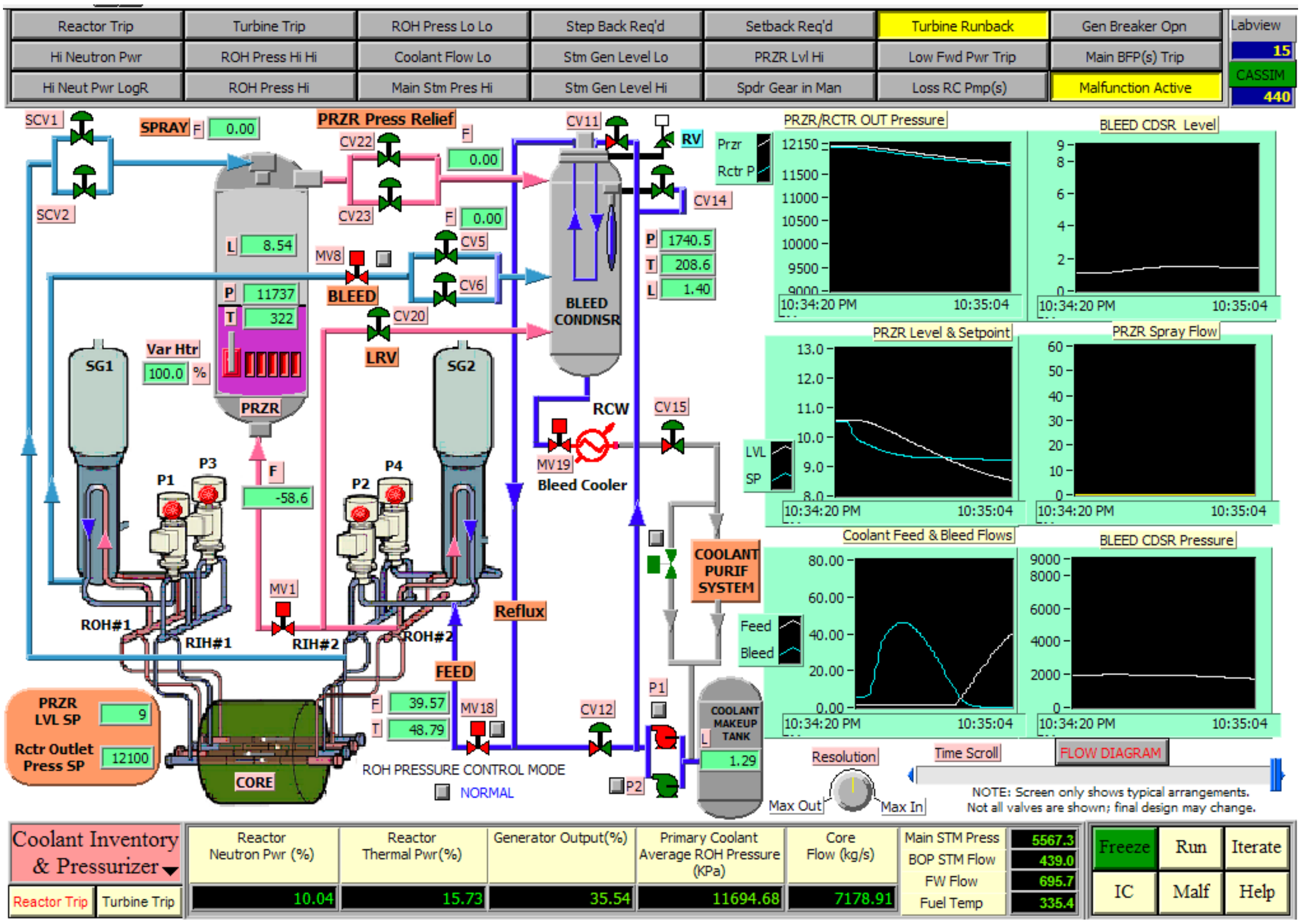
⇒ Go to “reactor power control” screen, observe that the ZCU rods are withdrawing. Record the reactivity (mk) change.

⇒ Go to “reactor coolant system” screen and observe the coolant pressure transient.

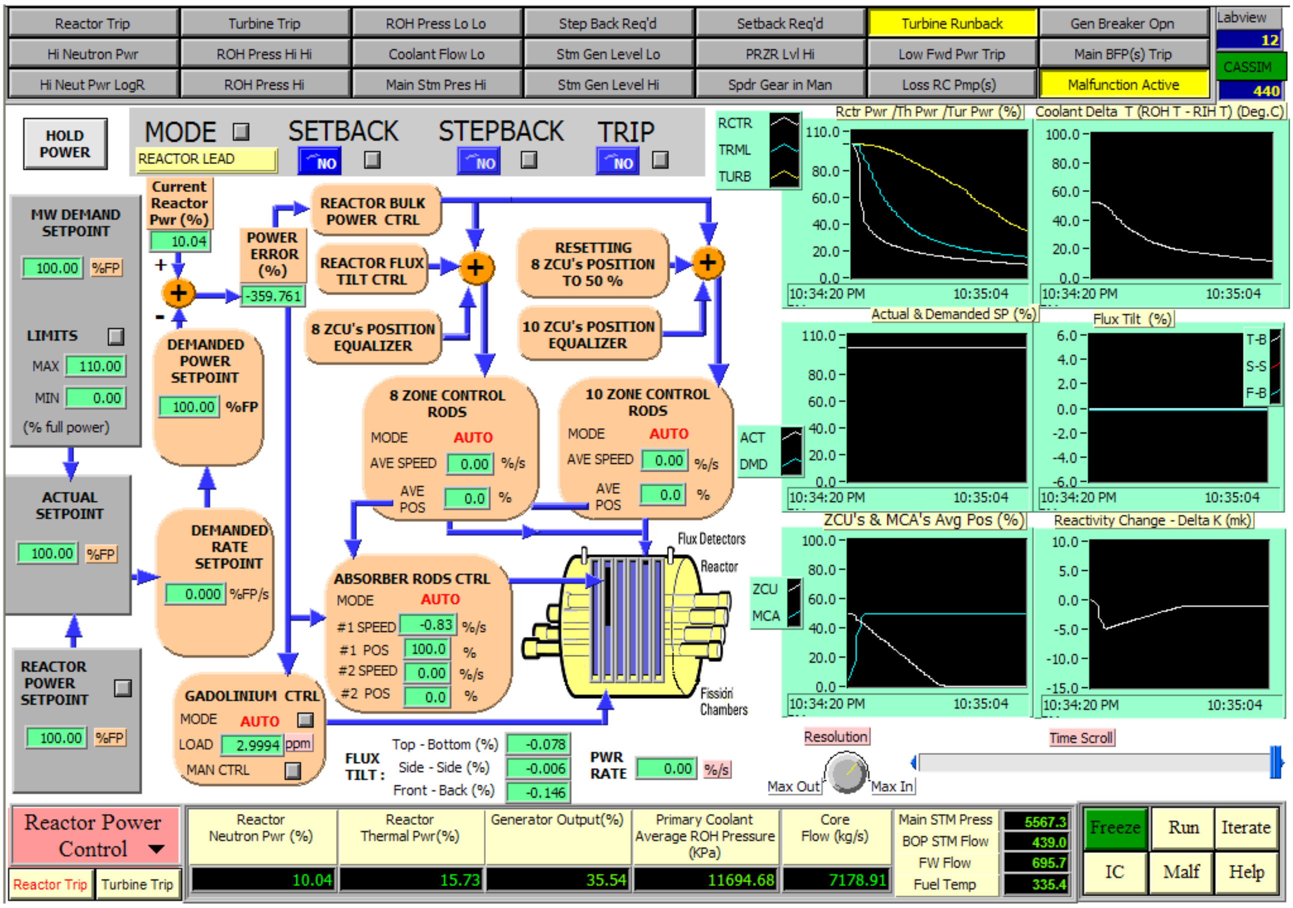
⇒ Go to “turbine generator” screen; observe the main steam pressure transient. Note the turbine runback is in progress.

⇒ Go back to “control rods & SD rods” screen; record the overall reactivity change again. Record the reactor power.

⇒ Describe and explain the long-term evolution of this transient.



Reactor Trip
Turbine Trip
ROH Press 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
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
Labview
15
CASSIM
440
SCV1
SCV2
SPRAY
PRZR Press Relief
CV22
CV23
CV5
CV6
CV20
CV11
RV
CV14
MV8
BLEED
LRV
BLEED CONDNSR
Var Htr
PRZR
SG1
SG2
P1
P2
P3
P4
ROH#1
RIH#1
RIH#2
ROH#2
MV1
Reflux
FEED
MV18
CV12
RCW
CV15
MV19
Bleed Cooler
COOLANT PURIF SYSTEM
COOLANT MAKEUP TANK
PRZR LVL SP
Rctr Outlet Press SP
CORE
ROH PRESSURE CONTROL MODE
NORMAL
PRZR/RCTR OUT Pressure
BLEED CDSR Level
PRZR Level & Setpoint
PRZR Spray Flow
Coolant Feed & Bleed Flows
BLEED CDSR Pressure
Resolution
Time Scroll
FLOW DIAGRAM
Max Out
Max In
NOTE: Screen only shows typical arrangements.
Not all valves are shown; final design may change.
Coolant Inventory & Pressurizer
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Primary Coolant Average ROH Pressure (KPa)
Core Flow (kg/s)
Main STM Press
BOP STM Flow
FW Flow
Fuel Temp
Freeze
Run
Iterate
IC
Malf
Help
10.04
15.73
35.54
11694.68
7178.91



HOLD POWER
MODE
SETBACK
STEPBACK
TRIP
REACTOR LEAD
MW DEMAND SETPOINT
LIMITS
ACTUAL SETPOINT
REACTOR POWER SETPOINT
Current Reactor Pwr (%)
POWER ERROR (%)
DEMANDED POWER SETPOINT
DEMANDED RATE SETPOINT
REACTOR BULK POWER CTRL
REACTOR FLUX TILT CTRL
8 ZCU's POSITION EQUALIZER
RESETTING 8 ZCU's POSITION TO 50 %
10 ZCU's POSITION EQUALIZER
8 ZONE CONTROL RODS
10 ZONE CONTROL RODS
ABSORBER RODS CTRL
GADOLINIUM CTRL
FLUX TILT :
Top - Bottom (%)
Side - Side (%)
Front - Back (%)
PWR RATE
Flux Detectors
Reactor
Fission Chambers
Rctr Pwr /Th Pwr /Tur Pwr (%)
Coolant Delta T (ROH T - RIH T) (Deg.C)
Actual & Demanded SP (%)
Flux Tilt (%)
ZCU's & MCA's Avg Pos (%)
Reactivity Change - Delta K (mk)
Reactor Power Control

5.15. All MCA rods "stuck" to MANUAL

This malfunction event impairs the capability of the reactor power control system to reduce reactor power at the *desired* rate during power maneuvering, due to the loss of control for the MCA rods.

When this malfunction transient occurs:

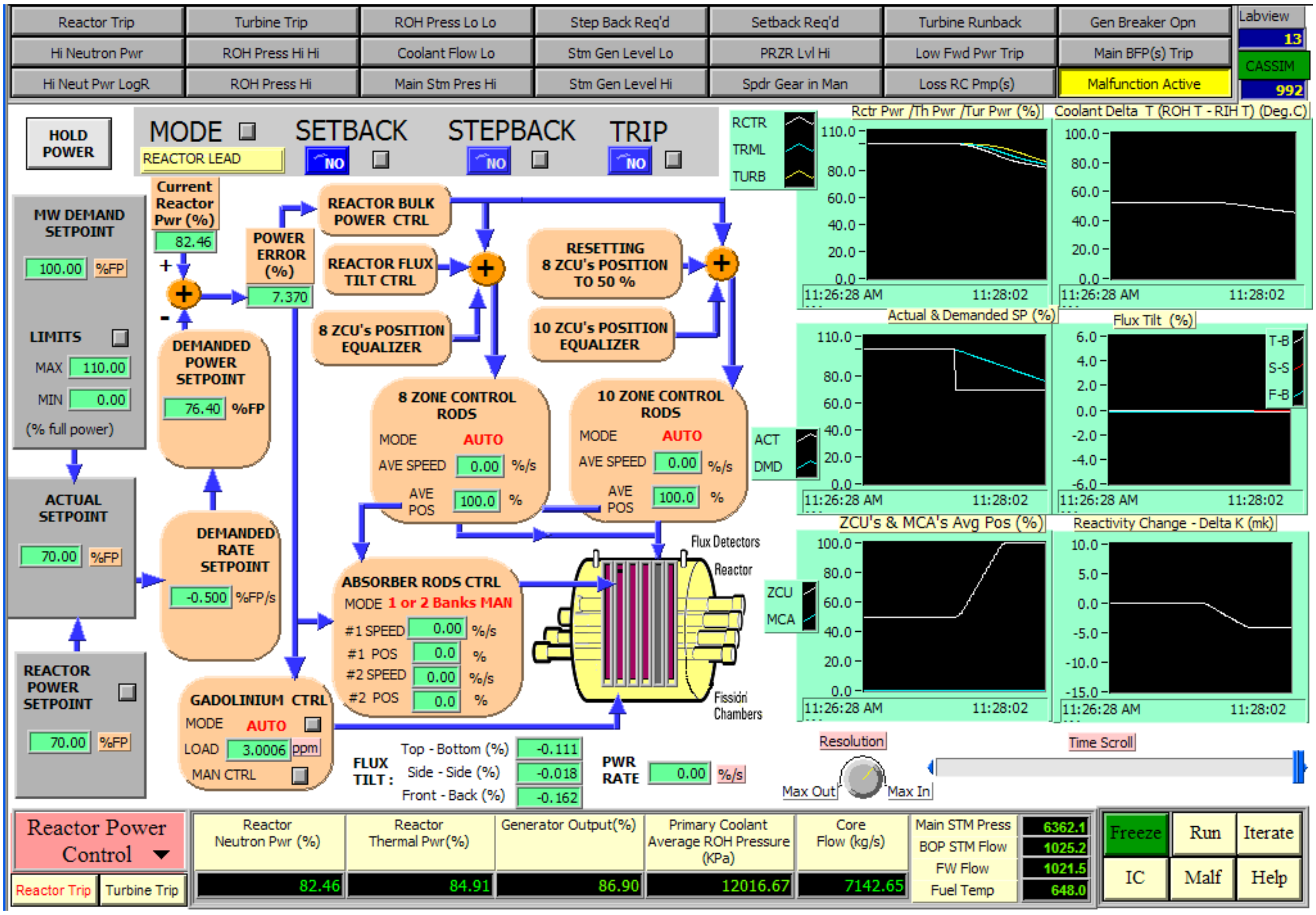
⇒ Go to "reactor power control" screen, set the mode to "reactor lead"

⇒ Enter target reactor power 70 %, and rate 0.5 % per sec. Accept the inputs.

⇒ Go to "ACR Control Rods & SD Rods" screen, observe the movement of ZCU rods, and the MCA rods, as the reactor power is decreasing towards the target power 70 %.

⇒ Monitor the Power Error (%) – note the peak positive power error (%) during this event. Discuss the behavior of the Power Error (%) - how it increases and then decreases ?

⇒ As you may notice, all the ZCUs will be inserted fully in core initially, where is the source of negative reactivity to reduce the reactor power to the target 70% FP ? Discuss.



5.16. Reactor setback/stepback both fail

This malfunction event impairs the first line of protective action initiated by the reactor control system (RRS), to decrease reactor power, in response to process conditions that exceed the alarm limits.

However, the reactor shutdown system (SDS) is always poised to act, should those alarm limits reach the trip setpoint.

⇒ Go to "control rods & SD rods" screen; insert the malfunction "reactor setback/stepback both fail".

⇒ Use the pop-up at bottom left to trip turbine.

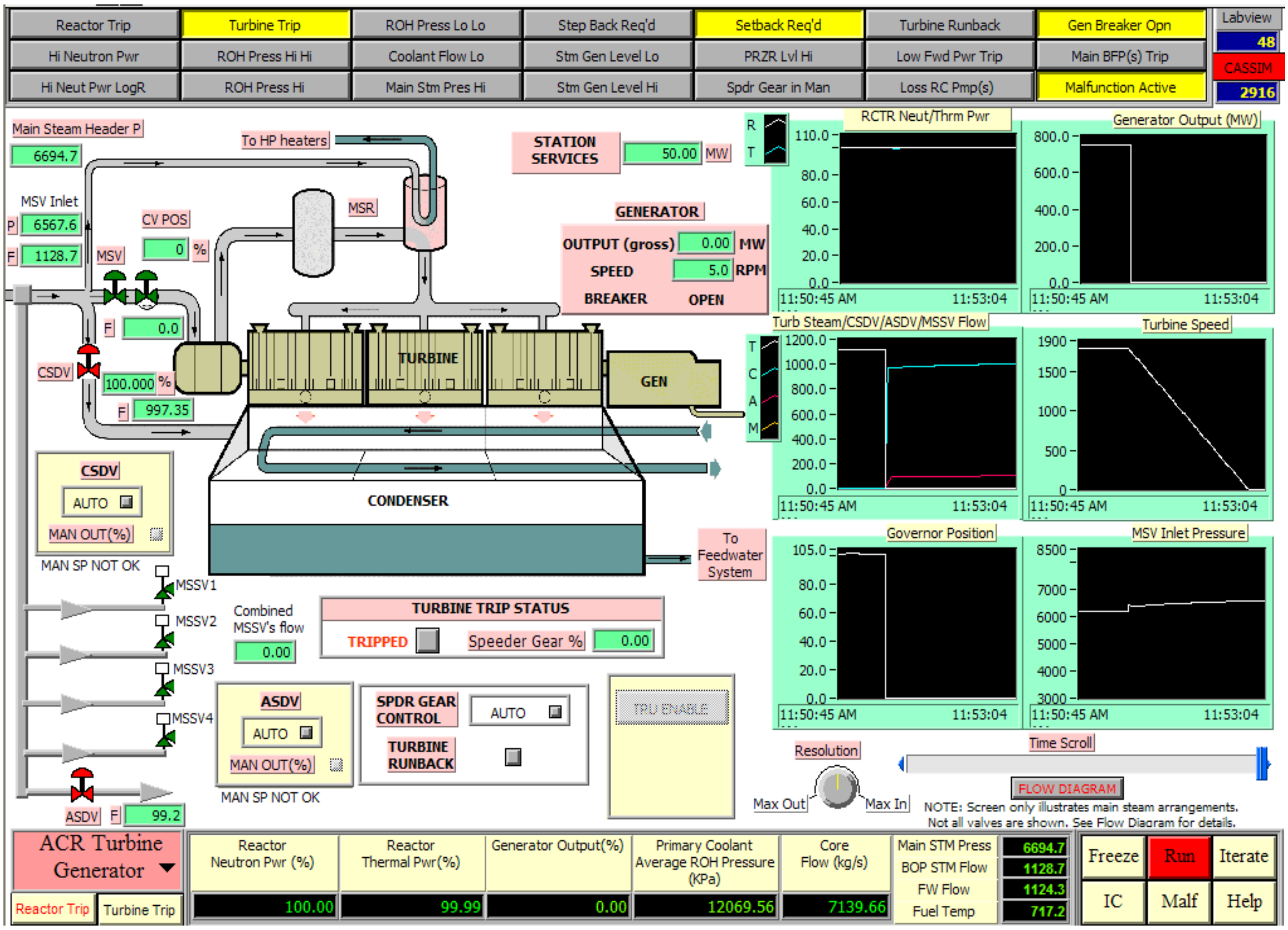
⇒ Observe that due to the malfunction, the reactor setback cannot be initiated; therefore ZCU and MCA Control Rods will not respond to turbine trip. Record reactor power after turbine trip.

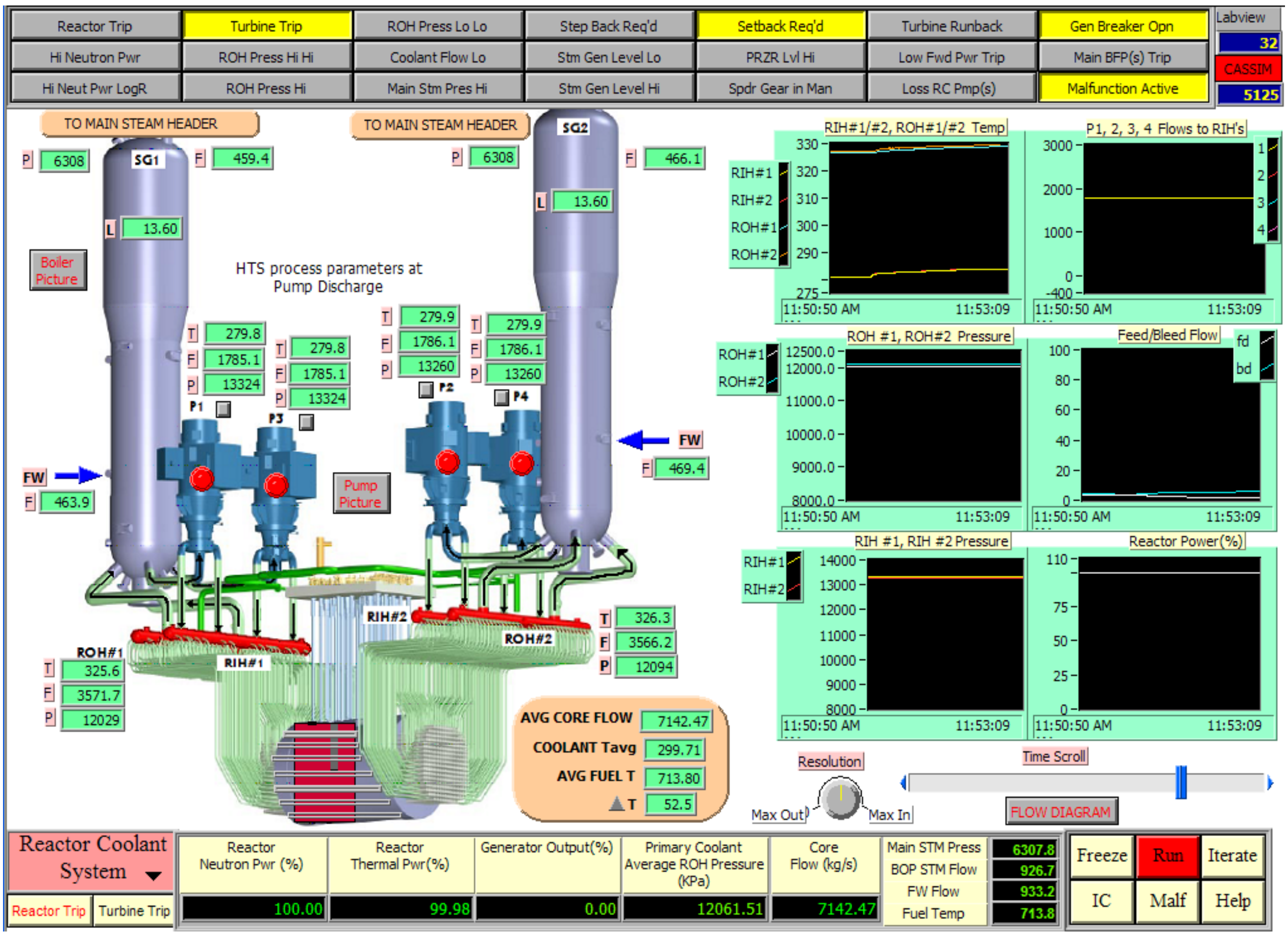
⇒ Go to "turbine generator" screen; observe the main steam pressure transient. The turbine bypass valve (CSDV, ASDV) should open to relieve steam pressure.

⇒ Go to "reactor coolant system" screen; observe the transient in coolant pressure and temperature.

⇒ With the reactor setback/stepback both failed, is the safety margin (e.g. coolant overpressure; fuel temperature, DNB etc.) of the system being challenged on a major transient like a turbine trip, on loss of first line of reactor protective action – malfunction of both the reactor setback and stepback?

⇒ Discuss the importance of steam bypass system, as well as the main steam safety features.


Reactor Trip
Turbine Trip
ROH Press 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
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
Labview
48
CASSIM
2916
Main Steam Header P
6694.7
To HP heaters
STATION SERVICES
50.00 MW
MSV Inlet
6567.6
1128.7
MSR
CV POS
0 %
MSV
GENERATOR
OUTPUT (gross) 0.00 MW
SPEED 5.0 RPM
BREAKER OPEN
0.0
CSDV
100.000 %
997.35
TURBINE
GEN
CSDV
AUTO
MAN OUT(%)
MAN SP NOT OK
CONDENSER
To Feedwater System
MSSV1
MSSV2
MSSV3
MSSV4
Combined MSSV's flow
0.00
TURBINE TRIP STATUS
TRIPPED
Speeder Gear %
0.00
ASDV
AUTO
MAN OUT(%)
MAN SP NOT OK
SPDR GEAR CONTROL
AUTO
TURBINE RUNBACK
TRU ENABLE
ASDV
99.2
RCTR Neut/Thrm Pwr
Generator Output (MW)
Turb Steam/CSDV/ASDV/MSSV Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
11:50:45 AM
11:53:04
Resolution
Time Scroll
Max Out
Max In
FLOW DIAGRAM
NOTE: Screen only illustrates main steam arrangements. Not all valves are shown. See Flow Diagram for details.
ACR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
100.00
Reactor Thermal Pwr(%)
99.99
Generator Output(%)
0.00
Primary Coolant Average ROH Pressure (KPa)
12069.56
Core Flow (kg/s)
7139.66
Main STM Press 6694.7
BOP STM Flow 1128.7
FW Flow 1124.3
Fuel Temp 717.2
Freeze
Run
Iterate
IC
Malf
Help


Reactor Trip
Turbine Trip
ROH Press 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
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
Labview
32
CASSIM
5125
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
6308
459.4
6308
466.1
13.60
13.60
Boiler Picture
HTS process parameters at Pump Discharge
Pump Picture
P1
P2
P3
P4
FW
463.9
FW
469.4
ROH#1
ROH#2
RIH#1
RIH#2
AVG CORE FLOW 7142.47
COOLANT Tavg 299.71
AVG FUEL T 713.80
ΔT 52.5
RIH#1/#2, ROH#1/#2 Temp
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
Feed/Bleed Flow
RIH #1, RIH #2 Pressure
Reactor Power(%)
11:50:50 AM
11:53:09
Resolution
Time Scroll
Max Out
Max In
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
100.00
Reactor Thermal Pwr(%)
99.98
Generator Output(%)
0.00
Primary Coolant Average ROH Pressure (KPa)
12061.51
Core Flow (kg/s)
7142.47
Main STM Press 6307.8
BOP STM Flow 926.7
FW Flow 933.2
Fuel Temp 713.8
Freeze
Run
Iterate
IC
Malf
Help

5.17. Loss of one RC pump P1

This malfunction event causes one primary heat transport (PHT) pump P1 to trip off line, due to pump failure such as rotor failure. The loss of one HTS pump will immediately initiate reactor power stepback.

⇒ Go to “reactor coolant system” screen; insert the malfunction for “loss of one PHT Pump P1”. Observe that PHT Pump 1 is tripped off, and the coolant discharge flow for P1 is decreasing rapidly. Observe coolant flow in the other PHT pumps.

⇒ Observe that reactor power is stepped back. Record the reactor power after malfunction is initiated.

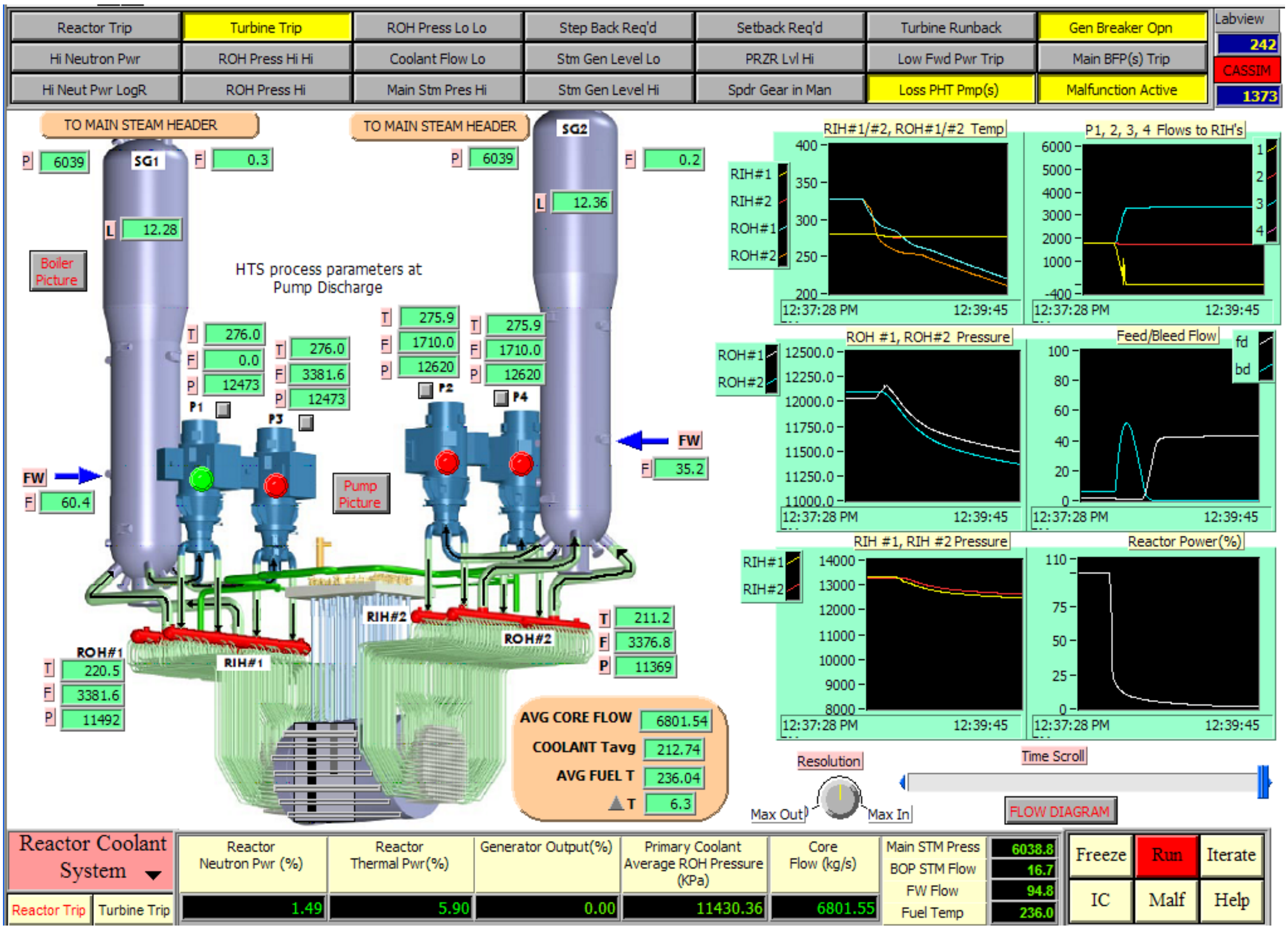
⇒ Observe the coolant pressure and temperature transients.

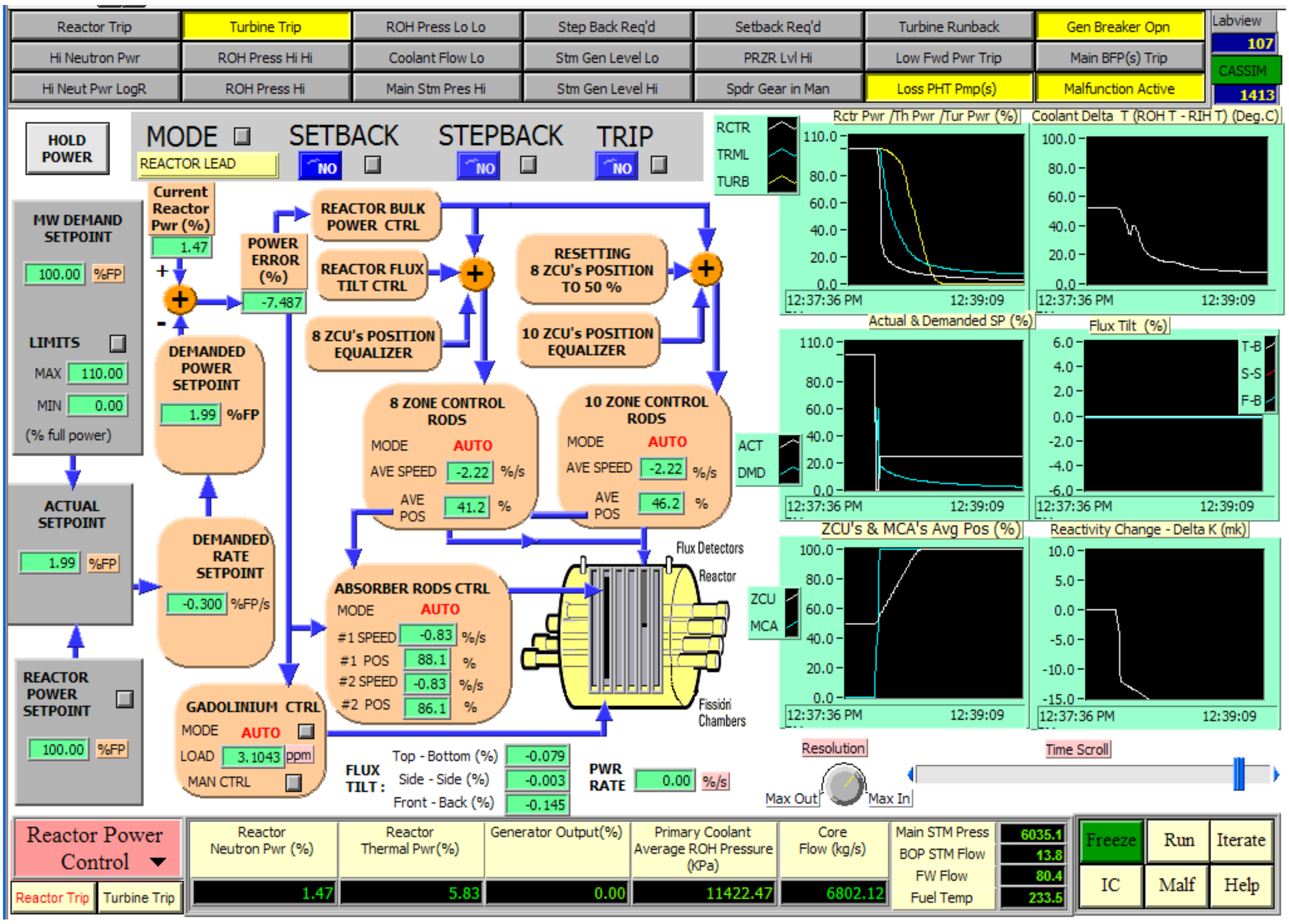
⇒ Observe the turbine steam flow and turbine power.

⇒ Repeat the malfunction event again with the use of the “reactor coolant system” screen, but before doing so, first insert another malfunction for “reactor setback & stepback both failed”. The purpose is to study how the system thermal margin is challenged without the initial reactor power stepback.

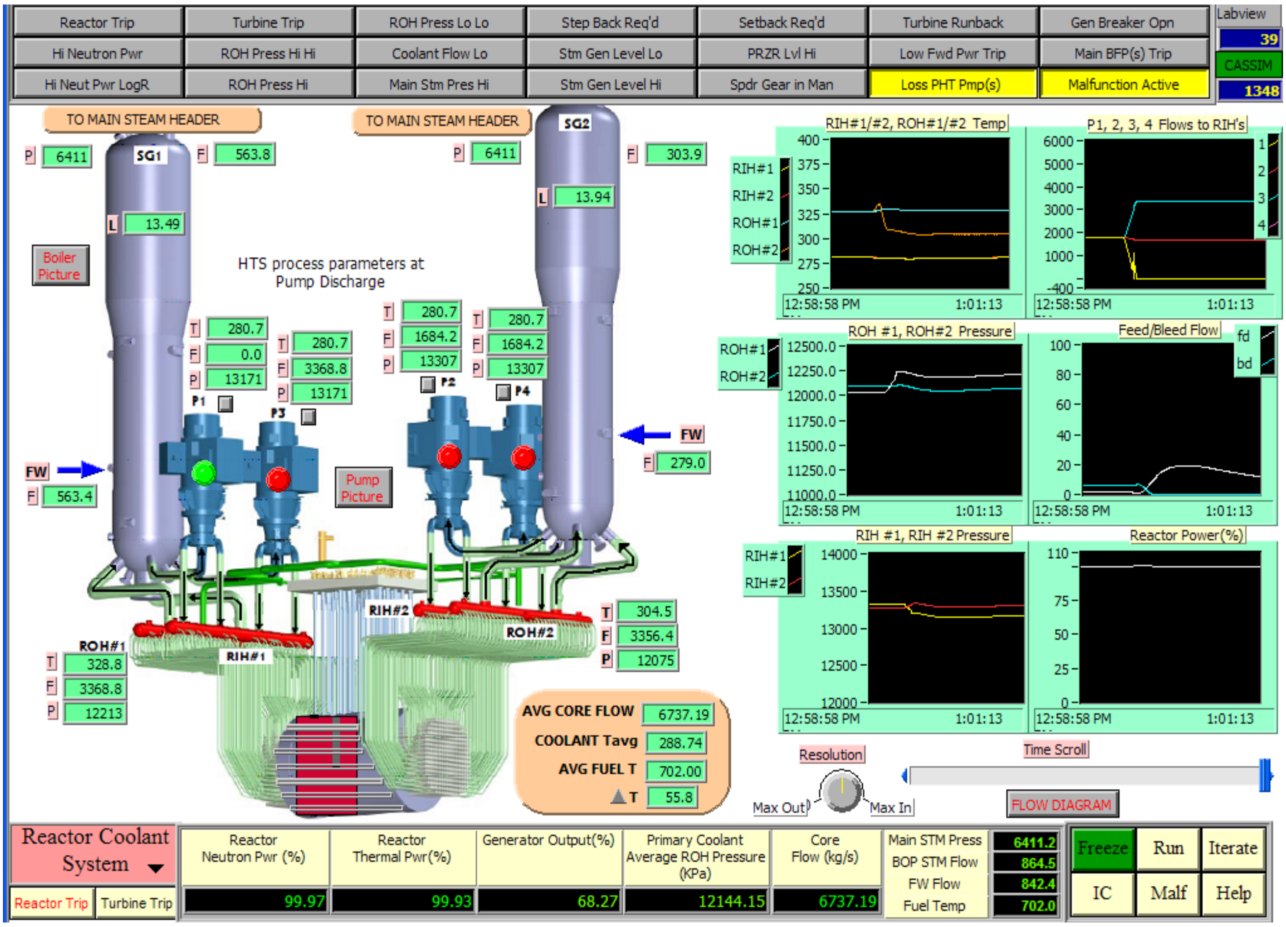
⇒ Observe the reactor power transient, coolant pressure and temperature transients, reactor neutron power, reactor thermal power, turbine power transients. Describe and explain the difference in responses, when compared with the previous malfunction transient.

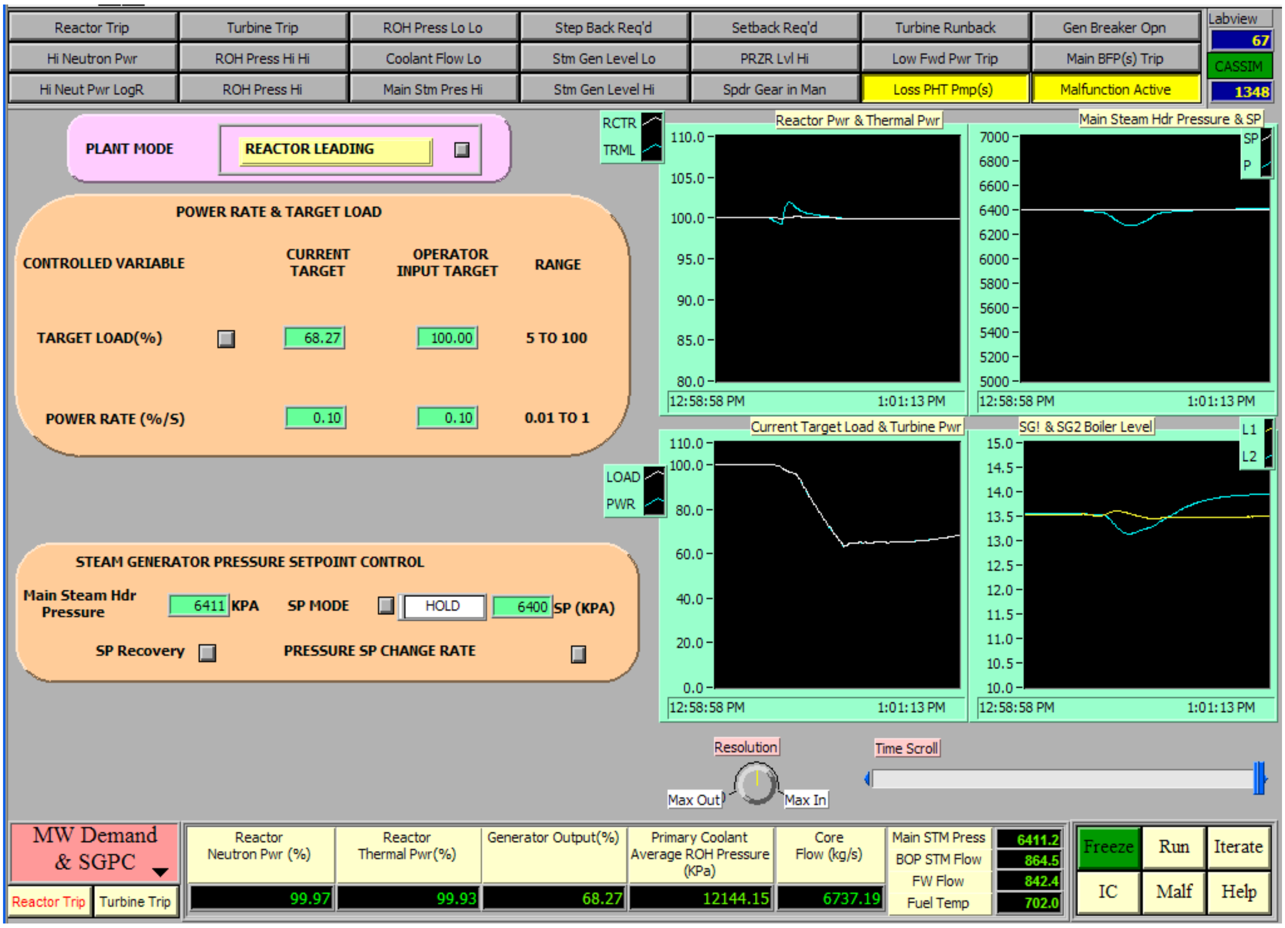
⇒ Discuss the thermal margin challenge in these cases, and how the safety and control systems can cope with these challenges.


Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
242
CASSIM
1373
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
Spdr Gear in Man
Loss PHT Pmp(s)
Malfunction Active
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
Boiler Picture
HTS process parameters at Pump Discharge
Pump Picture
FW
FW
ROH#1
ROH#2
RIH#1
RIH#2
P1
P2
P3
P4
AVG CORE FLOW 6801.54
COOLANT Tavg 212.74
AVG FUEL T 236.04
ΔT 6.3
RIH#1/#2, ROH#1/#2 Temp
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
Feed/Bleed Flow
RIH #1, RIH #2 Pressure
Reactor Power(%)
12:37:28 PM
12:39:45
Resolution
Time Scroll
Max Out
Max In
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 1.49
Reactor Thermal Pwr(%) 5.90
Generator Output(%) 0.00
Primary Coolant Average ROH Pressure (KPa) 11430.36
Core Flow (kg/s) 6801.55
Main STM Press 6038.8
BOP STM Flow 16.7
FW Flow 94.8
Fuel Temp 236.0
Freeze
Run
Iterate
IC
Malf
Help


Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
107
CASSIM
1413
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
Spdr Gear in Man
Loss PHT Pmp(s)
Malfunction Active
HOLD POWER
MODE
REACTOR LEAD
SETBACK
STEPBACK
TRIP
MW DEMAND SETPOINT 100.00 %FP
LIMITS
MAX 110.00
MIN 0.00
(% full power)
ACTUAL SETPOINT 1.99 %FP
REACTOR POWER SETPOINT 100.00 %FP
Current Reactor Pwr (%) 1.47
POWER ERROR (%) -7.487
DEMANDED POWER SETPOINT 1.99 %FP
DEMANDED RATE SETPOINT -0.300 %FP/s
REACTOR BULK POWER CTRL
REACTOR FLUX TILT CTRL
8 ZCU's POSITION EQUALIZER
RESETTING 8 ZCU's POSITION TO 50 %
10 ZCU's POSITION EQUALIZER
8 ZONE CONTROL RODS
MODE AUTO
AVE SPEED -2.22 %/s
AVE POS 41.2 %
10 ZONE CONTROL RODS
MODE AUTO
AVE SPEED -2.22 %/s
AVE POS 46.2 %
ABSORBER RODS CTRL
MODE AUTO
#1 SPEED -0.83 %/s
#1 POS 88.1 %
#2 SPEED -0.83 %/s
#2 POS 86.1 %
GADOLINIUM CTRL
MODE AUTO
LOAD 3.1043 ppm
MAN CTRL
Flux Detectors
Reactor
Fission Chambers
FLUX TILT :
Top - Bottom (%) -0.079
Side - Side (%) -0.003
Front - Back (%) -0.145
PWR RATE 0.00 %/s
Rctr Pwr /Th Pwr /Tur Pwr (%)
Coolant Delta T (ROH T - RIH T) (Deg.C)
Actual & Demanded SP (%)
Flux Tilt (%)
ZCU's & MCA's Avg Pos (%)
Reactivity Change - Delta K (mk)
12:37:36 PM
12:39:09
Resolution
Time Scroll
Max Out
Max In
Reactor Power Control
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 1.47
Reactor Thermal Pwr(%) 5.83
Generator Output(%) 0.00
Primary Coolant Average ROH Pressure (KPa) 11422.47
Core Flow (kg/s) 6802.12
Main STM Press 6035.1
BOP STM Flow 13.8
FW Flow 80.4
Fuel Temp 233.5
Freeze
Run
Iterate
IC
Malf
Help

Transients with additional malfunction: both Reactor Setback/Stepback failed.





5.18. Loss of 2 PHT pumps

This malfunction event is a more serious accident than that described in Section 5.17, because of drastic reduction of coolant flow in losing two PHT pumps.

⇒ Go to “reactor coolant system” screen; insert the malfunction for “Loss of 2 PHT pumps in Loop 1”. Observe that PHT Pumps 1 and 2 are tripped off, and the coolant flow is decreasing rapidly. Observe coolant flow in the other PHT pumps.

⇒ Observe that reactor power is stepped back. Record the reactor power after the malfunction is initiated.

⇒ Observe the coolant pressure and temperature transients, bleed and feed flow transients, reactor power, thermal power and turbine power transients.

⇒ Repeat the malfunction event again with the use of the “reactor coolant system” screen, but before doing so, first insert the malfunction for “Loss of PHT Pump P1”. Then on the “reactor coolant screen”, manually turn off PHT pump P3. The purpose is to study how the system thermal margin is challenged without two PHT pumps on the same loop supplying primary coolant to RIH #1. As well, what reactor protection is activated to mitigate the event.

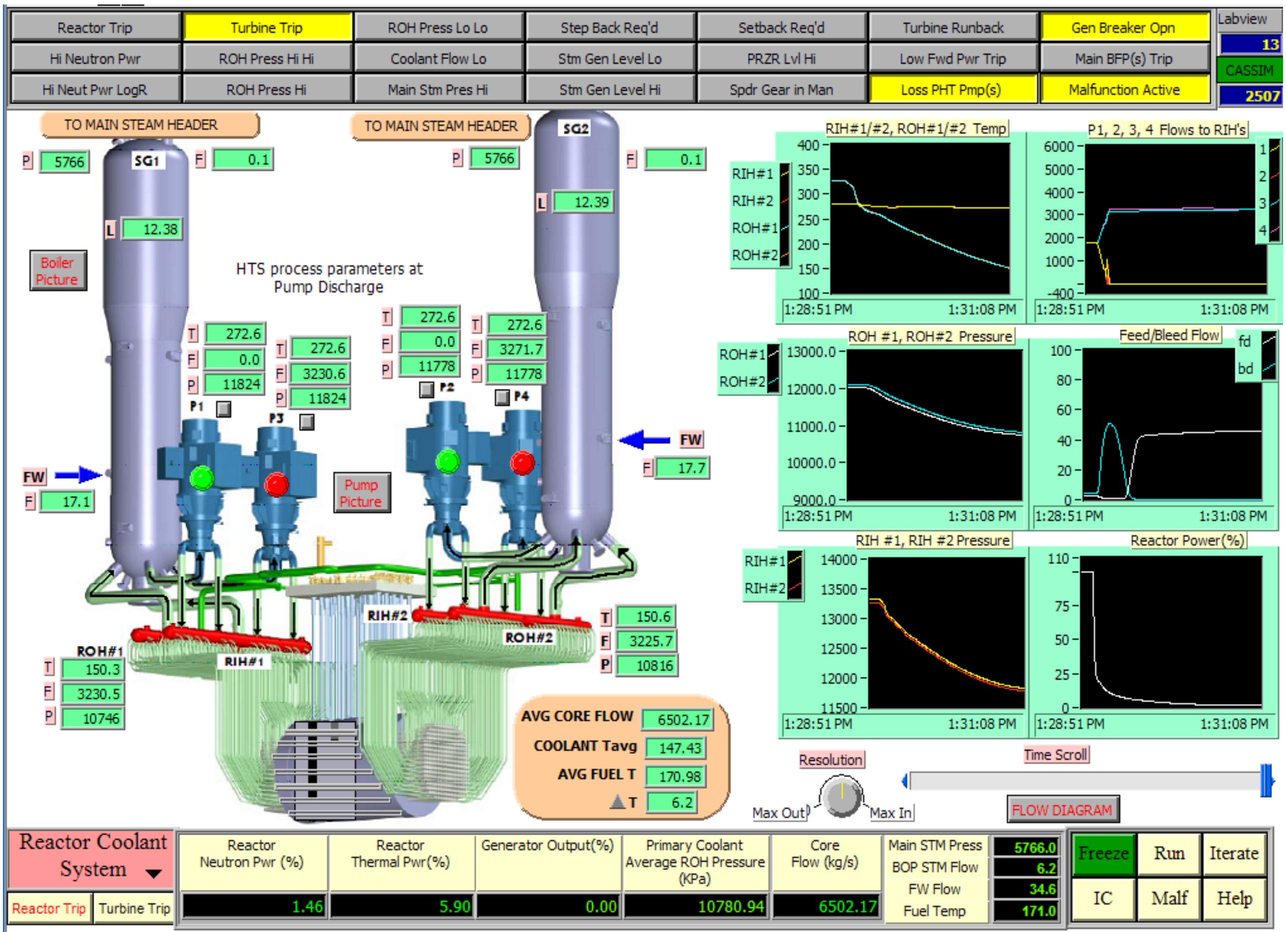
⇒ Observe the reactor power transient, coolant pressure and temperature transients, reactor neutron power, reactor thermal power, turbine power transients. Describe and explain the difference in responses, when compared with the previous malfunction transient.

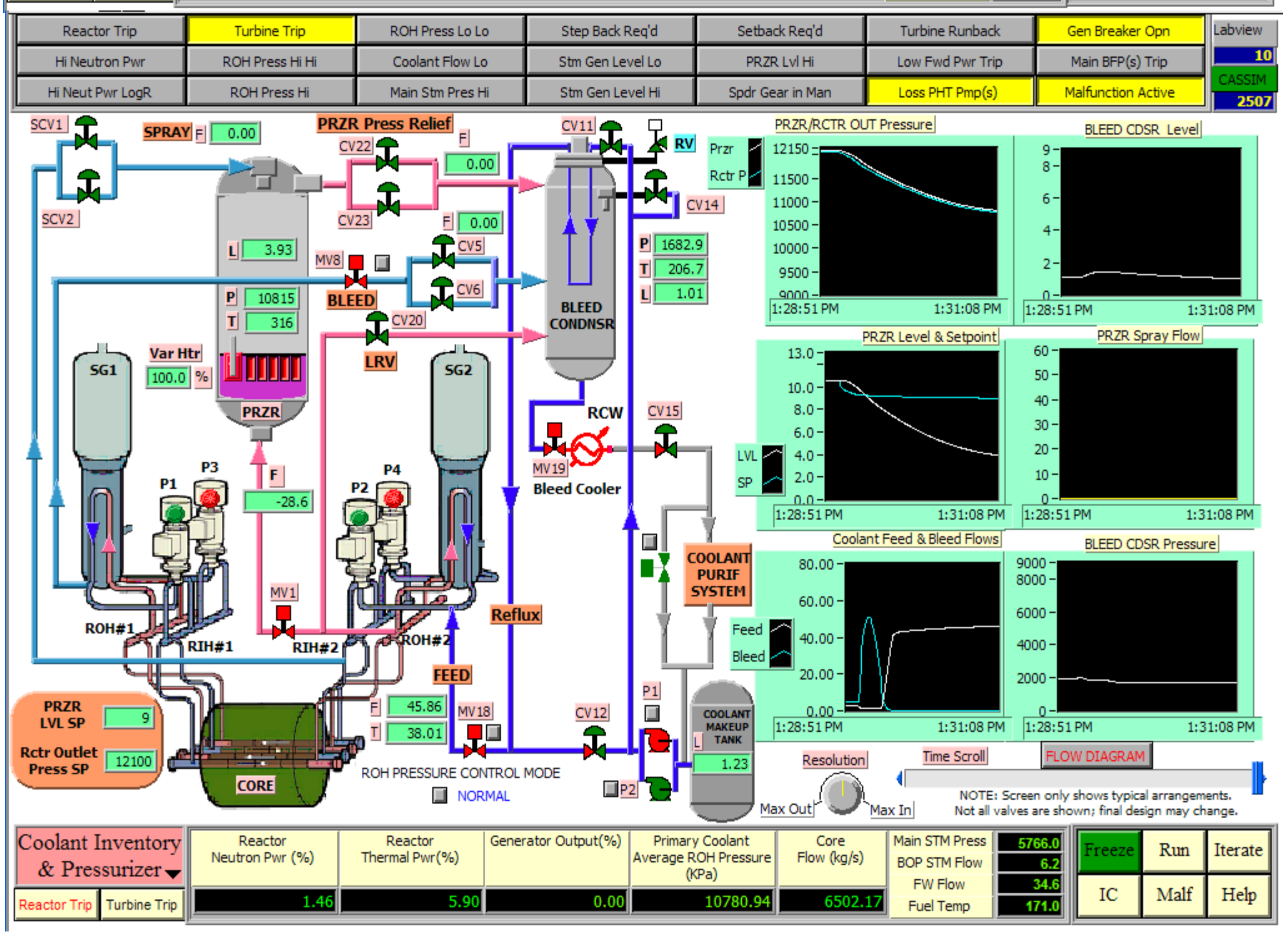
⇒ Repeat the malfunction event again, with the use of the “reactor coolant system” screen, but this time, before doing so, first insert the malfunction for “reactor setback & stepback both failed”. The purpose is to study how the system thermal margin is challenged without the initial reactor power stepback.

⇒ Observe the reactor power transient, coolant pressure and temperature transients. Describe and explain the difference in responses, when compared with the previous malfunction transient.

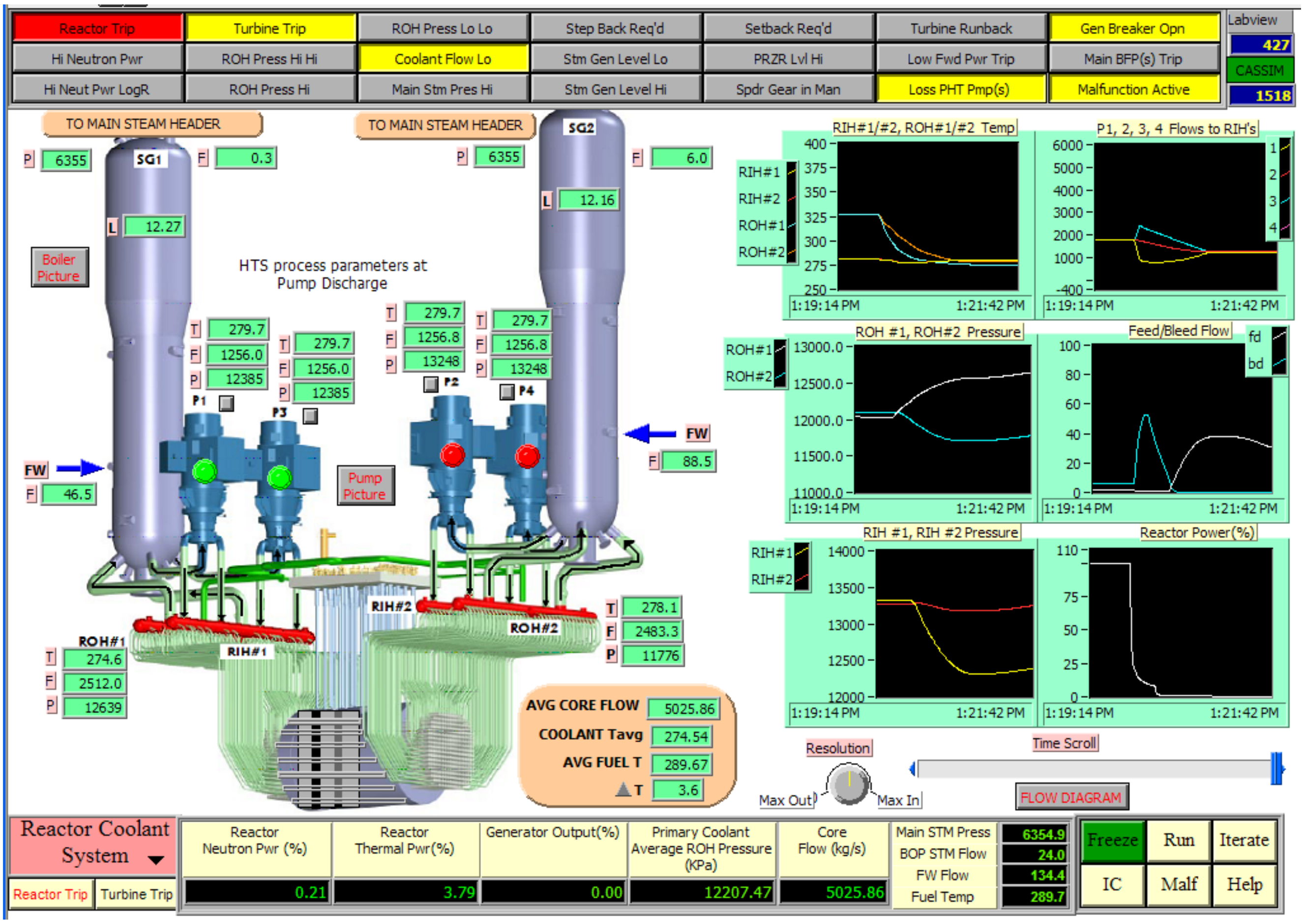
⇒ Discuss the thermal margin challenge in these cases, and how the safety and control systems can cope with these challenges.

Transients for the loss of PHT P1 and P2:





Transients for the loss of PHT P1 and P3:



5.19. 100% main steam header break

This malfunction event causes steam pipe break in the main steam line before the main steam stop valve (MSV) outside containment, leading to rapid depressurizaton of the main steam pressure. Turbine generator will be runback rapidly and will be tripped by zero forward power. The turbine trip initiates a reactor power setback.

The pipe break also results in increase in steam flow from the steam generators, leading to increase in heat removal from the reactor coolant system. Therefore, coolant temperature and pressure will drop.

⇒ Go to “reactor coolant system” screen; insert the malfunction “100 % main steam header break”. Observe and record the steam flows from the steam generators, and the main steam pressure.

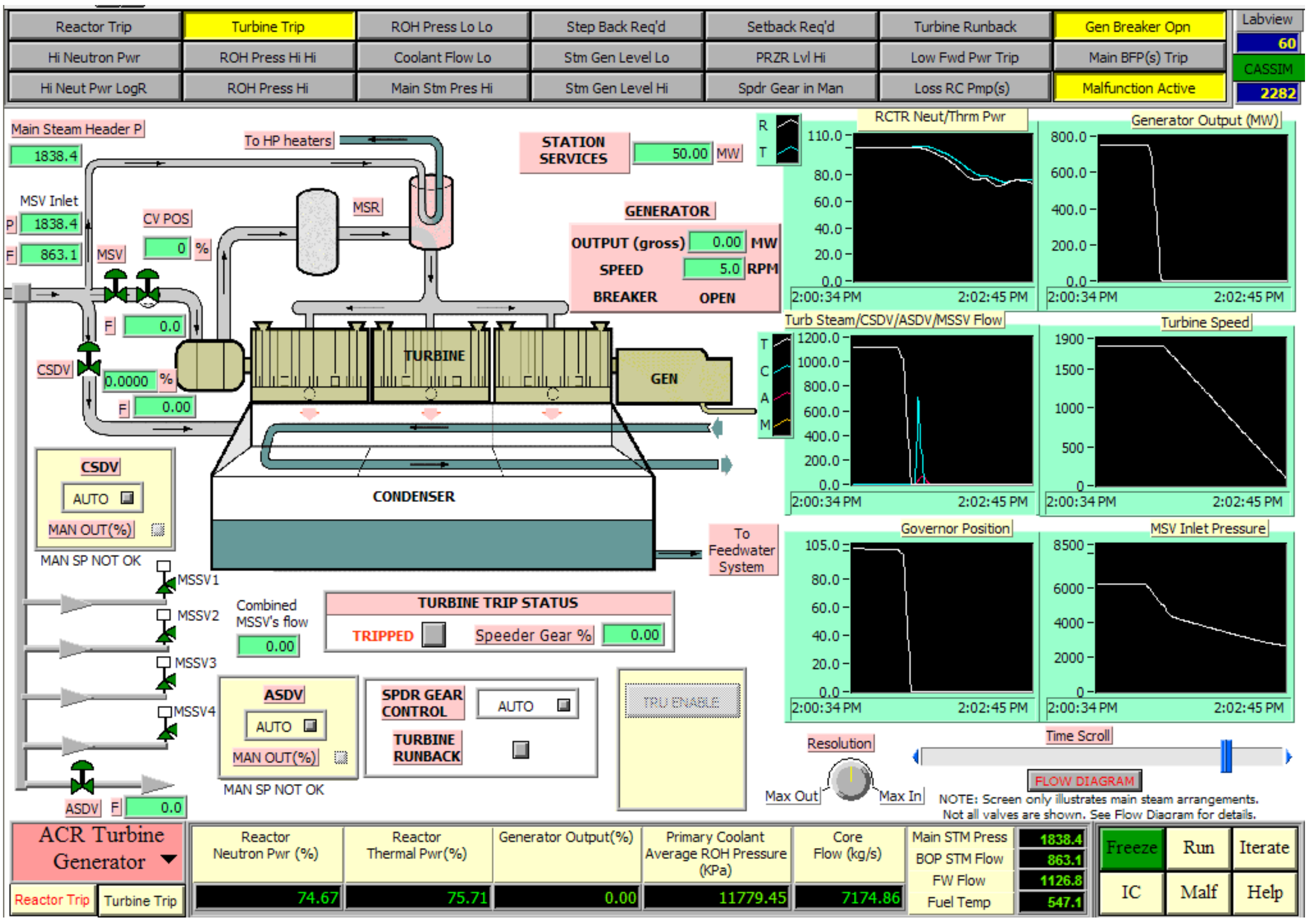
⇒ Observe the coolant temperature and pressure responses.

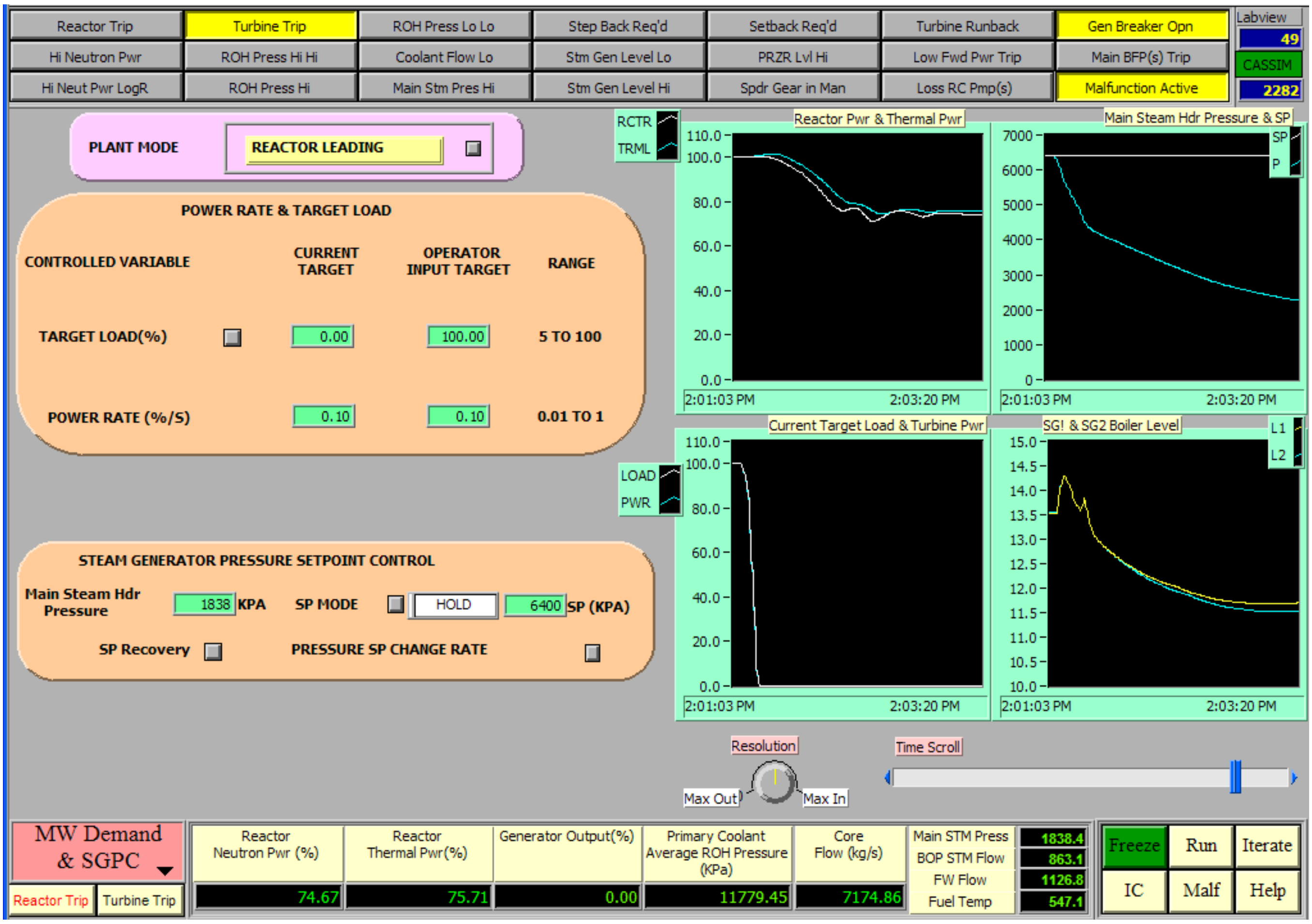
⇒ Observe that the turbine is running back to zero power. Confirm turbine is tripped.

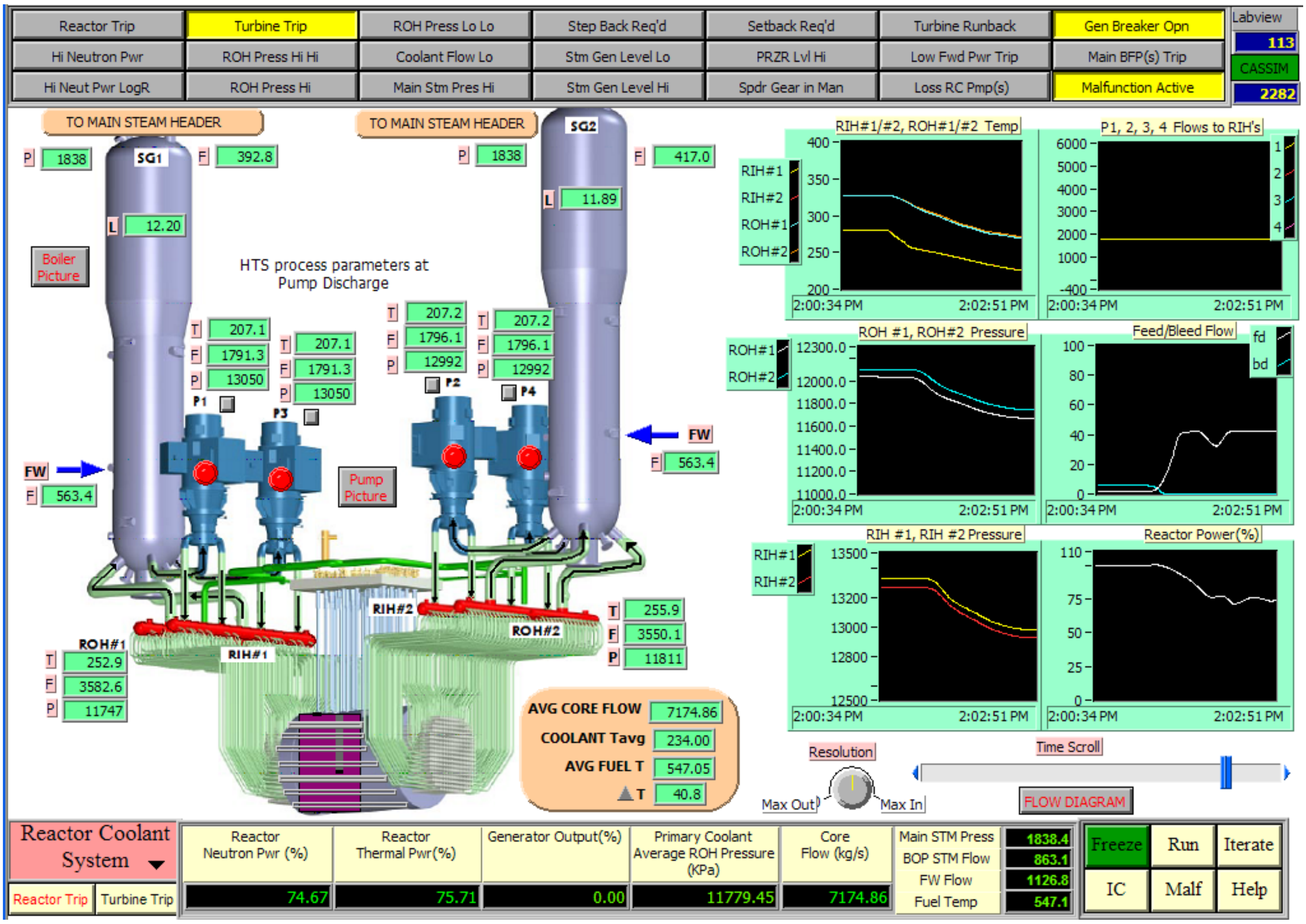
⇒ Record reactor power after setback.

⇒ Continue to monitor coolant pressure and temperature transients.

⇒ Discuss any safety margin challenge, if any, in this malfunction event, and how the safety and control systems can cope with these challenges




Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
49
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
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
2282
PLANT MODE
REACTOR LEADING
POWER RATE & TARGET LOAD
CONTROLLED VARIABLE
CURRENT TARGET
OPERATOR INPUT TARGET
RANGE
TARGET LOAD(%)
0.00
100.00
5 TO 100
POWER RATE (%/S)
0.10
0.10
0.01 TO 1
STEAM GENERATOR PRESSURE SETPOINT CONTROL
Main Steam Hdr Pressure
1838
KPA
SP MODE
HOLD
6400
SP (KPA)
SP Recovery
PRESSURE SP CHANGE RATE
RCTR
TRML
Reactor Pwr & Thermal Pwr
Main Steam Hdr Pressure & SP
SP
P
LOAD
PWR
Current Target Load & Turbine Pwr
SG! & SG2 Boiler Level
L1
L2
2:01:03 PM
2:03:20 PM
Resolution
Time Scroll
Max Out
Max In
MW Demand & SGPC
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
74.67
Reactor Thermal Pwr(%)
75.71
Generator Output(%)
0.00
Primary Coolant Average ROH Pressure (KPa)
11779.45
Core Flow (kg/s)
7174.86
Main STM Press
1838.4
BOP STM Flow
863.1
FW Flow
1126.8
Fuel Temp
547.1
Freeze
Run
Iterate
IC
Malf
Help


Reactor Trip
Turbine Trip
ROH Press Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
113
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
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
2282
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
Boiler Picture
HTS process parameters at Pump Discharge
Pump Picture
FW
ROH#1
ROH#2
RIH#1
RIH#2
AVG CORE FLOW
7174.86
COOLANT Tavg
234.00
AVG FUEL T
547.05
40.8
RIH#1/#2, ROH#1/#2 Temp
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
Feed/Bleed Flow
RIH #1, RIH #2 Pressure
Reactor Power(%)
2:00:34 PM
2:02:51 PM
Resolution
Time Scroll
Max Out
Max In
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
74.67
Reactor Thermal Pwr(%)
75.71
Generator Output(%)
0.00
Primary Coolant Average ROH Pressure (KPa)
11779.45
Core Flow (kg/s)
7174.86
Main STM Press
1838.4
BOP STM Flow
863.1
FW Flow
1126.8
Fuel Temp
547.1
Freeze
Run
Iterate
IC
Malf
Help

5.20. Primary Coolant RIH #1 LOCA break

This malfunction event causes a “crack” opening at the reactor inlet header #1. This break causes a loss of coolant accident (LOCA) event. Before the malfunction is inserted, it is recommended that the simulator user should be familiar with the design of the passive core injection system as described in Section 3.17 “ACR passive core cooling” screen, before performing this exercise.

⇒ First load the full power initial condition (IC) and “run” the simulator.

⇒ Go to “reactor coolant system” screen, and select the malfunction “Primary Coolant RIH #1 LOCA break”, then press “insert MF”, and press “return”.

⇒ Observe that the “malfunction active” alarm is “on”.

⇒ Note that all the trended parameters on the screen will change immediately. Record the break flow in Table V.

⇒ Record the primary coolant pressure when the reactor is tripped.

⇒ After the reactor is tripped, go to “ACR passive core cooling” screen. On this screen, the injection flow path by the passive core cooling system will be shown in “thick” blue lines, during the various stages of injection cooling.

⇒ Record the parameters in Table VI during the various stages of injection:

⇒ Explain the coolant pressure transients in the course of event evolution. When do pressure “bumps” occur? And why do they occur?

⇒ Explain why the accumulator is necessary? Can the accumulator be eliminated if we make use of the Reserve Water Tank (RWT) located at the top of the Containment building ?

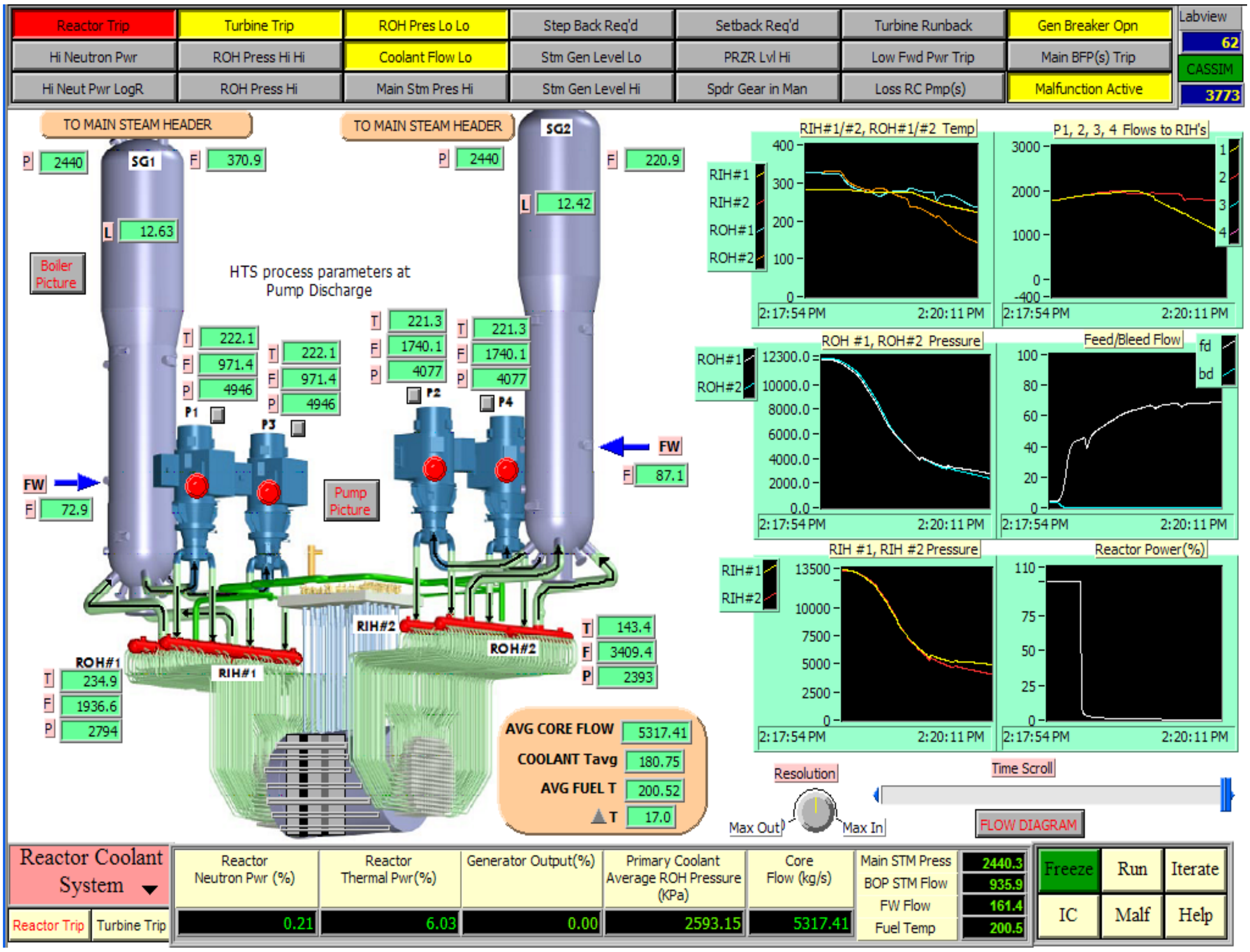
⇒ Explain why the opening of the MSSV is necessary — to serve what purpose?

-

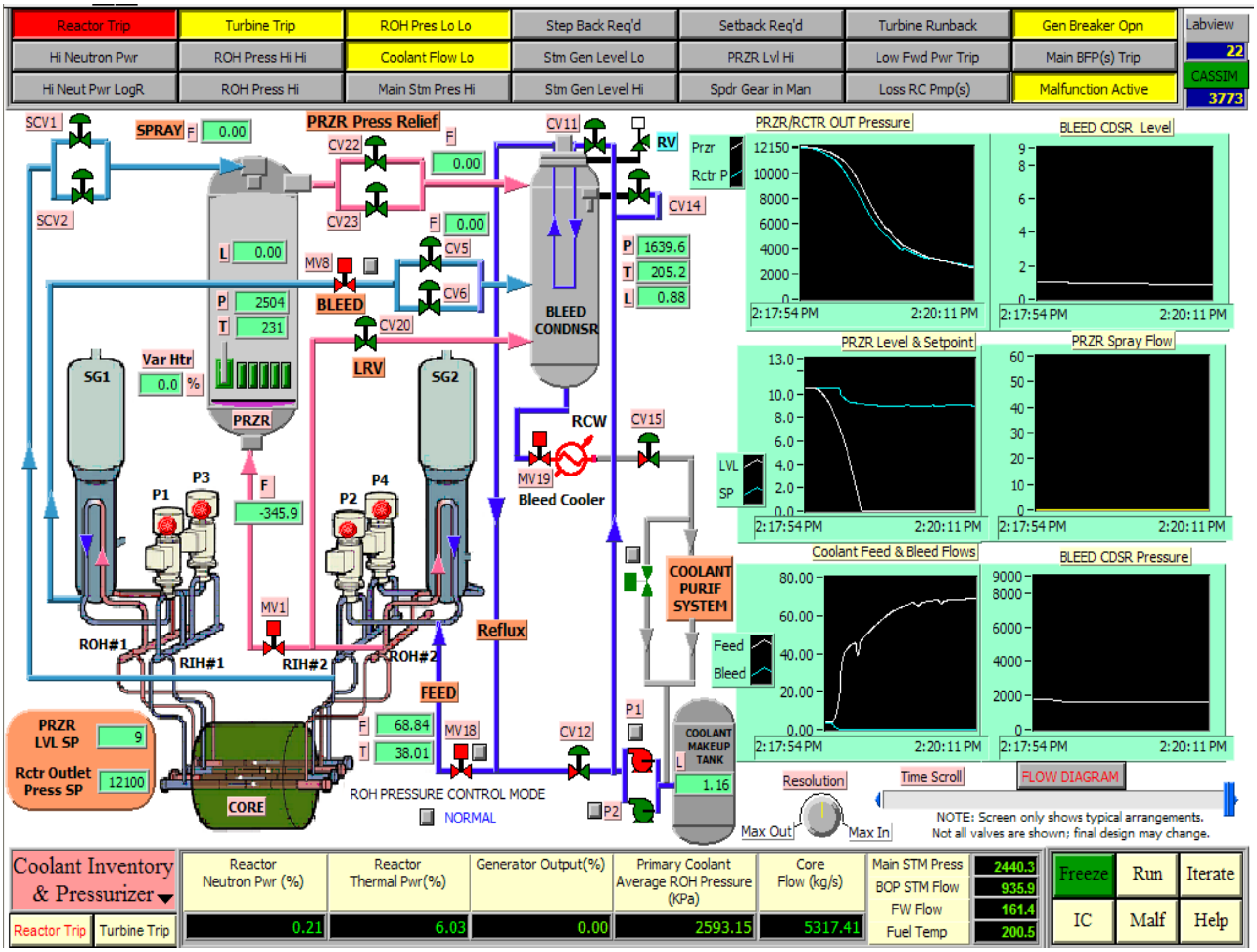
TABLE VI. RIH # 1 LOCA BREAK

Stages of Injection	**Accumulators in service**	**Steam Generators Crash Cool - MSSVs open**	**Reserve Water Tank in service**	**Long Term Cooing (LTC) in service**
Time elapsed after Break[6]	_____ sec after Break	_____ sec after Break	______ sec after Break	_____ sec after Break
Reactor Power (%)				
Turbine power (%)				
Reactor Thermal Power (%)				
Break Flow (Kg/s)				
Total Injection Flow (Kg/s)				
Core Flow (Kg/s)				
Average Coolant Temp. (°C)				
Fuel Temp (°C)				
PRZR level (m)				
PRZR Pressure (KPa)				
Coolant Pressure at RIH#1, #2 (KPa)				
Containment Pressure (KPa)				
Containment Temp (°C)				
ACC Level (% full)				
Flow from RWT (Kg/s)				

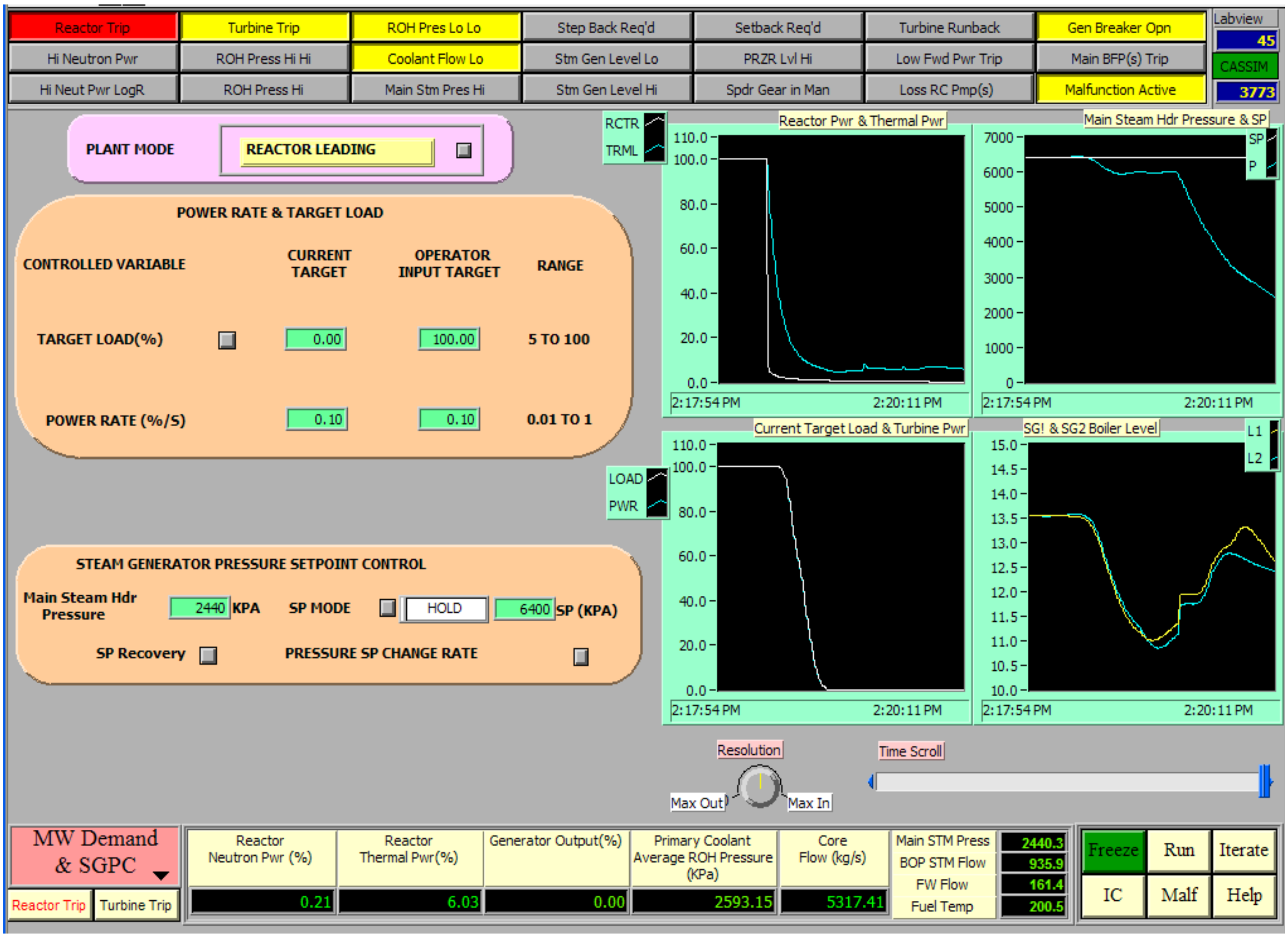
[6] To account for the time elapsed after the break, record the CASSIM iteration counts shown at the top right hand corner, multiply that number by the time step = 0.1 sec., to get the time in seconds. This calculation has assumed that the simulation iteration starts from 0 when the LOCA malfunction is initiated.

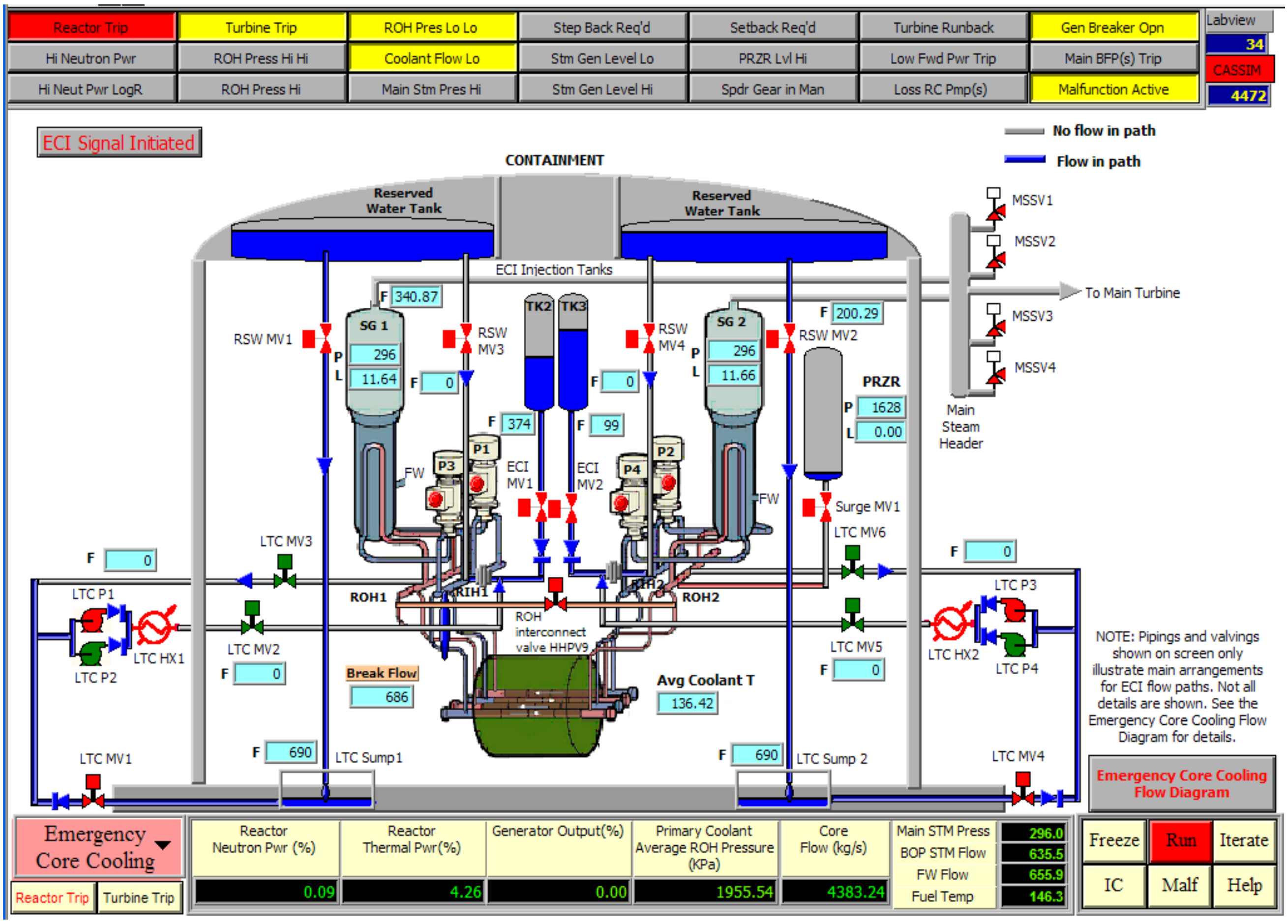


Reactor Trip
Turbine Trip
ROH Pres Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
62
Hi Neutron Pwr
ROH Press Hi Hi
Coolant Flow Lo
Stm Gen Level Lo
PRZR Lvl Hi
Low Fwd Pwr Trip
Main BFP(s) Trip
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
3773
TO MAIN STEAM HEADER
TO MAIN STEAM HEADER
SG1
SG2
Boiler Picture
HTS process parameters at Pump Discharge
Pump Picture
FW
FW
ROH#1
ROH#2
RIH#1
RIH#2
AVG CORE FLOW 5317.41
COOLANT Tavg 180.75
AVG FUEL T 200.52
ΔT 17.0
RIH#1/#2, ROH#1/#2 Temp
P1, 2, 3, 4 Flows to RIH's
ROH #1, ROH#2 Pressure
Feed/Bleed Flow
RIH #1, RIH #2 Pressure
Reactor Power(%)
2:17:54 PM
2:20:11 PM
Resolution
Time Scroll
Max Out
Max In
FLOW DIAGRAM
Reactor Coolant System
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.21
Reactor Thermal Pwr(%) 6.03
Generator Output(%) 0.00
Primary Coolant Average ROH Pressure (KPa) 2593.15
Core Flow (kg/s) 5317.41
Main STM Press 2440.3
BOP STM Flow 935.9
FW Flow 161.4
Fuel Temp 200.5
Freeze
Run
Iterate
IC
Malf
Help



Reactor Trip
Turbine Trip
ROH Pres Lo Lo
Step Back Req'd
Setback Req'd
Turbine Runback
Gen Breaker Opn
Labview
22
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
CASSIM
Hi Neut Pwr LogR
ROH Press Hi
Main Stm Pres Hi
Stm Gen Level Hi
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
3773
SCV1
SCV2
SPRAY
PRZR Press Relief
CV22
CV23
CV11
RV
CV14
CV5
CV6
MV8
BLEED
CV20
LRV
BLEED CONDNSR
Var Htr
PRZR
SG1
SG2
RCW
CV15
MV19
Bleed Cooler
COOLANT PURIF SYSTEM
MV1
Reflux
ROH#1
RIH#1
RIH#2
ROH#2
FEED
MV18
CV12
PRZR LVL SP
Rctr Outlet Press SP
CORE
ROH PRESSURE CONTROL MODE
NORMAL
COOLANT MAKEUP TANK
PRZR/RCTR OUT Pressure
BLEED CDSR Level
PRZR Level & Setpoint
PRZR Spray Flow
Coolant Feed & Bleed Flows
BLEED CDSR Pressure
2:17:54 PM
2:20:11 PM
Resolution
Time Scroll
FLOW DIAGRAM
Max Out
Max In
NOTE: Screen only shows typical arrangements.
Not all valves are shown; final design may change.
Coolant Inventory & Pressurizer
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.21
Reactor Thermal Pwr(%) 6.03
Generator Output(%) 0.00
Primary Coolant Average ROH Pressure (KPa) 2593.15
Core Flow (kg/s) 5317.41
Main STM Press 2440.3
BOP STM Flow 935.9
FW Flow 161.4
Fuel Temp 200.5
Freeze
Run
Iterate
IC
Malf
Help


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
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
Labview
45
CASSIM
3773
PLANT MODE
REACTOR LEADING
POWER RATE & TARGET LOAD
CONTROLLED VARIABLE
CURRENT TARGET
OPERATOR INPUT TARGET
RANGE
TARGET LOAD(%)
0.00
100.00
5 TO 100
POWER RATE (%/S)
0.10
0.10
0.01 TO 1
STEAM GENERATOR PRESSURE SETPOINT CONTROL
Main Steam Hdr Pressure
2440
KPA
SP MODE
HOLD
6400
SP (KPA)
SP Recovery
PRESSURE SP CHANGE RATE
RCTR
TRML
Reactor Pwr & Thermal Pwr
Main Steam Hdr Pressure & SP
SP
P
Current Target Load & Turbine Pwr
SG! & SG2 Boiler Level
L1
L2
LOAD
PWR
2:17:54 PM
2:20:11 PM
Resolution
Max Out
Max In
Time Scroll
MW Demand & SGPC
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Primary Coolant Average ROH Pressure (KPa)
Core Flow (kg/s)
0.21
6.03
0.00
2593.15
5317.41
Main STM Press
BOP STM Flow
FW Flow
Fuel Temp
2440.3
935.9
161.4
200.5
Freeze
Run
Iterate
IC
Malf
Help


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
Spdr Gear in Man
Loss RC Pmp(s)
Malfunction Active
Labview
34
CASSIM
4472
ECI Signal Initiated
CONTAINMENT
No flow in path
Flow in path
Reserved Water Tank
Reserved Water Tank
ECI Injection Tanks
MSSV1
MSSV2
To Main Turbine
MSSV3
MSSV4
Main Steam Header
SG 1
SG 2
340.87
200.29
296
11.64
296
11.66
TK2
TK3
RSW MV1
RSW MV3
RSW MV4
RSW MV2
PRZR
1628
0.00
374
99
P1
P2
P3
P4
ECI MV1
ECI MV2
FW
Surge MV1
LTC MV6
LTC MV3
LTC P1
LTC P2
LTC HX1
LTC MV2
LTC P3
LTC P4
LTC HX2
LTC MV5
ROH1
ROH2
RIH1
RIH2
ROH interconnect valve HHPV9
Break Flow
686
Avg Coolant T
136.42
690
LTC Sump1
690
LTC Sump 2
LTC MV1
LTC MV4
NOTE: Pipings and valvings shown on screen only illustrate main arrangements for ECI flow paths. Not all details are shown. See the Emergency Core Cooling Flow Diagram for details.
Emergency Core Cooling Flow Diagram
Emergency Core Cooling
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Primary Coolant Average ROH Pressure (KPa)
Core Flow (kg/s)
0.09
4.26
0.00
1955.54
4383.24
Main STM Press
BOP STM Flow
FW Flow
Fuel Temp
296.0
635.5
655.9
146.3
Freeze
Run
Iterate
IC
Malf
Help

6. REFERENCES:

1. AECL Document - ACR-700 Technical Description 10810-01371-TED-001 Revision 1 (March 2004).
2. AECL Document - PRELIMINARY DESIGN DESCRIPTION OF PRESSURE AND INVENTORY CONTROL SYSTEM ACR-700 10810-33310-200-001 Revision 1 (Feb., 2004).