Boiling Water Reactor Simulator with Active Safety Systems

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. This simulation is the result of a joint effort from the developer, the Agency's staff, and from Dr. Bharat Shiralkar, a thermal-hydraulics expert on boiling water reactors.

- 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 and later Ms. S. Bilbao y León 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/Education/Simulators/

EDITORIAL NOTE

<u>Note on running version 3 of the BWR Simulator:</u>

Due to the fact that this version of the BWR Simulator S/W incorporates intensive memory buffering for retaining trends history, some time is required for data initialization when the simulator is first loaded. To speed up this process, it is recommended that after the simulator is first loaded and the "BWR Plant Overview Screen" is displayed, first "RUN" the simulation for a few seconds, then "LOAD" the 100% FP IC again, before running the simulator.

IAEA BWR NPP Simulator (V.3) Revision Details:

Version 3 of the BWR Simulator incorporates changes recommended by a thermal hydraulics expert on BWR, who reviewed the simulator responses and provided suggestions in containment modeling. The scope of recommended changes is as follows:

(1) Improvement on the reactor level response. This is a key parameter for BWR operation as many alarms and trips are triggered by the level position. Level should drop for scram and pressurization events due to a decrease in core and downcomer voids, and increase for flow decreases and depressurization due to void fraction increases. The level control system has a time constant of ~ 10 seconds, and cannot mitigate the initial changes. The simulator trends often seem to be the opposite of what would be expected.

Implemented change: the suggested level response is implemented.

(2) The automatic power flow control system is usually not active in operating BWRs. The ABWR does have an APR, which is a power generation system that controls reactor power during reactor startup, power generation, and reactor shutdown, by appropriate commands to change rod positions, or to change reactor recirculation flow. It also controls the pressure regulator setpoint (or turbine bypass valve position) during reactor heatup and depressurization (e.g., to control the reactor cooldown rate). It would not be a factor during most rapid transients. The simulator should have the option of turning the automatic power controller off during transients. This will give a better picture of plant operation and also will not mask the transient trends.

Already implemented in version 2 – see Note 2 of P. 30:

"The APR has several important control components which include the RIP controls. One can find the user interface for RIP control on "Power Flow Map & Control Screen". On the right side of the screen, there is a button labeled as "RIP Crl". Upon pressing this button, one will see the typical PID controller faceplate for RIP. Currently the controller is at Remote Set Point (RSP), indicating a control mode where the setpoint for the controller is derived remotely from external computation. One can switch the controller to "Manual", and by manipulating the manual output signal, one can change the RIP head and hence speed (RPM), changing of core flow rate as a result."

(3) The simulator should implement the logic for ABWR reactor internal pump (RIP) trips and runbacks (summarized below):
RIP Trips and Runbacks -

- 4 RIPs are not connected to M/G sets; 6 are connected to M/G sets (larger coastdown inertia)
- 4 RIPs tripped on: High Pressure (7.76 MPaG); L3: and turbine stop valve closure or fast TCV closure but overruled if bypass valves open
- 6 RIPs tripped on: L2
- RIP runback (1% per sec) on scram

Changes implemented – Note the 10 RIPs are modeled on the basis of one integrated pump head, hence, in the event of RIPs Trips and Runback, one will see the corresponding pump head and speed decrease to match with the respective RIP trip or runback scenario. In addition, when RIP runback condition appears, one will see the Runback alarm shown on the pump in the BWR Power/Flow May and Control Screen.

(4) Changes implemented to reflect realistic system behavior due to various malfunction events introduced by the MALFUNCTION screen of the simulator. For example, regarding the loss of feedwater event, a rapid drop in level should initiate a core flow runback at L4 and a scram and trip of 4 RIP on L3. This has been implemented.

Assuming the measurement from Top of Fuel (TAF) to the bottom of the vessel = 9 m, the respective Level Trip Setpoints are implemented as follows:

Operating Setpoint for water level:

- normal water level setpoint is 13.5 m

Trip & Runback Setpoints:

- L 8 = 5.0 m TAF = 14.0 m – action: Turbine Trip.
- L 4 = 4.0 m TAF = 13.0 m – action: core flow runback.
- L 3 = 3.30 m TAF = 12.30 m – action: Reactor Scram.
- L 2 = 2.43 m TAF = 11.43 m – action : Trip 6 RIPs; Start RCIC in the ECC system.
- L 1.5 = 0.978 m TAF = 9.978 m – action: start HPCF pumps in the ECC system.
- L 1 = 0.153 m TAF = 9.153 m – action: start ADS blow down; start 2 RHR pumps in the ECC system.

The Level Trip Setpoints markers L1, L1.5, L2, L3, L4, L8 are shown on Power & Flow Map and Control Screen, and in the BWR Containment Screen.

(5) Modeling of the containment: The drywell and wetwell with the suppression pool are modeled. This would allow for better simulation of loss of coolant accidents (LOCA) and transients that lead to SRV openings and activation of the Automatic Depressurization System (ADS). Break or SRV flow pressurizes the containment. The suppression pool provides ECCS makeup to the reactor vessel. Heatup of the suppression pool affects the ECC temperature.

(6) Implement proper coefficients for all the reactor reactivity feedback effects: void, Doppler, coolant temperature.

Note on running version 2 of the BWR Simulator:

Due to the fact that this version of the BWR Simulator S/W incorporates intensive memory buffering for retaining trends history, some time is required for data initialization when the simulator is first loaded. To speed up this process, it is recommended that after the simulator is first loaded and the "BWR Plant Overview Screen" is displayed, first "RUN" the simulation for a few seconds, then "LOAD" the 100% FP IC again, before running the simulator.

IAEA BWR NPP Simulator (V.2) Revision Details:

(a) At Zero Power Hot condition, harmonize core flow and recirculation pumps speed.

At Zero Power Hot condition, the revised model now gives 33 % core flow, and recirculation pumps speed ~ 397 RPM.

(b) At Zero Power Cold condition, the operating point should be 0 % core flow and Zero RPM.

At Zero Power Cold condition, the revised model now gives ~ 0 % core flow (7.85 kg/s) and recirculation pumps speed is 1 RPM. This IC is considered to be close enough to the desired condition to start the reactor at sufficiently low power. In order to get to 0 kg/s core low, and 0 RPM, one can load this IC condition and let the simulation run for a long time. A no flow condition will be reached when the coolant density at the reactor downcomer becomes equal to the coolant density at the reactor core region. When the coolant densities at the two columns are equal, there will be no natural recirculation flow.

(c) Add an additional Reactor Scram Parameter – called High Steam Flow (>120 % nominal full power steam flow.)

The additional Reactor Scram Parameter is implemented. One can test this scram parameter as follows:

(1) Go to BWR Turbine Generator Screen. Click on Turbine Governor Control Button and switch the Control Mode to "Manual". Observe that the Alarm indicator "Turbine Governor in Man."

(2) Click on Bypass Valve Control Button and switch the Control Mode to "Manual". Select "MAN OUT(%)" and enter 100 % value to demand Bypass Valve opening to 100 %.

(3) Let the simulation run, and one will observe that the steam flow will increase from 2130 kg/s. When it reaches 2613 kg/s (> 120 % nominal full power steam flow), reactor scram will occur.

(d) Update BWR Simulator screens with memory buffer. The trend history will be memorized and maintained after changing screens. Update Simulator Freeze Control Design, so that on simulator freeze, all the screens with time trends will stop trending.

This is implemented. In addition, an “AUTOSCALE” button is implemented on the Trends Screen so that the user may enable or disable “Auto-Y-Scale” of all the 8 trends on display, at the press of the button. When “AUTOSCALE” is enabled, a red light will be turned on.

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1. INTRODUCTION

1.1 PURPOSE

The International Atomic Energy Agency (IAEA) has established an activity in nuclear reactor simulation computer programs to assist its Member States in education. The objective is to provide, for a variety of advanced reactor types, insight and practice in reactor 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 educational materials, sponsors workshops, and distributes documentation and computer programs.

This publication consists of course material for workshops using the boiling water reactor (BWR) 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 BWR 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 boiling water reactor. Therefore no attempt has been made to provide detailed descriptions of each individual BWR subsystem. Those descriptions are commonly found in nuclear engineering textbooks, BWR nuclear power plant (NPP) design manuals, and IAEA technical publications. However, details are provided where necessary to describe the functionality and the interactive features of the individual simulator screens, which relate to the specific BWR subsystems.

The manual covers basic NPP operations, such as plant load maneuvering, trips and recovery e.g. turbine trip and reactor scram. In addition, it covers plant responses to malfunction events. Some malfunction events lead to reactor scram or turbine trip. Other serious malfunction events (e.g. LOCA) lead to actuation of the core cooling safety system.

It should be mentioned that the equipment and processes modeled in the simulator represent realistic BWR 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 vendor's BWR 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 education. 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 HISTORICAL BACKGROUND

Boiling water reactor plants were designed in the 1950s and put into operation starting from the early 1960s. Many BWR plants have been constructed and operated safely worldwide. They constitute a significant electricity source from nuclear fission.

Although changes and improvements have been made in BWR designs throughout their history of operation, the basic concept is essentially unchanged since the first BWR design proposed by General Electric.

The basic feature of all BWR plants is the presence of a reactor pressure vessel (RPV) in which feedwater enters the vessel in subcooled conditions and saturated steam exits the vessel. The subcooled feedwater is heated by nuclear fission heat from the fuel bundles, as it travels up the various coolant channels in the reactor core. As boiling of the reactor coolant occurs at the upper region of the core coolant channels, a water-steam mixture exits the reactor core (into the upper plenum) at saturation temperature. Nominal core operating pressure is typically 7.0 MPa, which is nearly the same for all BWR designs. The water-steam mixture is then separated and dried in the upper plenum, with the saturated steam flowing directly to the turbine. The saturated liquid, separated from the water-steam mixture, is then recirculated back to the annular downcomer of the RPV where it mixes with the subcooled feedwater before entering the lower plenum of the reactor vessel and the coolant channels.

Originating from this basic BWR core design are the following reactor types that have been built and operated. Examples of existing plants are shown in parentheses:

- Natural circulation direct cycle BWR (Dodewaard)
- Forced Circulation Duel Cycle (Dresden 1, Garigliano)
- Forced Circulation Direct Cycle - external pumps (Ringhals, Oyster Creek)
- Forced Circulation Direct Cycle - jet pumps (Leibstadt, Dresden 2)
- Forced Circulation Direct Cycle - internal pumps (Kashiwazaki-Kariwa, Oskarshamn)

The evolution of the BWR design was started with a relatively complex dual cycle design (Dresden 1), involving an intermediate steam generator. Then the design evolved to a direct cycle using external pumps (e.g. Oyster Creek), to a jet pump reactor (e.g. Dresden 2). Nowadays, an example of a new BWR is the advanced boiling water reactor (ABWR) using internal pumps (e.g. Kashiwazaki-Kariwa), jointly designed by General Electric, Hitachi and Toshiba.

Historically, the dual cycle plant was designed and constructed in the early stages of BWR development, with the plant in the reactor-follow mode, i.e. the reactor power follows the turbine power. An intermediate steam generator was introduced to boil the feedwater utilizing the saturated liquid extracted from the primary circuit. The steam produced in the steam generator flowed to the turbine through a secondary steam line and was used to control the plant in reactor-follow mode. All the plants of this type are currently shutdown.

In order to increase the core cooling capability, pumps were introduced in the recirculation loop. The simplest configuration was with external pumps suctioning the fluid from the downcomer region and injecting it at higher pressure into the lower plenum of the core.

Further BWR development involved the recirculation loop configuration with jet pumps. The introduction of jet pumps satisfied two additional design objectives: (1) only a portion of the core coolant was recirculated externally to the vessel, and (2) no large pipe was connected to the bottom of the vessel, thus making core flooding easier in the unlikely event of a large pipe break in vessel-connected piping.

The configuration with internal pumps in the recirculation loop eliminates the piping and flows external to the vessel. Such a configuration is currently in the advanced BWR design.

The BWR fuel bundles are enclosed in rectangular boxes or channels. A large variety of fuel designs is currently available. While the fuel box dimension generally remained unchanged, the number of fuel rods and rods lattice changed from 7 × 7, to 8 × 8, up to 10 × 10. Latest fuel designs include part-length rods and "water channels" (locations where fuel rods are absent).

The control rods are inserted through the vessel from the bottom. As a consequence of larger moderation at the channel bottom, the axial power shape is typically bottom-skewed but the axial power distribution changes markedly during long-term reactor operation..

Finally, almost all BWR plants are equipped with a pressure suppression containment including a large pool of ambient temperature liquid (~ 5000 m^3) where the steam-liquid mixture lost from a LOCA (loss of coolant accident) can be condensed. The containment also serves as a protective shield and prevents release of radioactive contamination to the outside of the reactor building, in the unlikely event of a serious accident.

1.3 PROMINENT CHARACTERISTICS OF A TYPICAL BWR

A typical BWR is characterized by several prominent differences from other light water reactors (LWRs) such as the PWR:

(1) Under normal operating conditions the coolant in the core is subcooled liquid near the bottom of the core (the non-boiling region) and a two-phase saturated steam-water mixture in the boiling region downstream, up to the top of the core.

(2) Steam generation occurs in a direct cycle with steam separators and dryers inside the reactor pressure vessel. A separate steam generator is not required. Typical operating saturation temperature is around 280°C; steam pressure ~ 7 MPa.

(3) The reactor (steam dome) pressure is controlled by turbine inlet valves and turbine bypass valves.

(4) The BWR core consists of a number of fuel bundles (assemblies), each with a casing called a fuel channel. Each fuel bundle (assembly) contains a number of fuel rods arranged in a N × N square lattice, with slightly enriched Uranium fuel ~ 2% to 5% U-235 by weight.

(5) The control rods are of cruciform shape and enter the core from the bottom. Each control rod moves between 4 fuel assemblies.

(6) The reactor power control consists of control rods and recirculation flow control. Control rods are used to achieve the desired power level by adjustment of their positions in the core at a rate equivalent to a power change rate of up to 2% full power per second. The recirculation flow control also controls reactor power by causing the density of the water/steam mixture used as moderator to change. Indeed, the flow rate through the core affects the enthalpy of the coolant and its void (steam) content The flow rate is adjusted by a variable speed pump (such as the internal pumps of the ABWR) at a rate equivalent to a power change rate of up to 30% full power per minute.

(7) "Dried" steam from the reactor pressure vessel (RPV) enters the turbine plant through four steam lines connected to nozzles equipped with flow limiters. In the unlikely event of a steam line break anywhere downstream of the nozzle, the flow limiters limit the steam blowdown rate from the RPV to less than 200% rated steam flow rate at 7.07 MPa.

(8) There are safety relief valves (16 of them) connected to the four steam lines to prevent RPV overpressure, with a blow down pipe connected to the suppression pool.

(9) In the steam lines, isolation valves are provided inside and outside of the containment wall to isolate the RPV, if necessary.

(10) Saturated steam from the RPV main steam lines is admitted to the turbine HP cylinder via the governor valves. After the HP section, steam passes through the moisture separator reheater (MSR) to the LP turbine cylinders.

(11) A special steam bypass line, prior to the turbine governor valves, enables dumping the full nominal steam flow directly to the condenser in the event of plant upset such as a turbine trip, in order to avoid severe pressure surges and corresponding power peaks in the reactor.

(12) Typical balance of plant (BOP) systems for the BWR consists of the condenser, condensate pumps, deaerator, feedwater heaters, reactor feed pumps (RFP) and reactor level control valves.

(13) The containment is a cylindrical prestressed concrete structure with an embedded steel liner. It encloses the reactor, reactor coolant pressure boundary and important ancillary systems. The containment has a pressure-suppression type pool with a drywell and a wetwell. The drywell is the space around the RPV; the wetwell is a water pool where steam injected in the drywell may enter (via vents) and condense. The containment and pressure suppression pool configuration varies according to plant generation.

A typical BWR design with the above described features is shown in Figure 1. Descriptions of a number of BWR designs can be found in IAEA-TECDOC-1391 Status of Advanced Light Water Cooled Reactor Designs (2004).

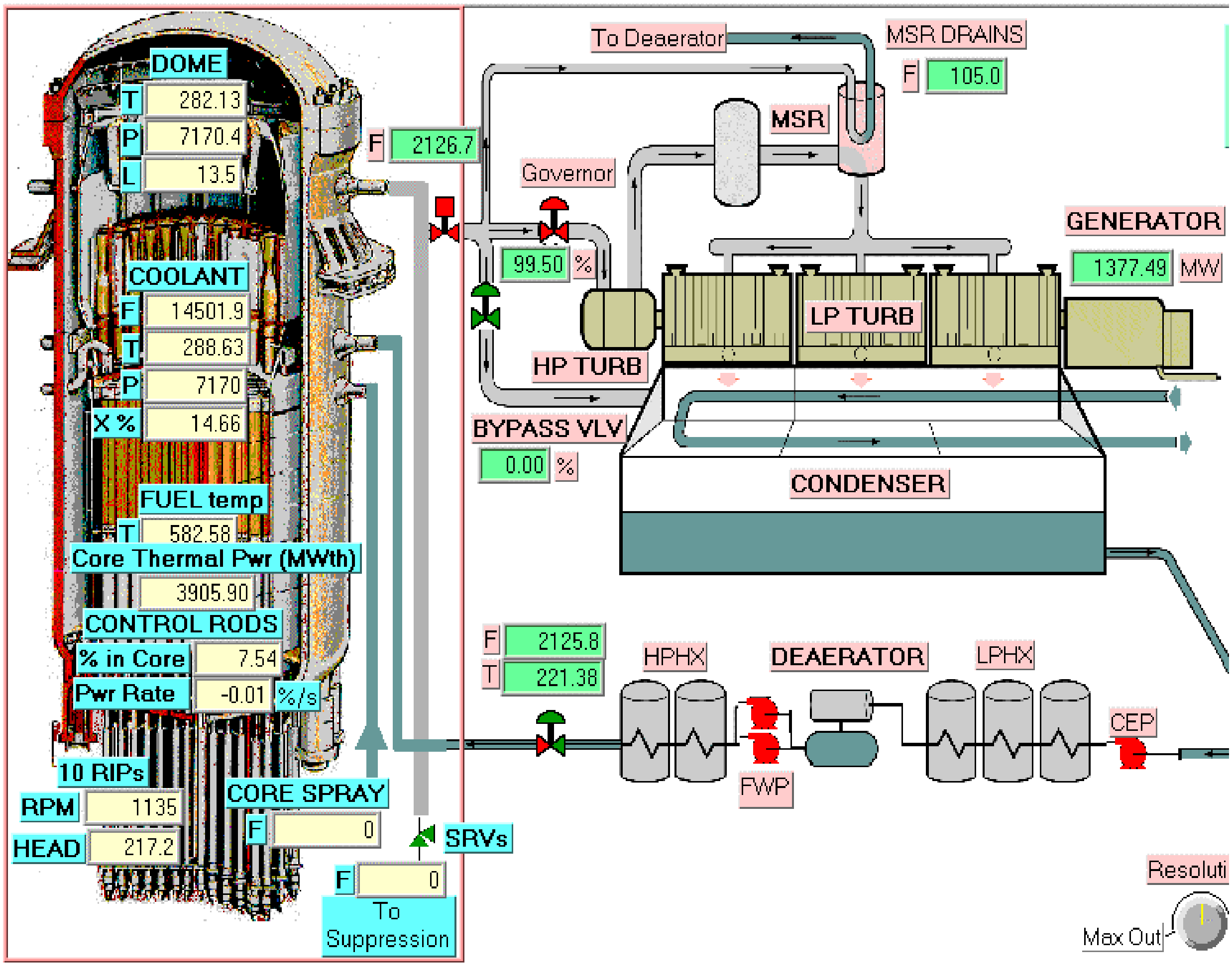


FIG. 1. *A typical 1300 MW(e) boiling water reactor NPP.*

2. 1300 MW(E) Boiling Water Reactor NPP Simulator

The purpose of the 1300 MW(e) boiling water reactor NPP simulator is educational — to provide a teaching tool for university professors and engineers involved in teaching topics in nuclear energy. As well, nuclear engineers, scientists and teachers in the nuclear industry may find this simulator useful in broadening their understanding of BWR NPP dynamics and transients.

The simulator can be executed on a personal computer (PC), to operate essentially in real time, and have a dynamic response with sufficient fidelity to provide BWR 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 interaction with the simulator during the operation of the simulated BWR 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 Mbytes display adaptor RAM, hi-resolution video card (capable of 1024 x 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 have 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. That 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 simulator configuration is able to respond to the operating conditions normally encountered in power plant operations, as well as to numerous malfunctions, as summarized in Table I.

The simulation development used a modular modeling approach: basic models for each type of device and process are represented as algorithms and 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 plant operator controls 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 nuclear power plants, as well as understanding the unique features of the BWR.

TABLE I. SUMMARY OF BWR SIMULATOR FEATURES

SYSTEM	SIMULATION SCOPE	DISPLAY PAGES	OPERATOR CONTROLS	MALFUNCTIONS
REACTOR CORE	• Neutron flux levels over a range of 0.001 to 110% full power, 6 delayed neutron groups • Decay heat (3 groups) • Reactivity feedback effects - void, xenon, fuel temperature, moderator temperature • 2 phase flow & heat transfer • Reactivity control rods • Essential control loops - Reactor Pressure Control; Core Recirculation Flow Control; Reactor Power Regulation; Reactor Water Level Control; Turbine Load/Frequency Control	• Plant Overview • BWR Reactivity & Setpoints • BWR Power /Flow Map & Controls	• Reactor power and rate of change (input to control computer) • Manual control of control rods • Reactor scram • Manual Control Rods "run-in" • Manual control of core recirculation flow rate • Manual adjustment of reactor water control level setpoint	• Increasing and decreasing core flow due to Flow Control malfunctions • Inadvertent withdrawal of one bank of control rods • Inadvertent insertion of one bank of control rods • Inadvertent reactor isolation • Power loss to 3 Reactor Internal Pumps (RIPs) • Reactor bottom break
STEAM & FEED-WATER	• Steam supply to turbine and reheater • Main Steam Isolation Valve • Turbine Bypass to condenser • Steam Relief Valves to Suppression Pool in containment • Extraction steam to feed heating • Feedwater system	• BWR Feedwater and Extraction Steam	• Reactor water level setpoint changes: computer or manual • Extraction steam to feedwater heating isolating valves controls • Deaerator main steam extraction pressure control • Feed pump on/off controls	• Loss of both feedwater pumps • Loss of feedwater heating • Reactor feedwater level control valve fails open • Safety valves on one main steam line fail open • Steam line break inside Drywell • Feedwater line break inside Drywell

SYSTEM	SIMULATION SCOPE	DISPLAY PAGES	OPERATOR CONTROLS	MALFUNCTIONS
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	• BWR Turbine-Generator	• Turbine trip • Turbine run-back • Turbine run-up and synchronization • Turbine Speeder Gear control: manual or computer control • Steam Bypass Valve Computer or Manual Control	• Turbine throttle pressure transmitter fails low • Turbine trip with Bypass Valve failed closed • Increasing and decreasing steam flow due to Pressure Control System failures
OVERALL UNIT	• Fully dynamic interaction between all simulated systems • Turbine-Following-Reactor load maneuvering • Unit annunciation • Major control loops	• BWR Plant Overview • BWR Reactivity & Setpoints		
CONTAINMENT	• Pressure and temperature respons of the drywell to the break flow discharge into the drywell. • Vent clearing accounting for the inertia of the water legs in the vertical and horizontal branches. • Discharge through the vents and suppression pool mass and energy balance. • Wetwell airspace pressurization due to noncondensibles added to this space and the increased vapor pressure corresponding to the suppression pool surface temperature.	• BWR Containment	• Manual Spray Controls for Drywell and Wetwell.	• LOCA Break flow ~ 800 kg/s

2.1 SIMULATOR STARTUP

- Select program 'BWR' for execution - the executable file is BWR.exe
- Click anywhere on 'BWR simulator" screen
- Click 'OK' to 'Load Full Power IC?'
- The simulator will display the "Plant Overview" screen with all parameters initialized to 100% Full Power
- At the bottom right hand corner click on 'Run' to start the simulator
- Due to the fact that the revised version of the BWR Simulator S/W incorporates intensive memory buffering for retaining trends history, some time is required for

data initialization when the simulator is first loaded. To speed up this process, it is recommended that after the simulator is first loaded and the ”Plant Overview Screen” is displayed, first “RUN” the simulation for a few seconds, then “LOAD” the 100% FP IC again, before running the simulator.

2.2 SIMULATOR INITIALIZATION

If at any time you need 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:\BWR_Simulator\FP_100.IC’
- Click ‘YES’ to 'Load C:\BWR_Simulator\FP_100.IC’
- Click ‘Return’
- Start the simulator operating by selecting ‘Run’.

2.3 LIST OF BWR SIMULATOR DISPLAY SCREENS

(1) BWR Plant Overview
(2) BWR Control Loops
(3) BWR Power/Flow Map & Controls
(4) BWR Reactivity & Setpoints
(5) BWR Scram Parameters
(6) BWR Turbine Generator
(7) BWR Feedwater & Extraction Steam
(8) BWR Containment
(9) BWR Trends

2.4 GENERIC BWR SIMULATOR DISPLAY COMMON FEATURES

The generic BWR simulator has 8 interactive display screens or pages. Each screen has the same information at the top and bottom, as follows:

- The top of the screen contains 21 plant alarms and annunciations; these indicate important status changes in plant parameters that require operator actions;
- The top right hand corner shows the simulator status:
  - ⇒ The window under ‘Labview’ (this is the proprietary graphical user interface software that is used to generate 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 simulation engine software that computes the simulation model 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' (red "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 (%) – The reactor thermal power (%) is the percentage of the rated thermal output from the reactor which is 3926 MWth at full power.
  - ⇒ Turbine generator output power (Gross) (%)
  - ⇒ Reactor pressure (kPa)
  - ⇒ Core flow (kg/s)
  - ⇒ Reactor water level (m)
  - ⇒ Balance of plant (BOP) steam flow (kg/s) — that means steam flow after the main steam isolation valve
  - ⇒ Feedwater flow (kg/s)
  - ⇒ Average fuel temperature (°C)

- The bottom left hand corner allows the initiation of two major plant events:
  - ⇒ 'Reactor trip' or 'reactor scram'
  - ⇒ 'Turbine trip'

These correspond to hardwired push buttons in the actual control room.

- The box above the trip buttons shows the display currently selected (i.e. 'plant overview'); by clicking and holding on the arrow in this box the titles of the other displays will be shown, and a new one can be selected by highlighting it;

- The remaining buttons in the bottom right hand corner allow control of the simulation one iteration at a time ('Iterate'); the selection of initialization points ('IC'); insertion of malfunctions ('Malf'); and calling up the 'Help' screen (online hyperlinked "Help" is not available yet).

As a general rule, all dynamic display values shown in display boxes on the screens follow the following conventions:

- All pressure values are designated as "P" next to the display box, and have units of kPa;

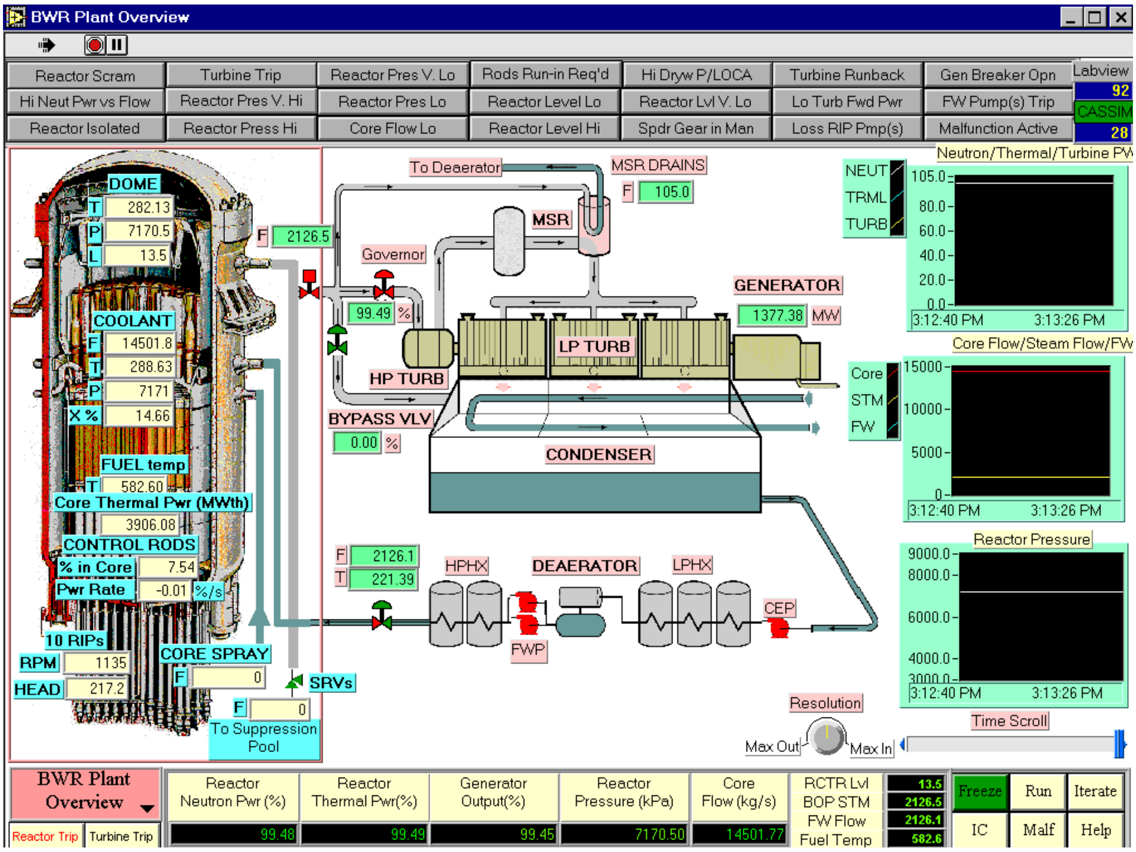
- All temperature values are designated as “T” next to the display box and have values of ºC;
- All flow values are designated as “F” next to the display box and have values of kg/s;
- 2 phase qualities are indicated as “X” next to the display box and have % as units.

Valve status and pump status as shown by dynamic equipment symbols are represented as follows:

- Valve status — red for valve fully open; green for valve fully closed; partial red and green indicates partial valve opening;
- Pump status — red for running; green for stopped.

3. SIMULATOR DISPLAY SCREENS

3.1 BWR PLANT OVERVIEW SCREEN



This screen 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 point kinetic model with six groups of delayed neutrons; the decay heat model uses a three-group approximation; 2-phase flow and heat transfer. Reactivity calculations include the reactivity of the control rods — FMCRD, fine motion control rod drives, reactivity feedback effects due to Xenon, two-phase voiding in channels; fuel temperature (Doppler) and moderator (light water) temperature.

- The reactor parameters displayed are:

  Reactor dome section

  ⇒ Dome steam temperature (°C)

  ⇒ Dome pressure (kPa)

  ⇒ Steam flow from core (kg/s)

  ⇒ Reactor water level (m)

  Reactor core section

  ⇒ Neutron power rate (%/s)

⇒ Thermal power generated by core (MW(th))

⇒ Average fuel temperature (ºC)

⇒ Coolant flow rate in core (kg/s)

⇒ Coolant pressure at core exit (kPa)

⇒ Coolant temperature at core exit (ºC)

⇒ Coolant quality at core exit (X%)

⇒ Control rods position in core (% of total length in core). Note that control rods reactivity worth is as follows: 100% in core - negative 170 mk (milli-k); 100% out-of-core - positive 120 mk.

Reactor downcomer section

⇒ Reactor internal pumps head (kPa)

⇒ Reactor internal pumps speed (RPM)

- Outside the reactor pressure vessel (RPV) and still inside the containment are shown:

  ⇒ Main steam isolation valve status: red means fully open;

  ⇒ The main steam lines have branch connection to the safety relief valves (SRVs) that are connected to the suppression pool inside the containment. Here all the SRVs are shown in "one equivalent valve" symbol; in fact there are 8 SRVs, with 2 SRVs associated with each main steam line; and there are four separate main steam lines. So the steam flow shown is for total steam flow through all the SRVs.

  ⇒ Emergency core cooling (ECC) injection is shown here as "total ECC core injection" flow (from LP flooder and HP flooder sources) in case of loss of coolant accident. Note that in this screen, no distinction is made between LP flooder and HP flooder. They are all treated as one source and go directly to the core.

  Note: The containment drywell and wetwell are modeled in this simulator. In the event of major accidents inside the drywell, such as feedwater line break, steam line break, and reactor vessel bottom break (LOCA), these breaks will cause high pressure in the drywell, which in turn will trigger the LOCA signal. As a result, ECC will be started, the reactor will be scrammed, and "isolated". See detailed descriptions for BWR Containment Screen.

- Outside containment is the balance of plant systems — turbine generator, feedwater & extraction steam. The following parameters are shown:

  ⇒ Status of control valves is indicated by their colour: green is closed, red is open; the following valves are shown for the steam system:
  Turbine governor valve opening (%)
  Steam bypass valve opening (%)

  ⇒ Moisture separator and reheater (MSR) drains flow (kg/s)

  Generator output (MW) is calculated from the steam flow to the turbine

- Condenser and condensate extraction pump (CEP) are not simulated but the pump status is shown.

- Simulation of the feedwater system is very much simplified; the parameters displayed on the plant overview screen are:
  - ⇒ Total feedwater flow to the steam generators (kg/s)
  - ⇒ Average feedwater temperature after the high pressure heaters (HPHX)
  - ⇒ Status of feedwater pumps (FWP) is indicated as red if any pumps are 'ON' or green if all the pumps are 'OFF'

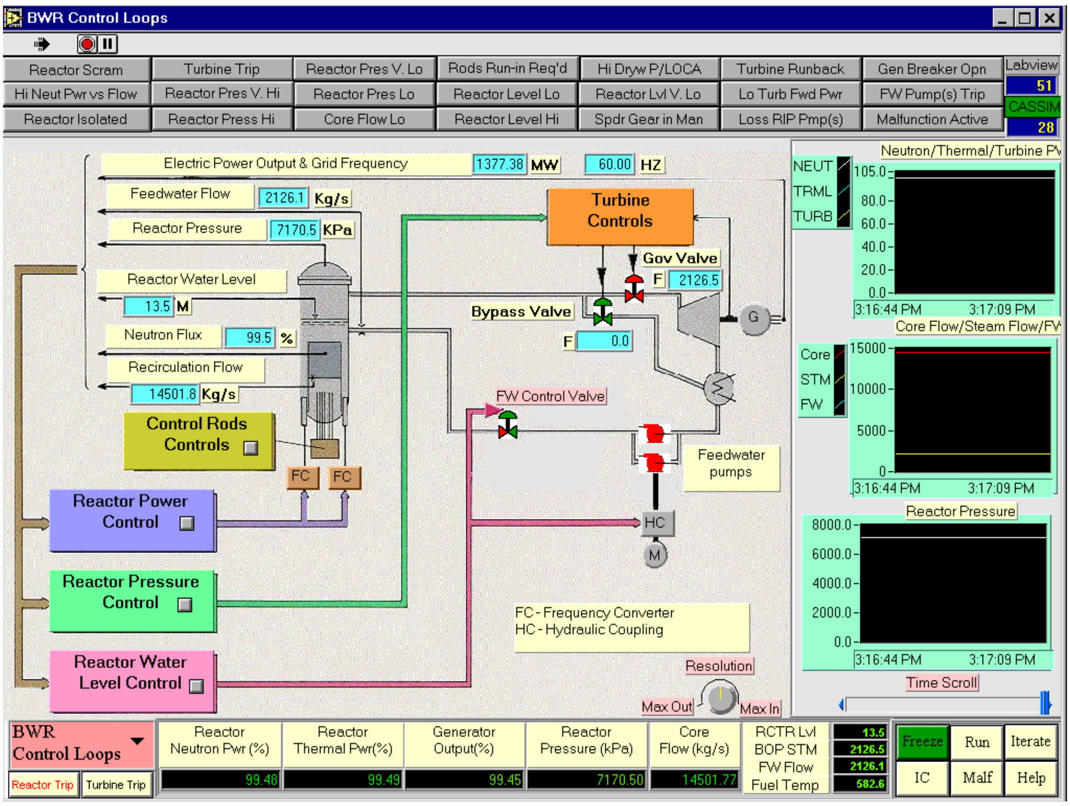
Three trend displays show the following parameters:

- Reactor neutron power, reactor thermal power and turbine power (0-100%)
- Core flow, steam flow, feedwater flow (kg/s)
- Reactor pressure (kPa)

  The upper and lower limits of the parameter trends can be altered while the simulator is running by clicking on the number to be changed and editing it.

  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.

3.2 BWR CONTROL LOOPS SCREEN



This screen shows all the essential control loops for the generic BWR plant, and the essential control parameters for these loops. The parameters are:

- Generator output and frequency
- Feedwater flow
- Reactor pressure
- Reactor water level
- Neutron flux
- Core flow

The essential control loops are:

- **Control rods control** — press the button to display a pop-up window, which describes the functions of the control system. The control rod drive system is composed of three major elements: the fine motion control rod drive, FMCRD mechanisms; the hydraulic control unit (HCU) assemblies; the control rod drive hydraulic subsystem (CRDH). The FMCRDs, together with the other components are designed to provide:

  (1) Electric-motor-driven positioning for normal insertion and withdrawal of the control rods;

  (2) Hydraulic-powered rapid control rod insertion (scram) in response to manual or automatic signals from the reactor protection system (RPS);

  (3) Electric-motor-driven "Run-Ins" of some or all of the control rods as a path to rod insertion for reducing the reactor power by a sizable amount.

  For manual control of control rods and recirculation pumps, go to Screen "BWR Power/Flow Map & Controls".

- **Reactor power control** — press the button to display a pop-up window, which describes the functions of the control system. The reactor power output control system consists of control rods, rod drive system and recirculation flow control system. The control rods and their drive system maintain a constant desired power level by adjusting the position of the rods inside the core. The recirculation flow control also controls the reactor power level by changing the recirculation flow to alter the void density of the two-phase water in the core, which leads to a change in reactivity in the core due to the altered neutron moderation efficiency of the coolant. The recirculation flow is controlled by recirculation pumps known as reactor internal pumps (RIPs). The pump speed changes according to the change of frequency of the induction motor that drives the pump. Different pump speed will give rise to different pump dynamic head in the core recirculation flow path, resulting in different core flow. This recirculation flow control system is capable of changing the reactor output rapidly over a wide range. Go to Screen "BWR Reactivity & Setpoints" for changing the reactor power setpoint, and observe the Power & Recirculation Flow relationship in Screen "BWR Power/Flow Map & Controls"

- **Reactor pressure control** — press the button to display a pop-up window, which describes the functions of the control system. When the reactor is in power-level operation, the reactor pressure is automatically controlled to be constant. For that purpose, a pressure controller is provided and is used to regulate the turbine inlet steam pressure by opening and closing the turbine governor control valve and the turbine bypass valve. Currently, the reactor pressure setpoint is set at plant design pressure of 7170 kPa.

- **Reactor water level control** — press the button to display a pop-up window, which describes the functions of the control system. In order to suppress the water carry-over in the steam going to the turbine as well as to prevent the core from being exposed, three signals detecting the feedwater flow, the main steam flow, and the water level inside the reactor pressure vessel are provided. The flow of feedwater is automatically controlled to maintain the specified water level by a "three element" control scheme: steam flow, feedwater flow, and water level. The valve opening of the feedwater control valve provided at the outlet of the feedwater pumps is regulated by the control signal as a result of this "three-element" control scheme. To modify the reactor water level setpoint, go to Screen "BWR Feedwater & Extraction Steam", and call up the related Pop-Up Window.
- **Turbine control** — the turbine control employs an electrohydraulic control system (EHC) to control the turbine valves. Under normal operation, the reactor pressure control (RPC) unit keeps the inlet pressure of the turbine constant, by adjusting the opening of the turbine "governor" which controls the opening of the turbine governor valve opening. Should the generator speed increase due to sudden load rejection of the generator, the speed control unit of the EHC has a priority to close the turbine governor valve over the reactor pressure control (RPC) unit.
- **Turbine steam bypass system** — the simulated BWR plant is designed with turbine steam bypass capacity of over 75% rated steam flow. Hence, in the event of any reactor pressure disturbances, either caused by reactor power sudden increases, or due to turbine load rejection, or frequency changes, when the reactor pressure control unit cannot cope with these pressure increases fast enough, the turbine bypass valve will open up to pass steam to condenser to reduce sudden reactor pressure increases. The setpoint for the bypass valve to "come in" when the turbine is not "tripped" is — 130 kPa (called bias) over the normal reactor setpoint of 7170 kPa. That means the bypass valve will not open until the reactor pressure increases to > 7300 kPa; this gives room for the turbine control valve to act in an attempt to control pressure back to 7170 kPa. However, if the turbine is tripped, the bias will be removed and the setpoint for the bypass valve is 7170 kPa.

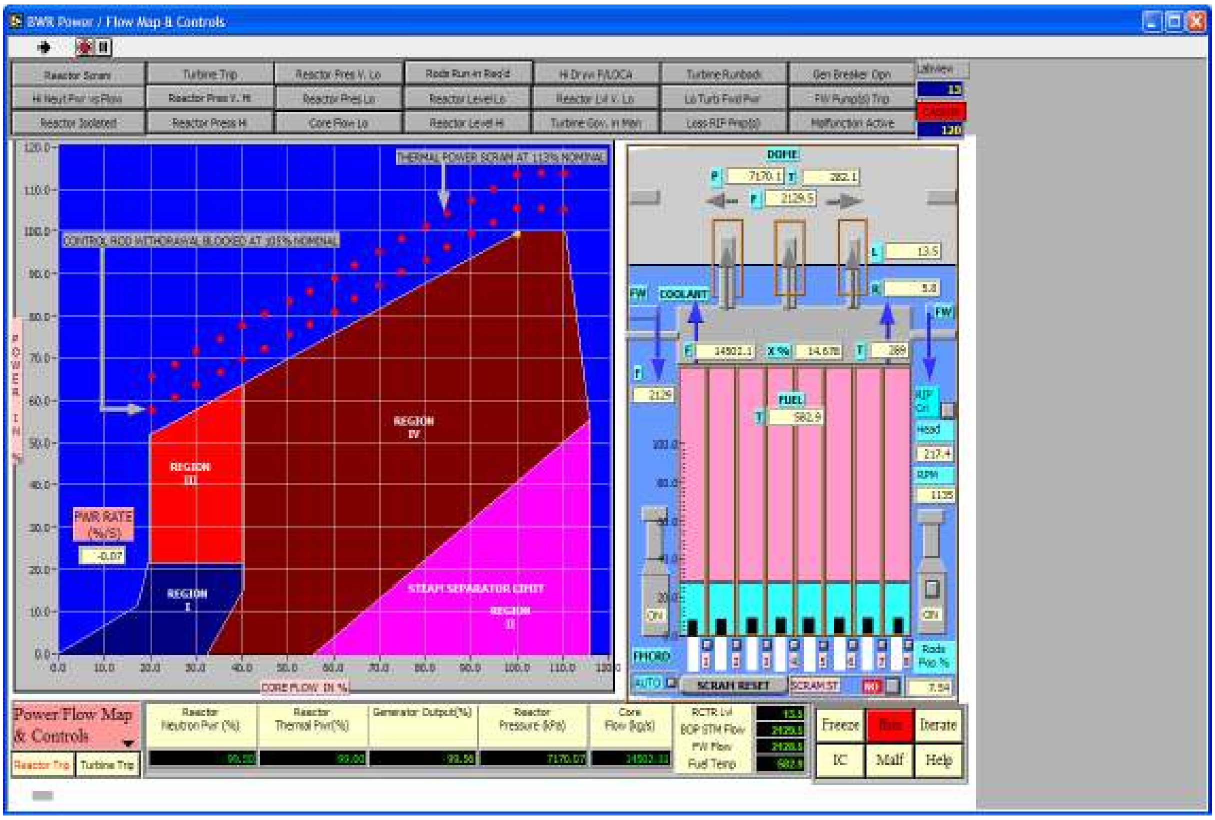
3.3 BWR POWER/FLOW MAP & CONTROLS

This screen shows

(a) The relationship between reactor neutron power versus core flow;

(b) The reactor core conditions with respect to boiling height; water level; fuel temperature; coolant temperature, pressure and flow; steam pressure, flow and temperature;

(c) Controls for scramming the reactor, as well as for resetting the scram; the AUTO/MANUAL controls for the control rods (FMCRD) and for the reactor internal pumps (RIPs) drive unit.



- **POWER/FLOW MAP**
- The power flow map is a representation of reactor power vs. recirculation flow. The horizontal axis is the core flow in % of full power flow. The vertical axis is reactor neutron power in % of full power.
- Any operation path that changes the power and the flow from one condition to another condition through control rod maneuver and/or recirculation flow change can be traced on this map. Under normal plant start-up, load maneuvering, and shutdown, the operation path through REGION IV is recommended. In fact, the line which borders between REGION I & IV, REGION III & IV, the blue region and REGION IV is the “maximum power-flow” path to be followed for power increases and decreases, and usually operation of the plant is “below” this “**maximum**” power-flow line.
- Limits are imposed to prevent operation in certain areas of the Power - Flow Map:

  (1) To maintain core thermal limits and to avoid operation above licensed power level - there are three measures to prevent that:

  (a) **Control rods withdrawal “Blocked**” (red dotted line) — if at any time, the current power exceeds 105% of the power designed for the current flow rate (in accordance with the maximum power-flow line as described above), the Control Rods withdrawal will be “blocked” until the power drops to 5% less than the current value. Should this occur, the alarm **“Hi Neut Pwr vs Flow**” will be in “Yellow” color, as well, in the BWR Reactivity & Control Screen, there will be a “yellow color message” saying “Controls Rods Out Blocked”.

(b) **Control rods "Run-in"** — if any time the current power exceeds 110% of the power designed for the current flow rate (in accordance with the maximum power-flow line as described above), the control rods will be inserted into the core to reduce power quickly and the "Rods Run-in" will be stopped until the power has been reduced to 10% less than the current value. Should this occur, the alarm **"Hi Neut Pwr vs Flow**" will be in "Yellow" color, as well as the alarm "**Rods Run-in Req'd**".

(c) **Reactor scram** (red dotted line) — if any time the current power exceeds 113% of the power designed for the current flow rate (in accordance with the maximum power-flow line as described above), the reactor will be scrammed.

(2) To avoid operation where core instability may occur — in REGION III.

It is a well-known and well-documented phenomenon in the BWR that oscillations in neutronic and thermal-hydraulic parameters occur during operation in the "low flow - high power" region identifiable in the Power/Flow Map as REGION III.[1]

Research has shown that such oscillations are characterized as "density wave" oscillations. From a physical point of view, the removal of thermal power by boiling water in a vertical channel, in a closed or open loop configuration, may cause instability in the operation owing to density changes and various thermalhydraulic feedback mechanisms. Since the coolant is also a neutron moderator, an oscillation in the coolant density (void content) is reflected as a variation of the thermal neutron flux, which in turn, via the heat flux, affects the void. This may cause a coupled neutronic-thermalhydraulic oscillation under certain power and core flow conditions. The details of core instability in Region III belong to an advanced topic that is beyond the scope of this manual.

(3) To avoid operation where excess moisture in the steam may be carried to the main turbine — in REGION II.

- **REACTOR CORE GRAPHICS**

  The right side of the screen depicts the reactor core conditions at all operations. The control devices for control rods and the reactor internal pumps are provided as well. Starting from the bottom:

- **FMCRD auto/manual button** — this button when pressed will allow the user to switch the control rods to be either under the "**automatic**" control scheme or under "**manual**" control. If they are in "manual" mode, the switch status will be indicated as "MAN", and the user can then control the rods by pressing the button above the designated number of the control rod bank #1 to #8 respectively. A control pop-up will appear when the button is pressed, allowing the user to "insert" or to "withdraw" each "bank" of rods separately, by using the "in", or "out" pushbutton respectively in the pop-up. To stop the movement of the rods, use the "stop" pushbutton in the pop-up.

  When the FMCRDs are in "AUTO", the automatic control scheme is in control, and its details are described in the "BWR Reactivity & Controls Screen" section. In Auto mode, all the controls rods move together as controlled by the reactor power regulating system.

[1] OECD Report OCDE/GD(97)13: "State of the Art Report on boiling water reactors stability (SOAR on BWRS), January 1997.

Note:

- There are approximately 208 FMCRDs in total, they are positioned and calibrated with reactivity worth of -170 mk when all of them are 100% in core, and +120 mk when all of them are 100% out of core; 0 mk when they are at the reset line.
- For the purpose of this generic simulator, the rods are grouped in 8 banks, so each bank of rods has + 15 mk when fully out of core; and -21.25 mk when fully in core. The full-speed travel time for the rod movement during power maneuvering is typically 60 s.
- The FMCRDs will be fully inserted into the core in the event of **a reactor scram**. In such case, the fast insertion time is typically 3 s. for 100% insertion.
- The average rods position in core is shown on the right hand bottom corner.

- **SCRAM status indication, manual scram/reset button, SCRAM reset button —** when the reactor is scrammed, and if scram conditions still exist, there will be a "YES" sign next to the "SCRAM ST" indicator and the alarm "Reactor Scram" will be in "red" color. Assuming the scram conditions have already disappeared, and the user wishes to reset the scram, the button to the right of the "YES" indicator is pressed, which will bring up a control pop-up. The user can then press the "Reset" pushbutton on the pop-up. If the reactor scram conditions do not exist at that time, then the "YES" sign will be changed to "NO" sign, meaning that the SCRAM Status indicates "NO" scram conditions.

  At this point, the user can proceed to press the "SCRAM RESET" button on the left side of the "SCRAM ST" indicator. When this button is pressed, the "Reactor Scram" alarm will disappear, and the rods withdrawal will begin, as can be seen from the downward arrows shown for the rods banks. The rods withdrawal will stop at the "reset line", pending on control actions taken by the reactor regulating system.

- **ON/OFF control for RIP pump motor** — there are 10 reactor internal pumps (RIPS), but they are modeled as one "lumped" pump, so the ON/OFF control button is used to turn "ON" or "OFF" all the pump motors. When the motor power is "OFF", the speed drive will go to the minimum position, giving "zero" pump head. When the power is "ON", the speed drive signal is subject to the flow controller signal that is described later. The RIP's speed changes according to the change of frequency of the induction motor that drives the pumps. Different pump speed will give rise to different pump dynamic head in the core recirculation flow path, resulting in different core flow. The automatic flow control scheme is handled by a flow controller. First, based on the reactor power setpoint, there is a pre-programmed flow rate schedule according to that power setpoint. The pre-programmed schedule is typically to follow the "maximum power — flow" path as described in the POWER-FLOW map section. Given the flow setpoint, the flow controller will drive the speed drive to provide enough pump head until the desired flow rate is achieved.

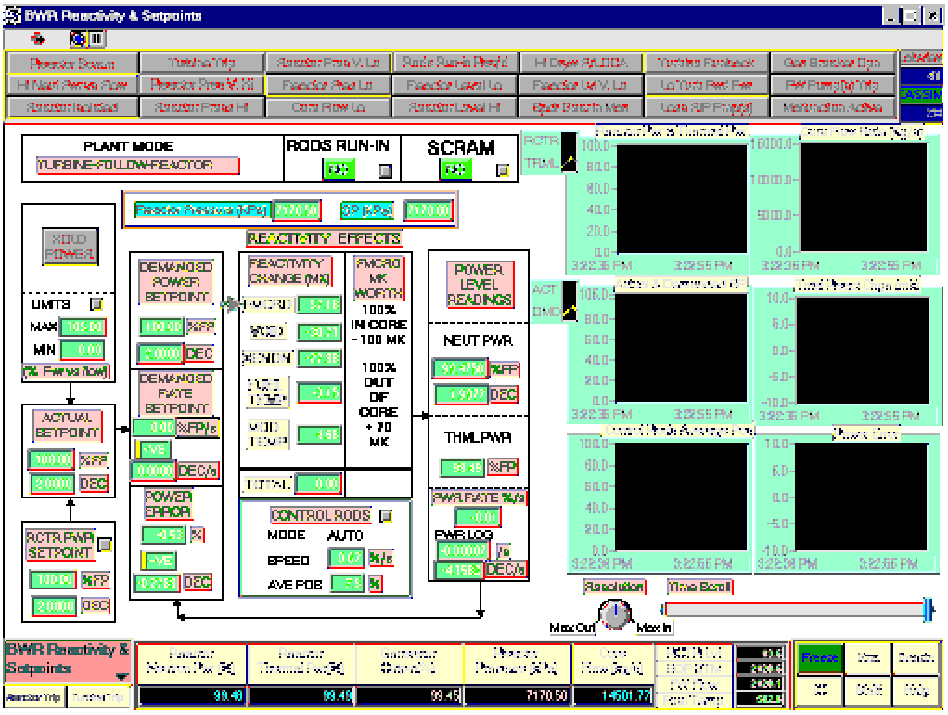
  RIP Trips and Runback -

  - 4 RIPs tripped on: High Pressure (7.76 MPaG); L3: and turbine stop valve closure or fast TCV closure but overruled if bypass valves open
  - 6 RIPs tripped on L2
  - RIP runback (1% per second) on scram

As noted above, the 10 RIPs are modeled on the basis of one lumped pump head, hence, in the event of RIP Trips and Runback, one will see the corresponding pump head and speed decrease to match the respective RIP trip or runback scenario. In addition, when RIP runback condition appears, one will see the Runback alarm shown on the pump in the BWR Power/Flow May and Control Screen.

- **Average pump head indicator, average pump speed indicator, flow controller control button** — the average pump head is shown in kPa, and the average pump speed is shown in RPM. The flow controller button is labeled as RIPCrl. When this button is pressed, it will show a typical controller template. The setpoint is on remote setpoint (RSP), meaning that it receives the reactor core flow setpoint (kg/s) from the pre-programmed flow rate scheduler computer control program. The horizontal "blue" bar is for indication of the current core flow rate; the "green" pointer is the core flow setpoint indicator. When the core flow rate is at setpoint, the "green" pointer can be seen to be at the tip of the "blue" bar. When the "auto" button is pressed, the controller is no longer subject to remote setpoint, rather it will be subject to "local" setpoint, and the user can enter a "new" core flow setpoint (kg/s) into the box under "SP". Alternatively, the user can switch the controller to manual. In that case, the output (%) box will show an "up/down" ("increase/decrease") arrow. One can press the "up/down" arrow to change the out (%) value incrementally, or, a % value can be entered into the out (%) box to change the controller output directly. The new out (%) value will change the speed of the pumps' drive, thus changing the head produced by the pumps and consequently the core flow (kg/s).
- **Core conditions display -** the following parameters are shown for the core conditions:
  - Average fuel temperature
  - Coolant flow rate, temperature at core exit, and quality (%) at core exit**;** feedwater flow rate; coolant recirculation ratio "R" are shown. The "blue" arrows show the flow path of the coolant out from the core channels, as it goes to the core upper plenum, enters the dome space, mixes with incoming feedwater, and goes down to the downcomer, to enter the suction path of the reactor internal pumps. Then from the RIPs discharge, the coolant enters the core lower plenum generally subcooled. As the subcooled coolant enters the core channels again, it receives heat from the fuel bundles, saturates and then becomes a two-phase mixture that exits the core with a certain quality. The saturated steam escapes from the mixture; the remaining saturated water is recirculated back to the downcomer and mixes with the incoming feedwater.
  - The boiling height – the two-phase boiling region of the core is shown in a "pink" color. It is animated, so as the boiling height changes according to the core conditions, the "pink" section boundary also changes. The same applies to the "light blue" subcooled section - or non-boiling section of the core.
  - The water level in the core is indicated with "blue" color and is animated. As the level changes, the "blue" section boundary changes.
  - For the Dome space, the gray arrows show the flow path of saturated steam; the flow, pressure and temperature are shown.

3.4 BWR REACTIVITY & CONTROLS SCREEN



This screen shows input devices which facilitate reactor power setpoint entry, as well as to facilitate reactor 'Manual Scram', or 'Manual Rods Run-in". These inputs interact with an underlying reactor regulating system.

- The user can enter **new reactor power target** and **power change rate** by pressing the button located near the bottom left side of the screen next to "RCTR PWR SETPOINT". When this button is pressed, a control pop-up will allow the user to enter the reactor power target in %, and the rate in % full power per second (if current power is > 20% FP), or % present power per second (if the current power is < 20% FP). The purpose is to allow higher power rate change only at higher power.

  Note on power change rate: according to generic information obtained from the Advanced BWR vendor, control rods are used to achieve the desired power level, *from 0% FP to 65 % FP*, by adjustment of their positions in the core at a rate equivalent to a power change rate of **maximum** *1 % full power per second.*

  <u>The suggested nominal rate is 0.5 % per second or lower, particularly if one observes fluctuations in level, and power during load changes</u>.

  The recirculation flow control also controls reactor power *from 65 % FP to 100 % FP* by causing the density of the water/steam mixture used as moderator to change. The flow rate is adjusted by a variable speed pump (such as the internal pumps of the ABWR) at a rate equivalent to a power change rate of **maximum** *30% full power per minute (0.5 % FP per second).*

  These maximum power rates could be higher than that for a conventional BWR, typically 2.5% FP per minute, below 65 % FP, according to feedback from experienced

BWR personnel. This could be due to the technology advance made in Advanced BWR, with the use of digital controls and FMCRD, etc.

For realism, it is suggested that the simulator user should "observe" this maximum power change rates guidelines during simulator exercises. However, recognizing the fact that this is an educational simulator, the rate control in the simulator may be different than that of the actual BWR in operation.

As well, one may observe that the power rate entered by the user may not be the same power rate being displayed. The reason is that the power rate being displayed is an instantaneous value of the power rate at any time. To get an average power rate, one should integrate the instantaneous values over a specific time.

- After the setpoint and rate are entered, the "ACTUAL SETPOINT" section reflects the setpoint actually accepted by the regulating system. Then the incremental demanded setpoint is computed in the "DEMANDED SETPOINT" section; as well the rate is shown in "DEMANDED RATE SETPOINT" section. The POWER ERROR is computed as:

  POWER ERROR = ACTUAL POWER - DEMANDED POWER

- The reactor regulating system will check if the current power is < 65%. If it is, then the control rods movement is necessary. Based on the power error - whether it is positive or negative, the rods will be inserted or withdrawn accordingly, so that the power error becomes zero. If the current power is > 65%, then usually rod movement is not required; the new incremental demanded power setpoint signal is sent to the flow rate scheduler (as described in previous section) which will provide a flow rate setpoint to the flow controller. If the flow rate increase/decrease cannot provide enough reactivity change causing sufficient reactor power increase/decrease so that the power error is less than a pre-determined dead-band, the rods movement will become necessary at that time so that the power error is within limits.

- This screen provides the important information regarding reactivity changes as shown by the various reactivity feedback effects - void density, xenon, fuel temperature, coolant temperature, as well as the control rods reactivity changes as a result of their movement in the core. Note that reactivity is a computed not a measured parameter, it can be displayed on a simulator but is not directly available at an actual plant. Also note that when the reactor is critical the total reactivity must be zero.

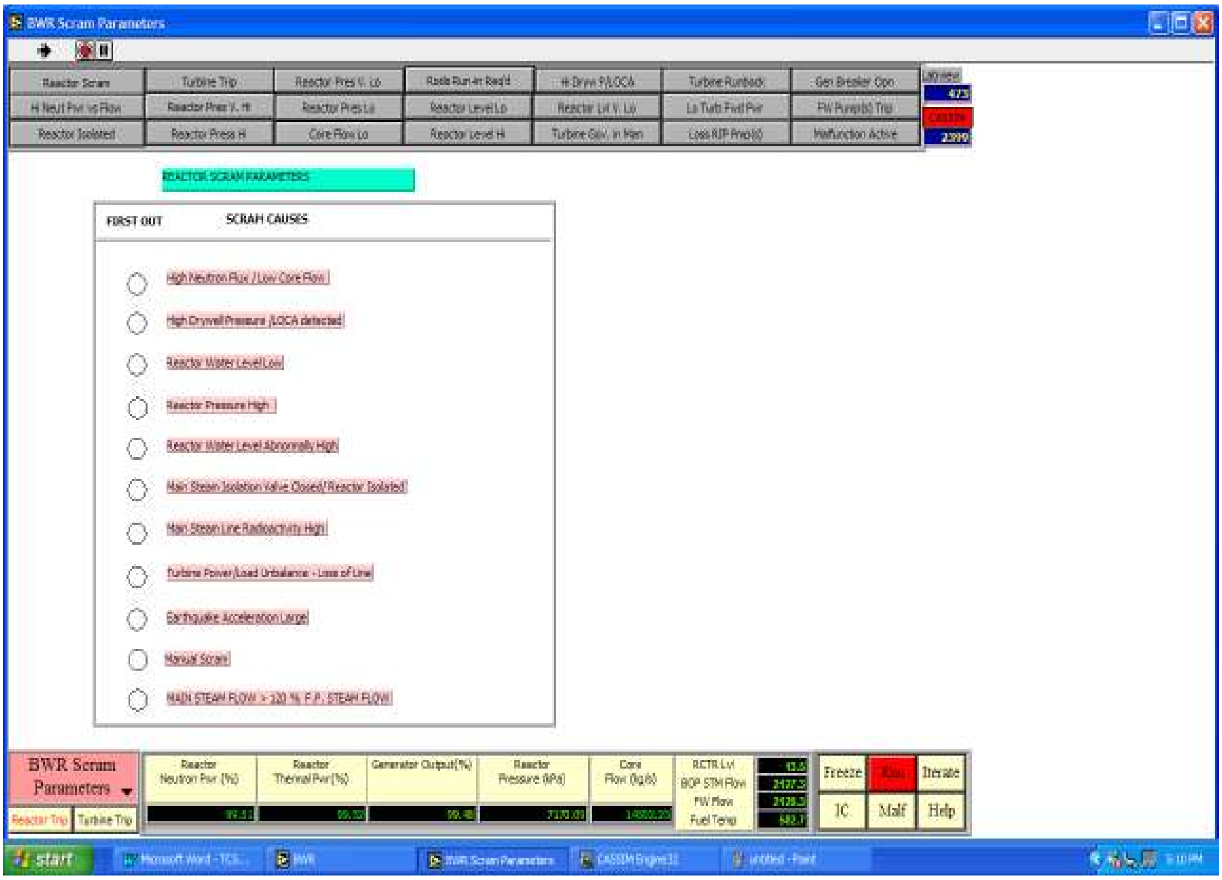
- Note that the BWR plant is always operating in a turbine-following-reactor mode.

- The buttons at the top of the screen allow the user to perform a "manual" "rods run-in", as well as a "manual" reactor scram.

- The "HOLD POWER" button near the top left hand corner allows the user to "suspend" reactor power changes at any time. Just pressing the button once will result in the Demanded Power Setpoint being set to "frozen", if it was increasing or decreasing initially.

- Near the bottom of the middle section of the screen is the button that can switch the controls rods "AUTO/MANUAL".

3.5 BWR SCRAM PARAMETERS SCREEN

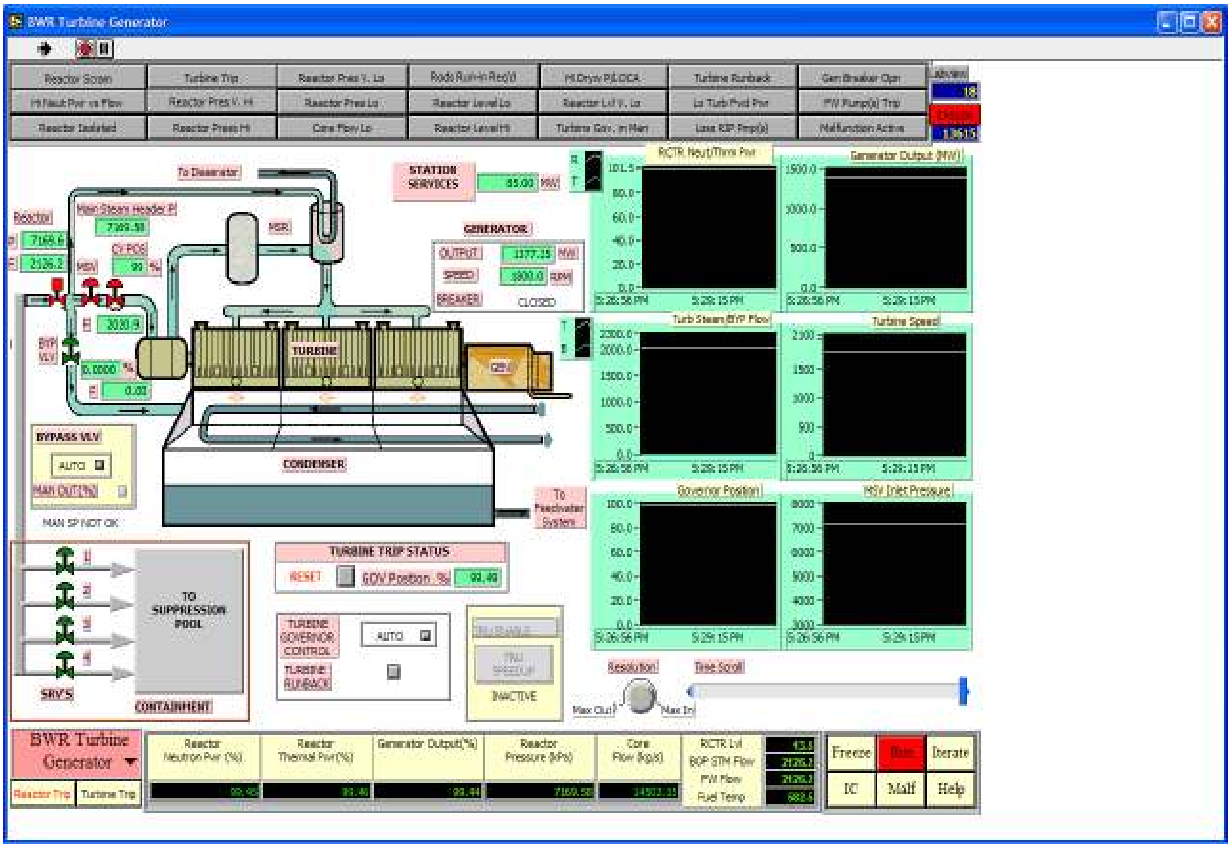


This screen shows all the parameters that will cause reactor scrams:

- High neutron flux/low core flow - as described previously, if at any time the current power exceeds 113% of the power designed for the current flow rate (in accordance with the maximum power-flow line as described above), the reactor will be scrammed.
- High drywell pressure/LOCA detected — if the drywell pressure exceeds 114.6 kPaa, then the LOCA logic senses that a LOCA condition has occurred.
- Reactor water level low — the scram setpoint is 12.30 meters above reactor bottom; L3 = 3.30 TAF. Normal level is 13.5 meters above reactor bottom.
- Reactor pressure high — the scram setpoint is 7870 kPa. Normal reactor pressure is 7170 kPa.
- Reactor water level very high — the scram setpoint is 14.0 meters above reactor bottom. L8 = 5.0 TAF.
- Main steam isolation valve closed/reactor isolated.
- Main steam line radioactivity high.

- Turbine power/load unbalance or loss of line (load rejection).
- Earthquake acceleration large.
- Manual scram.
- Main Steam Flow > 120 % FP Steam Flow. When the main steam flow exceeds 2613 kg/s (> 120 % nominal full power steam flow), the reactor will be scrammed.

3.6 BWR TURBINE GENERATOR SCREEN



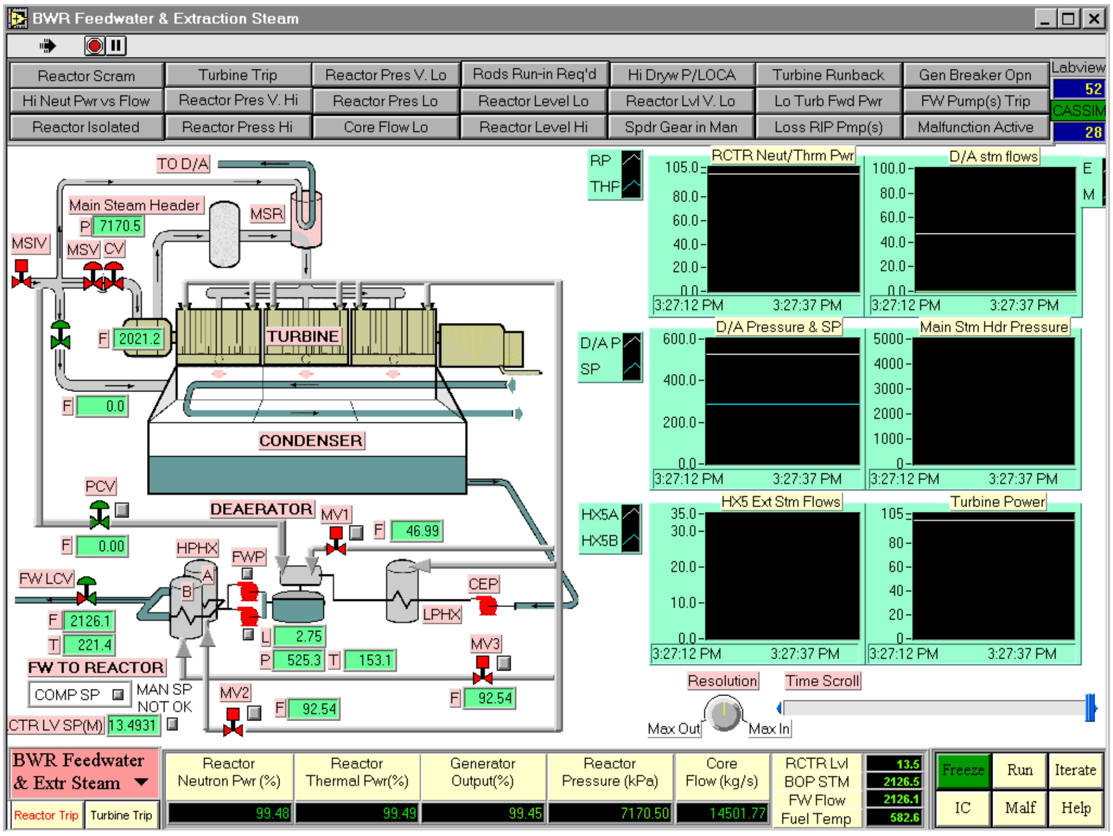
This screen shows the main parameters and controls associated with the turbine and the generator. The parameters displayed are:

- Reactor side main steam pressure and main steam flow (before the isolation valve); main steam isolation valve status
- Main steam header pressure after the main steam isolation valve.
- Status of main steam safety relief valves (SRVs)
- Status, opening and flow through the steam bypass valves
- Steam flow to the turbine (kg/s)
- Governor control valve position (% open)
- Generator output (MW)
- Turbine/generator speed of rotation (rpm)
- Generator breaker trip status
- Turbine trip status
- Turbine control status
- All the trend displays have been covered elsewhere or are self explanatory

The following pop-up menus are provided:

- TURBINE RUNBACK — sets target (%) and rate (%/s) of runback when ‘Accept’ is selected
- TURBINE TRIP STATUS — trip or reset
- Steam bypass valve ‘AUTO/MANUAL’ control — AUTO select allows the pop-up window that appears to transfer to MANUAL control, following which the manual position of the valve may be set.
- Computer control of the turbine governor can be in the “AUTO” mode or “MANUAL” mode. The normal control is in AUTO mode. When the turbine governor is in MANUAL mode, use the pop-up controls “INCREASE/STOP/DECREASE” to change the governor valve position (%) manually. Note: press the “STOP” button first to stop any governor valve movement, then either press “INCREASE” or “DECREASE”. The governor valve will move accordingly upon command, until “STOP” is pressed again.
- Turbine runup/speedup controls

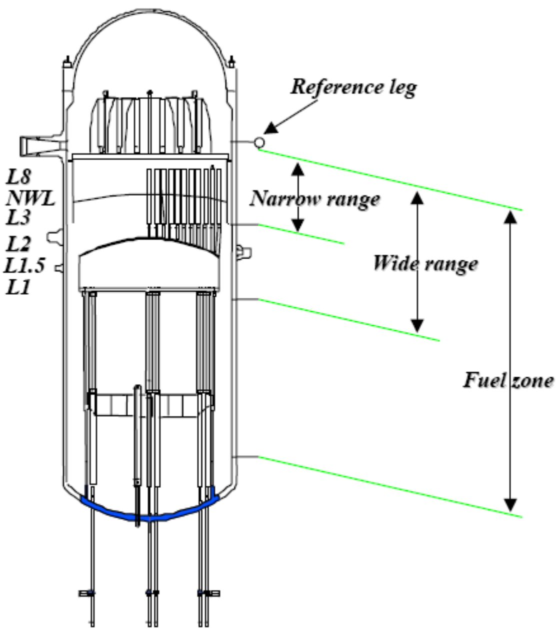
3.7 BWR FEEDWATER AND EXTRACTION STEAM SCREEN



This screen shows the portion of the feedwater system that includes the deaerator, the boiler feed pumps, the high pressure heaters and associated valves, with the output of the HP heaters going to the reactor water level control valves. The following parameters are displayed:

- Main steam header pressure after the main steam isolation valve, steam flow through the turbine governor valve and the bypass valve.
- Deaerator level (m) and deaerator pressure (kPa); extraction steam motorized valve status and controls from turbine extraction, as well as pressure controller controls for main steam extraction to deaerator. The extraction steam flows 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.
- Flow rate at reactor level control valve outlet and feedwater temperature.

BWR Water Level Measurement



- **L8 - Turbine trip, MSIV close**
- **L3 - Scram**
- **L2 - RCIC start**
- **L1.5 - HPCF start**
- **L1 - Remaining ECCS start (i.e., LPFL, ADS)**

Assume Top of Fuel (TAF) = 9 m:normal water level setpoint is 13.5 m, the respective Level Trip Setpoints are implemented as follows:

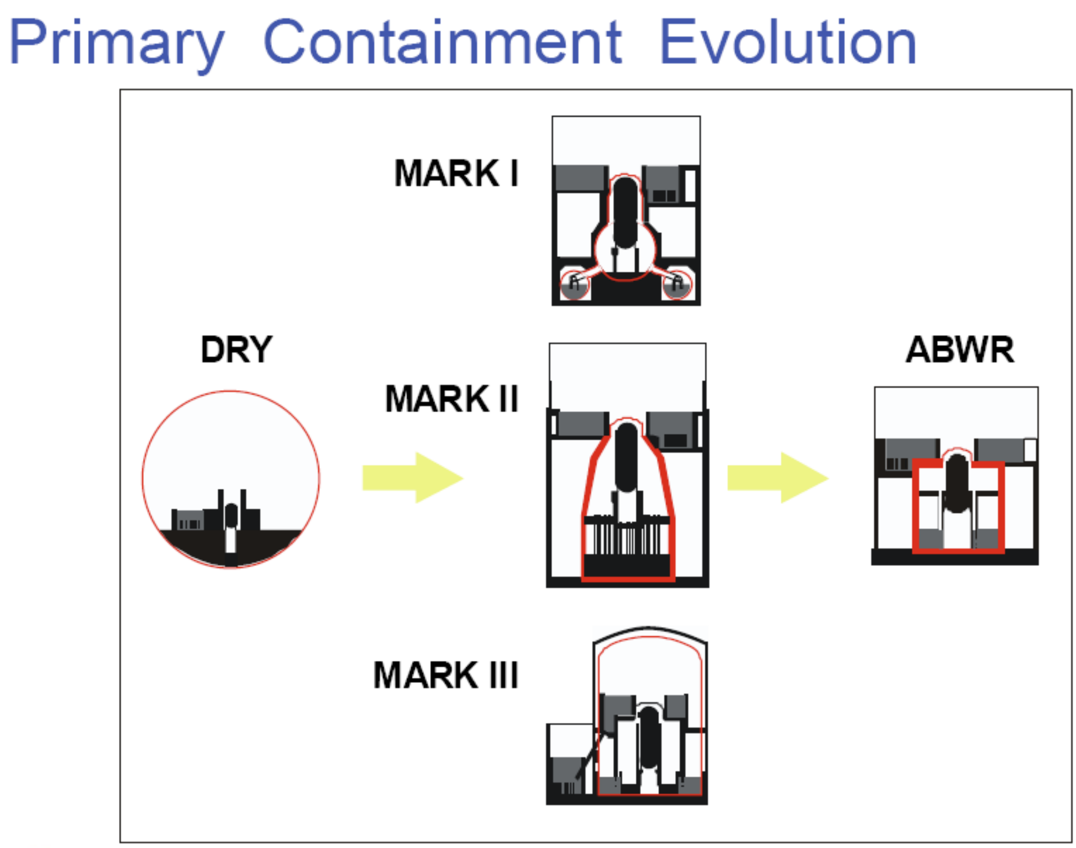
- L 8 = 5.0 m TAF = 14.0 m – action: Turbine Trip.
- L 4 = 4.0 m TAF = 13.0 m – action: core flow runback.
- L 3 = 3.30 m TAF = 12.30 m – action: Reactor Scram.
- L 2 = 2.43 m TAF = 11.43 m – action : Trip 6 RIPs; Start RCIC in the ECC system.
- L 1.5 = 0.978 m TAF = 9.978 m – action: start HPCF pumps in the ECC system.
- L 1 = 0.153 m TAF = 9.153 m – action: start ADS blow down; start 2 RHR pumps in the ECC system.

The Level Trip Setpoints are shown on the Power/Flow Map and Control Screen, and in the BWR Containment Screen.

3.8 BWR Containment

3.8.1 Introduction

The BWR containment configuration has evolved through the Mark 1, Mark 2 and Mark 3 product lines. As this simulator models a plant with internal pumps similar to the ABWR, the ABWR containment is described here. The ABWR containment is similar to a Mark 3 containment in the configuration of the horizontal vents connecting the drywell to the suppression pool, but the wetwell volume is smaller and the arrangement is different from a traditional Mark 3 containment.



3.8.2 ABWR Containment Configuration

The ABWR pressure suppression primary containment system comprises the drywell (DW), wetwell (WW), and supporting systems. The arrangement is shown schematically in Figure 1.

The DW is comprised of two volumes:

• An upper drywell volume surrounding the reactor pressure vessel (RPV) and housing the steam and feedwater lines and other connections of the reactor primary coolant system, safety/relief valves (SRVs) and the drywell HVAC coolers.

• A lower drywell volume housing the reactor internal pumps, fine motion control rod drives (FMCRDs) and under vessel components and servicing equipment.

The upper drywell is a cylindrical, reinforced concrete structure with a removable steel head and a reinforced concrete diaphragm floor. The cylindrical RPV pedestal, which is connected rigidly to the diaphragm floor, separates the lower drywell from the wetwell. Ten drywell connecting vents are built into the RPV pedestal and connect the upper drywell and lower drywell. The drywell connecting vents are extended downward via steel pipes, each of which has three horizontal vent outlets into the suppression pool.

The WW is comprised of a gas volume and a suppression pool filled with water to rapidly condense steam from a reactor vessel blowdown via the SRVs or from a break in a major pipe inside the drywell through the vent system. The wetwell boundary is a cylindrical reinforced concrete wall which is continuous with the upper drywell boundary. A reinforced concrete mat foundation supports the entire containment system and enclosed structures.

The containment structure includes a steel liner to reduce fission product leakage. All normally wetted surfaces of the liner in the suppression pool are made of stainless steel. The allowable leakage is 0.5% per day from all sources, excluding main steam isolation valve (MSIV) leakage.

The drywell is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the drywell and also the rapid reversal in pressure when the steam in the drywell is condensed by the containment sprays that are part of the Emergency Core Cooling System (ECCS)/Residual Heat Removal System (RHR).

A redundant vacuum breaker system connects the drywell and wetwell. The purpose of the wetwell-to-drywell vacuum relief system is to prevent backflooding of the suppression pool water into the lower drywell and to protect the integrity of the diaphragm floor slab between the drywell and wetwell, and the drywell structure and liner.

In the event of a pipe break within the drywell, the increased pressure inside the drywell forces a mixture of noncondensible gases, steam and water through the drywell connecting vents and horizontal vents into the suppression pool, where the steam is rapidly condensed. The noncondensible gases transported with the steam escape from the pool and are contained in the free gas volume of the wetwell. There is sufficient water volume in the suppression pool to provide submergence of the upper row of horizontal vents when water is removed from the pool during post-LOCA drawdown by the ECCS. The design pressure of the containment is 45 psig (4 bar absolute pressure). The suppression pool is sized to accommodate the stored energy within the RPV during a LOCA without exceeding its design temperature.

During isolation transients, when the MSIVs close, the SRVs discharge steam from the relief valves through their exhaust piping and quenchers into the suppression pool which has many hours of decay heat absorption storage capability.

For beyond-design-basis events, piping with temperature actuated valves connect the DCV with the lower drywell. This provides a passive flooding capability. The ABWR containment is normally inerted with nitrogen containing < 3.5% oxygen to avoid hydrogen burning or detonation after a severe accident.

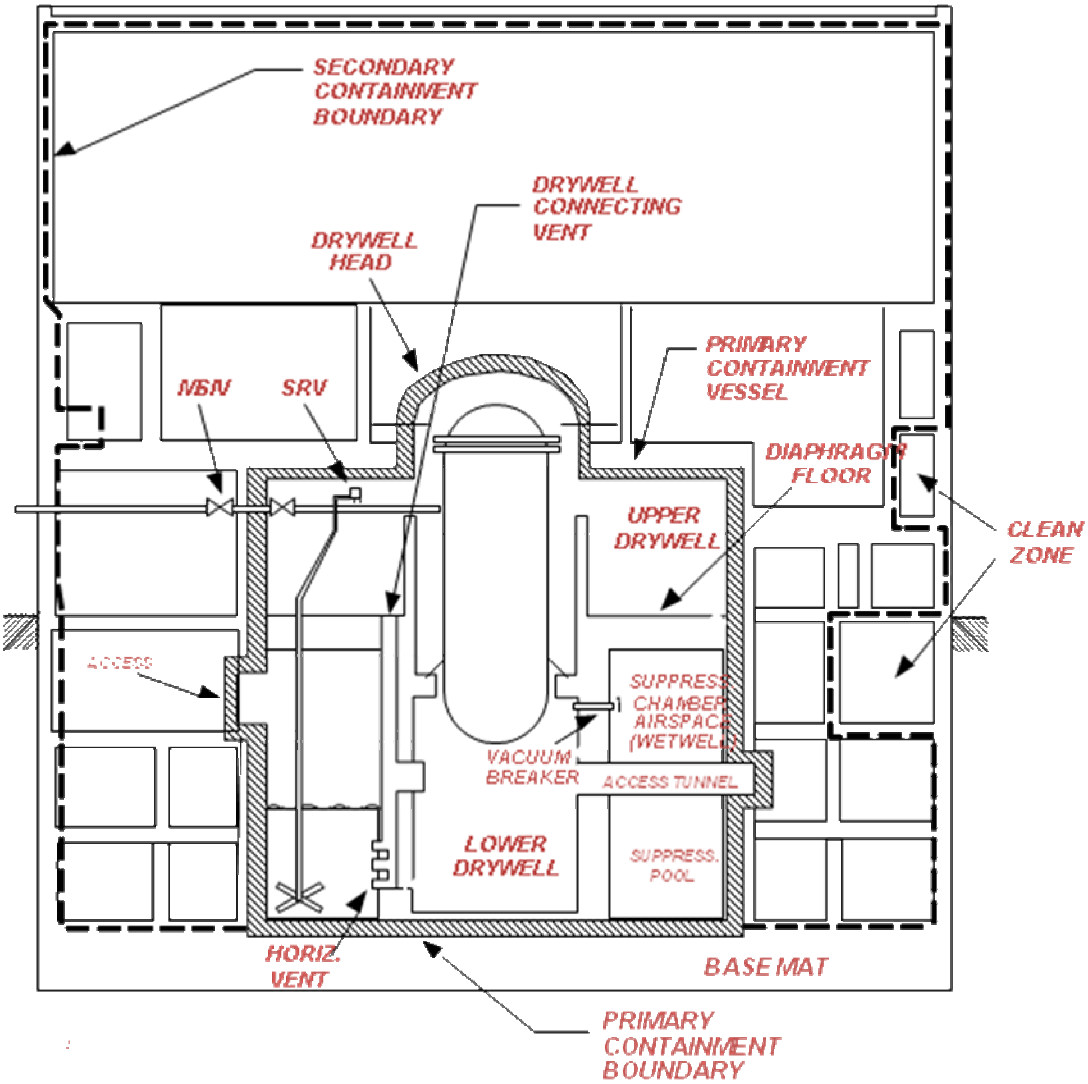


Figure 1: Schematic of Typical ABWR Containment

3.8.3 Containment Cooling Systems

Suppression Pool Water Cooling: This is one of the modes of operation of the Residual Heat Removal System (RHR). The Suppression Pool Cooling system cools the suppression pool water following reactor coolant blowdown in a LOCA event. It is automatically initiated on high suppression pool temperature, and can also be manually initiated. Suction is taken from the suppression pool, the flow goes through the RHR heat exchangers and is returned to the suppression pool

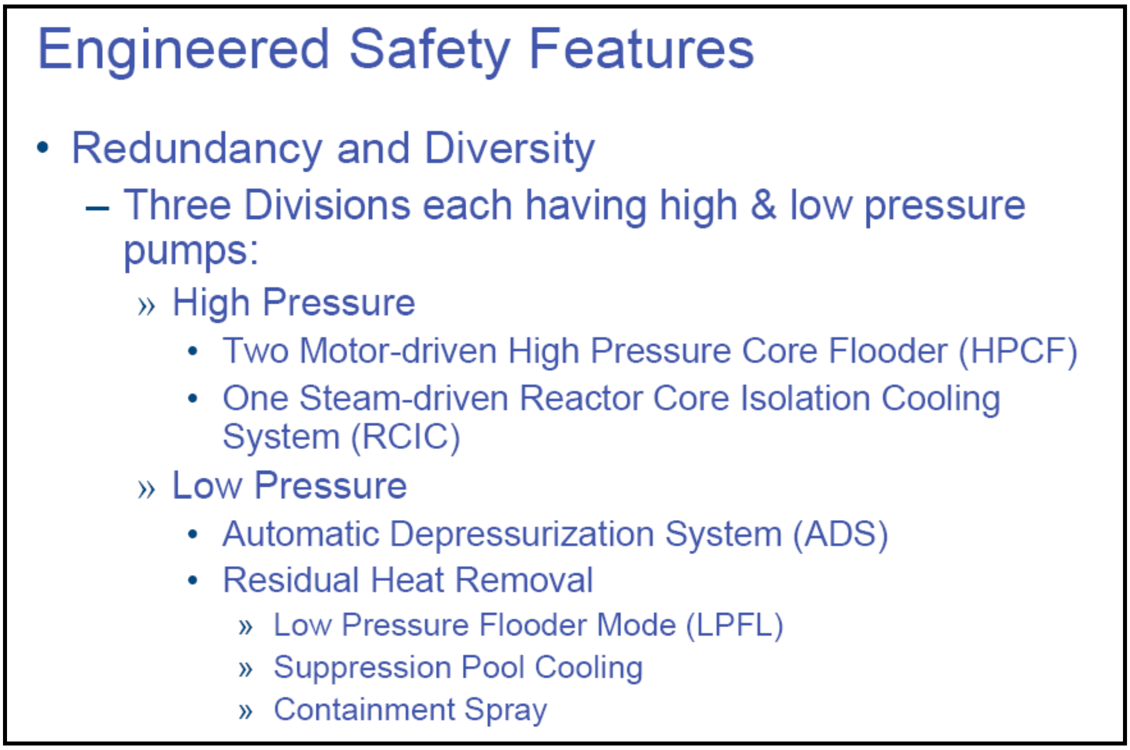
Primary Containment Spray Cooling: This is another mode of the RHR system which is manually initiated and sprays the water from the suppression chamber pool into the drywell and wetwell after the event of a LOCA. This sprayed water in the drywell returns to the suppression chamber through vent pipes after the drywell water level reaches the vent pipe inlet level. It is mixed with the sprayed water in the wetwell and cooled by the RHR system heat exchangers.

Typical ABWR Containment Parameters

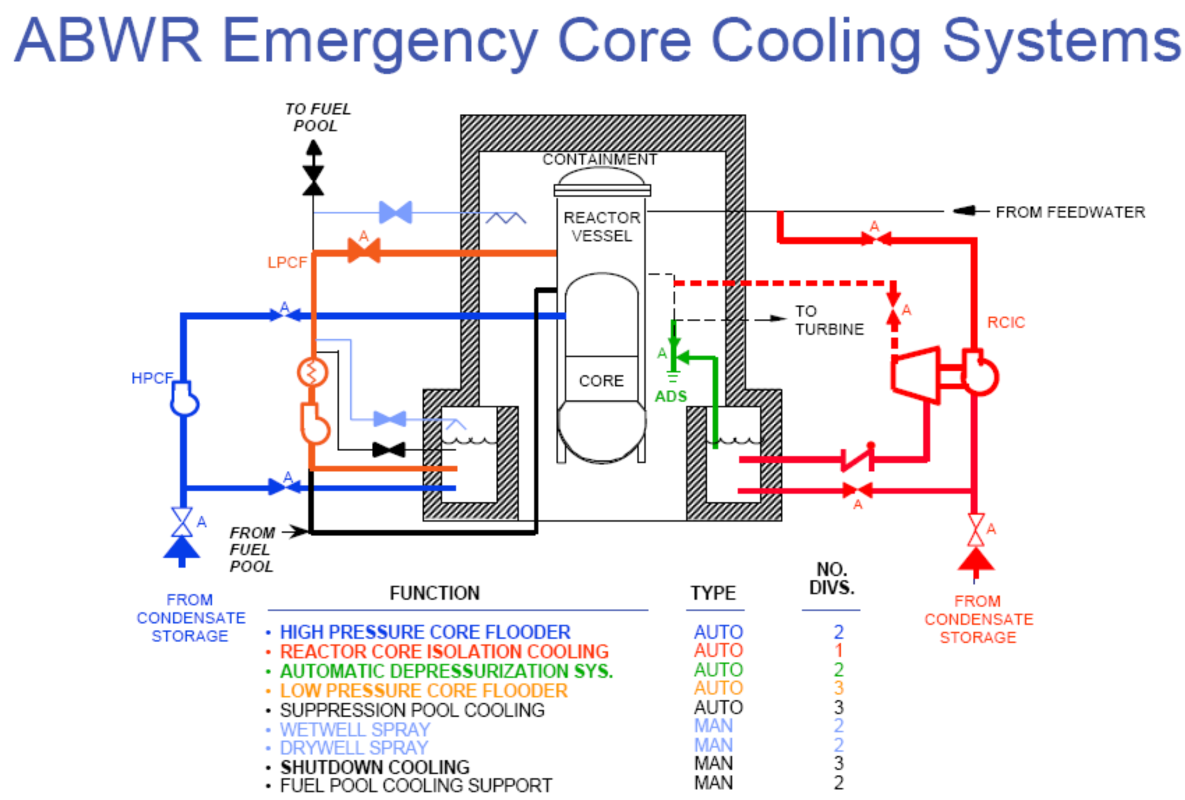
Design pressure, MPa	0.41
Drywell free volume, m^3	7350
Upper drywell volume, m^3	5490
Lower drywell volume, m^3	1860
Area of drywell connecting vents, m^2	11.3
Wetwell free volume, m^3	5960
Suppression pool water volume, m^3	3580
Suppression pool water height, m	7.05
Vents	
Number of vertical vents	10
Vertical vent diameter, m	1.2
Number of horizontal vents/vertical vent	3
Horizontal vent diameter, m	0.7
Horizontal vent length, m	1.0
Initial submergence of top horizontal vents, m	3.5
Initial submergence of middle horizontal vents, m	4.9
Initial submergence of lower horizontal vents, m	6.2
Vent loss coefficient	2.5 – 3.5
Residual Heat Removal System (Pool Cooling Mode)	
Loops	3
Flow rate per loop at 275 kPa, m^3/hr	950
Automatic initiation on pool temperature, K	322
Required NPSH, m	2.4
Residual Heat Removal System (Containment Spray Mode)	
Loops	2
Drywell Flow Rate (kg/hr)	0.84E6
Wetwell Flow Rate (kg/hr)	1.14E6
Manual initiation	
Residual Heat Removal System Heat Exchangers	
Type	U-tube
Overall heat transfer coefficient, $kW/^0C$	370
Reactor cooling water flow rate, kg/hr	1.2E6
Service water temperature, ^0C	30
Drywell Coolers	
Heat removal capability, MWt	1.25
Drywell pressure increase scram, kPa	13.6

High pool temperature scram, K	316.6
Vacuum breakers	
Number	8
Opening differential pressure setpoint, kPa	0.69; 3.43 (fully open)
Diameter, cm	50.8
Loss coefficient	3
Required minimum A/sqrt(k), m^2	0.77
Initial Conditions	
Maximum initial drywell temperature, K	308
Drywell humidity , %	20
Maximum initial wetwell/suppression pool temperature, K	308
Wetwell humidity, %	100

In summary, the ECC engineering safety features for the Passive BWR plant are:



The schematic diagram for the HP, LP ECCS and ADS is shown below, illustrating Auto and Manual initiation type logic and number of divisions.



3.8.4 Phenomena Modeled

From the perspective of the simulator, the containment needs to be modeled for transients involving safety/relief valve discharge into the suppression pool, and pipe breaks inside the drywell. Both types of events have a short-term impact as well as an effect on long-term containment response. Many of the short-term transient loads are complex and their understanding relies on empirical experimental data; these are beyond the scope of a training simulator. They will be mentioned, however, here for the sake of completeness.

(a) SRV Discharge Transient

A pressurization transient in the RPV will cause SRVs to open and discharge steam into the suppression pool via the discharge line and quencher which is attached at the discharge end. Prior to SRV actuation, the SRV discharge line above the water level is filled with non-condensable gas. Sudden opening of the SRV and the ensuing rapid steam discharge results in pressurization of the line. This pressurization creates a large force which pushes the gas and water leg out of the discharge line through the quencher and into the suppression pool. This gas then forms bubbles which oscillate and impart loads to the submerged boundaries and structures in the suppression pool. This phenomenon is known as SRV air-clearing and takes place over a time period of 1 second following SRV actuation.

After the air-clearing phase, steam is discharged into the pool. The rapid condensation of the steam also causes a condensation loading on submerged boundaries and structures. As the steam flow drops, the condensation process becomes periodic (chugging flow). These loads are, however, small when quencher discharge devices are used.

The main phenomena to be modeled for this transient would be:

1. Discharge of the initial inventory of noncondensibles and water

2. Condensation of steam in the suppression pool resulting in pool temperature increase and wetwell gas space pressure increase due to increased steam partial pressure corresponding to the higher pool temperature. Because the SRVs discharge near the bottom of the pool, the pool is well mixed and there is little temperature stratification.

3. If the wetwell pressure exceeds the drywell pressure by the vacuum breaker set point pressure differential, vacuum breakers can open.

(b) Pipe Break in Drywell

Following a pipe break in the drywell, steam discharged through the break starts to pressurize the drywell. The increased pressure depresses the water level in the drywell connecting vents and uncovers the horizontal vents. Once the top horizontal vents open, a mixture of noncondensibles and steam discharges into the suppression pool. This reduces the pressurization rate of the drywell. Typically, the drywell pressure turns around when the second row of horizontal vents is uncovered. Vent clearing occurs over a time period of approximately 1 second. This initial pressure peak is the highest drywell pressure reached during the transient.

Most of the noncondensibles are swept out of the drywell in 5 to 10 seconds. Discharge of this gas volume through the horizontal vents into the pool results in pool swell phenomena. After the vent is cleared of water, a bubble forms at the vent exit, and the noncondensibles/steam mixture starts to flow into the suppression pool. The bubble at the vent exit expands to suppression pool hydrostatic pressure as noncondensible/steam flow continues from the pressurized drywell. The water mass above the expanding bubble is accelerated upward by the difference between the bubble and the airspace pressures. The pool water surface is moved upwards until the bubble breaks through the surface. During the pool swell phase, the wetwell region is subjected to hydrodynamic loads due to pressure, drag, and impact forces.

The wetwell gas space is pressurized due to the accumulation of the noncondensibles that bubble through to this space and due to increased steam vapor pressure as the pool temperature rises.

As the steam flow through the vents drops, the condensation process becomes periodic. First Condensation Oscillation (CO) phenomena occur when the vent flow is predominantly steam at relatively high mass flux. The steam-water interface at the vent exit oscillates as the steam is condensed, with steam mass flux sufficient to prevent pool water flow back into the vent. The steam condensation process at the vent exit induces oscillatory and steady pressure loads on the suppression pool boundary and structures submerged in the pool.

As the steam flow decreases further, typically at a low vent steam mass flux of 2-10 $lb/s\text{-}ft^2$ or 10-50 $kg/s\ m^2$), Chugging oscillations occur. A steam bubble forms at the vent exit, grows and ultimately collapses, when the heat transfer to the suppression pool water is greater than the steam energy feeding the bubble. A collapsing bubble produces a pressure spike, followed by damped oscillation, which is transmitted to the submerged boundaries and structures. The chugging pressure amplitude is found to decrease with higher pool temperature, and there is no significant chugging loading with pool temperature above 60 °C.

As the steam flow drops and steam is discharged through only the top row of horizontal vents, steam is condensed primarily in the region above the top vents. This results in pool temperature stratification, with the pool surface temperature being higher than that calculated for a well-mixed pool. However, this is offset by condensation and chugging oscillations which promote good mixing.

The main phenomena to be modeled for the LOCA are:

1. Pressure and temperature response of the drywell to the break flow discharging into the drywell.
2. Vent clearing accounting for the inertia of the water legs in the vertical and horizontal branches.
3. Discharge flow through the vents and suppression pool mass and energy balance.
4. Wetwell air space pressurization due to the noncondensibles added to this space from the drywell and the increased vapor pressure corresponding to the suppression pool surface temperature.

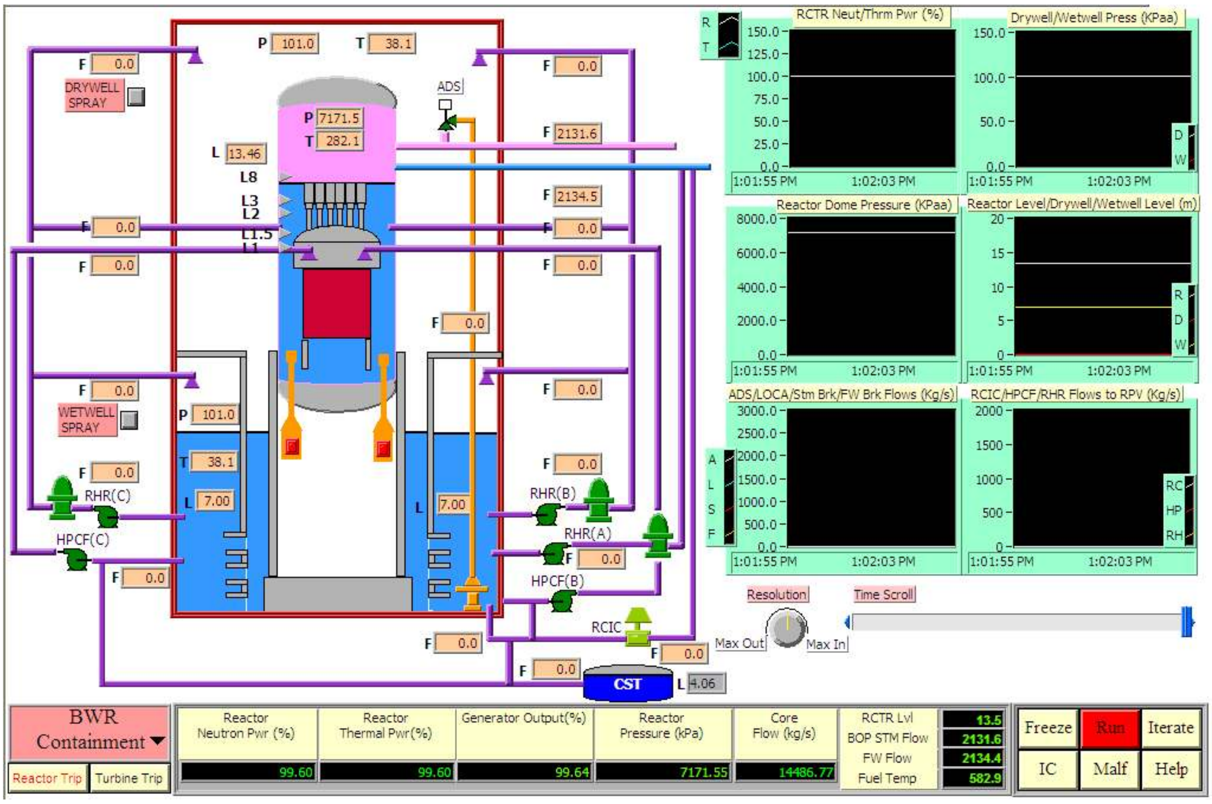
3.8.5 Interactions between the RPV and Containment

1. Steam from pipe break is discharged into either the upper drywell (steamline, feedwater line breaks) or lower drywell (RIP seal, bottom drain line breaks). Flow transitions from choked flow to Bernoulli flow at low pressure.
2. SRV flow is discharged to suppression pool.
3. Water from the suppression pool is injected into the RPV by the RHR system
4. Drywell pressure increase of 13.6 kPa initiates reactor scram
5. Suppression pool temperature of 316.6 K initiates reactor scram

3.8.6 Sources:

1. ABWR General Description
2. Lungmen PSAR
3. Non-proprietary GE presentations

3.8.7 BWR Containment Screen



The BWR Containment Screen and its underlying model attempt to present the Drywell, Wetwell, HP, LP ECC and ADS functions, system behavior and parameters as described above:

The parameters shown on the screen are:

- Drywell pressure, temperature
- Wetwell pressure, pool temperature, level
- Dynamic simulation of vent clearing in the event of LOCA.
- Reactor core water level; dome steam pressure, temperature
- RIPs running status.
- Pump status for RCIC, HPCF (division A, B, C), RHR (division B, C), and respective flows.
- Status for ADS, and steam relief flows to Suppression Pool.
- ECC sprays flow to Drywell and Wetwell.
- CST tank level, and flow from CST. The switch-over from CST to Wetwell on CST low level is also modeled.

The trends represented are:

- Reactor Power, Thermal Power (top left)
- Drywell, Wetwell Pressures (top right)
- Reactor Dome Pressure (middle left)
- Reactor Level, Drywell Level, Wetwell Level (middle right)
- ADS steam relief flow, LOCA break flow, Steam Line Break Flow (inside Drywell), Feedwater Line Break Flow (inside Drywell) (bottom left)
- ECC flows to Reactor Pressure Vessel from RCIC, HPCF, RHR

4. SIMULATOR EXERCISES

Included with the simulator is a number of Initial Conditions (IC) which are stored with reference to various states of the BWR plant during unloading and reloading. Please use them as necessary to assist you in the simulator exercises:

Plant Unloading	***IC Point***	***Plant State***
100 % Full Power	FP_100.ic	All systems running as required
68 % Full Power	FP_68.ic	All systems running as required.
10 % Full Power	FP_10.ic	All systems running as required.
Zero Power Hot	Zero_hot.ic	Reactor is subcritical. Other systems are running as required. To reload at this point, follow Section 4.1.2 Note 3 for steps in reloading the reactor power.
Zero Power Hot – RPV depressurized to 1800 kPa (low pressure).	Zero_hot_LP.ic	Reactor Scrammed; Turbine Tripped; RIPs runback; FW pumps off. Reactor pressure at 1800 kPa; Fuel and Coolant still hot > 200 ºC. Water level lower than L4. To restart the plant at this point, require knowledge on resetting Reactor Trip; resetting Turbine Trip; Resetting RIPs, Restarting the FW pumps.
Plant Startup and Reloading	***IC Point***	***Plant State***
Reactor Power at 5 % FP; RPV pressure increasing	FP_5_SU.ic	Turbine Tripped; RPV pressure at 3800 kPa, increasing – pressurization phase. Other systems running.
Reactor Power at 5 % FP; RPV pressure increasing.	FP_5_SU2.ic	Turbine Tripped; RPV pressure at 5380 kPa, increasing – pressurization phase. Other systems running.

Reactor Power at 20 %	FP_20_SU.ic	All systems are running, except that the turbine governor is on "Manual". This is because on turbine reloading, the turbine is taking more steam from the reactor steam dome (due to the reactor pressure controller's response – e.g. large pressure integral error). As a result, the turbine load is larger than the reactor power (temporarily). To reload the plant from this point onward, firstly, close down the governor valve manually to match with the reactor power. Let the pressure stabilize to 7100 kPa, then return the governor to AUTO control. After the systems are responding properly, increase reactor power setpoint and rate, and monitor the turbine governor response and feedwater level response closely.

4.1 INTRODUCTORY EXERCISES

4.1.1 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.
  - ⇒ Go to "Reactivity & Setpoint" Screen
  - ⇒ Press RCTR PWR SETPOINT button
  - ⇒ In pop-up menu lower 'target' to 90.00% at a 'Rate' of 0.5% FP/s
  - ⇒ 'Accept' and 'Return'
- Observe the response of the displayed parameters until the transients in reactor power and steam pressure are completed. Observe the power flow path on "Power Flow Map & Controls" screen.
- Continuing the above operation, raise "UNIT POWER" to 100% at a rate of 0.5% FP/s..

4.1.2 Reduction to 0% full power and back to 100% full power

- Initialize the simulator to 100%FP, reduce power using 25% steps at 0.5% FP/s from 100 % to 65 %. **Monitor the Reactor Water Level at all times**
- From 65 % to 20 %, use the rate of 0.5 % FP/s (**Note: choose a slower rate, e.g. 0.3 % per second, if there are fluctuations in core flow, power, and level**).
- From 20 % to 0 %, us the rate 100 % present power (PP)/s. See explanation in Section 3.4.
- Record the following values:

Parameter	Unit	100%	75%	50%	25%	0%	Comments
Reactor Power	%						
Core Flow Rate	kg/s						
Coolant Temperature	°C						
Coolant Pressure at core exit	kPa						
Coolant Quality at core exit	%						
Reactor Level	m						
Reactor Steam Pressure	kPa						
Reactor Steam Flow	kg/s						
Feedwater Flow	kg/s						
Turbine-Generator Power	%						

Under “Comments” please note type of parameter change as a function of reactor power 0% → 100% FP: constant, linear increase or decrease, non-linear increase or decrease. In particular, comment on the power — flow path during the power evolution, and comment if the path enters into particular regions in the power flow map.

- Increase reactor power back to 100% after ~ 0% is reached, using 25% steps
- From 0 % to 20 %, use 100 % present power/s.
- From 20 % to 65 % , use the rate 0.5% per second (maximum rate is 1 % FP/s). Watch out for level swing, when it is near to the trip/runback setpoint.
- From 65 % to 100%, use the rate of 0.5 % FP/s or slower
- Repeat the above recordings and comments.

Notes:

1. During power maneuvering with the simulator, you may see the following anomalies: trajectory falls below rated line; power change rate could be slower than demand; rods would be used to assist flow control, if it is too slow to meet power demand setpoint. Note to users: the real BWR plant operation may not have these anomalies. These anomalies may be due to modeling assumptions made in various reactivity feedback coefficients (e.g. void), and/or assumptions made in controls tunings for reactor power control and feedwater control. The control functions as modeled in this simulator are based on simplified design descriptions available in the public domain. For all intent and purposes, they are considered functionally correct for educational training. However, in absence of detailed plant control documentation implementation, it is difficult to get the same performance as the real plant. Henceforth, caution is advised for the simulator users regarding these anomalies.

2. The automatic power flow control system (as described above by using the Reactivity & Setpoints Screen) is provided for users in power maneuvering. In the real ABWR plant, this control system is known as the Automatic Power Regulation (APR). It is a power generation system that controls reactor power during reactor startup, power generation, and reactor shutdown, by appropriate commands to change rod positions, or to change reactor recirculation flow. It also controls the pressure regulator setpoint (or turbine bypass valve position) during reactor heatup and depressurization (e.g. to control the reactor cooldown rate).

   The APR has several important control components which include the RIP controls. One can find the user interface for RIP control on “Power Flow Map & Control Screen”. On the right side of the screen, there is a button labeled as “RIP Crl”. Upon pressing this button, one will see the typical PID controller faceplate for RIP. Currently the controller is at Remote Set Point (RSP), indicating a control mode where the setpoint for the controller is derived remotely from external computation. One can switch the controller to “Manual”, and by manipulating the manual output signal, one can change the RIP head and hence speed (RPM), changing of core flow rate as a result.

3. The reactor could be subcritical at zero power. When reloading the subcritical reactor at zero power, at times even though power target and rate setpoints have been entered, the reactivity does not seem to change. If this happens, re-enter the target power and rates again at the Reactivity & Setpoint Screen, and then go to Power/Flow Map & Control Screen, **press and hold for a few seconds, then release the “SCRAM RESET” Button, repeatedly until you see the “down” arrows showing the rods movement.** This is necessary to give a temporary higher power setpoint (hence higher power error signal) to get the Reactor Power Control system into the control range in order to respond accordingly.

   Note: This procedure is only applicable for this Educational Simulator, and does not apply for the real plant.

4.1.3 Turbine trip and recovery

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are moisture separator and heater drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum and reactor high water level. The event sequence is as follows:

1. After the main turbine is tripped, turbine bypass valves are opened.

2. Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system.

3. Turbine stop valves closure initiates a trip of four RIPs, thereby reducing the core flow. The pressure relief system, operates the relief valves independently when system pressure exceeds relief valve (SRV) lift setpoints due to high pressure.

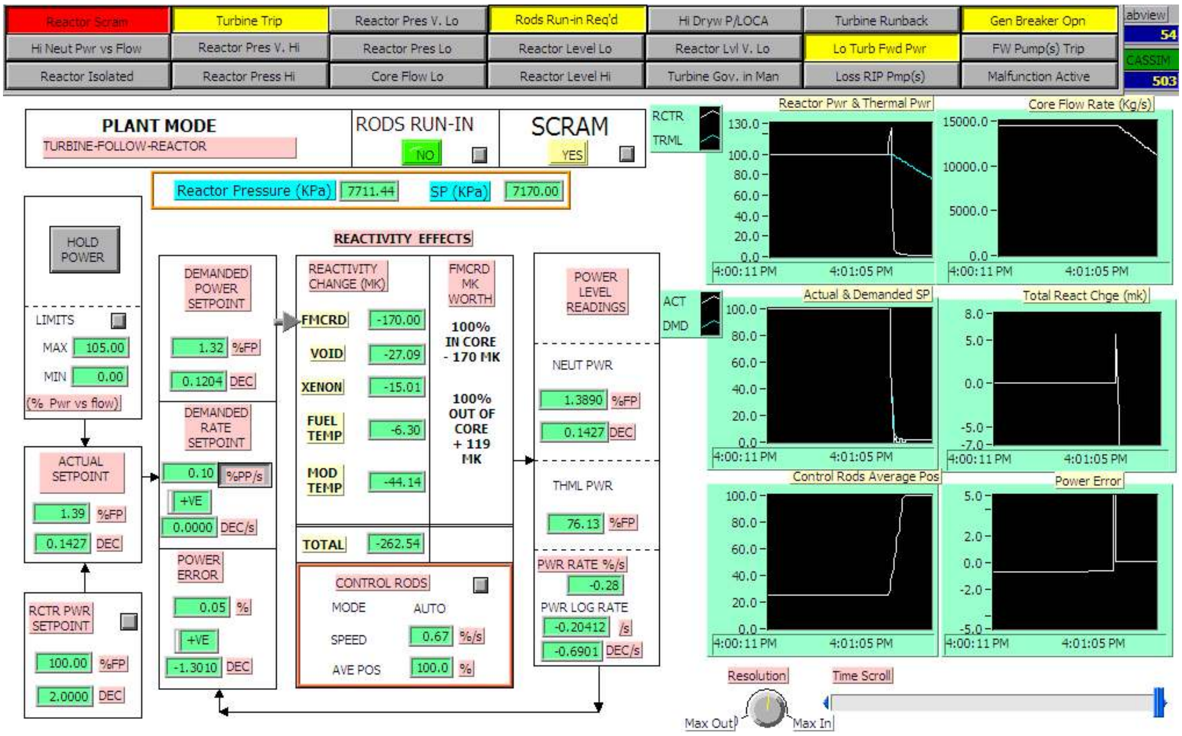
In the actual plant, in the event of Turbine Trip, the plant operator would normally perform the following:

(1) Verify auto-transfer of buses supplied by generator to incoming power (if automatic transfer does not occur, manual transfer must be made). Explain why.
(2) Monitor and maintain reactor water level at required level. Does it go up or down? Explain.
(3) Check turbine for proper operation of all auxiliaries during coastdown.
(4) Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes.
(5) Secure the RCIC operation if auto initiation occurred due to low water level.
(6) Monitor control rod drive positions.
(8) Cool down the reactor per standard procedure if a restart is not intended.

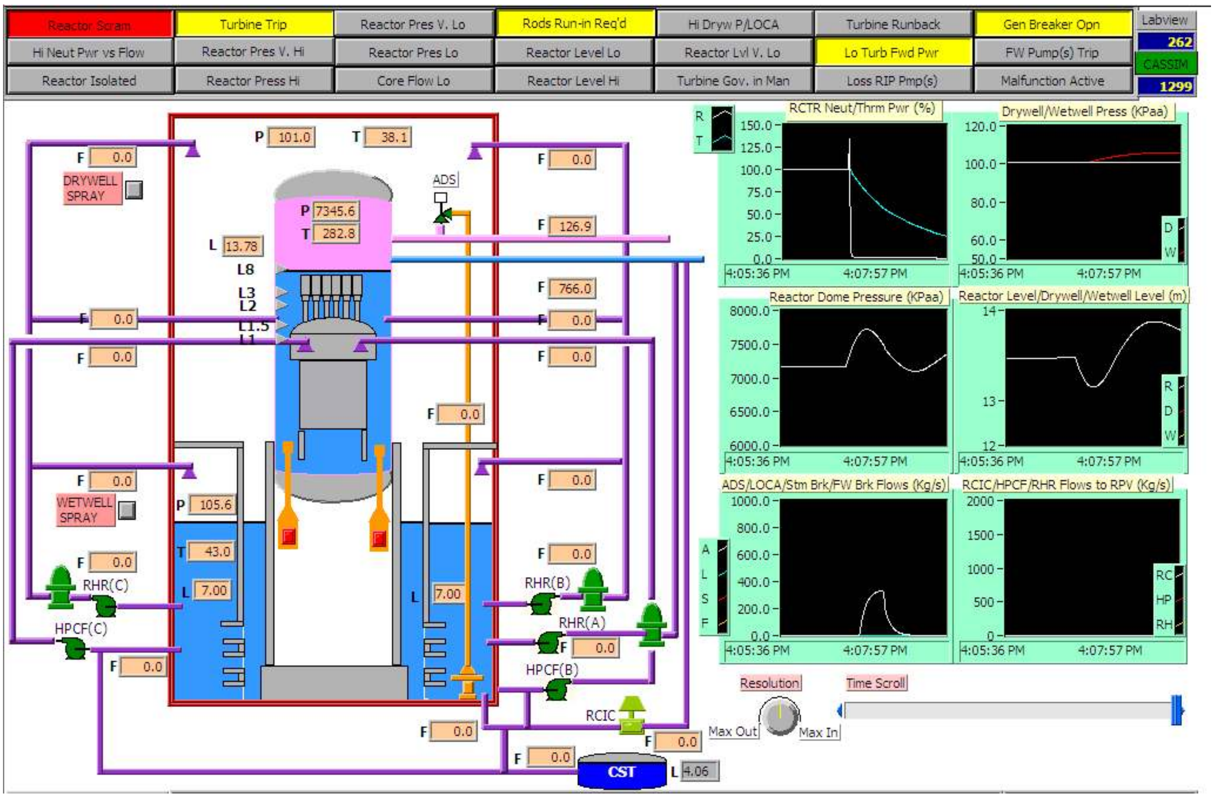
To observe the transients for Turbine Trip in the simulator, first load the simulator at Initial state of 100% full power, press the turbine trip button on the left-hand bottom corner of the screen, and confirm turbine trip.

- Notice the power flow path on “Power Flow Map and Controls” screen and monitor the reactor neutron power on the BWR REACTIVITY & CONTROLS SCREEN.
- Explain why there is a sudden increase in reactor power.
- Monitor the core flow rate.
- What is the steam flow through the bypass valve on the turbine generator screen?
- Does any SRV open? What is the steam flow through the SRV on the turbine generator screen?
- Since a Turbine Trip leads to Reactor Scram in this exercise, go to the next section to restart the reactor, followed by turbine restart.

The Following snap shot shows the reactor power pulse due to void collapse from high reactor dome pressure, causing surge in reactivity increase. However, the closing of the Turbine Stop Valve initiates Reactor Scram almost immediately.



The following snap shot shows that the reactor level drops immediately after turbine trip, due to bubble collapse caused by high dome pressure. The SRV is lifted to relieve steam to the Suppression Pool.



4.1.4 Reactor scram and recovery

- Initialize the simulator to 100%FP
- Manually scram the reactor
- Observe the response of the overall unit. Describe and explain the following responses upon reactor scram:
  - (a) Recirculation core flow;
  - (b) Reactor pressure;
  - (c) The sub-cooled & boiling region boundary of core — explain the changes;
  - (d) Turbine load & bypass system.
  - (e) Reactor water level.
- Notice that the RIP runback alarm is on the Power/Flow Map and Control Screen.
- Wait until generator power is zero.
- Note the reactor neutron power. Reset reactor scram using control devices provided on the "Power Flow Map & Controls" screen. Press the "YES" button next to SCRAM ST.
- The "YES" Button now turns to "NO", meaning that the Reactor Scram Status is "NO SCRAM".
- Press the "ON" button on the RIPs. A pop-up appears, press RESET. The RIP Runback Alarm should be clear.
- Press the Button "RIP Crl", ensure that RIP control is at Remote Control Setpoint mode (RSP).
- Monitor the yellow cursor on the Power/Flow Map. It should be roughly at 0 % power, 35 % core flow.
- Now pull the Control Rods out of the core by pressing the "SCRAM RESET" button to begin rod withdrawal.
- Observe any changes in control rod reactivity. **The rods should be withdrawing. Otherwise, go to the "Power Flow Map & Controls" screen, and press the "SCRAM RESET" button again; you should see the "down" arrows showing the movement of the rods**.
- When the rods are withdrawing, go to the "BWR Reactivity & Setpoint" screen. Enter a new power setpoint say 5 %, at a rate of 100% present power per second, so that the Reactor power control system has registered a new power setpoint and rate.

  Note: if the reactor neutron power is at 1% FP, 100% present power per second will yield an *effective* power rate of 1 % FP/s. So the value of present power selected by user should depend on the present neutron power at that time. The maximum *effective* power rate should not exceed 1 % FP/s, for power range less than 65 % FP.
- Record the time (using the display under the chart recorders) needed to withdraw all rods to Reset line.
- Go to "Reactivity & Setpoint" screen; record the net total reactor reactivity. Is the reactor subcritical, critical or supercritical?

Go to “Reactivity & Setpoint” screen, and observe the response of the reactor regulating system and the reactivity changes that take place. Monitor Total Reactivity Change, Power Error, Demanded Setpoint, Current Neutron Power.

Look out for the following:

(a) Water level in the reactor vessel may go as high as L8, in which case the feedwater pumps will be tripped. Continue to raise power to 10 % FP, to blow off steam to condenser via the Steam Bypass Valve. When the level drops below L8, go to Feedwater and Extraction Screen and restart the feedwater pumps (use the pop-up button next to the pump).

- When the rods have reached the RESET line, the reactor may still be subcritical. Even though power target and rate setpoints have been entered, the reactivity does not seem to change. If this happens, re-enter the target power and rates again at the Reactivity & Setpoint Screen, and then go to Power/Flow Map & Control Screen, **press and hold for a few seconds, then release the “SCRAM RESET” Button, repeatedly until you see the “down” arrows showing the rods movement.** This is necessary to give a temporary higher power setpoint (hence higher power error signal) to get the Reactor Power Control system into the control range in order to respond accordingly.

  Note: This procedure is only applicable for this Educational Simulator, and does not apply for the real plant.

- Continue to raise power to 10% FP, 20 % in 10 % steps at a rate not more than maximum effective power rate of 1 % FP/s (by entering the appropriate value of present power per sec). Note after 20% FP is reached, you can enter % FP/s as the rate, instead of “present” power per second. Note that the maximum rate is 1 % FP/s, the suggested rate is 0.5 % FP/s.

- If the turbine is tripped as a result of low power, reset turbine trip, synchronize and reload as follows:

(a) Go to BWR Turbine Generator screen, reset turbine trip, select ‘TRU ENABLE’, and select “TRU Speedup” to synchronize the generator and load to match with the reactor/thermal power.

(b) After turbine is in service, what happens to the steam bypass valve as the turbine power increases? Note the reactor pressure.

(c) Note if the turbine power is increased to a value more than the reactor power, due to the controller overshooting. If that happens, go to Turbine Generator Screen, and turn the turbine governor from “AUTO” to "MANUAL”. Let the pressure control system stabilize, and then switch the governor control to “AUT”".

(d) After the turbine power is equal to reactor power, go to “Reactivity & Setpoints” to increase reactor power to 100% in 10% steps at 0.5% FP/s.

4.2 Malfunction Exercises

4.2.1 Loss of feedwater - both FW pumps trip

A loss of feedwater flow could occur from pump failures, loss of electrical power, operator errors, or reactor system variables such as a high vessel water level (L8) trip signal.

- Loss of feedwater flow results in a reduction of vessel inventory, causing the vessel water level to drop.
- Water level continues to drop and the recirculation flow is runback at Level 4 (L4).
- When the level reaches the vessel level (L3), scram trip setpoint is reached, whereupon the reactor is shut down and the four RIPs are tripped.
- Feedwater flow terminates due to loss of FW pumps. Subcooling decreases, causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure.
- Vessel water level continues to drop to the L2 trip. At this time, the remaining six RIPs are tripped and the RCIC operation is initiated.

In the real plant, the operator should ensure RCIC actuation so that water inventory is maintained in the reactor vessel. Additionally, the operator should monitor reactor water level and pressure control and T-G auxiliaries during shutdown. The following is the sequence of operator actions expected during the course of the event when no immediate restart is assumed. The operator should:

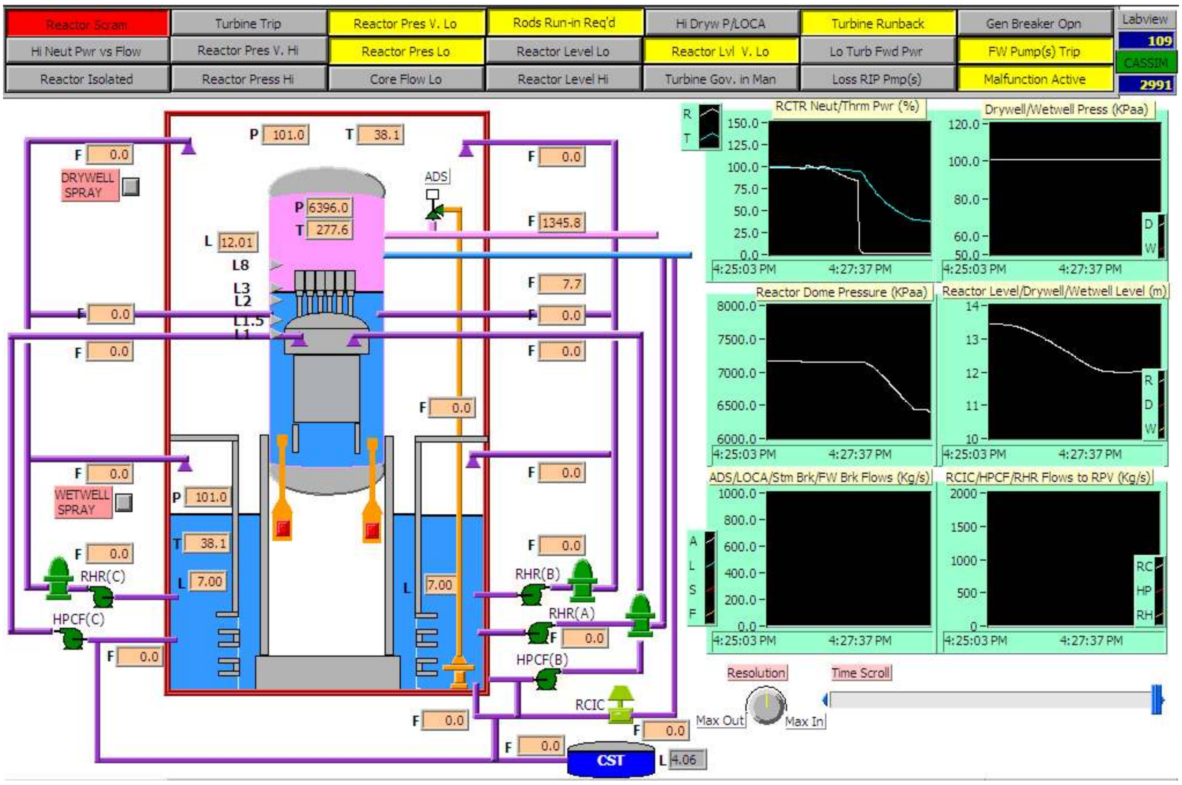
(1) Verify all rods in, following the scram
(2) Verify trip of four RIPs
(3) Verify RCIC initiation
(4) Verify that the remaining recirculation pumps trip on reactor low level (L2)
(5) Continue operation of the RCIC System until decay heat diminishes to a point where the RHR System can be put into service
(6) Monitor turbine coastdown, break vacuum as necessary.
(7) Complete scram report and survey maintenance requirements.

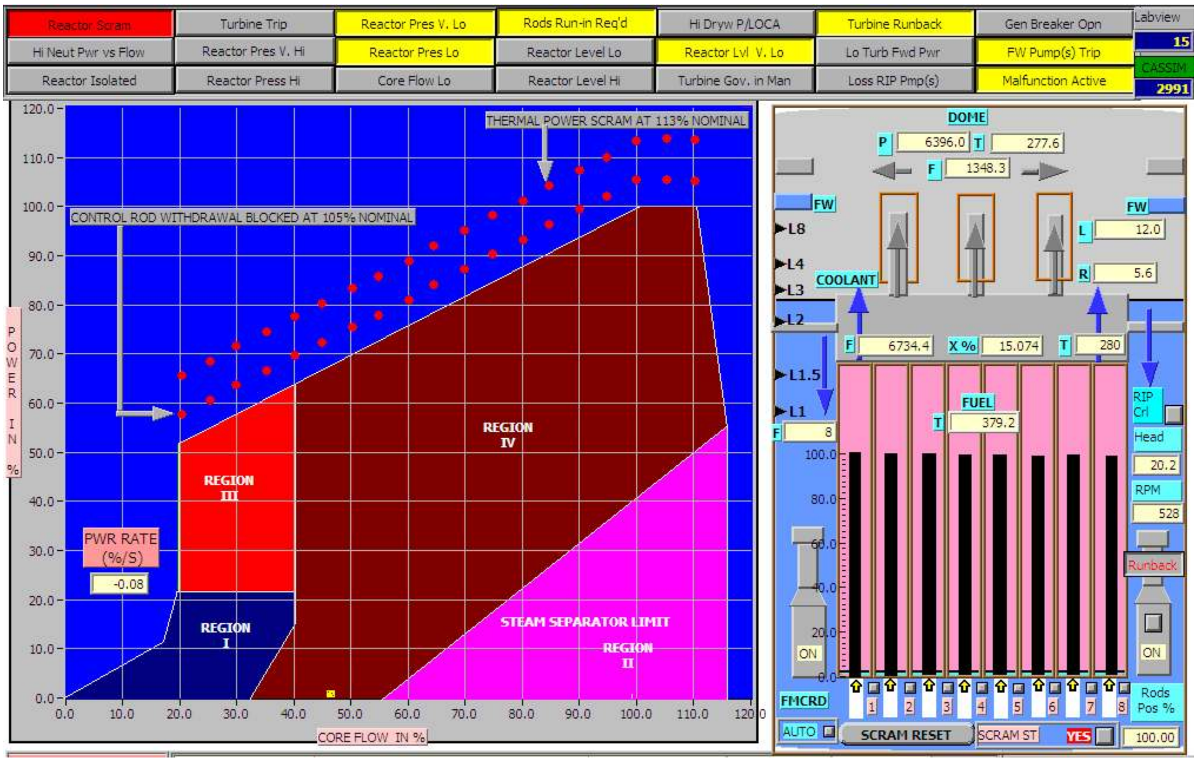
To observe this transient, go to BWR Feedwater & Extraction Steam Screen. Load the 100% FP IC, then insert the above malfunction. This malfunction leads to total loss of feedwater to the Reactor Pressure Vessel.

When this malfunction transient occurs:

- On the BWR Feedwater & Extraction Steam Screen, observe that both feedwater pumps stop.
- Go to Power/Flow Map Screen. The reactor level drops quickly due to loss of feedwater flow.
- Dome pressure is decreasing gradually as reactor water level drops. Provide explanation why this is happening.
- As dome pressure drops, the turbine inlet pressure also drops. In order to restore the Reactor Pressure at setpoint, the Reactor Pressure Controller closes the turbine governor valve slightly. As a result, the generator power (MW) drops.
- As the water level drops below L4, the recirculation flow is runback.
- Note the movement of the yellow cursor in the Power/Flow Map.

- After a short while, the Reactor will be scrammed by Low Water Level L3.
- When level reaches L2, the remaining RIPs will be tripped. Emergency Core Cooling (ECC) from RCIC will be activated on a Low Reactor Level L2.



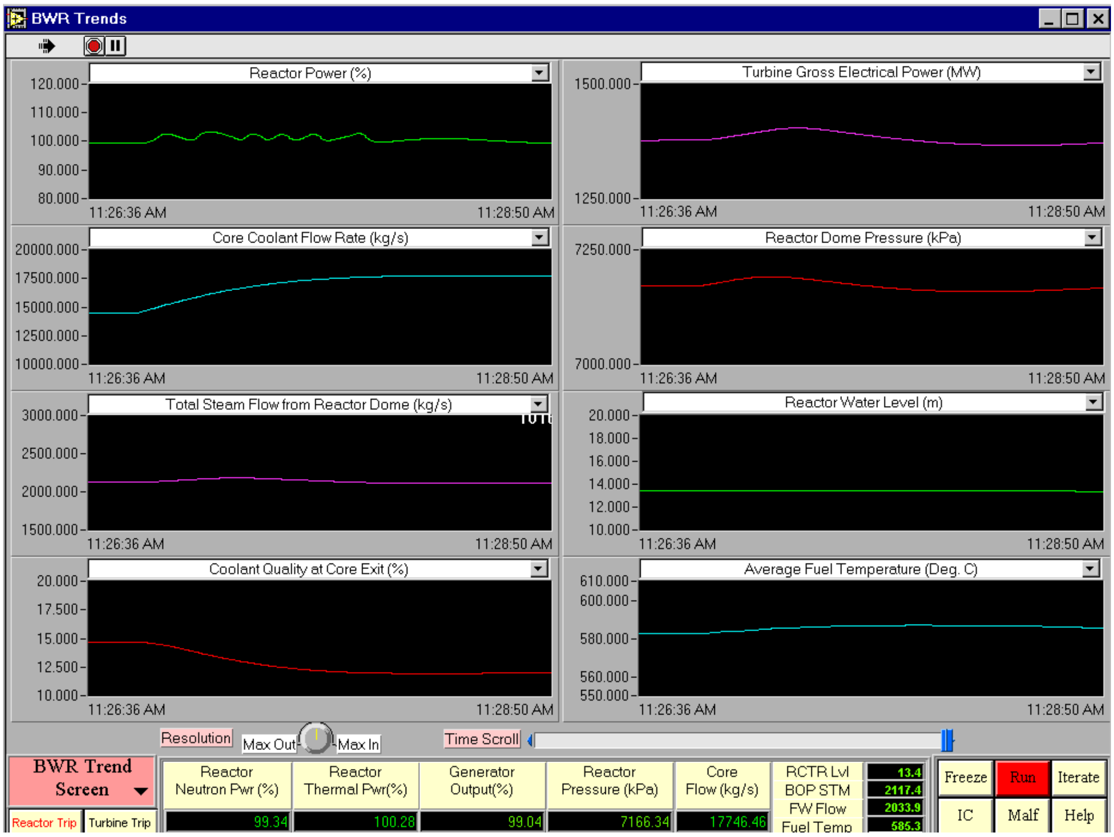


4.2.2 Increasing core flow due to flow control failure

This malfunction will cause the recirculation flow controller to fail in such a way that the process variable input for the controller fails in "low" value. But the flow transmitter reading for display is normal. The consequence is that the Recirculation Flow Control System is "fooled" into thinking that the recirculation flow is lower than the flow setpoint, hence it will increase the speed of the Reactor Internal Pumps (RIPs) to increase flow.

Go to Power/Flow Map. Load the 100% FP IC. Run the simulator. Now insert the malfunction. When this malfunction transient occurs:

- On the Power/Flow Map Screen, observe the movement of the yellow cursor.
- Observe that the coolant flow is increasing. As coolant flow increases, core quality X (%) decreases. Provide explanation.
- As core quality decreases, so does void fraction. As a result, the reactor neutron flux increases. Provide explanation.
- As soon as the reactor power exceeds the target setpoint, the Reactor Power Control system will attempt to decrease reactor power by inserting control rods.
- Observe that the cursor moves beyond the operating path on the Power/Flow Map.

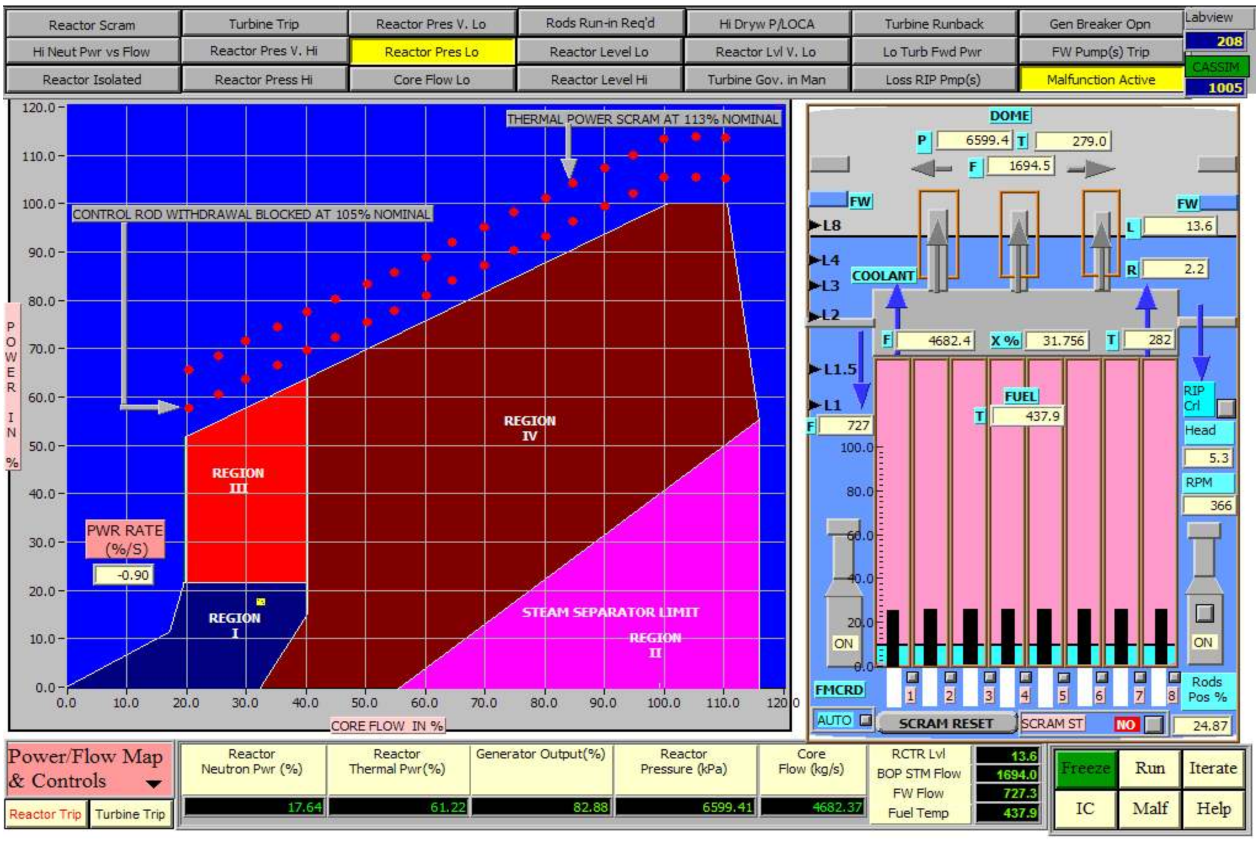


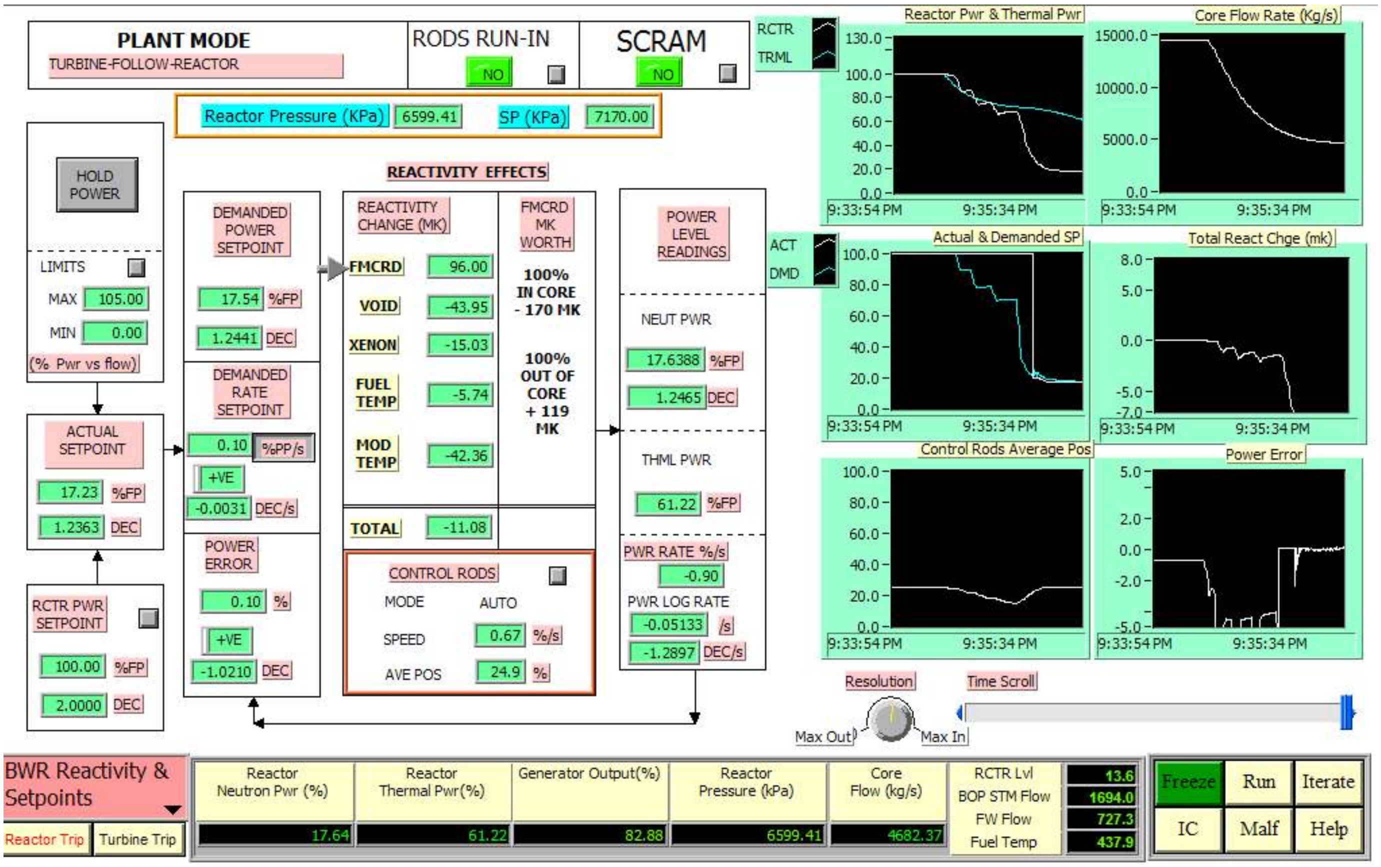
4.2.3 Decreasing core flow due to flow control failure

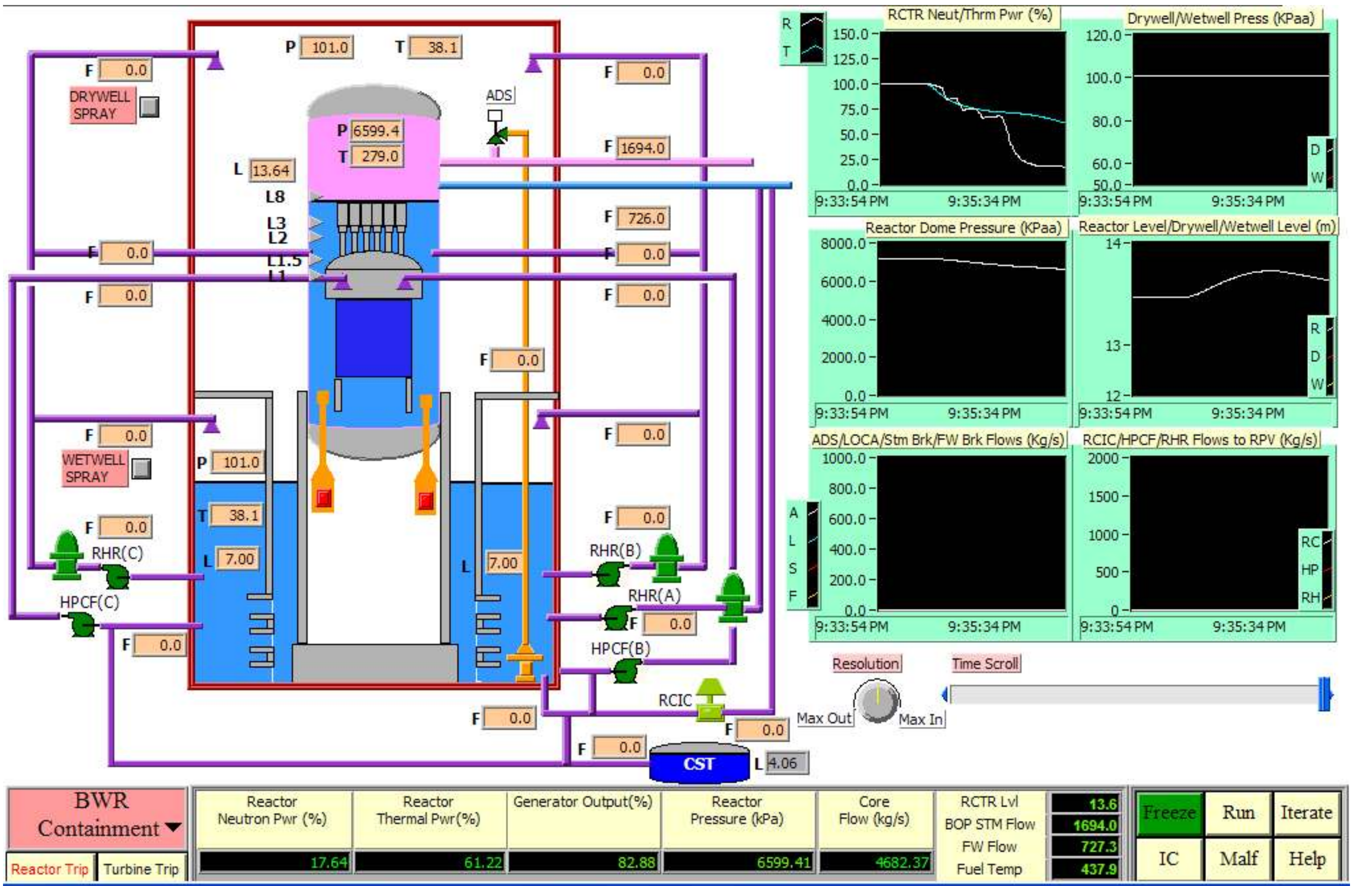
This malfunction will cause the recirculation flow controller to fail in such a way that the process variable input for the controller fails in "high" value. But the flow transmitter reading displayed is correct. The consequence is that the Recirculation Flow Control System is "fooled" into thinking that the recirculation flow is higher than the flow setpoint, hence it will decrease the speed of the Reactor Internal Pumps (RIPs) to decrease flow.

Go to Power/Flow Map Screen. Load the 100% FP IC. Run the simulator. Now insert the malfunction. When this malfunction transient occurs:

- On Power/Flow Map screen, observe the movement of the yellow cursor.
- Observe that the coolant flow is decreasing. As coolant flow decreases, core quality X (%) increases. Provide explanation.
- As coolant flow is decreasing, the level increases.
- Also as core quality increases, so does void fraction. As a result, the reactor neutron flux decreases. Provide explanation.
- As soon as the reactor power is lower than the target setpoint, the Reactor Power Control system will attempt to increase reactor power by withdrawing control rods.
- But the rate of flow decrease is faster than the reactor power adjustment; hence the cursor is above the normal operating path on the Power/Flow Map. Observe the alarm "Hi Neut Pwr vs Flow".
- As the malfunction event evolves, the cursor moves into the "region I".
- The reactor pressure is decreasing and low-pressure alarms come on. Explain why Reactor Pressure Control cannot increase reactor pressure back to setpoint.
- There is also an alarm "Turbine runback". Explain why there is a turbine runback.




PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN
NO
SCRAM
NO
Reactor Pressure (KPa) 6599.41
SP (KPa) 7170.00
HOLD POWER
LIMITS
MAX 105.00
MIN 0.00
(% Pwr vs flow)
ACTUAL SETPOINT
17.23 %FP
1.2363 DEC
RCTR PWR SETPOINT
100.00 %FP
2.0000 DEC
DEMANDED POWER SETPOINT
17.54 %FP
1.2441 DEC
DEMANDED RATE SETPOINT
0.10 %PP/s
+VE
-0.0031 DEC/s
POWER ERROR
0.10 %
+VE
-1.0210 DEC
REACTIVITY EFFECTS
REACTIVITY CHANGE (MK)
FMCRD 96.00
VOID -43.95
XENON -15.03
FUEL TEMP -5.74
MOD TEMP -42.36
TOTAL -11.08
FMCRD MK WORTH
100% IN CORE - 170 MK
100% OUT OF CORE + 119 MK
CONTROL RODS
MODE AUTO
SPEED 0.67 %/s
AVE POS 24.9 %
POWER LEVEL READINGS
NEUT PWR
17.6388 %FP
1.2465 DEC
THML PWR
61.22 %FP
PWR RATE %/s
-0.90
PWR LOG RATE
-0.05133 /s
-1.2897 DEC/s
RCTR
TRML
Reactor Pwr & Thermal Pwr
Core Flow Rate (Kg/s)
ACT
DMD
Actual & Demanded SP
Total React Chge (mk)
Control Rods Average Pos
Power Error
9:33:54 PM
9:35:34 PM
Resolution
Time Scroll
Max Out
Max In
BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 17.64
Reactor Thermal Pwr(%) 61.22
Generator Output(%) 82.88
Reactor Pressure (kPa) 6599.41
Core Flow (kg/s) 4682.37
RCTR Lvl 13.6
BOP STM Flow 1694.0
FW Flow 727.3
Fuel Temp 437.9
Freeze
Run
Iterate
IC
Malf
Help


P 101.0
T 38.1
DRYWELL SPRAY
ADS
P 6599.4
T 279.0
L 13.64
L8
L3
L2
L1.5
L1
F 1694.0
F 726.0
WETWELL SPRAY
P 101.0
T 38.1
L 7.00
L 7.00
RHR(C)
HPCF(C)
RHR(B)
RHR(A)
HPCF(B)
RCIC
CST
L 4.06
RCTR Neut/Thrm Pwr (%)
Drywell/Wetwell Press (KPaa)
Reactor Dome Pressure (KPaa)
Reactor Level/Drywell/Wetwell Level (m)
ADS/LOCA/Stm Brk/FW Brk Flows (Kg/s)
RCIC/HPCF/RHR Flows to RPV (Kg/s)
9:33:54 PM
9:35:34 PM
Resolution
Time Scroll
Max Out
Max In
BWR Containment
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 17.64
Reactor Thermal Pwr(%) 61.22
Generator Output(%) 82.88
Reactor Pressure (kPa) 6599.41
Core Flow (kg/s) 4682.37
RCTR Lvl 13.6
BOP STM Flow 1694.0
FW Flow 727.3
Fuel Temp 437.9
Freeze
Run
Iterate
IC
Malf
Help

4.2.4 Decreasing steam flow from dome due to pressure control failure

This malfunction will cause the Reactor Pressure Controller to fail in such a way that the process variable input for the controller fails in “low” value. But the pressure transmitter reading displayed is correct. The consequence is that the Reactor Pressure Control System is “fooled” into thinking that the reactor pressure is lower than the pressure setpoint of 7170 kPa, therefore it will increase coolant flow rate to raise power in order to increase steam flow. At the same time, the Turbine Control system, thinking that main steam pressure is “low”, will decrease opening of the turbine governor valve. Hence steam flow to turbine will decrease. Thus the generator load is decreasing.

Go to the Power/Flow Map. Load the 100% FP IC. Run the simulator. Now insert the malfunction. When this malfunction transient occurs:

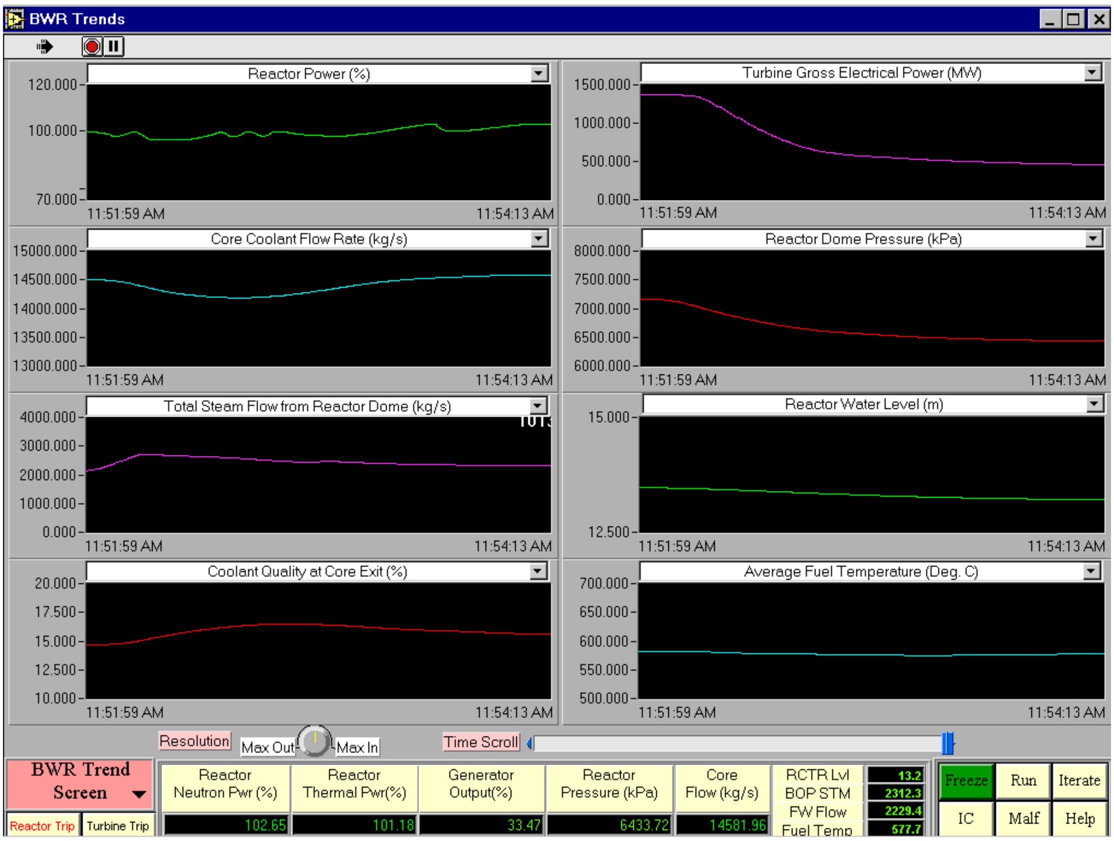
- On the Power/Flow Map Screen, observe the movement of the yellow cursor and record the reactor pressure reading - increasing or decreasing, as the malfunction event evolves.
- Observe that the coolant flow is increasing. Provide explanation.
- As coolant flow increases, core quality X (%) decreases. Provide explanation.
- As core quality decreases, so does void fraction. As a result, the reactor neutron flux increases. Provide explanation.
- As soon as the reactor power is higher than the target setpoint, the Reactor Power Control system will attempt to decrease reactor power by inserting control rods.
- Note the Generator Output (%) reading at the bottom of the screen - increasing or decreasing? Provide explanation. Compare this with the Reactor Thermal power (%). What is the difference? Where does this power difference go to?
- As the malfunction event evolves, the reactor pressure increases, and an alarm “Reactor Press Hi” will come on. But after a short while, it will disappear, then sometime later it will reappear again. Provide explanation why this is the case.
- As this malfunction further evolves, alarm “Rods Run-in Required” will come on. Observe the movement path of the yellow cursor. Provide explanation.
- Discuss what should the operator do in this event.

4.2.5 Increasing steam flow from dome due to pressure control failure

This malfunction will cause the Reactor Pressure Controller to fail in such a way that the process variable input for the controller fails in “high” value. But the pressure transmitter reading displayed is correct. The consequence is that the Reactor Pressure Control System is “fooled” into thinking that the reactor pressure is higher than the pressure setpoint of 7170 kPa, therefore it will decrease coolant flow rate to decrease power in order to decrease steam flow. At the same time, the Turbine Control system, thinking that main steam pressure is “high”, will increase opening of the turbine governor valve. But the turbine governor valve is almost at the 100% fully open position.

Go to Power/Flow Map Screen. Load the 100% FP IC. Run the simulator. Now insert this malfunction. When this malfunction transient occurs:

- On the Power/Flow Map Screen, observe the movement of the yellow cursor and record the reactor pressure reading - increasing or decreasing, as the malfunction event evolves.
- Observe that the coolant flow is decreasing. Provide explanation why this is so.
- As coolant flow decreases, core quality X (%) increases. Provide explanation.
- As core quality increases, so does void fraction. As a result, the reactor neutron flux decreases. Provide explanation.
- Reactor pressure will be decreasing further, causing low-pressure alarm "Reactor Pres Lo", as well as turbine runback. Provide explanation why turbine is running back.
- As soon as the reactor power is lower than the target setpoint, the Reactor Power Control system will attempt to increase reactor power by withdrawing control rods.
- Note the Generator Output (%) reading at the bottom of the screen - increasing or decreasing? Provide explanation. Compare this with the Reactor Thermal power (%). What is the difference? Where does this power difference go to?
- As the malfunction event evolves, the reactor pressure further decreases, and an alarm "Reactor Pres V. Lo" will come on.
- Discuss what should the operator do in this event.

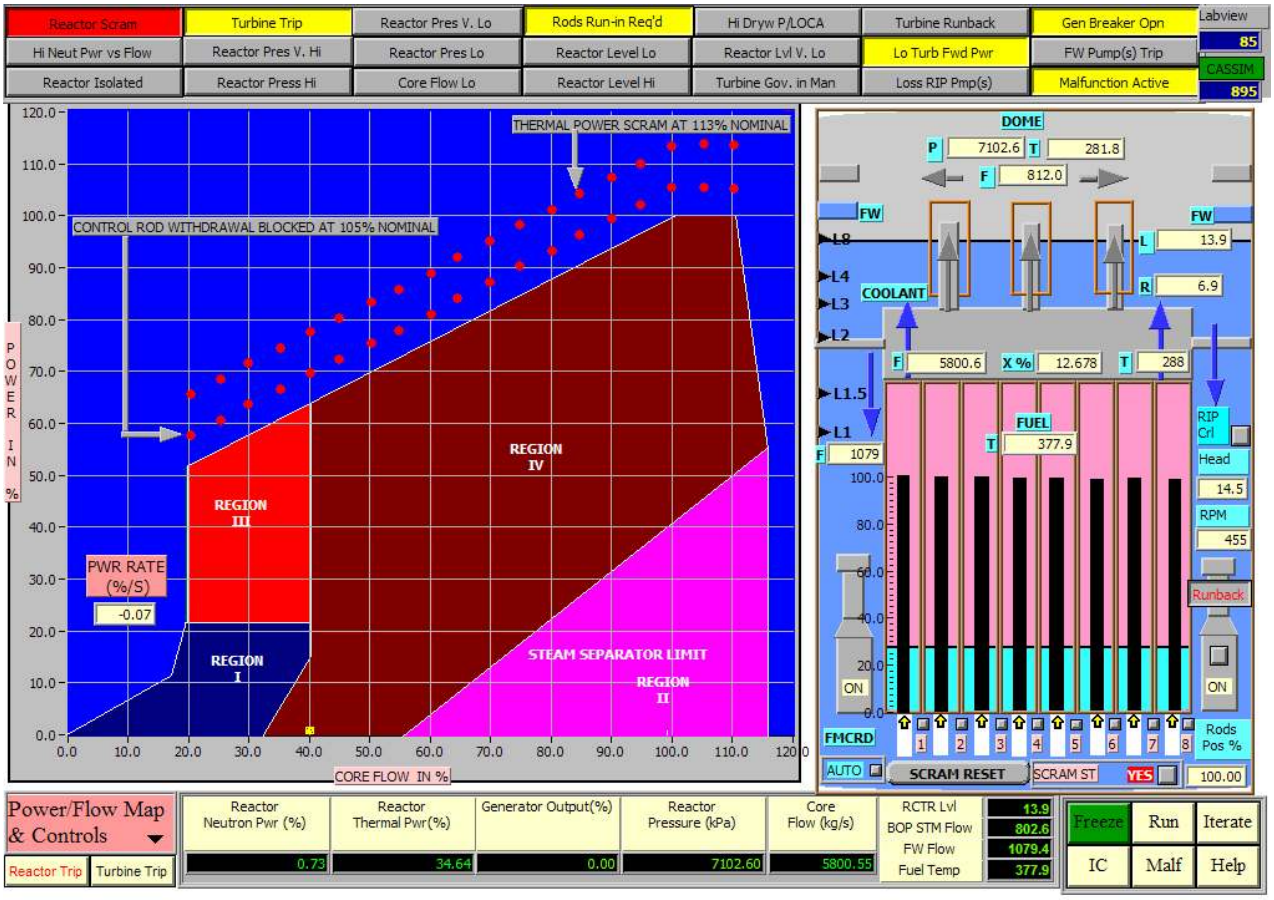


4.2.6 Turbine throttle PT fails low

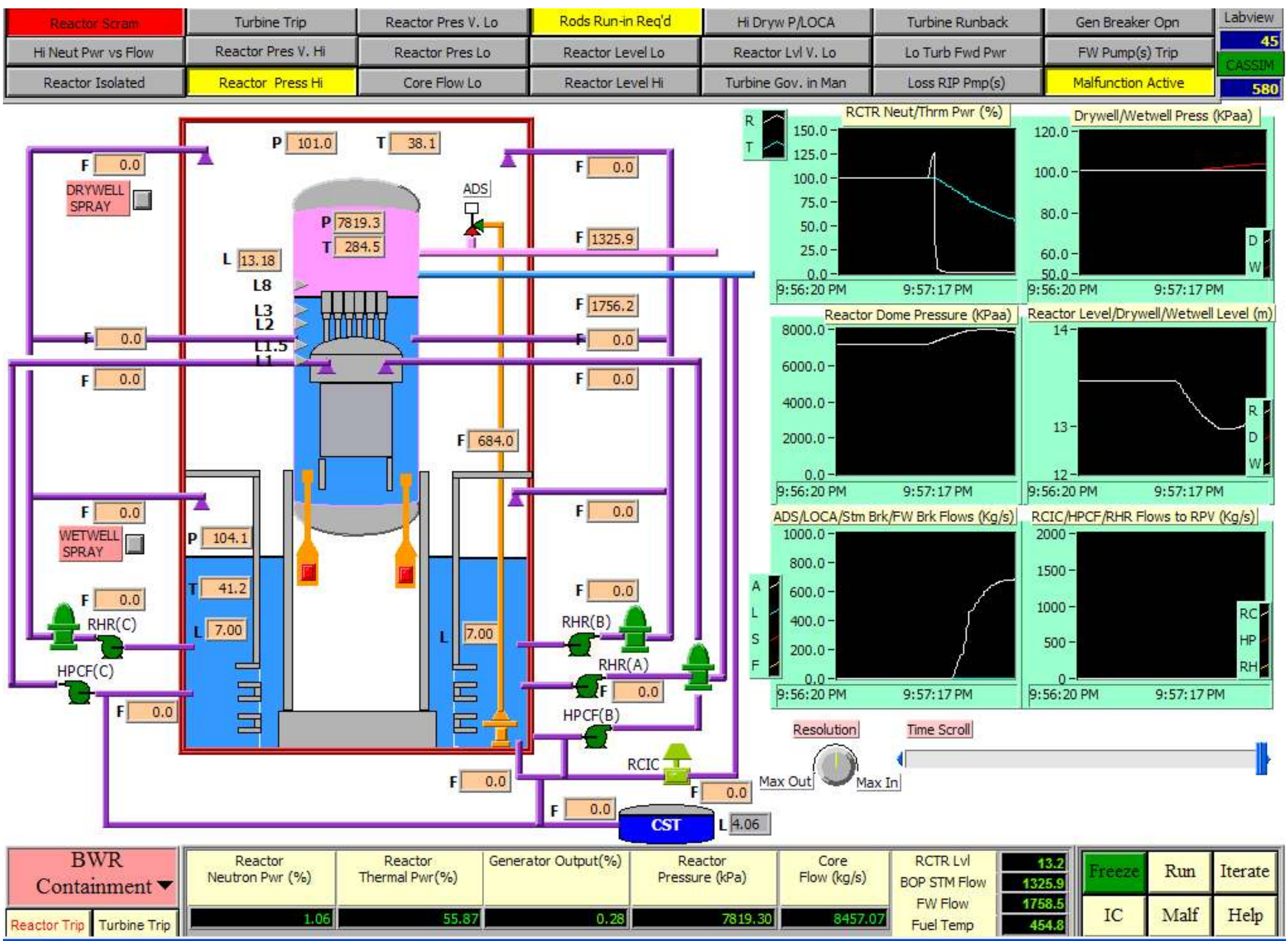
This malfunction will cause the turbine throttle pressure transmitter to fail “low”. The consequence is that the turbine governor control system is “fooled” into thinking that the main steam pressure is rapidly decreasing, hence as a regulation control action, the turbine governor will run back turbine load immediately in order to maintain main steam pressure, which in actual fact, is not “low”. The consequence is that the reactor pressure immediately shoots up rapidly.

Go to Power/Flow Map. Load the 100% FP IC and run the simulator. Now insert this malfunction. When this malfunction transient event occurs:

- On the Power/Flow Map Screen, observe the movement of the yellow cursor on the screen.
- Provide the explanation why the power increases so rapidly.
- As the power increases, it exceeds the maximum allowable limit given by the current core flow as per the Power/Flow Map. Therefore, a “Hi Neut Pwr vs Flow” alarm is generated. While this is happening, the Reactor Power Control System is inserting the control Rods, trying to reduce reactor power.
- But the reactor power increase is so fast that it exceeds the safety protection limits - first the “Rods Run-in Req’d” is activated, followed by Reactor Scram by “High Neutron Flux/Low Core Flow”, and “Reactor Pressure High”. Turbine Trip follows as a result of turbine runback on low reactor pressure.



Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
THERMAL POWER SCRAM AT 113% NOMINAL
CONTROL ROD WITHDRAWAL BLOCKED AT 105% NOMINAL
REGION III
REGION IV
REGION I
STEAM SEPARATOR LIMIT
REGION II
PWR RATE (%/S)
-0.07
POWER IN %
CORE FLOW IN %
DOME
COOLANT
FUEL
SCRAM RESET
Power/Flow Map & Controls
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
Freeze
Run
Iterate
IC
Malf
Help



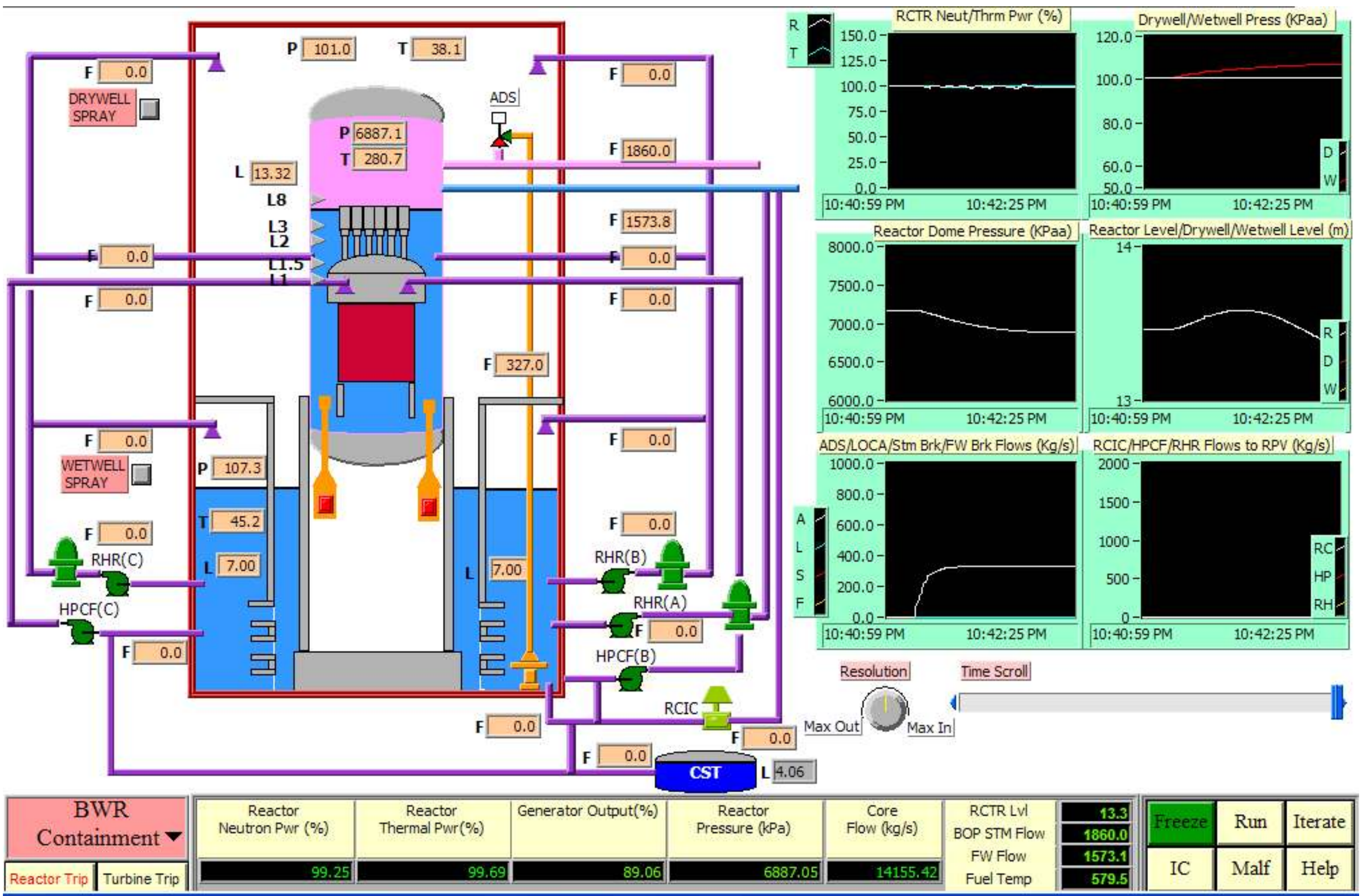
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
DRYWELL SPRAY
WETWELL SPRAY
ADS
RHR(C)
HPCF(C)
RHR(B)
RHR(A)
HPCF(B)
RCIC
CST
RCTR Neut/Thrm Pwr (%)
Drywell/Wetwell Press (KPaa)
Reactor Dome Pressure (KPaa)
Reactor Level/Drywell/Wetwell Level (m)
ADS/LOCA/Stm Brk/FW Brk Flows (Kg/s)
RCIC/HPCF/RHR Flows to RPV (Kg/s)
Resolution
Time Scroll
Max Out
Max In
BWR Containment
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
Freeze
Run
Iterate
IC
Malf
Help

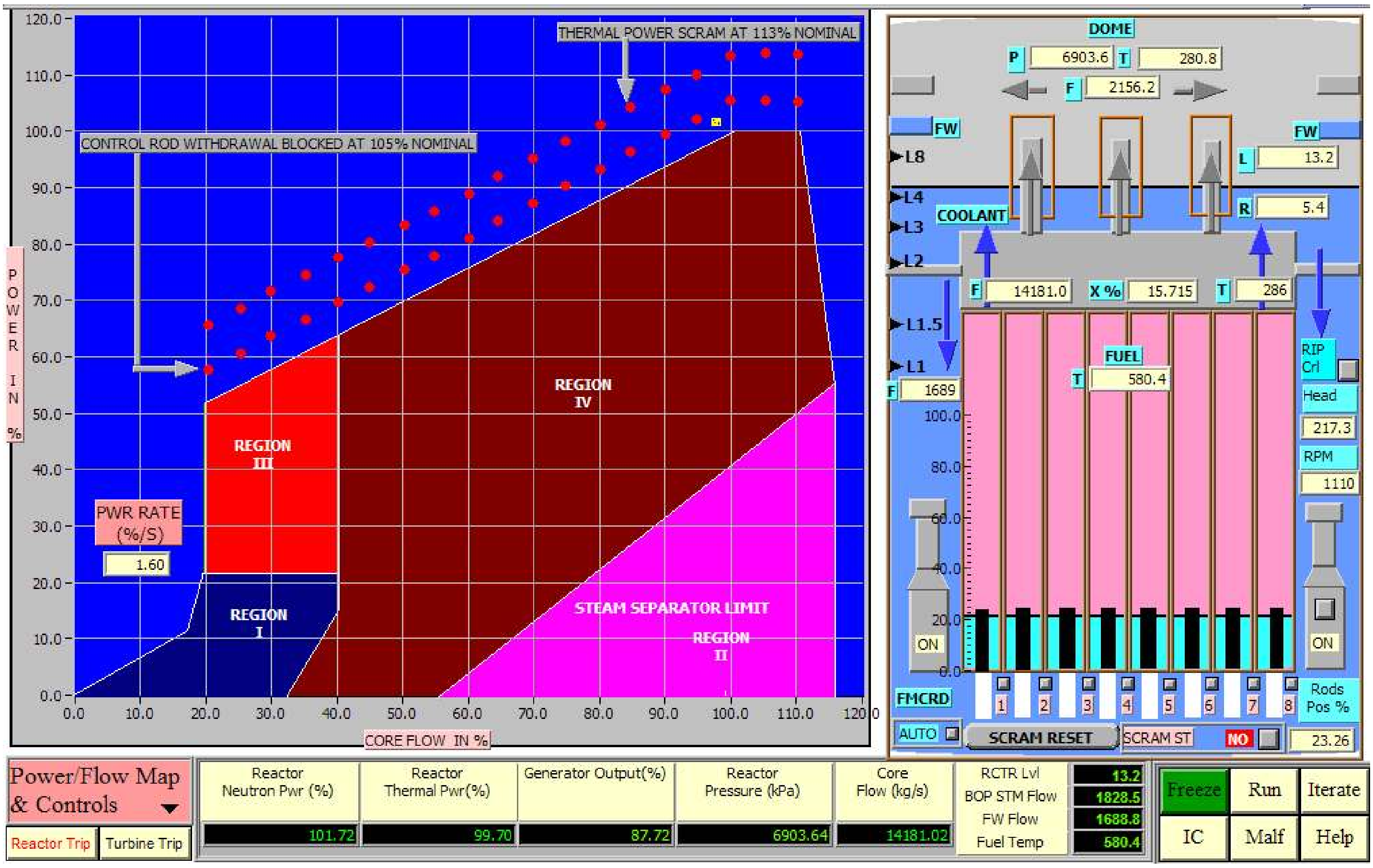
4.2.7 Safety relief valve (SRV) on one main steam line fails open

This malfunction will cause the safety relief valve (SRV) on one main steam line to fail open. The consequence is that reactor pressure falls immediately.

Go to BWR Turbine Generator Screen. Load the 100% FP IC. Run the simulator. Insert the malfunction. When this malfunction transient occurs:

- On the BWR Turbine Generator Screen, observe that SRV #1 opens to the suppression pool, and also note that turbine runback is initiated.
- Freeze the simulator. Go to Power/Flow Map. Reinitialize the simulator to 100% FP Initial Condition (IC). Run the simulator.
- Insert the malfunction again and observe the movement of the yellow cursor on the screen.
- Reactor power decreases initially in response to pressure drop. Why?
- Because the reactor power drops below setpoint, the reactor power control system will withdraw the control rods, trying to restore power. But the pressure drop is too fast and alarm "Reactor Pres Lo" is generated.
- When the transient settles down a bit, note the following parameters: reactor pressure, generator output (%), reactor neutron power (%).
- Discuss what should the operator do.





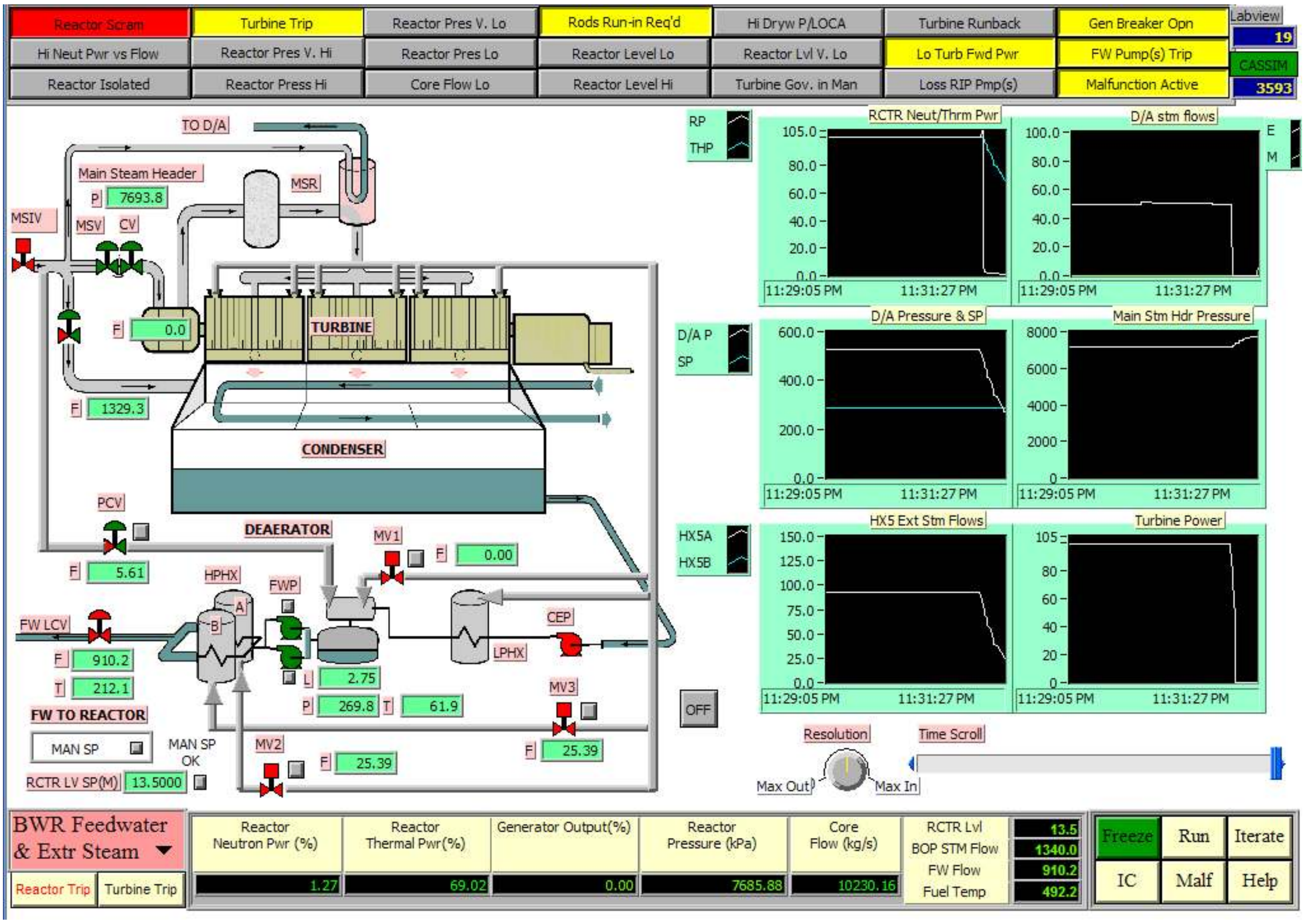
THERMAL POWER SCRAM AT 113% NOMINAL
CONTROL ROD WITHDRAWAL BLOCKED AT 105% NOMINAL
POWER IN %
CORE FLOW IN %
REGION I
REGION II
REGION III
REGION IV
STEAM SEPARATOR LIMIT
PWR RATE (%/S)
1.60
DOME
P 6903.6
T 280.8
F 2156.2
FW
L 13.2
R 5.4
COOLANT
F 14181.0
X % 15.715
T 286
FUEL
T 580.4
F 1689
RIP Crl
Head
217.3
RPM
1110
ON
FMCRD
AUTO
SCRAM RESET
SCRAM ST
NO
Rods Pos %
23.26
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 101.72
Reactor Thermal Pwr(%) 99.70
Generator Output(%) 87.72
Reactor Pressure (kPa) 6903.64
Core Flow (kg/s) 14181.02
RCTR Lvl 13.2
BOP STM Flow 1828.5
FW Flow 1688.8
Fuel Temp 580.4
Freeze
Run
Iterate
IC
Malf
Help

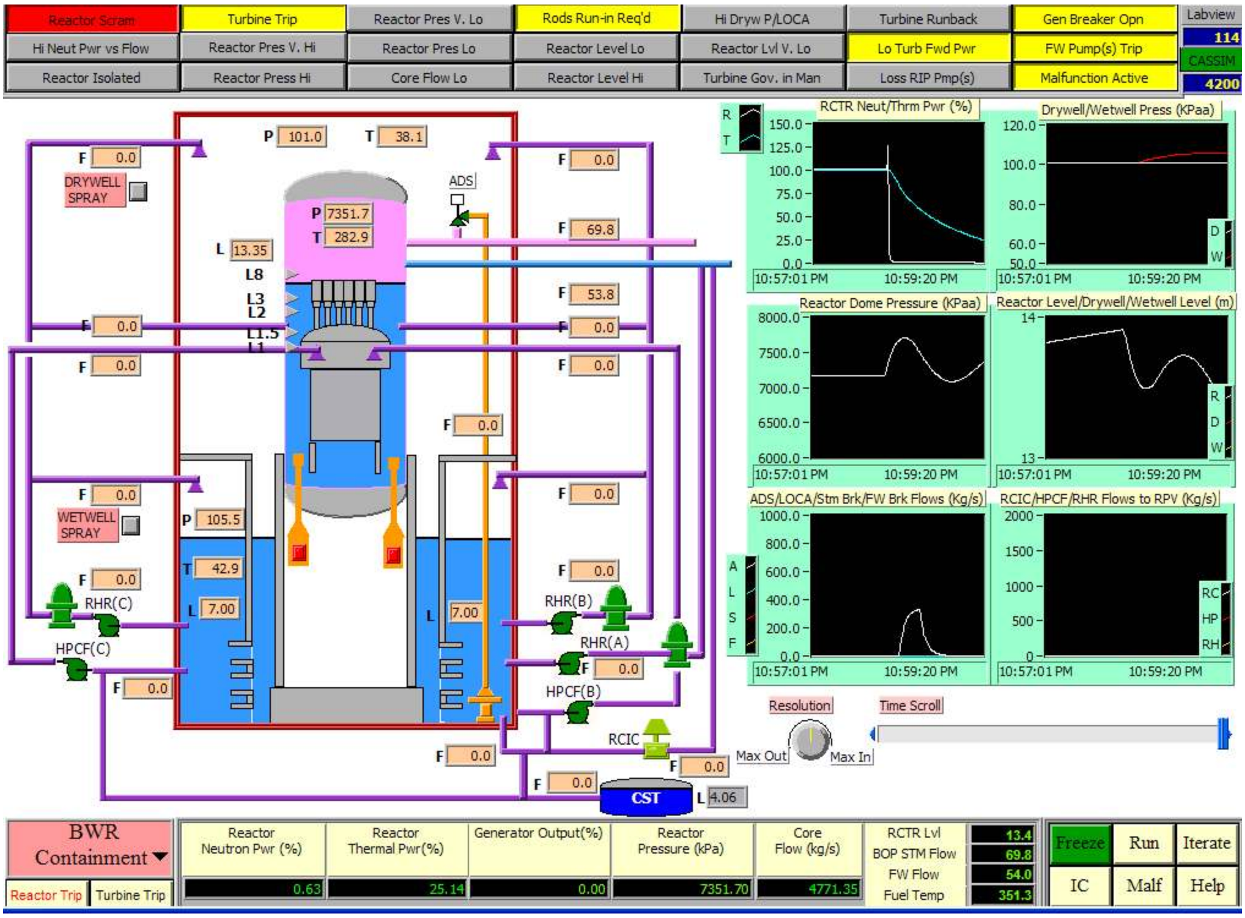
4.2.8 Feedwater level control valve fails open

This malfunction will cause the Feedwater Level Control Valve to fail open 100%. The consequence is that the reactor water level will increase immediately.

Go to BWR Feedwater & Extr Steam Screen. Load the 100 % FP IC. Run the simulator. Insert the malfunction. When this malfunction transient occurs:

- On the BWR Feedwater & Extr Steam Screen, observe that FW LCV flow is increasing.
- Go to Power/Flow Map Screen; observe the movement of the yellow cursor on the screen. Note the reactor water level, reactor neutron power, reactor pressure, FW flow.
- Let the malfunction event run for about 5 minutes. Note the reactor water level, reactor neutron power, reactor pressure, FW flow.
- Reactor water level has increased, due to the imbalance between the steam flow and the feedwater flow. Has the Reactor pressure increased or decreased? Has reactor neutron power increased or decreased? Provide explanation.
- Observe any movement of control rods and provide explanation why this is happening.
- If the malfunction is left running for some time, there will be a "Reactor Level Hi" alarm, followed by reactor scram on reactor high level. FW will also be tripped on L8. Confirm this by going to BWR Scram Parameters Screen, after the reactor scram has occurred.





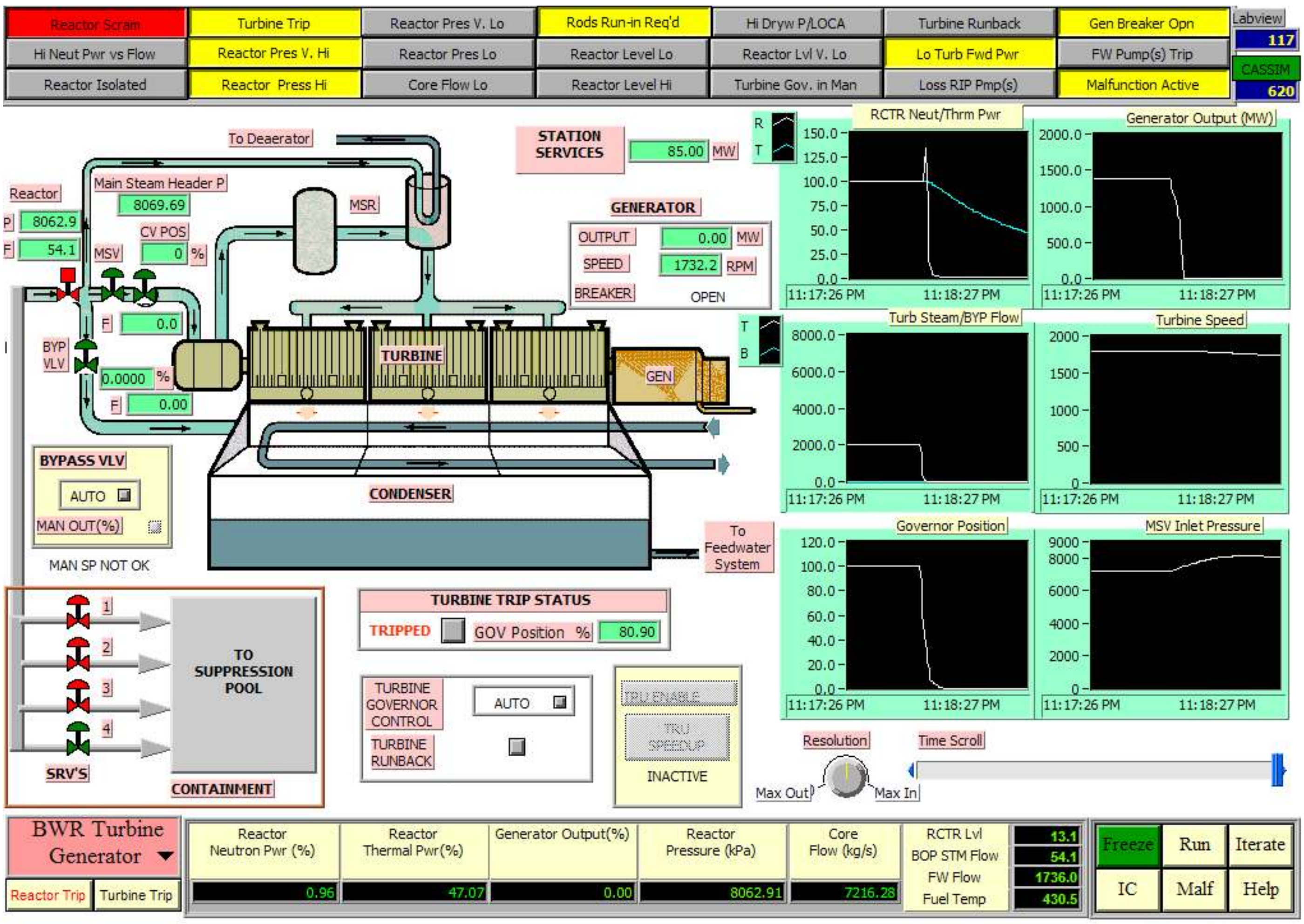
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
114
CASSIM
4200
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
DRYWELL SPRAY
WETWELL SPRAY
ADS
RHR(C)
HPCF(C)
RHR(B)
RHR(A)
HPCF(B)
RCIC
CST
RCTR Neut/Thrm Pwr (%)
Drywell/Wetwell Press (KPaa)
Reactor Dome Pressure (KPaa)
Reactor Level/Drywell/Wetwell Level (m)
ADS/LOCA/Stm Brk/FW Brk Flows (Kg/s)
RCIC/HPCF/RHR Flows to RPV (Kg/s)
10:57:01 PM
10:59:20 PM
Resolution
Time Scroll
Max Out
Max In
BWR Containment
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
RCTR Lvl
BOP STM Flow
FW Flow
Fuel Temp
0.63
25.14
0.00
7351.70
4771.35
13.4
69.8
54.0
351.3
Freeze
Run
Iterate
IC
Malf
Help

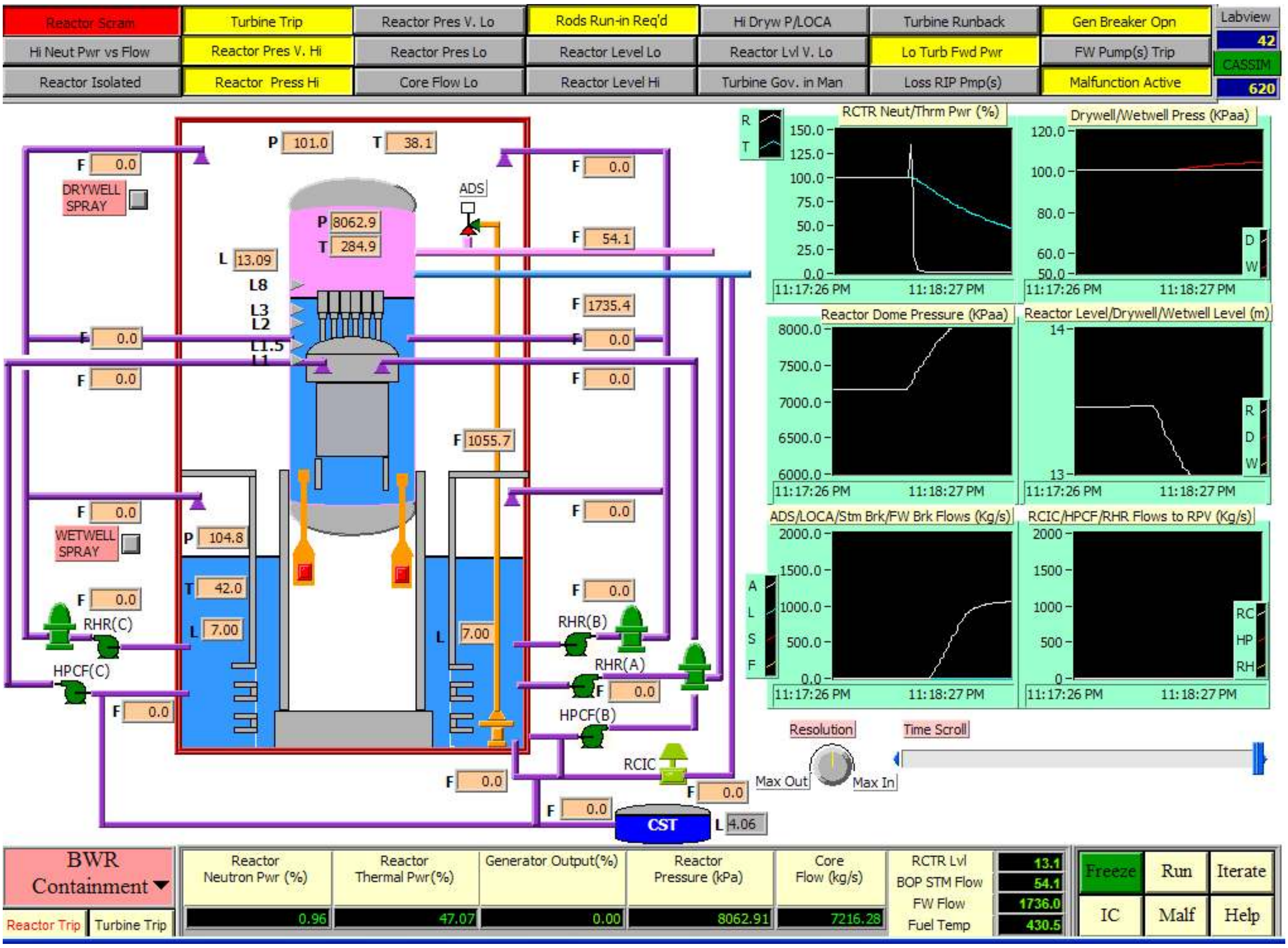
4.2.9 Turbine trip with bypass valve failed closed

This malfunction will cause two failures to occur at the same time: (1) turbine trip (2) the turbine steam bypass valve failed closed. The consequence is that the reactor will have high steam pressure, and the safety relief valves (SRVs) will open to the suppression pool in order to relieve the steam pressure.

Go to BWR Turbine Generator Screen. Load the 100% FP IC. Run the simulator. Insert this malfunction. When this malfunction transient occurs:

- On the BWR Turbine Generator Screen, observe that turbine main stop valve (MSV) and the bypass valve (BYP VLV) are closed. Alarms indicate that the turbine is tripped.
- Note the reactor pressure at the bottom of the screen and the “Reactor Press Hi” alarm.
- Note any SRV opening to suppression pool.
- Reactor pressure will rise rapidly, leading to “Reactor Pres. V. Hi” alarm, followed by reactor scram.
- Confirm the reactor scram parameter by going to the BWR Scram Parameter screen.





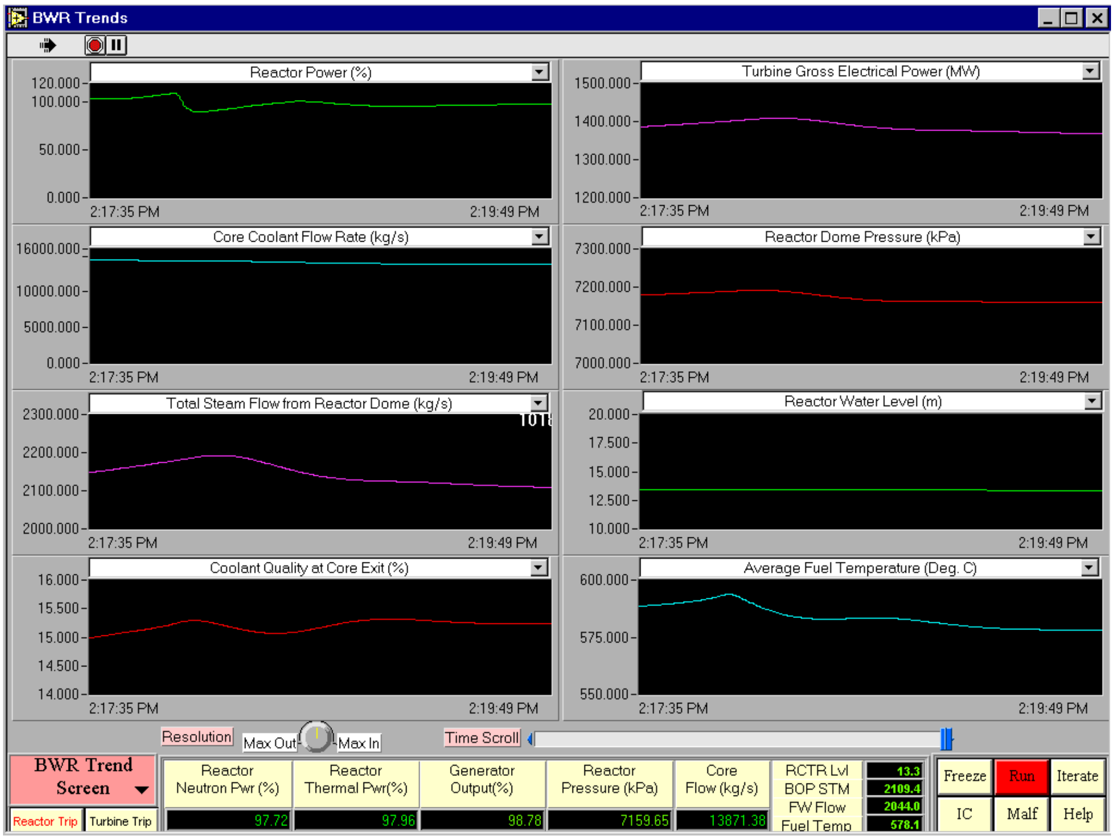
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
42
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
620
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
DRYWELL SPRAY
WETWELL SPRAY
ADS
RHR(C)
HPCF(C)
RHR(B)
RHR(A)
HPCF(B)
RCIC
CST
RCTR Neut/Thrm Pwr (%)
Drywell/Wetwell Press (KPaa)
Reactor Dome Pressure (KPaa)
Reactor Level/Drywell/Wetwell Level (m)
ADS/LOCA/Stm Brk/FW Brk Flows (Kg/s)
RCIC/HPCF/RHR Flows to RPV (Kg/s)
11:17:26 PM
11:18:27 PM
Resolution
Time Scroll
Max Out
Max In
BWR Containment
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.96
Reactor Thermal Pwr(%) 47.07
Generator Output(%) 0.00
Reactor Pressure (kPa) 8062.91
Core Flow (kg/s) 7216.28
RCTR Lvl 13.1
BOP STM Flow 54.1
FW Flow 1736.0
Fuel Temp 430.5
Freeze
Run
Iterate
IC
Malf
Help

4.2.10 Inadvertent withdrawal of one bank of rods

This malfunction will cause inadvertent withdrawal of control rod Bank #1. The consequence is that the reactor will suddenly have a positive reactivity addition.

Go to Power/Flow Map Screen. Load the 100% FP IC. Run the simulator. Insert this malfunction. When this malfunction transient occurs:

- On the Power/Flow Map Screen, observe the movement of the yellow cursor.
- Note the arrow pointer for control rods Bank #1 and note its position in the core. Observe that reactor power is increasing.
- In response to the reactor power increase, what would the reactor power control system do in order to compensate the sudden increase in power? Note the coolant flow rate and provide explanation.
- After the malfunction event runs for a while, the "Hi Neut Pwr vs Flow" alarm is generated. This is followed by the insertion of the control rods in the core to compensate for the sudden increase in reactor power.
- Note the movement of the yellow cursor and note the control rods Bank # 1 position.
- Eventually Bank #1 control rods are fully withdrawn, and the transient is stabilized.

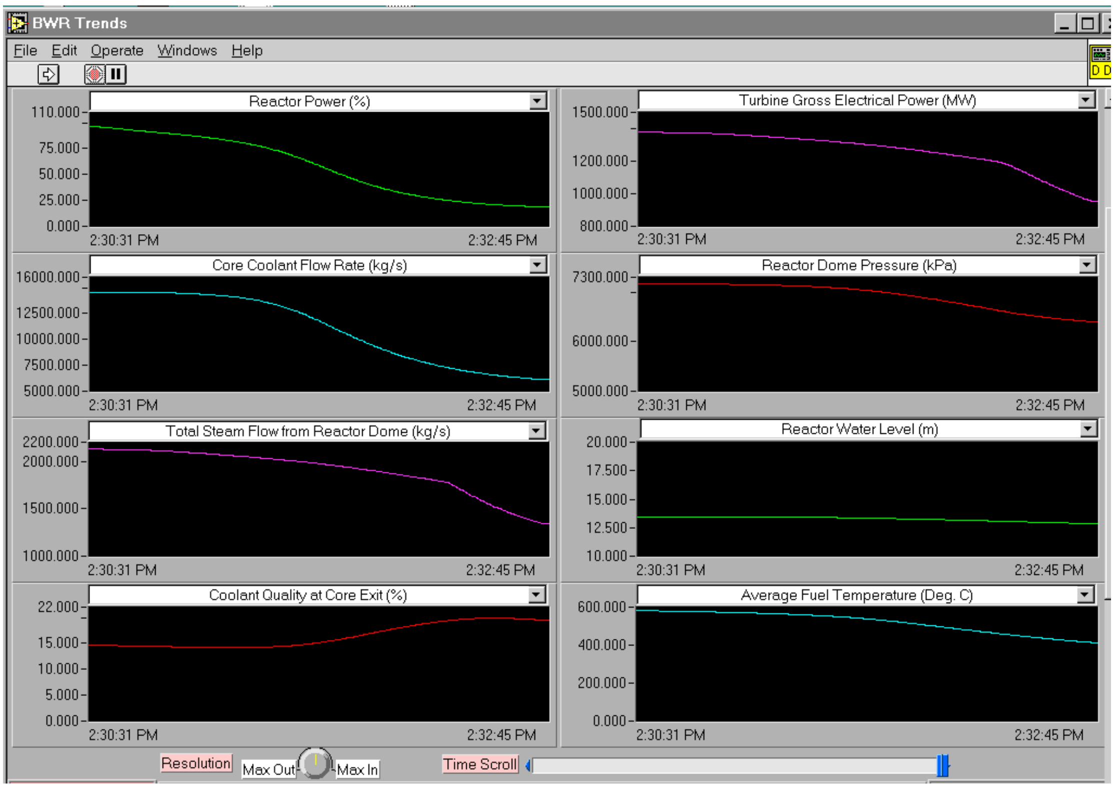


4.2.11 Inadvertent insertion of one bank of rods

This malfunction will cause inadvertent insertion of control rods Bank #1. The consequence is that the reactor will suddenly have a negative reactivity addition.

Go to Power/Flow Map Screen. Load the 100% FP IC. Run the simulator. Insert this malfunction. When this malfunction transient occurs:

- On the Power/Flow Map Screen, observe the movement of the yellow cursor.
- Note the arrow pointer for control rods Bank #1 and note its position in the core. Observe that reactor power is decreasing.
- In response to the reactor power decrease, what would the reactor power control system do in order to compensate the sudden decrease in power? Note the coolant flow rate and provide explanation.
- As the reactor power is decreasing steadily due to the continuous insertion of Bank #1 control rods, note the coolant flow rate again and compare with previous measurement.
- The reactor power control system recognizes that increasing the coolant flow rate is unable to maintain the reactor power that is decreasing steadily. Therefore it will ramp down the coolant flow rate in step with the reactor power decrease, in accordance with the recommended unit shutdown path in the Power/Flow Map.
- Because the reactor power is reduced at a fast pace, reactor low pressure results, as shown by the alarm “Reactor Pres Lo”. Note the discrepancies between reactor neutron power (%), reactor thermal power (%) and generator output (%) and provide explanation for the discrepancies.
- In response to low pressure in reactor dome, the turbine control system will runback the turbine in order to restore main steam pressure.



4.2.12 Inadvertent reactor isolation

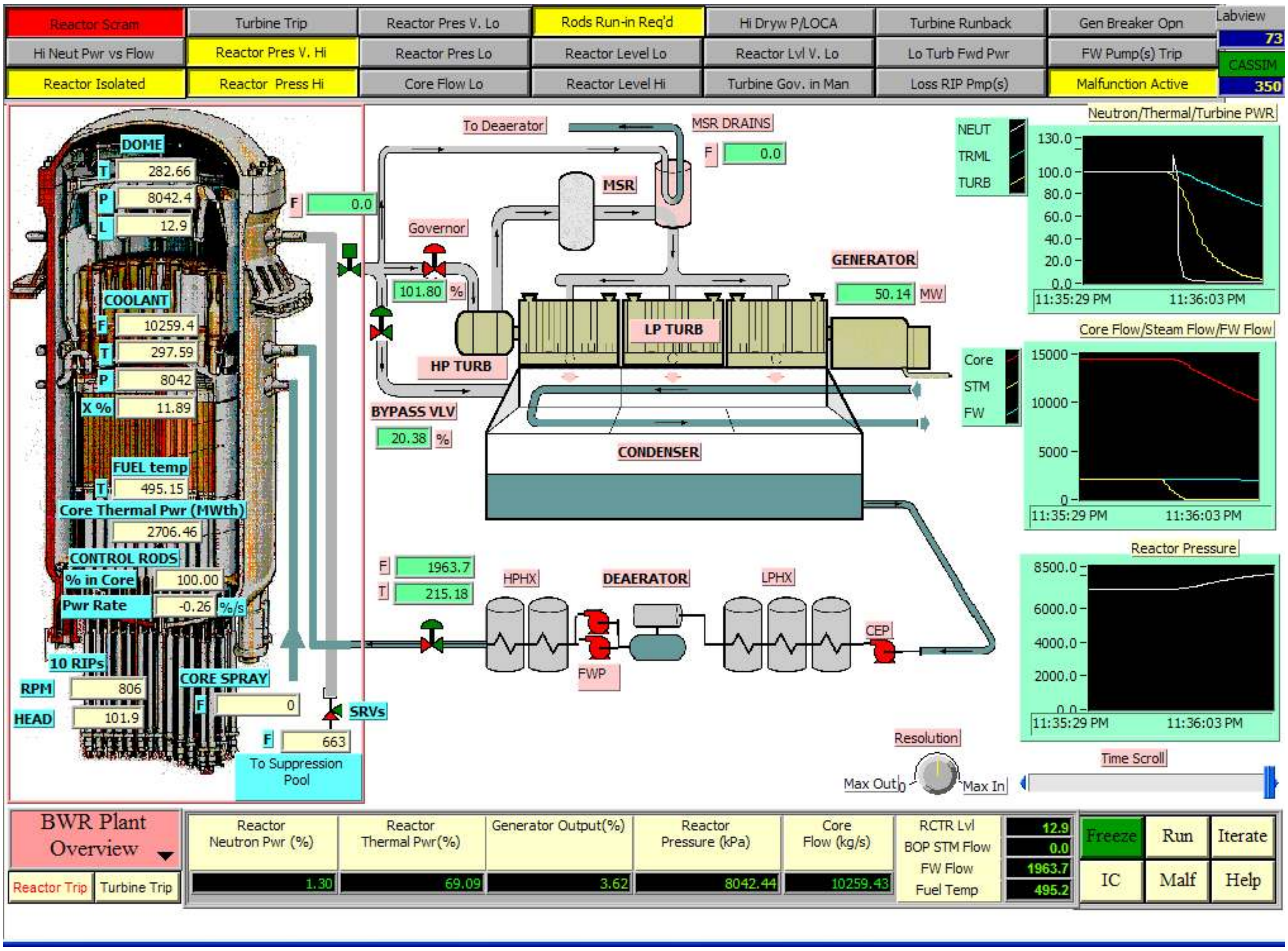
This malfunction will cause inadvertent closing of the reactor vessel isolation valve. The consequence is that the reactor vessel steam is not supplied to the turbine generator.

Go to BWR Plant Overview Screen. Load the 100% FP IC and run the simulator. Insert the malfunction. When this malfunction transient occurs:

- On the BWR Plant Overview Screen, observe the status of the reactor isolation valve.
- Note the steam flow from the reactor dome. Observe the status of SRV to suppression pool.
- At the bottom of the screen, note the following parameters, as the event evolves:

	10 seconds	30 seconds	1 minute	5 minutes
Reactor Power				
Generator Output				
Reactor Pressure				
Core flow				

- As the reactor isolation valve closes, the reactor pressure increases rapidly. As a result, “Reactor Press Hi” alarm is generated.
- Due to vapor void collapses due to the high pressure - hence less void fraction in core, there is a positive reactivity insertion. Therefore, reactor power increases.
- Because reactor power increases beyond the power limit allowable by the Power/Flow Map, “Rods Run-in “ protection is activated, hence the alarm is generated. The reactor power is decreased in response to control rod insertion.
- Because of losing steam flow to the turbine generator, the generator output (%) is decreasing, and subsequently the turbine is tripped due to low Turbine forward power.
- When the reactor isolation valve is fully closed, the reactor is scrammed due to reactor isolation. Confirm this by reviewing the BWR scram parameter screen.
- Repeat this malfunction by re-initializing to 100% FP IC. But this time go to the Power/Flow Map screen and observe the movement of the yellow cursor.



Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
73
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
350
DOME
T 282.66
P 8042.4
L 12.9
COOLANT
F 10259.4
T 297.59
P 8042
X % 11.89
FUEL temp
T 495.15
Core Thermal Pwr (MWth)
2706.46
CONTROL RODS
% in Core 100.00
Pwr Rate -0.26 %/s
10 RIPs
RPM 806
HEAD 101.9
CORE SPRAY
F 0
SRVs
F 663
To Suppression Pool
F 0.0
To Deaerator
MSR DRAINS
F 0.0
MSR
Governor
101.80 %
HP TURB
LP TURB
GENERATOR
50.14 MW
BYPASS VLV
20.38 %
CONDENSER
F 1963.7
T 215.18
HPHX
DEAERATOR
LPHX
CEP
FWP
Neutron/Thermal/Turbine PWR
NEUT
TRML
TURB
130.0
100.0
80.0
60.0
40.0
20.0
0.0
11:35:29 PM
11:36:03 PM
Core Flow/Steam Flow/FW Flow
Core
STM
FW
15000
10000
5000
0
11:35:29 PM
11:36:03 PM
Reactor Pressure
8500.0
6000.0
4000.0
2000.0
0.0
11:35:29 PM
11:36:03 PM
Resolution
Max Out
Max In
Time Scroll
BWR Plant Overview
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 1.30
Reactor Thermal Pwr(%) 69.09
Generator Output(%) 3.62
Reactor Pressure (kPa) 8042.44
Core Flow (kg/s) 10259.43
RCTR Lvl 12.9
BOP STM Flow 0.0
FW Flow 1963.7
Fuel Temp 495.2
Freeze
Run
Iterate
IC
Malf
Help

4.2.13 Loss of feedwater heating

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

(1) Steam extraction line to heater is closed.
(2) Feedwater is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the feedwater bypasses the heater and no heating of that feedwater occurs. In either case, the reactor vessel receives cooler feedwater.

A loss of up to 55.6 °C of the feedwater heating capability of the plant will cause an increase in core inlet subcooling. This can increase core power by 18 % due to the negative void reactivity coefficient. However, the power increase is slow. The Feedwater Control System (FWCS) includes a logic intended to mitigate the consequences of a loss of feedwater heating capability. The system will be constantly monitoring the actual feedwater temperature and comparing it with a reference temperature. When a loss of feedwater heating is detected (i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint, which is currently set at 16.7 ° C), the FWCS sends an alarm to the operator.

The operator can then take actions to mitigate the event. This will avoid a scram and reduce the Δ CPR during the event. The same signal is also sent to the RCIS to initiate the SCRRI (selected control rods run-in) to automatically reduce the reactor power and avoid a scram. This will prevent the reactor from violating any thermal limits.

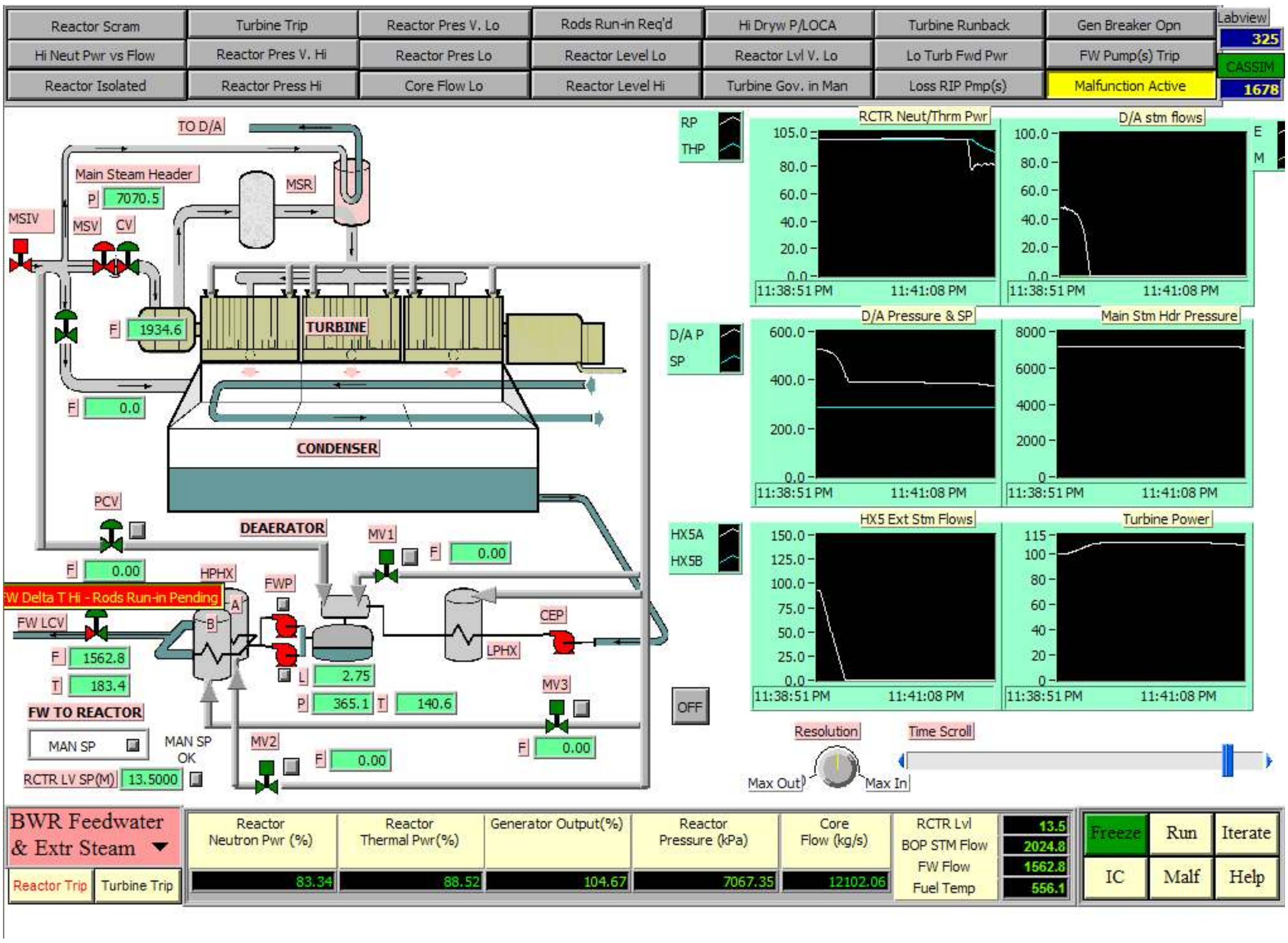
Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

Load the 100% FP IC. Run the simulator.

- Go to BWR Feedwater & Extr Steam Screen. Record the extraction steam flows to Deaerator and HP heaters. Record the feedwater temperature going to the reactor.
- Insert the above malfunction. This malfunction causes all the extraction steam valves to close — namely, MV1, MV2, and MV3. The consequence is total loss of feedwater heating.
- Record the following parameters as the malfunction event evolves:

	1 minute	3 minutes	5 minutes	7 minutes	10 minutes
Reactor Power (%)					
Generator Power (%)					
FW Temp (°C)					
Reactor Pressure (kPa)					
Core Flow (kg/s)					

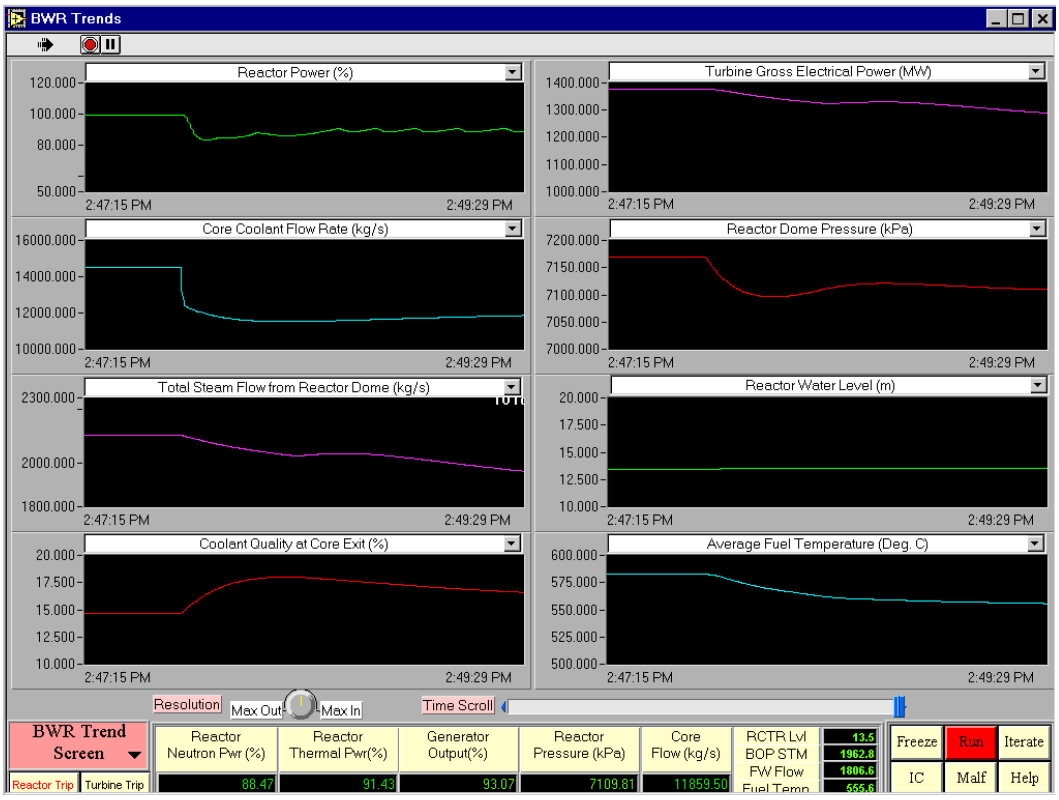
- Observe that generator output (%) is increasing. This is due to the fact that the steam, which is supposed to be used for feedwater heating, is now used to do work in the turbine. Hence the generator output (%) increases.
- Does the reactor power change at this time?
- How much did the feedwater temperature drop since the initiation of the malfunction ?
- Colder feedwater temperature into the reactor core means that the coolant becomes more subcooled. Hence the non-boiling height will increase. When the two-phase coolant mixes with a more subcooled feedwater in the downcomer, the effect is less channel quality, and therefore less void fraction. The result is a positive reactivity change. Hence the reactor power increases after some time.
- But the reactor power increase is above the target setpoint, therefore control rods are inserted momentarily to decrease reactor power. One can put the control rods in Manual mode and observe the reactor power increase trend.
- Over time, the system will reach a new thermodynamic equilibrium state with the new steady state feedwater temperature, where reactor power and the generator power will be equal again.



4.2.14 Power loss to three reactor internal pumps (RIPs)

Note that there are 10 reactor internal pumps (RIPs). This malfunction causes the loss of power to three RIPS, resulting in a sudden and large reduction of core flow.

- Go to Power/Flow Map Screen. Load the 100% FP IC and run the simulator. Record the core flow.
- Then insert the above malfunction. Observe the movement of the yellow cursor on the screen. Record the core flow as the malfunction event evolves.
- How does the yellow cursor travel on the Power/Flow Map? The alarm “Hi Neut Pwr versus Flow” is generated. Provide explanation for this alarm.
- After a step reduction in core flow, there is a substantial increase in voids in the core, hence a negative reactivity change. The reactor power will decrease accordingly.
- One can observe that the reactor power drop is more than the power setpoint given by the core flow at that time, as determined by the recommended path on the Power/flow map. Therefore, the reactor power control system will increase reactor power by withdrawing control rods, in order to bring the cursor back to the optimum path on the Power/Flow Map.
- The transient will settle at about 89% FP, with core flow of 11,965 kg/s.



4.2.15 Steam line break inside drywell

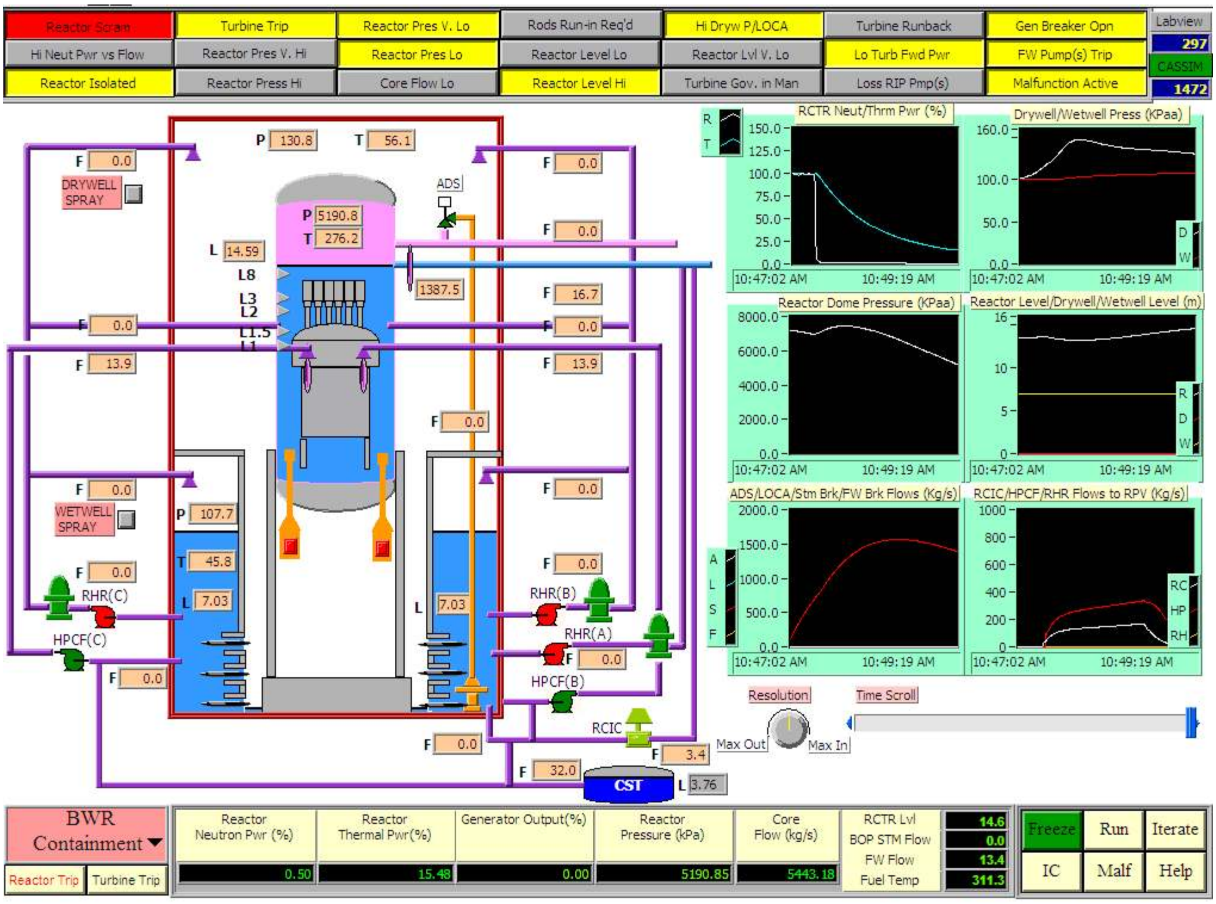
This malfunction causes a main steam line break (before the main steam isolation valve) inside the containment drywell.

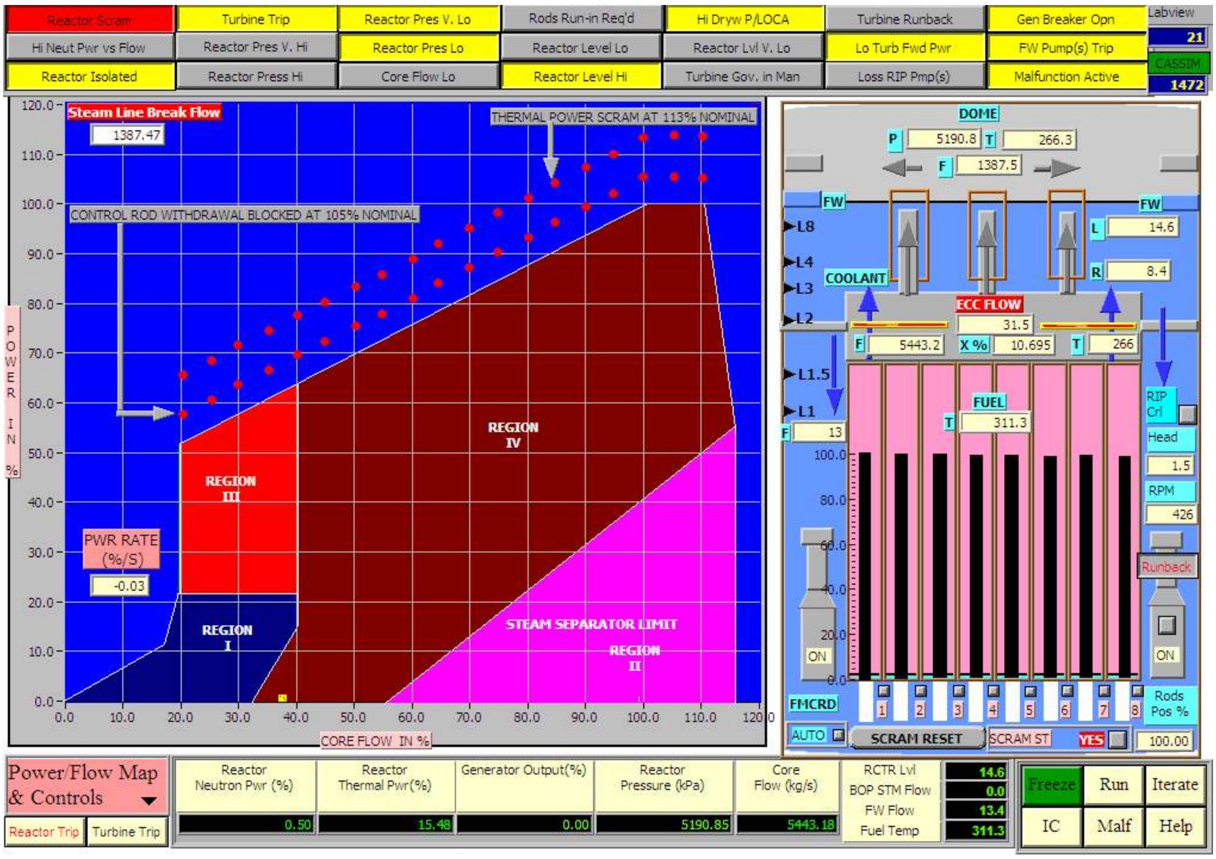
Go to Power/Flow Screen and load the 100% FP IC. Run the simulator. Introduce the malfunction. When this malfunction transient occurs:

- The break flow into the drywell will increase rapidly, resulting in depressurization of the reactor dome, as well as pressurization of the drywell.
- The consequence is the detection of a LOCA condition, which automatically activates the emergency core cooling (ECC) system to spray cooling water into the core for emergency cooldown.
- Record the following parameters as the malfunction event evolves:

	5 s	10 s	30 s	1 minute	5 minutes or longer
Break Flow					
Reactor pressure					
Steam flow from dome					
Feedwater flow					
ECC activation - note which ECC system is activated: ADS, RCIC, HPCF, LPCF and the respective flow.					
Coolant Temp					
Fuel Temp					
Reactor Power					
Drywell Pressure					
Wetwell Pressure					

- As the reactor dome depressurizes, there is more boiling in the core, hence more voids. The result is a negative reactivity change, leading to reactor power decrease. Observe that there is "Reactor Lo Pres" alarm.
- At the same time, the drywell is pressurized very quickly by the steam flow from the break. In a short time, a LOCA is detected, and reactor is scrammed by "high drywell pressure/LOCA detected" trip logic.
- When LOCA is detected, emergency core cooling (ECC) is activated. FW pumps will also be tripped, when ECC is in service. At the same time, main steam isolation valve will close, in order to isolate the containment.
- Due to lack of steam flow, the turbine throttle pressure decreases rapidly, leading to turbine runback by the turbine control system. The result is turbine trip by low turbine forward power (alarm "Lo Turb Fwd pwr")
- Go to BWR plant overview screen. Re-initialize the simulator with the 100% IC.
- Repeat the malfunction event, and observe the evolution of all the trended variables.





4.2.16 Feedwater line break inside drywell

This malfunction causes a feedwater line break inside the containment drywell. The feedwater break flow into the drywell will increase rapidly, resulting in pressurization of the drywell. The consequence is the detection of a LOCA condition, which automatically activates the emergency core cooling (ECC) system to spray cooling water into the core for emergency cooldown.

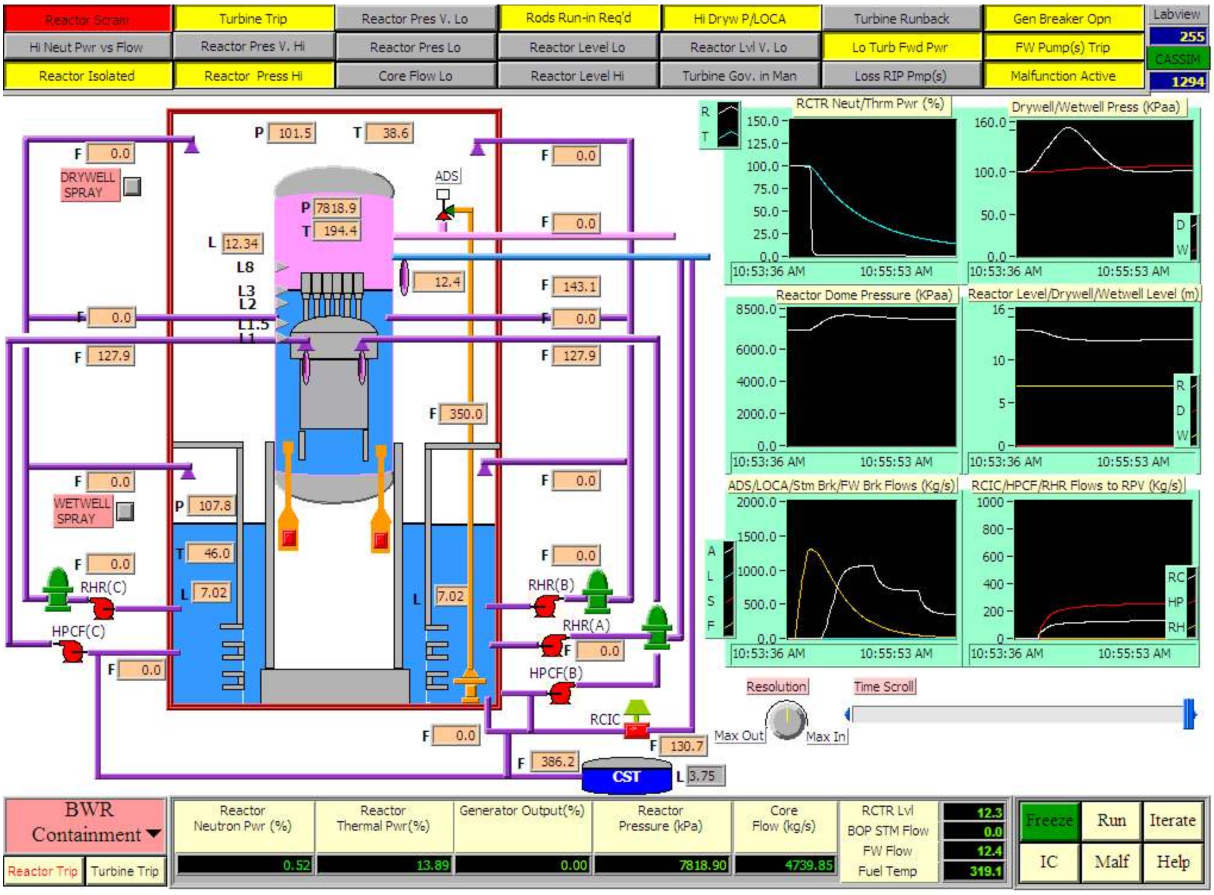
First go to Power/Flow Screen and load the 100% FP IC. Run the simulator. Now insert the above malfunction. When this malfunction transient occurs:

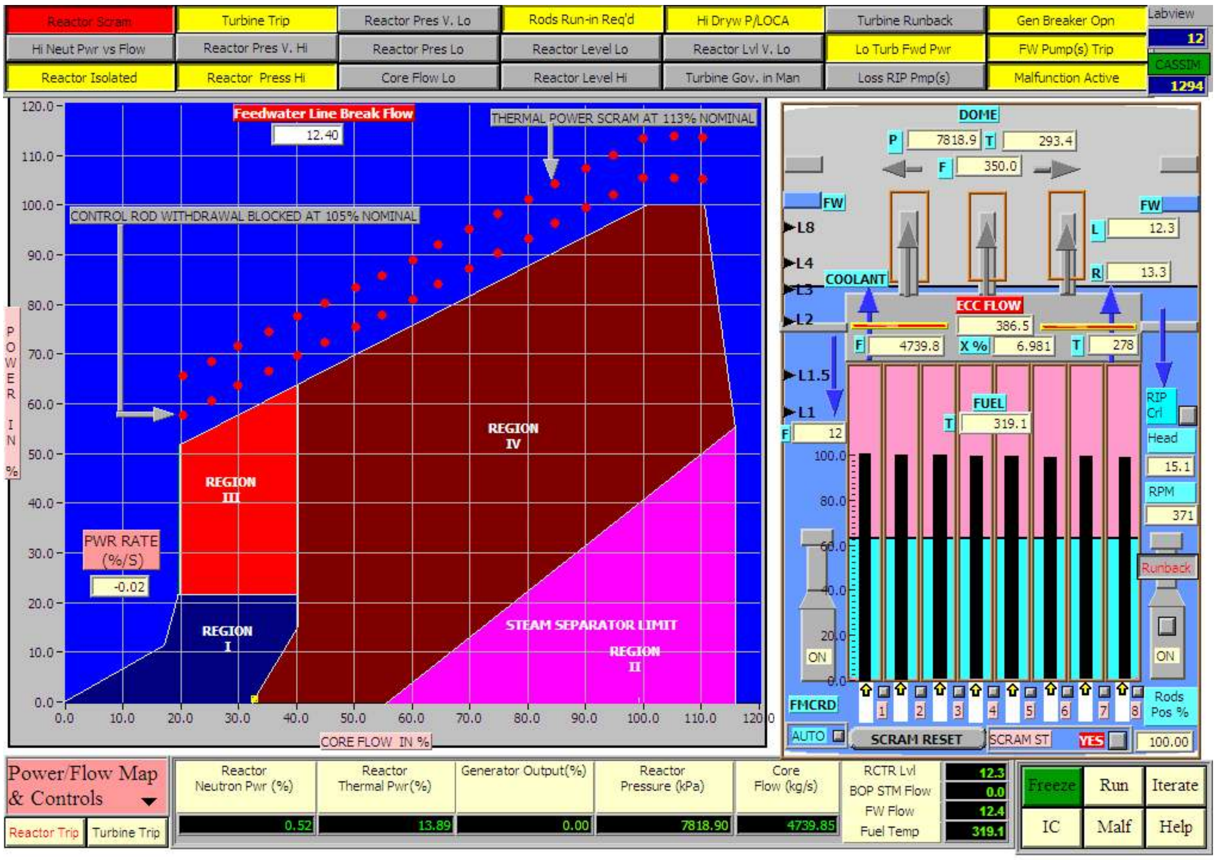
- Record the following parameters as the malfunction event evolves:

	5 s	10 s	30 s	1 minute	5 minutes or longer
FW Break Flow					
Reactor pressure					
Steam flow from dome					
FW flow to core					
ECC activation - note which ECC system is activated: ADS, RCIC, HPCF, LPCF and the respective flow.					
Coolant Temp					
Fuel Temp					
Reactor Power					
Drywell Pressure					
Wetwell Pressure					

- At the onset of the feedwater line break, the reactor water level drops, thus the steam volume in the dome increases. This in turn decreases the dome pressure slightly. As a result, the void fraction in the core increases slightly, leading to slight decrease in reactor power. That is why the yellow cursor on the screen moves downward.
- At the same time, the drywell is pressurized very quickly by the feedwater flow from the break. In a short time, LOCA is detected, and the reactor is scrammed by “high drywell pressure/LOCA detected” trip logic.
- When LOCA is detected, the ECC is activated. FW pumps will also be tripped, when ECC is in service. At the same time, the main steam isolation valve will close, in order to isolate the containment.
- Due to the closing of the reactor isolation valve, the reactor pressure increases rapidly, resulting in “Reactor Press Hi” alarm, and the SRV will be opened to the suppression Pool.
- As there is no steam flow going to the turbine, the turbine throttle pressure decreases rapidly, leading to turbine runback by the turbine control system. The result is turbine trip by low turbine forward power (alarm “Lo Turb Fwd Pwr”)
- Go to BWR Plant Overview Screen. Re-initialize the simulator with the 100% IC.

- Repeat the malfunction event, and observe the evolution of all the trended variables.





4.2.17 Reactor Vessel Medium Size Break ~ 500 kg/s LOCA

This malfunction causes a “crack” opening at the Reactor Vessel bottom, resulting in a break flow ~ 800 kg/s LOCA event. The break flow will pressurize the drywell rapidly. The consequence is the detection of a LOCA condition, which automatically activates the ECC system to spray cooling water into the core for emergency cooldown.

Go to Power/Flow Screen and load the 100% FP IC. Run the simulator. Now insert the malfunction.

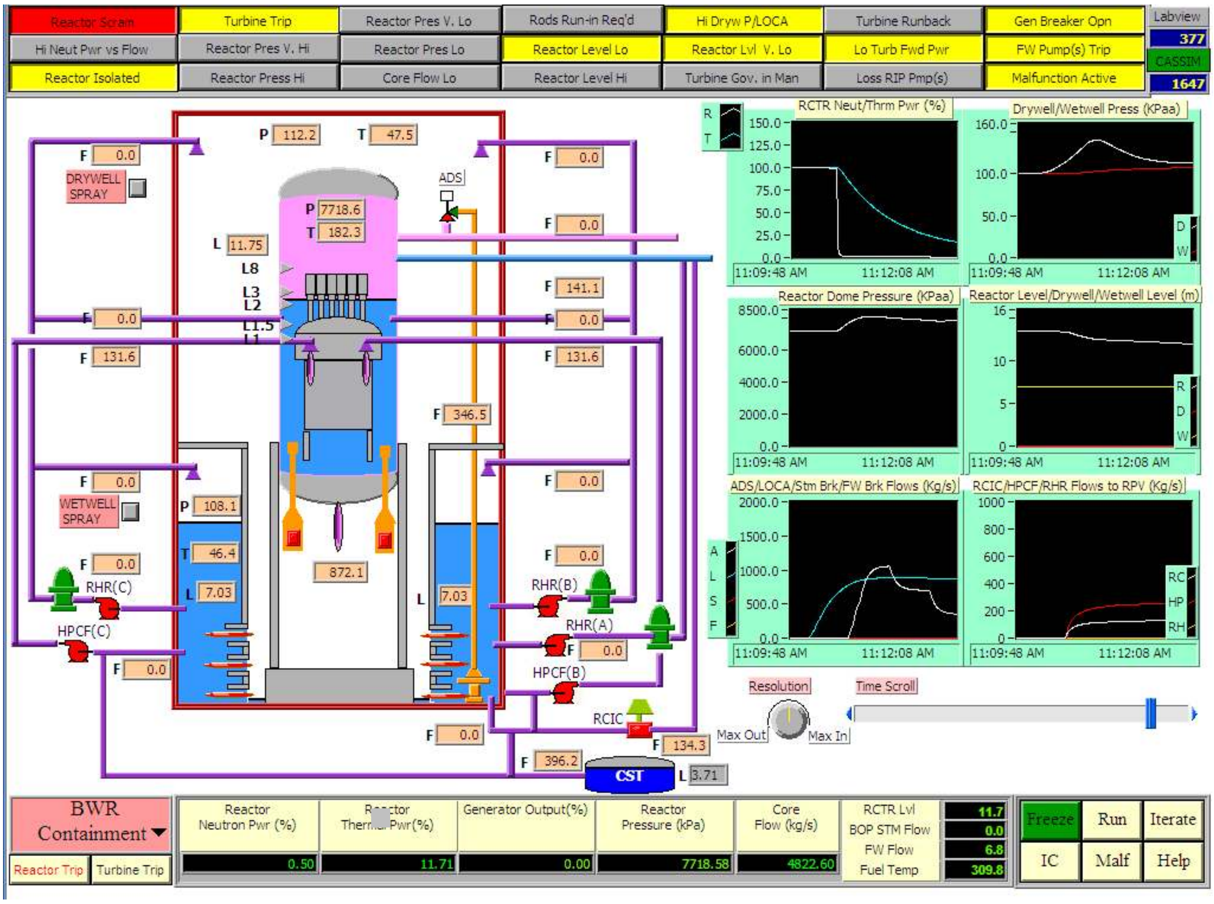
- Record the following parameters as the malfunction event evolves:

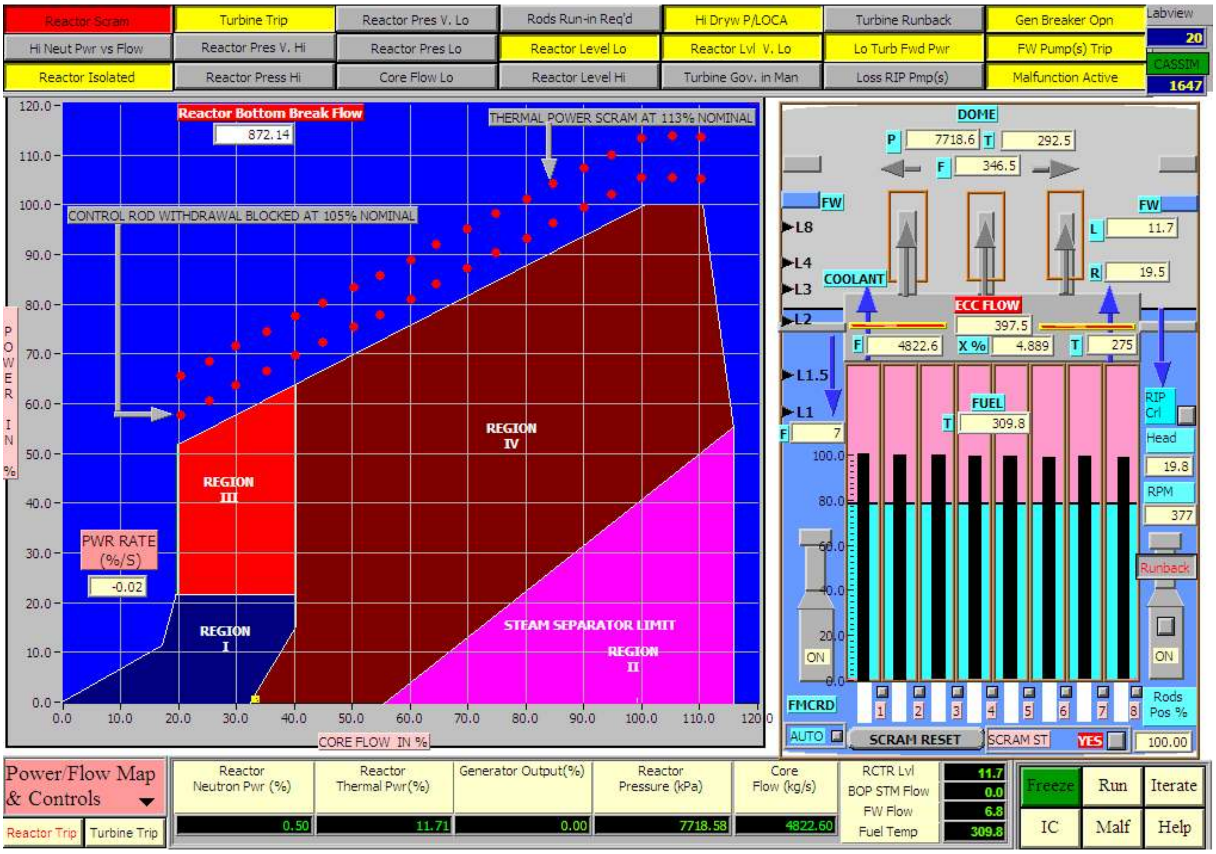
	5 s	10 s	30 s.	1 minute	5 minutes or longer
Break flow					
Reactor pressure					
Steam flow from dome					
Feedwater flow					
ECC activation - note which ECC system is activated: ADS, RCIC, HPCF, LPCF and the respective flow.					
Coolant temp					
Fuel temp					
Reactor water level					
Reactor power					

- At the onset of the reactor vessel bottom break, the reactor water level drops, thus the steam volume in the dome increases. This in turn decreases the dome pressure. As a result, the void fraction in the core increases, leading to slight decrease in reactor power. That is why the yellow cursor on the screen moves downward.
- Recognizing that the reactor power is below the setpoint, the reactor power control system will withdraw the control rods in order to restore reactor power.
- At the same time, the drywell is pressurized very quickly by the LOCA break flow. In a short time, LOCA is detected, and reactor is scrammed by “high drywell pressure/LOCA detected” trip logic.
- When LOCA is detected, ECC is activated. FW pumps will also be tripped, when ECC is in service. At the same time, the main steam isolation valve will close, in order to isolate the containment.
- Due to the closing of the reactor isolation valve, the reactor pressure increases rapidly, resulting in “Reactor Press Hi” alarm. But due to the LOCA break, and the ECC core

injection's cooling effect, the reactor pressure quickly decreases and stabilizes, so the "Reactor Press Hi" alarm will disappear.

- As there is no steam flow going to the turbine, the turbine throttle pressure decreases rapidly, leading to turbine runback by the turbine control system. The result is turbine trip by low turbine forward power (alarm "Lo Turb Fwd Pwr")
- Go to BWR Plant Overview Screen. Re-initialize the simulator with the 100% IC.
- Repeat the malfunction event, and observe the evolution of all the trended variables.



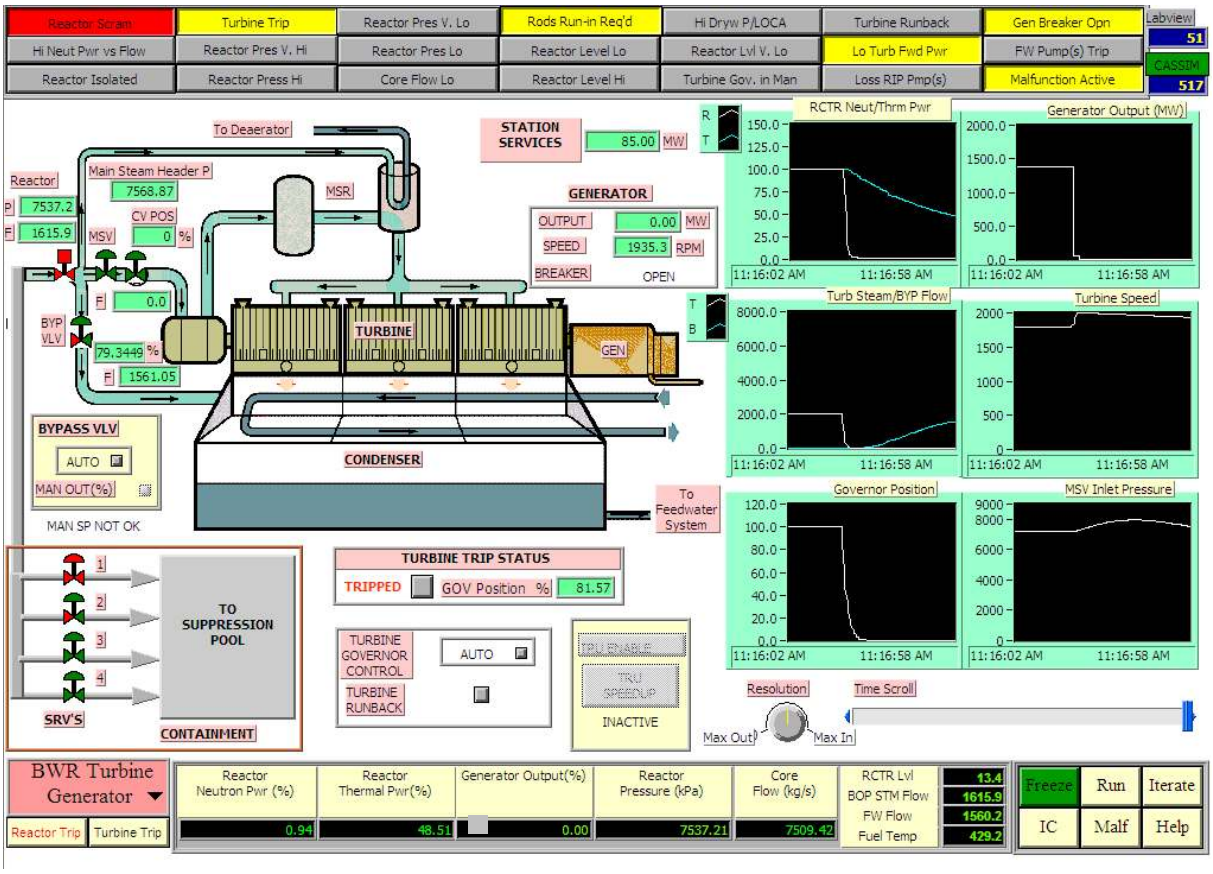


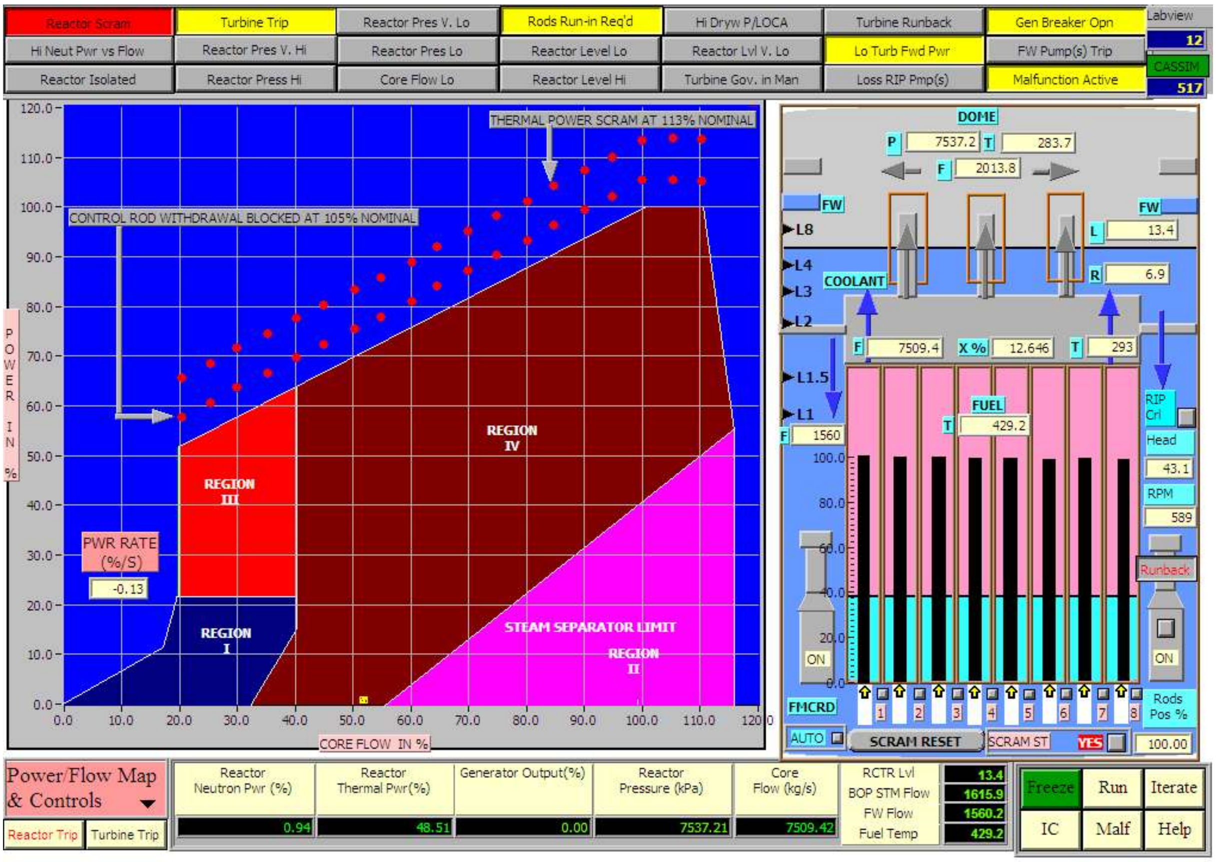
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
20
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
1647
Reactor Bottom Break Flow
872.14
THERMAL POWER SCRAM AT 113% NOMINAL
CONTROL ROD WITHDRAWAL BLOCKED AT 105% NOMINAL
POWER IN %
REGION IV
REGION III
PWR RATE (%/S)
-0.02
REGION I
STEAM SEPARATOR LIMIT
REGION II
CORE FLOW IN %
DOME
P 7718.6
T 292.5
F 346.5
FW
L8
L4
L3
L2
L1.5
L1
COOLANT
ECC FLOW
397.5
L 11.7
R 19.5
F 4822.6
X % 4.889
T 275
FUEL
T 309.8
F 7
RIP Crl
Head
19.8
RPM
377
Runback
ON
FMCRD
AUTO
SCRAM RESET
SCRAM ST
YES
Rods Pos %
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
0.50
Reactor Thermal Pwr(%)
11.71
Generator Output(%)
0.00
Reactor Pressure (kPa)
7718.58
Core Flow (kg/s)
4822.60
RCTR Lvl 11.7
BOP STM Flow 0.0
FW Flow 6.8
Fuel Temp 309.8
Freeze
Run
Iterate
IC
Malf
Help

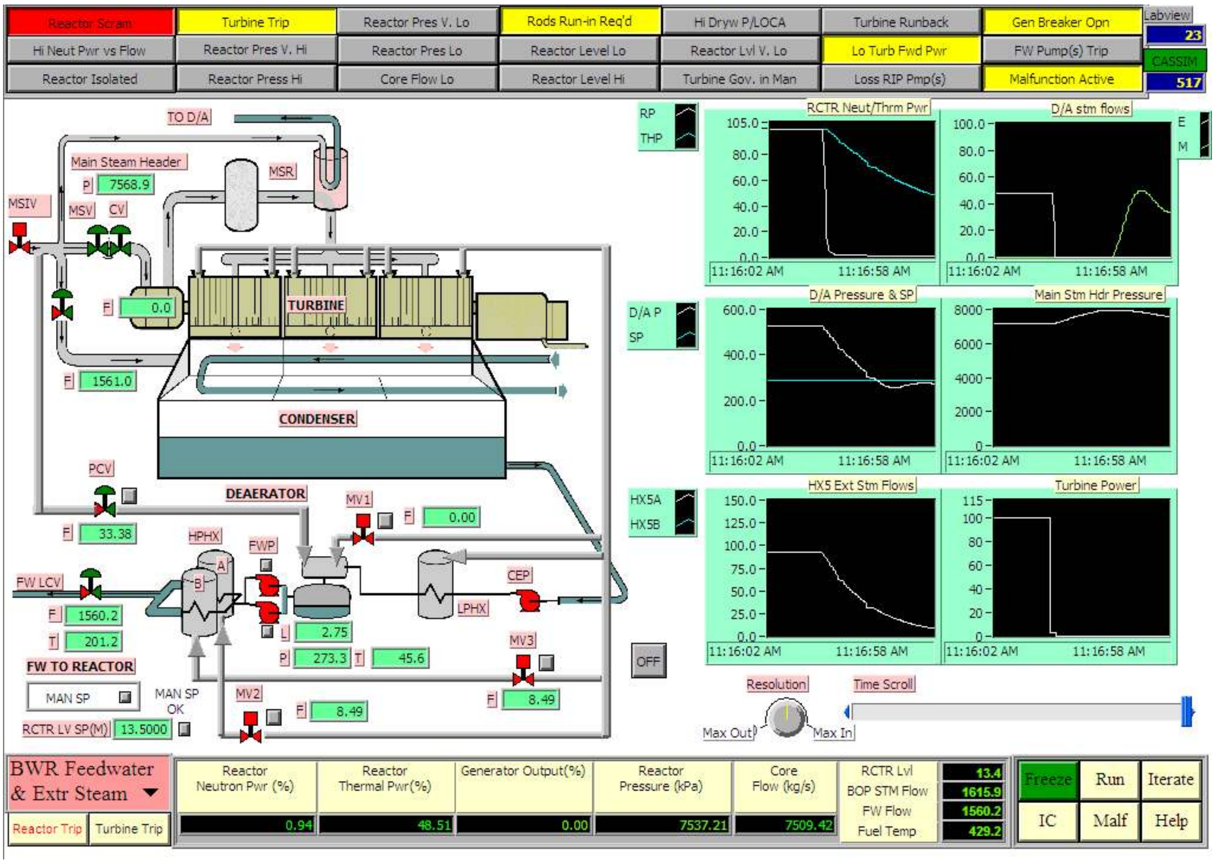
4.2.18 Load rejection

This malfunction causes sudden opening of the electrical switchyard breaker. This breaker connects the electrical power from the generator to the grid. The consequence is that the generator suddenly loses electrical load; this trips the generator, and produces a subsequent trip of the turbine.

- Go to the Power/Flow Map Screen. Load the 100% FP IC. Run the simulator. Insert the malfunction, and observe the movement of the yellow cursor.
- Turbine trip occurs very quickly, followed by reactor scram by "Turbine Power/Load Unbalance — Loss of Line" trip logic.
- This transient is very similar to turbine trip.




Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
12
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
517
THERMAL POWER SCRAM AT 113% NOMINAL
CONTROL ROD WITHDRAWAL BLOCKED AT 105% NOMINAL
POWER IN %
REGION III
REGION IV
REGION I
REGION II
STEAM SEPARATOR LIMIT
PWR RATE (%/S)
-0.13
CORE FLOW IN %
DOME
P 7537.2
T 283.7
F 2013.8
FW
L8
L4
L3
L2
L1.5
L1
L 13.4
R 6.9
COOLANT
F 7509.4
X % 12.646
T 293
FUEL
T 429.2
F 1560
RIP Crl
Head 43.1
RPM 589
Runback
ON
FMCRD
Rods Pos %
AUTO
SCRAM RESET
SCRAM ST
YES
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.94
Reactor Thermal Pwr(%) 48.51
Generator Output(%) 0.00
Reactor Pressure (kPa) 7537.21
Core Flow (kg/s) 7509.42
RCTR Lvl 13.4
BOP STM Flow 1615.9
FW Flow 1560.2
Fuel Temp 429.2
Freeze
Run
Iterate
IC
Malf
Help


Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
23
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
517
TO D/A
Main Steam Header
P 7568.9
MSR
MSIV
MSV
CV
F 0.0
TURBINE
F 1561.0
CONDENSER
PCV
F 33.38
DEAERATOR
MV1
F 0.00
HPHX
FWP
FW LCV
F 1560.2
T 201.2
L 2.75
P 273.3
T 45.6
CEP
LPHX
MV3
FW TO REACTOR
MAN SP
MAN SP OK
MV2
F 8.49
F 8.49
RCTR LV SP(M) 13.5000
RP
THP
RCTR Neut/Thrm Pwr
D/A stm flows
E
M
D/A P
SP
D/A Pressure & SP
Main Stm Hdr Pressure
HX5A
HX5B
HX5 Ext Stm Flows
Turbine Power
11:16:02 AM
11:16:58 AM
OFF
Resolution
Time Scroll
Max Out
Max In
BWR Feedwater & Extr Steam
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.94
Reactor Thermal Pwr(%) 48.51
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BOP STM Flow 1615.9
FW Flow 1560.2
Fuel Temp 429.2
Freeze
Run
Iterate
IC
Malf
Help

5. STEADY STATE MODEL

5.1 PURPOSE

The previous chapters describe the features of a BWR dynamic simulator. The simulator exercises and the malfunction responses in Chapter 3 are intended to provide students some insight and practice in BWR operational characteristics.

However, in order to have a deeper understanding of the BWR characteristics, it is necessary to understand the physical model behind this BWR simulator. The understanding of the physical model starts with an analysis of the steady state model, which is the focus of this chapter. What then follows is the understanding of the dynamic model, which is the focus of the next chapter.

The objective of the steady state model analysis is to understand the fundamental BWR operational question:

- In a BWR, why control rods are only used to raise reactor power up to 65% full power, and then further reactor power increase can be achieved by just increasing the core recirculation flow, without the movement of control rods?

5.2 BOILING WATER REACTOR MASS AND ENERGY BALANCE

A simplified BWR mass and energy flow diagram is shown in Figure 2.

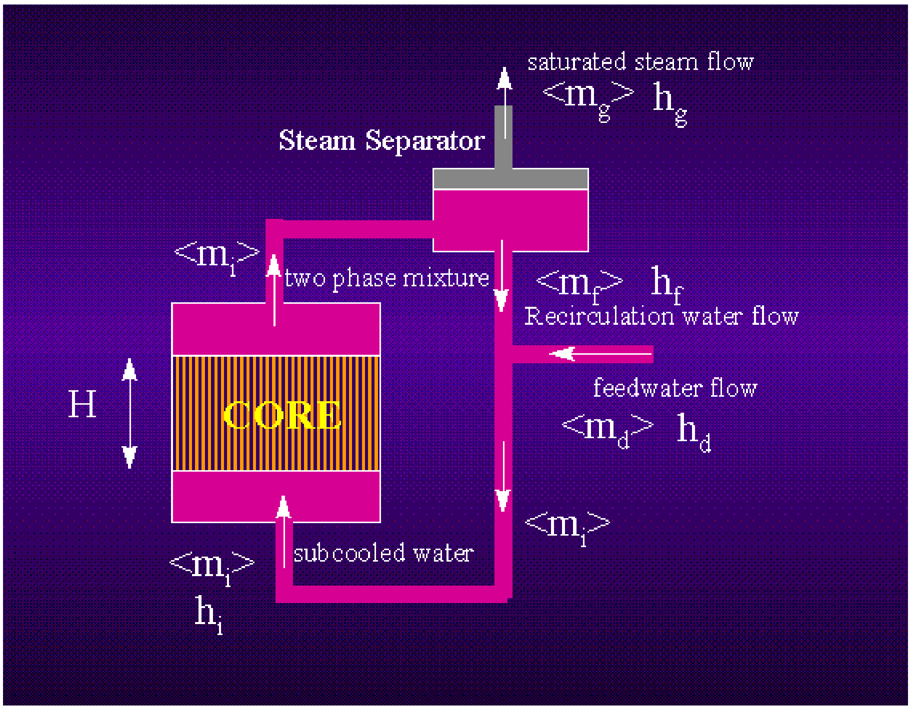


FIG. 2. Simplified BWR mass and energy flow diagram.

The following symbols are used to designate the variables:

- $<m>$ = mass flow rate
- h = specific enthalpy
- H = core boiling height
- $< m_d >$ = feedwater flow rate ; h_d = feedwater enthalpy
- $< m_i >$ = coolant flow rate at core inlet; h_i = coolant enthalpy at core inlet
- $< m_g >$ = saturated steam flow rate; h_g = saturated steam enthalpy
- $< m_f >$ = recirculated liquid flow rate; h_f = recirculated liquid enthalpy

(1) **Overall mass balances in reactor system:**

The steady state overall mass balances in the system are:

- Steam flow = feedwater flow

Hence $< m_g > \; = \; < m_d >$ .... (5.2-1)

- Subcooled water flow at reactor core inlet = feedwater flow + recirculation liquid flow.

Hence $< m_i > = < m_d > + < m_f >$ .... (5.2-2)

using equation (5.2-1),

$< m_i > = < m_f > + < m_g >$ .... (5.2-3)

(2) **Core exit quality:**

By definition, the average core exit quality is

$$X = \frac{< m_g >}{< m_g > + < m_f >} \quad \text{.......................(5.2-4)}$$

Using equations (5.2-1) and (5.2-3),

$$X = \frac{< m_d >}{< m_d > + < m_f >} = \frac{< m_d >}{< m_i >} = \frac{< m_g >}{< m_i >} = \frac{\text{Steam Flow Rate}}{\text{Core Flow Rate}} \quad \text{..............(5.2-5)}$$

(3) **Recirculation ratio:**

By definition, the recirculation ratio R is the recirculation flow rate/steam flow rate:

$$R = \frac{< m_f >}{< m_g >} \quad \text{....................................(5.2-6)}$$

Applying equation (5.2-5),

$$R = \frac{< m_f >}{< m_g >} = \frac{(1-X)}{X} \quad \text{.........................(5.2-7)}$$

(4) **Core flow rate:**

Using equation (5.2-3), and applying equation (5.2-7)

$$< m_i > = < m_f > + < m_g > = < m_f > + < m_f > * \frac{X}{(1-X)} = \frac{< m_f >}{(1-X)} \quad \text{.............(5.2-8)}$$

(5) **Enthalpy balance at reactor core inlet:**

$$< m_i > h_i = < m_f > h_f + < m_d > h_d \quad \text{..............(5.2-9)}$$

Dividing both sides by < m_i >,

$$h_i = \frac{< m_f >}{< m_i >} h_f + \frac{< m_d >}{< m_i >} h_d \quad ,$$

Applying equations (5.2-5) and (5.2-8),

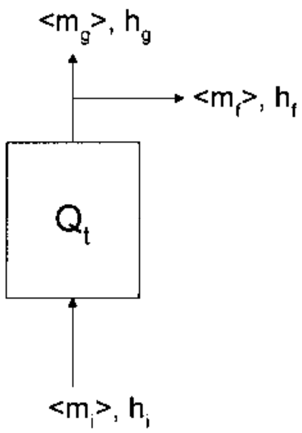
$$h_i = (1-X)\, h_f + X\, h_d \quad \text{........................(5.2-10)}$$

or solving for X,

$$X = \frac{(h_f - h_i)}{(h_f - h_d)} \quad \text{.......................... (5.2-11)}$$

(6) **Energy balance at the core**:

Consider Q_t, the energy transferred from the core to the coolant,



Applying an energy balance to the core,

$$< m_g > h_g + < m_f > h_f = Q_t + < m_i > h_i \quad \text{........................(5.2-12)}$$

Re-arranging,

$$Q_t = < m_g > h_g + < m_f > h_f - < m_i > h_i \quad \text{..........................(5.2-13)}$$

Or,

$$Q_t = < m_i > \{\frac{< m_g >}{< m_i >} h_g + \frac{< m_f >}{< m_i >} h_f - h_i\} \text{ ……………………….(5.2-14)}$$

Applying equations (5.2-5) and (5.2-8),

$$Q_t = < m_i > [X h_g + (1 - X)\, h_f - h_i] \text{ ……………………….(5.2-16)}$$

Starting from equation (5.2-12) again and applying (5.2-2),

$$Q_t = < m_g > h_g + < m_f > h_f - < m_f > h_f - < m_d > h_d$$

$$Q_t = < m_g > (h_g - h_d) \text{ ……………………………………….(5.2-17)}$$

5.3 BOILING WATER REACTOR SPREADSHEET MODEL

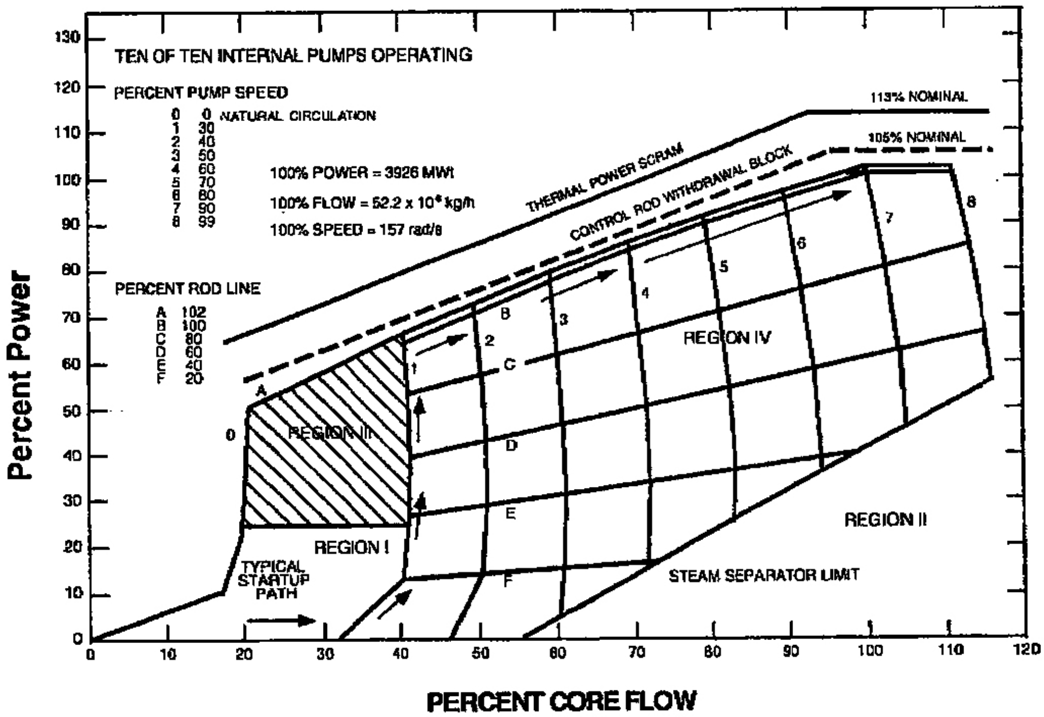
Using the basic equations for BWR steady state mass and energy balance derived above, it is intended in this section to build an EXCEL model for the BWR. In building this spreadsheet model, it is necessary to utilize the following Technical Data:

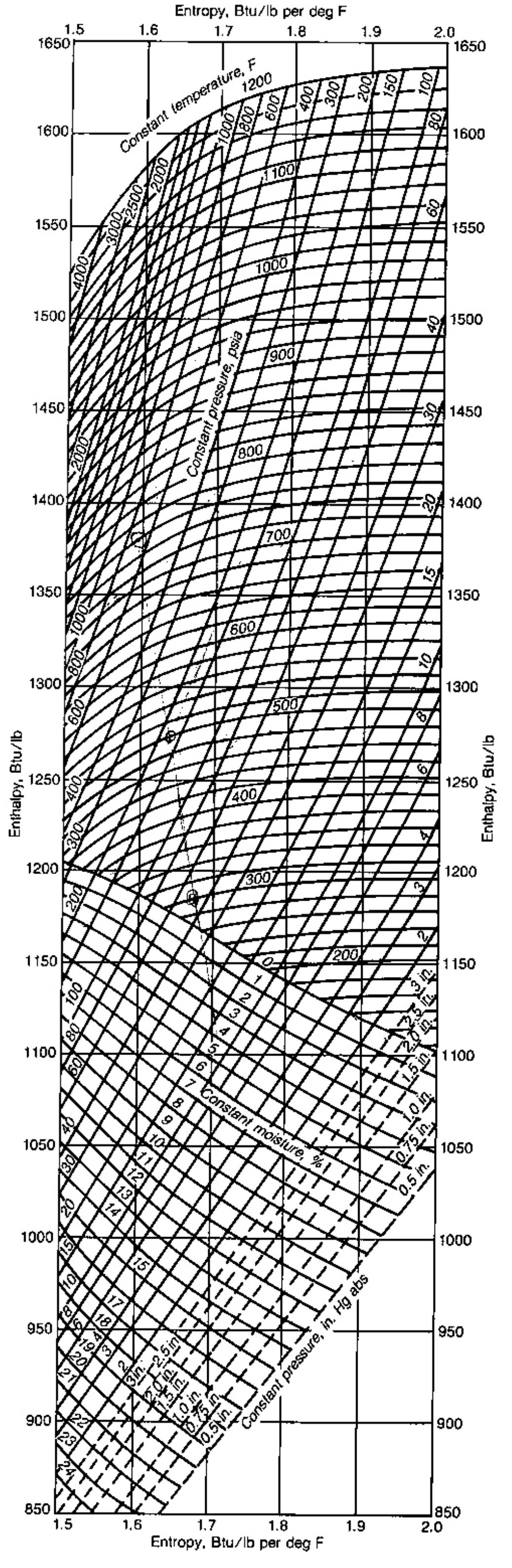
(1) BWR technical data sheet — in appendix

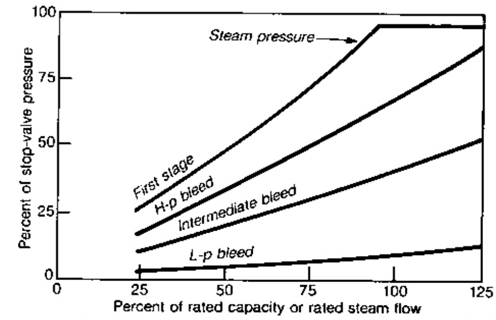
(2) Technical data for BWR power/flow map — see below

(3) Technical data for available energy for condensing turbine — see below

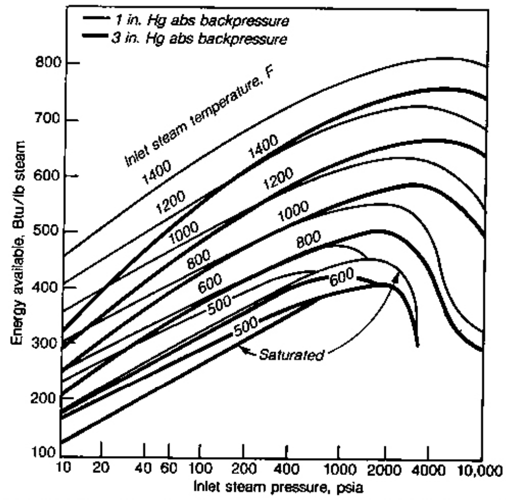
ABWR Power Flow Map



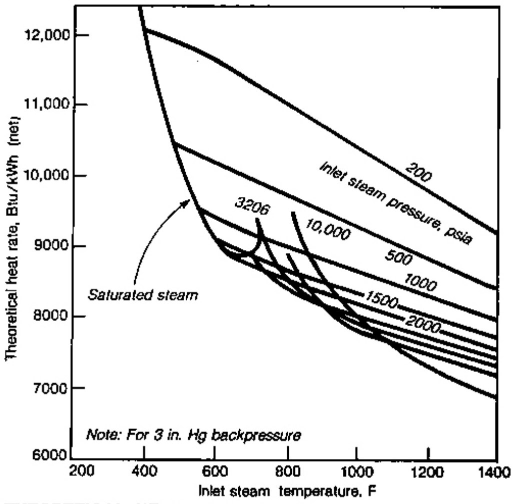




TURBINE CURVES, plotted when machine is new, are watched for deviations that indicate fouling, plugging, blade wear, etc



AVAILABLE ENERGY for condensing turbines depends on inlet steam pressure and temperature as well as backpressure.



THEORETICAL HEAT RATE of condensing turbines generally improves with rising throttle-steam pressure and temperature

◄MOLLIER CHART is handy for estimating turbine performance. Section shown is the most-used area of the complete chart

5.3.1 Procedures for spreadsheet model

(1) Create an EXCEL spreadsheet called BWR.
(2) Name Column A % FP, and put in % numbers: 100%, 90%, 80%, …, 20%, 10%, 5%, 3%, 0%.
(3) Name Column B MW — in the first cell, enter the value for MW (gross) at 100% FP. This number should be obtained from the BWR data sheet (Appendix)[2].
(4) Compute the rest of the cells in Column B using the % numbers in Column A.
(5) Name Column C KBTU/hr. To convert MW in Column B to KBTU/hr in Column C, the conversion factor is 3413. Compute all the cells in Column C using this conversion factor.
(6) Column D is the steam flow (kg/s). To compute steam flow kg/s at various % FP for the BWR plant, follow the steps below:
    (a) From the BWR data sheet (Appendix), get the turbine inlet pressure, the turbine backpressure (the same as condenser pressure), and the inlet steam temperature.[3]
    (b) Using the above turbine plant parameters (a), and from the technical data curves for available energy for the condensing turbine, obtain the available energy BTU per lb of steam for the turbine.[4]
    (c) Multiply this number by the efficiency of the turbine (assume 74%) to get the "actual" BTU/lb. [5]
    (d) Divide the Column C numbers (kBTU/hr) by the "actual" BTU/lb to get klb/hr.
    (e) Then multiply this number by 0.126 to convert klb/hr to kg/s.
    (f) Verify the result by checking the BWR plant data sheet — the 100 FP steam flow is 2122 kg/s. It may be necessary to adjust turbine efficiency slightly in order to match this number.
(7) Column E is core flow (kg/s). From the BWR data sheet (Appendix), find the 100% FP Core Flow, and enter it into the first cell.[6]
(8) Column F is core flow in %. From the BWR Power/Flow Map and following the typical plant startup curve, obtain the % core flow corresponding to the respective % full power of the plant, and enter the number into the corresponding cell in Column F. For example,
    100% FP → 100% core flow;
    90% FP → 80% core flow
    70% FP → 65% core flow
    …… etc.
(9) After all the % numbers have been entered for all cells in Column F, compute the core flow (kg/s) in all remaining cells in Column E.
(10) Column G is Quality X. Compute quality for all cells in Column G using the equations in Section 4.2.
(11) Column H is Recirculation Flow (kg/s) – calculate the Recirculation Flow for all cells in Column H.

---

[2] 100% FP MW is 1385 MW.
[3] Turbine Inlet Pressure is 6.8 MPa = 1000 PSIA; Condenser Pressure is 11.75 kPa = 3 inch Hg; Turbine Inlet Temperature is 284 ^{o}C = 543 ^{o}F.
[4] Available energy for condensing turbine is 380 BTU/lb.
[5] After taking into account the turbine efficiency, the "actual" available energy for condensing turbine is 281.1 BTU/lb.
[6] The 100% FP Core Flow is 14,502 kg/s.

(12) Plot a curve for the Quality X versus Power (%), and comment on the quality values as reactor power increases, with particular reference to "Void Feedback" on reactor reactivity".

(13) Try to answer the fundamental BWR operational question posed :

"In a BWR, why control rods are only used to raise reactor power up to 65% full power, and then further reactor power increase can be achieved by just increasing the core recirculation flow, without the movement of control rods? "

(14) In the steps below, it is intended to compute the reactor thermal power (MWth),and then compare the computed value versus that in the BWR data sheet, in order to verify the steady state model.

(a) Name Cell A23 "Reactor Pressure MPa". From the BWR data sheet (Appendix), find the reactor pressure, and enter the value into Cell B23.[7]

(b) Name Cell A24 "Saturated Coolant Enthalpy h_f (kJ/kg)".

(c) In Cell B24, enter the following equation to compute saturated coolant enthalpy (kJ/kg) as a function of reactor pressure (MPa)

= 373.7665*POWER(B23, 0.4235532) + 415

(d) Name Cell A25 "Saturated Vapor Enthalpy h_g (kJ/kg).

(e) In Cell B25, enter the following equation to compute the saturated vapor enthalpy (kJ/kg) as a function of reactor pressure (MPa):

= - 0.9219176 * POWER((B23 - 9), 2) - 16.38835*(B23 - 9) + 2742.03

(f) Name Column I "Reactor Thermal Power (MWt)".

(g) Given the feedwater temperature provided in the BWR data sheet (Appendix), the feedwater enthalpy at that temperature is 932.077 kJ/kg. Now using values in Cell B24 for h_f and Cell B25 for h_g, and other column's values, compute all the cell values for reactor thermal power in Column I.

(h) Verify the calculated 100% FP reactor thermal power versus the one provided in the BWR data sheet (Appendix).[8]

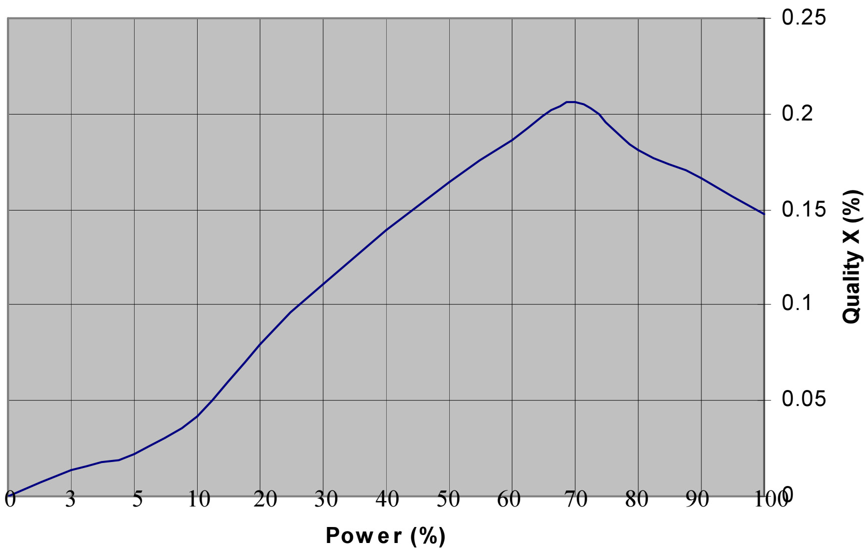
[7] Reactor operating pressure is 7.17 MPa.

[8] The 100% FP reactor thermal power per BWR data sheet is 3926 MWt.

5.3.2 Steady state model solutions

% FP	MW	kBTU/hr	Steam flow kg/s	Core flow kg/s	Core flow %	Quality	Recir Flow kg/s	Thermal Power MWth
100%	1385	4,727,005.00	2137.35	14502	100.00	0.1474	12364.7	3926.0
90%	1246.5	4,254,304.50	1923.61	11601.6	80.00	0.1658	9678.0	3533.4
80%	1108	3,781,604.00	1709.88	9426.3	65.00	0.1814	7716.4	3140.8
70%	969.5	3,308,903.50	1496.14	7251	50.00	0.2063	5754.9	2748.2
60%	831	2,836,203.00	1282.41	6888	47.50	0.1862	5605.6	2355.6
50%	692.5	2,363,502.50	1068.67	6526	45.00	0.1638	5457.3	1963.0
40%	554	1,890,802.00	854.94	6163	42.50	0.1387	5308.1	1570.4
30%	415.5	1,418,101.50	641.20	5800	39.99	0.1106	5158.8	1177.8
20%	277	945,401.00	427.47	5365	36.99	0.0797	4937.5	785.2
10%	138.5	472,700.50	213.73	5070	34.96	0.0422	4856.3	392.6
5%	69.25	236,350.25	106.87	4855	33.48	0.0220	4748.1	196.3
3%	41.55	141,810.15	64.12	4640.64	32.00	0.0138	4576.5	117.8
0%			0.00	2900.4	20.00	0.0000	2900.4	0.0

Reactor Press 7.17



6. DYNAMIC MODEL DESCRIPTION

The dynamic model for BWR Plant consists of a number of modules, which are interconnected simulation subsystem models. The modular structure of the dynamic model is best illustrated using a modeling diagram shown below:

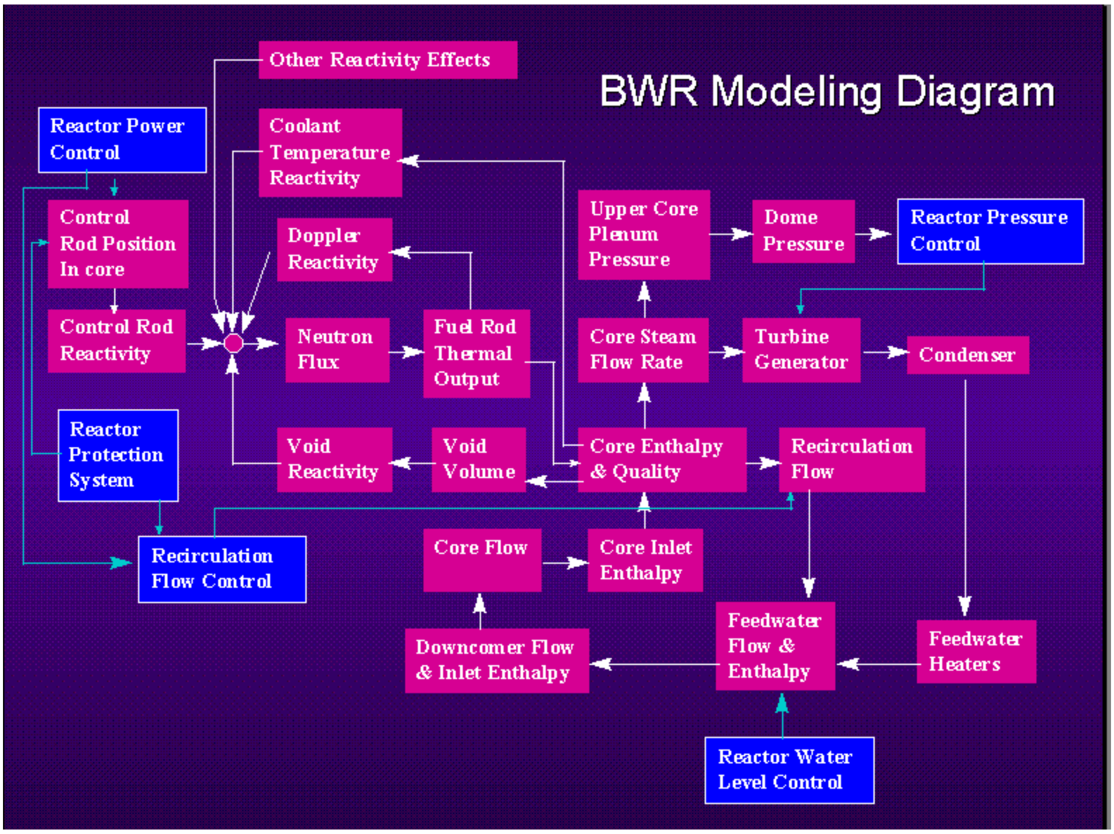


FIG. 3. Modeling diagram for the BWR illustrating the interconnected simulation subsystem models.

- The focal point of the model is the reactor block that simulates the reactor transient neutronics. It is represented as a small circle in the diagram.
- The inputs to the reactor block are reactivity values from various external blocks: (1) control rods reactivity (2) coolant temperature reactivity (3) Doppler reactivity (4) coolant void reactivity.
- The control rods reactivity block calculates their reactivity worth based on their positions in the core. The coolant temperature reactivity block calculates the reactivity feedback based on the average coolant temperature in the core. The Doppler reactivity block calculates the reactivity feedback based on average fuel temperature. The coolant void reactivity block calculates the reactivity feedback based on average void fraction in the core.
- The output of the reactor block is neutron flux. Based on the neutron flux as input, the fuel rod thermal output block computes the fuel cladding temperature, the fuel temperature and the heat transfer to coolant. The fuel temperature output in turn provides input to the Doppler reactivity block.
- The heat transfer to the coolant provides input to a core channel flow model that consists of a number of blocks to calculate coolant thermal-hydraulic behavior and void

fraction. The blocks included in the core channel flow model are: (1) core flow (2) core inlet enthalpy (3) core enthalpy and quality (4) core steam flow rate.

- The reactor vessel model consists of a number of blocks that calculate pressure, flow, and enthalpy at various points within the vessel. The blocks included in the reactor vessel model are: (1) upper core plenum pressure; (2) dome pressure; (3) recirculation flow; (4) feedwater flow and enthalpy; (5) downcomer flow and inlet enthalpy.
- The balance of plant (BOP) model consists of the following blocks: (1) turbine generator; (2) condenser; (3) feedwater heaters.
- The control system model includes the following controller blocks: (1) reactor power control; (2) reactor protection system; (3) recirculation flow control; (4) reactor water level control; (5) reactor pressure control.

6.1 REACTOR MODEL

The reactor model utilizes the point kinetics formulation to simulate neutronic behavior. The neutron power is based on six delayed neutron groups and the overall change in reactivity.

1. The total delayed neutron fraction is the summation of the neutron fractions of the 6 neutron groups

$$\beta = \sum_{i=1}^{6} \beta_i \quad \text{.......................……........................(6.1-1)}$$

β = total delayed neutron fraction

β_i = group i delayed neutron fraction (i =1,…6);

2. The delayed neutron group precursor concentration balance equations are:

$$\frac{dC_i}{dt} = \frac{\beta_i N_{FLUX}}{T_{NEUTRON}} - \lambda_i C_i \qquad \text{(i = 1,...6)} \quad \text{...........(6.1-2)}$$

C_i = concentrations of the six delayed neutron group precursors

λ_i = decay constants of the delayed neutron precursors

N_{FLUX} = total neutron flux in zone (norm)

$T_{NEUTRON}$ = mean neutron lifetime (s)

3. The rate of change of the neutron flux can be expressed as:

$$\frac{dN_{FLUX}}{dt} = \frac{(\Delta K - \beta)N_{FLUX}}{T_{NEUTRON}} + \sum_{i=1}^{6} \lambda_i C_i \quad \text{…..........(6.1-3)}$$

ΔK = overall neutron reactivity change

4. N_{FLUX} can be calculated by solving the above equations using a backward Euler expansion.

$$N_{FLUX} = \frac{N'_{FLUX} + \sum_{i=1}^{6} A_i}{1 - \Delta t\left(\frac{\Delta K - \beta}{T_{NEUTRON}} + \sum_{i=1}^{6} B_i\right)}$$

where ....(6.1-4)

$$A_i = \frac{\lambda_i C_i \Delta t}{1 + \lambda_i \Delta t}$$

$$B_i = \frac{\lambda_i \beta_i \Delta t}{(1 + \lambda_i \Delta t) T_{NEUTRON}}$$

N_{FLUX}' = total neutron zonal flux from previous iteration (norm)

5. The change in reactivity in the reactor is a function of control rod positions in the core, average concentration of xenon, average fuel temperature, average moderator temperature, and average void fraction in coolant. The overall reactivity change is expressed as:

$\Delta K = \Delta K_C + \Delta K_M + \Delta K_{XE} + \Delta K_{FUEL} + \Delta K_{VOID}$ ....(6.1-5)

ΔK = overall neutron reactivity change (k)

ΔK_C = neutron reactivity change due to control rods (k)

ΔK_M = overall neutron reactivity change due to moderator temperature (k)

ΔK_{XE} = overall neutron reactivity change due to xenon build-up (k)

ΔK_{VOID} = overall neutron reactivity change due to void fraction in core (k)

ΔK_{FUEL} = overall neutron reactivity change due to fuel temperature (k)

6. The reactivity change due to xenon poisoning is assumed to be:

$\Delta K_{XE} = 0.001*(27.93 - C_{XE})$ ....……………….(6.1-6)

with C_{XE} = xenon concentration

Equation 6.1-6 assumes that at 100 % FP, the steady state Xenon load is approximately 28 mk (typical equilibrium xenon load for a water cooled reactor).

The formation of xenon is assumed to be from the decay of iodine as well as from the initial fission products. The concentration of xenon can be found using the following rate equations.

$$\frac{dX}{dt} = \gamma_X \Sigma_f \phi + \lambda_I I - \lambda_X X - \sigma_X \phi X \quad \text{...................(6.1-7)}$$

$$\frac{dI}{dt} = \gamma_I \Sigma_f \phi - \lambda_I I \quad \text{...(6.1-8)}$$

where X, I = xenon, iodine concentrations, nuclei/cm^3

ϕ = neutron flux in neutrons/cm^2.s

Σ_f = macroscopic fission cross section

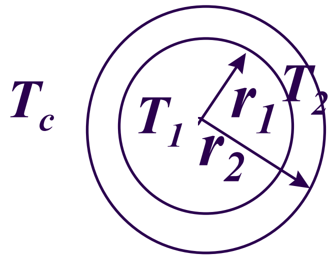
γ_X, γ_I = fractional yields of xenon and iodine

λ_X, λ_I = decay constants of xenon and iodine

σ_X = the microscopic capture cross-section of xenon for thermal neutrons

6.2 FUEL HEAT TRANSFER

A lumped parameter technique is used for calculating the heat transfer from the UO_2 fuel rods (cross-section with dimensions shown below):



Cross-section of a fuel pellet, enclosed by metalic clad. The reactor coolant gets heat transfer from the fuel clad.

The transient fuel "meat" temperature and the fuel clad temperature are given by:

$$C_1 \frac{dT_1}{dt} = \dot{Q}_n - \frac{T_1 - T_2}{R_1} \quad \ldots\ldots\ldots\ldots\ldots(6.2\text{-}1)$$

$$C_2 \frac{dT_2}{dt} = \frac{T_1 - T_2}{R_1} - \frac{T_2 - T_c}{R_2} \quad \ldots\ldots\ldots\ldots(6.2\text{-}2)$$

Where

$\dot{Q}_n$ = nuclear heating rate of the fuel rod

C_1 = thermal capacity for fuel pellet = $\pi\ r_1^2 c_{p1} \rho_1$

C_2 = thermal capacity for fuel clad = $2\pi r_2 (\Delta r) c_{p2} \rho_2$

R_1 = resistance of UO_2 and gap $= \dfrac{1}{4\pi\ k_1} + \dfrac{1}{2\pi\ r_1 h_g}$

k_1 = UO_2 thermal conductivity;

h_g = gap conductance

T_1 = average fuel pellet temperature

T_2 = average fuel clad temperature

T_c = average coolant temperature

R_2 = the resistance between the clad and coolant = $1/(2\pi r_2 h)$,

where h is the conductance between clad and coolant and r_2 is the outside radius of the clad.

6.3 DECAY HEAT MODEL

The buildup and decay of fission products in an operating BWR lead to a decay power source in the core, even after the chain reaction has been stopped, and the delayed neutrons have disappeared, as would be the case after a reactor scram. There is a large number of fission product isotopes which contribute to this decay heat source, and it is difficult to model individual decay sources. However, from measured data, empirical formulas can be constructed to fit the decay source following a reactor scram to a sum of contributions from the various sources as shown below.

The decay heat calculation for the reactor assumes that three separate decay product groups exist, each with a different decay time constant.

$$P = N_{FLUX} + \sum_{i=1}^{3} \lambda_i D_i \quad \text{.....................(6.3-1)}$$

$$\frac{dD_i}{dt} = \gamma_i N_{FLUX} - \lambda_i D_i \quad \text{...................(6.3-2)}$$

Where

P = total thermal power released from fuel (normalized)

N_{FLUX} = neutron flux power (normalized)

D_i = fission product concentration for decay group i

γ_i = fission product fraction for decay group i

λ_i = decay time constant for decay group i

The reactor power P is used by the "Fuel Heat Transfer to Coolant" module to calculate the coolant and fuel temperatures.

6.4 COOLANT HEAT TRANSFER

The BWR core is composed of parallel channels that enclose the fuel elements. These channels are connected to the lower plenum at the bottom of the core inlet, and to the upper plenum at the core exit. The flow enters the channel at the bottom through an inlet orifice. Additional flow enters the bypass region of the core through fuel casting holes.

Under steady state (normal operating) conditions, the core inlet flow is subcooled. As the flow in the channel passes by the fuel, it is heated until boiling occurs. Thus the channel flow can be considered as having three separate regions:

- Subcooled, non-boiling
- Nucleate boiling in subcooled liquid and
- Boiling in saturated liquid

With further heating, under transient conditions, “bulk” (transition) and film boiling may occur. The heat transfer characteristics for steady state and transient conditions have been well researched and documented as shown in the following diagram:

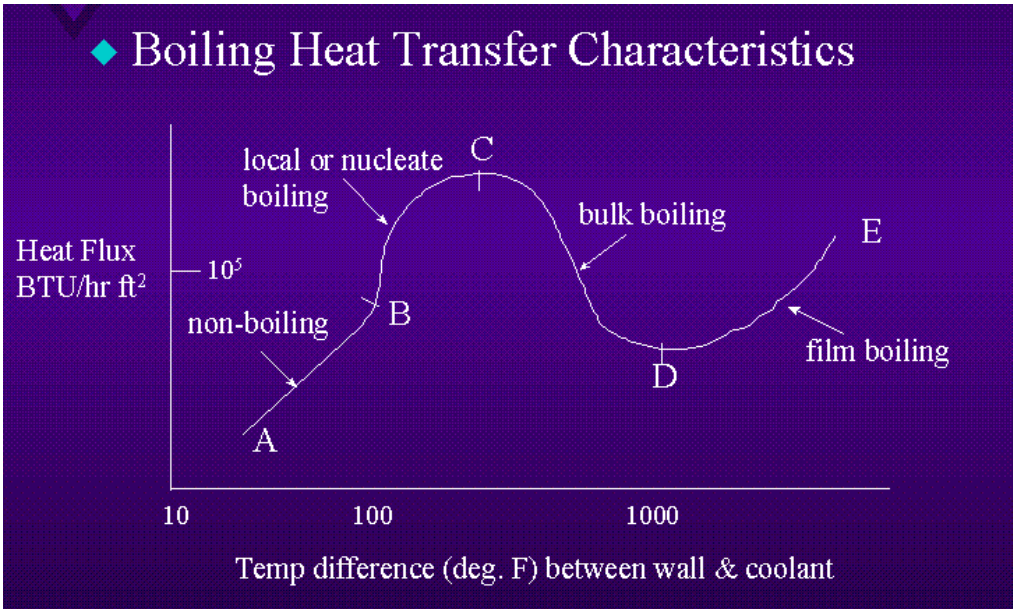


FIG. 4. Boiling heat transfer characteristics.

- A-B: non-boiling heat transfer by single-phase liquid convection.
- B-C: local or nucleate boiling. The heated surface temperature exceeds saturation temperature by a few degrees; bubbles are formed; there is large increase in heat flux due to the evaporation of water and mixing of the liquid by bubbles.
- C: dry out: water film at fuel rod surface disappears
- C-D: bulk (transition) boiling. The heated surface is blanketed by unstable, irregular film in violent motion. The heat flux decreases with surface temperature substantially.
- D-E: film boiling. At D, the vapor film becomes stable, and heat transfer improves, as the surface gets hotter. However, very high surface temperature is reached with high heat flux in this region, usually resulting in the destruction of the fuel and sheath. Dryout brings the operating point to region E.

For the BWR, the core for the most part operates in B-C nucleate boiling region, away from C. For calculating the heat transfer coefficient to be used in the heat transfer model, Thom's nucleate boiling heat transfer at pressures from 750 to 2000 psia is used:

$$(T_w - T_{sat}) = 0.7123 \frac{\sqrt{q''}}{e^{(\frac{P}{8690})}} \quad \text{……………………(6.4-1)}$$

where

T_w = fuel wall temperature (ºC)

T_{sat} = saturated coolant temperature (ºC)

q'' = heat flux (MW/m^2)

P = pressure (kPa)

Several axial zones of fuel along the channel can be considered. The average fuel rod energy equation is given by:

$$\rho_f V_f C_f \frac{dT_f}{dt} = P - UA(T_{Cladding} - T_c) \quad \text{…………………(6.4-2)}$$

where

ρ_f= volume average fuel rod density

V_f = fuel rod volume (in the zone considered)

C_f = average fuel rod specific heat capacity

T_f= average fuel rod temperature

T_c = average coolant temperature (in the zone considered)

P = reactor power

U = overall heat transfer coefficient. In the non-boiling region, the Dittus-Boelter correlation for forced convection is used; the heat transfer rate is proportional to the coolant flow to the power 0.8. In the boiling region, the heat transfer coefficient is derived from Thom's nucleate boiling correlation (equation 6.4-1).

A = overall heat transfer area for the fuel channel (in the zone considered)

An approximate average core coolant energy equation is given by:

$$\rho_c V_c \frac{dE_o}{dt} = W_i h_i - W_o h_o + UA(T_{Cladding} - T_c) - \rho_c V_c \frac{d(P_c V_c)}{dt} \quad \text{..............(6.4-3)}$$

where

ρ_c = volume average coolant density
V_c = coolant volume in the zone considered
P_c = pressure of coolant at outlet of core.
h_i = average coolant specific enthalpy at inlet of the core
E_o = internal energy of the coolant at outlet of the core
$T_{cladding}$ = average fuel cladding temperature
T_c = average coolant temperature
W_i = coolant mass flow rate at fuel channel inlet
W_o = coolant mass flow rate at fuel channel outlet

6.5 CORE HYDRAULICS AND HEAT TRANSFER

The core hydraulics for the BWR involves the solution of the mass, energy and momentum equations. Since these equations are coupled in a relatively weak fashion, it is possible to de-couple the mass and momentum equations from the energy equation as far as their simultaneous solution is concerned. This allows a much simpler solution of the equations in the core. For this purpose, the core is divided into flow regions or nodes axially, and pressures and flows are calculated along the reactor coolant flow regions through the core, as shown in the following flow network diagram. Calculation of energy transfer is handled separately and is described in the next section.

As the core has a non-boiling region, and a boiling region, appropriate hydraulic flow equations should be used for these regions. For the non-boiling region, incompressible flow equations will be used. For the boiling region, two-phase, variable-density flow equations will be used. The details are provided below.

Note: The two-phase flow modeling approach has been grossly simplified in this model and it is not suitable for design and safety analysis. However, the IAEA reviewers consider that the steady state and dynamic transient performance of this simulator with the model simplification still fulfills educational purposes.

6.5.1 Incompressible flow for non-boiling region

In the core's subcooled non-boiling region, the numerical solution technique used for solving the network pressures and flows employs the hydraulic equations for incompressible flow.

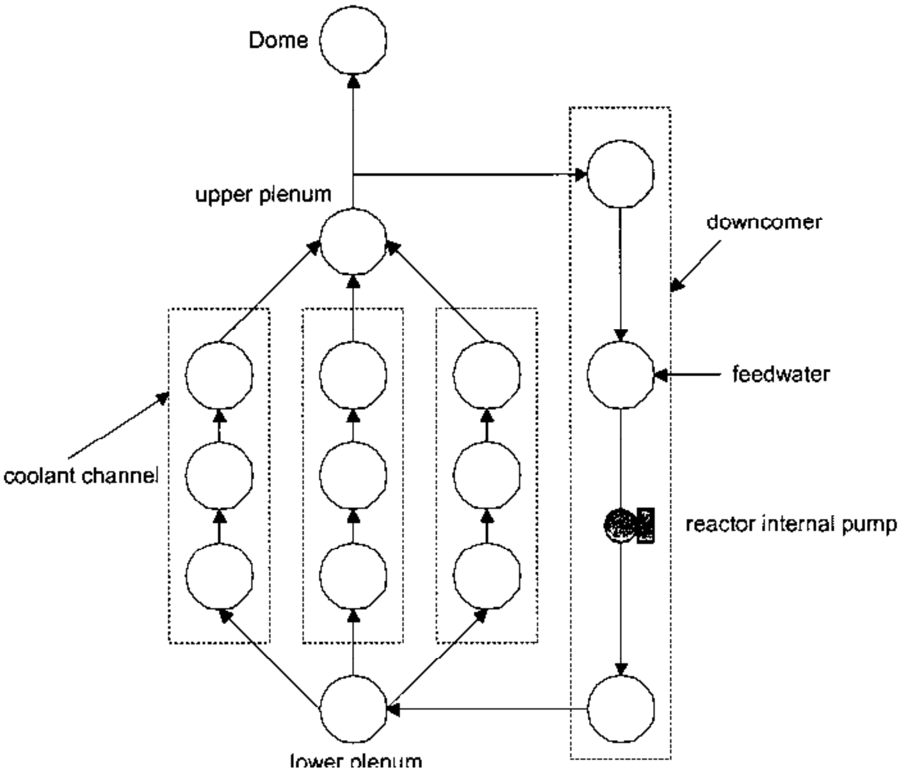


FIG. 5. Flow network diagram for core hydraulics.

For example, in the above flow network, the "circles" are pressure nodes, where pressures are calculated based on the coolant mass balance at the nodes. For example, the pressure at Node 1 is given by:

$$C_{N_1} \frac{dP_{N_1}}{dt} = W_{IN} - W_{OUT} \quad \ldots\ldots\ldots\ldots\ldots\ldots(6.5\text{-}1)$$

Where

C_{N1} = Node 1 Capacitance, which is a function of volume expansion due to vapor generation from boiling and flashing from the energy equation. But for this simplified model, the node capacitance is assumed constant.

P_{N1} = Node 1 Pressure

W_{IN} = total flows into the node 1

W_{OUT} = total flows out of the node 1

Similarly for all other nodes in this region.

The "arrow" paths joining adjacent nodes are called "links", where flows are calculated based on the square root of pressure difference between adjacent nodes, known as the momentum equation for incompressible flow. For example, the flow between node 1 and node 2 is given by:

$$W_{N_1N_2} = K_{N_1N_2}\sqrt{P_{N_1} + P_{DYH} - P_{N_2}} \quad \text{...........(6.5-2)}$$

Where

W_{N1N2} = flow from node 1 to node 2

K_{N1N2} = link flow conductance, which includes effects of valve C_v (if applicable). For accurate representation of two phase flow, the two phase multiplier should be included.

P_{N1} = Node 1 pressure

P_{N2} = Node 2 pressure

P_{DYH} = Pump dynamic head, if applicable

Similarly for all other links in this region.

By specifying the "nodes" and connecting them by "links" as in the above diagram, a nodal representation of the core hydraulic flow network problem can be defined. Then a matrix numerical method is employed to solve the system of node pressure equations (as in 6.5-1) and link momentum equations (as in 6.5-2) to obtain the pressures and flows. If the coolant heat transfer in the flow network results in fluid density changes, these changes will be taken into consideration by the link's conductance calculations (see above K_{N1N2} term).

6.5.2 Compressible flow for boiling region

In the core's boiling region, the following momentum equations are used:
For non-choked flow:

$$Flow = K_c.V_1.(P_{up}^2 - P_{down}^2)^{0.5} \quad \text{..........................(6.5-3)}$$

For choked flow:

$$Flow = K_c.V_1.P_{up} \quad \text{...................................... (6.5-4)}$$

Where K_c = link conductance

V_1 = valve port area in the link between the upstream and downstream nodes.

P_{up} = upstream pressure

P_{down} = downstream pressure

As noted in equation (6.5-3), the compressible flow equation is non-linear, thus making the analytical numerical solution complex and difficult to solve. Therefore, an approximation method technique is employed. It is called the "*variable conductance*" technique, in which the same set of *incompressible* flow equations (6.5-1, 6.5-2) could be used for a compressible flow system.

In this technique, when the flows are calculated using the incompressible flow equations (6.5-1, 6.5-2), they have to be substituted back to the compressible momentum equations (6.5-3, 6.5-4), using initial pressure conditions, to back calculate the link conductance K_c, hence the name "variable conductance".

Then in the next simulation iteration, because of the changes in link conductance obtained by the back-calculation, new values for flow will result from another iteration of the incompressible flow equations. Again the new flow values will give rise to new "link conductance" from back-calculation using the compressible flow equations. So after a few iterations in this manner, the flow obtained from incompressible flow equations, with the calculation of variable conductance in each iteration, will converge to the flow supposed to be computed from the compressible flow equations.

In other words, this technique "allows" the link momentum equation for "compressible" type flow systems to be approximated by the "incompressible" flow system momentum equation. But instead of a fixed conductance, the conductance is calculated in each iteration from the compressible momentum equation using back-substitution. The application of this technique allows the same numerical algorithm formulated for "incompressible" flow systems (Network Solver) to be adapted for use by "compressible" flow systems, simply with the additional computation of "variable link conductance".

6.5.3 Boiling boundary

As the reactor core conditions (e.g. pressure) change, the core boiling boundary will change.

Applying the following notations:

- H_o = non-boiling height;
- H_B = boiling height;
- H = total active height of core

The height ratio Ho/H is related to the ratio of sensible heat, q_s added per unit mass of incoming coolant (kJ/kg) to the total heat q_t added in the channel per unit mass of coolant channel (kJ/kg), assuming uniform heat addition:

$$\frac{q_s}{q_t} = \frac{H_o}{H} \quad \text{.......................... (6.5-5)}$$

The ratio q_s/q_t can be computed using enthalpies

$$\frac{q_s}{q_t} = \frac{h_f - h_i}{(h_f + X\,h_{fg}) - h_i} \quad \text{............(6.5-6)}$$

Where

h_f = saturated coolant enthalpy, kJ/kg

h_i = coolant enthalpy at inlet of channel kJ/kg

$h_{fg} = h_g - h_f$ = latent heat of vaporization kJ/kg

Thus using equations (6.5-5) and (6.5-6), the non-boiling height Ho and the boiling height $H_B = H - Ho$ can be calculated in terms of the respective enthalpies.

6.5.4 Summary of multi-nodal approach for simulating core hydraulics and heat transfer

The above-described approach in calculating the core hydraulics and heat transfer is to divide the coolant channels into a number of "control volumes" known as "nodes" connected by "links". Mass and momentum conservation equations are solved on a nodal basis, while the energy conservation equation is solved for each link. Incompressible and compressible hydraulic network solution techniques are used to simulate single phase and homogeneous two-phase flows. The calculation procedures are as follows:

- Each lumped channel is divided vertically into a number of zones, called nodes — for simplicity, assume lower, middle and upper zones. The nodalization can be as granular as the model requires.
- Each coolant channel zone is assumed to have its own coolant flow, its own lumped fuel element.
- The fuel heat transfer to coolant calculations (equation 6.2-1, 6.2-2; equations 6.4-1, 6.4-2, 6.4-3) start with the lower zones, with zone inlet temperatures derived from the core lower plenum temperatures; and with coolant flows derived from the hydraulic flow network computation at the lower plenum (equations 6.5-1, 6.5-2).
- After obtaining the lower zone coolant outlet temperatures and average fuel temperatures, the calculations proceed to the middle zones. It should be noted that a program check is performed in each zone to determine if the coolant outlet enthalpy exceeds saturated coolant enthalpy. If so, there is quality in the channel, and the "*variable link conductance*" approach (see above) is used to approximate compressible flow and pressure in the zone. (equation 6.5-3, 6.5-4).
- The core hydraulics and heat transfer for middle zones are computed in the same manner using parameters obtained from the lower zones, and then the calculations proceed to the upper zones.
- At the core exit upper plenum, the coolant temperatures from all the lumped channels are mixed together by the flow turbulence to determine the average mixing temperature at the upper plenum.

References:

(a) "Modular Modeling System: A Code for Dynamic Simulation of Fossil and Nuclear Power Plants. Overview and General Theory" EPRI CS/NP-2989, March 1983.

(b) "RETRAN-02, A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", Volume 1: Theory and Numerics, EPRI NP-1850-CCMA.

(c) Tong, L.S., "Boiling Heat Transfer and Two-Phase Flow", John Wiley & Sons, Inc. New York.

6.6 SATURATED ENTHALPY, SATURATED LIQUID DENSITY

Using the steam table correlation within the operating range for pressure, the following enthalpies can be obtained as a function of pressure P:

- The saturated liquid enthalpy :

  $h_f = f(P)$ ……….…………(6.6-1)

- The saturated steam enthalpy :

  $h_g = f(P)$ ……………….….(6.6-2)

- The latent heat of vaporization:

$$h_{fg} = h_g - h_f \quad \text{………………(6.6-3)}$$

- The saturated liquid density:

$$\rho_{sat} = f(P) \quad \text{…………………(6.6-4)}$$

6.7 CORE EXIT ENTHALPY, CORE QUALITY, VOID FRACTION

- The core exit enthalpy based on a quasi-steady-state approximation:

$$h_{core} = h_{dc} + \frac{Q_t}{W_{dc}} \quad \text{………………..(6.7-1)}$$

where

W_{dc} = coolant flow at downcomer

h_{dc} = enthalpy of liquid at downcomer

Q_t = heat transfer from core according to:

$$Q_t = W_{dc}.(h_f + X\,h_{fg} - h_{dc}) \quad \text{………(6.7-2)}$$

- The core exit quality:

$$X = \frac{h_{core} - h_f}{h_{fg}} \quad \text{………………(6.7-3)}$$

- The corresponding void fraction :

$$\alpha = \frac{1}{1 + (\frac{1-X}{X}).\psi} \quad \text{………………(6.7-4)}$$

where

X = core exit quality

$\psi = (\rho_g/\rho_f).S$; S = Slip Ratio ……(6.7-5)

ρ_g = saturated steam density

ρ_f = saturated liquid density

6.8 DOME MASS BALANCE AND ENERGY BALANCE

Mass balance at dome:

$$\frac{dV_d}{dt} = \frac{1}{\rho_f}\left[(1-X)W_r - W_{dc} + W_{fw}\right] \quad \text{………(6.8-1)}$$

Where

V_d = fluid volume in dome

ρ_f = saturated fluid density

X = core exit quality

W_r = core flow

W_{dc} = downcomer flow

W_{fw} = feedwater flow

Energy Balance at Dome:

$$\frac{dh_d}{dt} = \frac{1}{\rho_f V_d}\left[(1-X)W_r(h_f - h_d) + W_{fw}(h_{fw} - h_d)\right] \quad \text{.......(6.8-2)}$$

Where

h_d = enthalpy of saturated liquid in dome after mixing with feedwater

h_f = saturated liquid enthalpy

h_{fw} = feedwater enthalpy

V_d = liquid volume in dome

ρ_f = saturated liquid density

X = core exit quality

W_r = core flow

W_{fw} = feedwater flow

6.9 SATURATED STEAM DENSITY AND DOME PRESSURE

The saturated steam density:

$$\frac{d\rho_g}{dt} = \frac{X W_r - W_g + \rho_g \frac{dV_d}{dt}}{V_D - V_d + V_{SM} + V_r\alpha} \quad \text{........................(6.9-1)}$$

Where

V_d = liquid volume in dome

ρ_g = saturated steam density

X = core exit quality

Wr = core flow

V_D = volume of dome

V_{SM} = volume of steam main

Vr = coolant volume of core

α = void fraction

Dome pressure is computed from the saturated steam density,

$$P_D = f(\rho_g) \quad \text{........................(6.9-2)}$$

6.10 DRIVING PRESSURE IN BOILING CORE

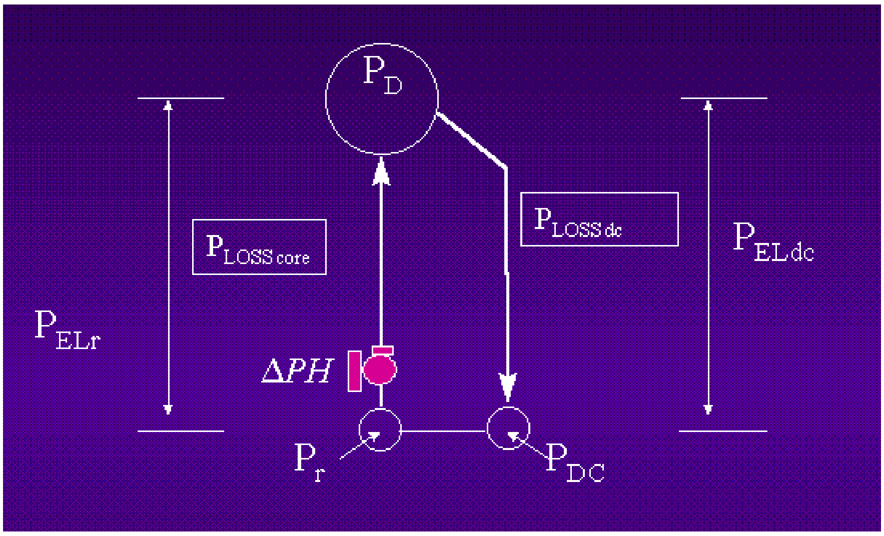


FIG. 6. Driving pressure in boiling core.

- Referring to the above diagram, the pressure at the downcomer (before pump suction) is given by:

$$P_{DC} = P_D + P_{ELdc} - P_{LOSSdc} \quad \text{..................................(6.10-1)}$$

where P_D = pressure of dome

P_{ELdc} = static pressure due to elevation of dome above downcomer

P_{LOSSdc} = pressure losses due to fluid flow from dome to downcomer

- The pressure at the dome is given by

$$P_D = P_{DC} - P_{LOSScore} - P_{ELr} + \Delta PH \quad \text{............................(6.10-2)}$$

where P_{DC} = pressure at downcomer (before pump suction)

$P_{LOSScore}$ = pressure loss due to core flow from lower plenum to dome

P_{ELr} = static pressure due to dome above lower plenum

ΔPH = recirculation pump head

or re-arranging (6.10-2),

$$P_{DC} = P_D + P_{LOSScore} + P_{ELr} - \Delta PH \quad \text{...................... (6.10-3)}$$

- Equating (6.10-1) and (6.10-3),

$$P_{LOSSdc} + P_{LOSScore} = (P_{ELdc} - P_{ELr}) + \Delta PH \quad \text{............... (6.10-4)}$$

- Note that the elevation pressure difference between the two flow paths:

  (a) from the downcomer to the dome, through the core, and
  (b) from the dome to downcomer, after mixing with feedwater

  is directly related to the mean fluid density difference between the downcomer and the core as given (in a consistent system of units) below:

  $(P_{ELdc} - P_{ELr}) = g\, Z_{EL}\, (\rho_{dc} - \rho_r)$ .... (6.10-5)

  Where $g = 9.81\ m/s^2$ is the acceleration of gravity

  Z_{EL} = elevation height of the dome from the bottom of the reactor pressure vessel (m)

  ρ_{dc} = mean fluid density at downcomer

  ρ_r = mean fluid density at core.

- It can be observed from (6.10-4) that:

  If the fluid density in the downcomer column is heavier than the fluid density in the core fluid column, so that the pressure difference in these two fluid columns "overcomes" the total pressure losses in these two columns, then natural circulation can be sustained without a recirculation pump. Otherwise forced circulation with the recirculation pump is required.

6.11 RECIRCULATION FLOW & PRESSURE LOSSES

Applying the momentum equation for incompressible flow (in a consistent system of units) to the downcomer:

$$\frac{dW_{dc}}{dt} = \frac{A_{dc}}{Z_{EL}}(\Delta PH + \Delta P_{EL} - P_{LOSSdc} - P_{LOSScore}) \text{ .... (6.11-1)}$$

Where

W_{dc} = recirulation flow

A_{dc} = cross-sectional area of the downcomer (m^2)

Z_{EL} = elevation height of dome from bottom of reactor vessel (m)

The pressure loss calculation is important for reactor design. It involves the following calculations:

- sum of the single phase and two-phase frictional pressure losses in the core and downcomer, according to the direction of the flow
- sum of the acceleration pressure losses
- sum of all pressure losses due to flow area contractions and expansions
- sum of gravitational pressure heads.

The topic of pressure loss calculation is commonly found in nuclear engineering text books (e.g. *Nuclear Heat Transport* - El. Wakil, ISBN 0-7002-2309-6), therefore details are not provided here.

6.12 COOLANT RECIRCULATION PUMPS

The main force driving the flow of the coolant in the primary loops of a BWR is the recirculation pump. The basic formulation for the pump model is as follows:

The torque balance (angular momentum) equation for the shaft and rotating assembly is:

$$\frac{2\pi}{60} I \frac{d\Omega}{dt} = T_M - T_h - T_f \text{ ....(6.12-1)}$$

where I = pump moment of inertia
Ω = pump speed (RPM)
T_M = motor torque
T_h = hydraulic torque
T_f = friction torque

The head and torque characteristics of a pump as a function of flow rate and rotor speed, are determined using the homologous theory as given by Stepanof [9]. In this theory, the pump parameters are represented by their normalized values. The shape of the homologous curves depends only on the rated speed of the pump. The homologous modeling relates normalized head, h, and normalized hydraulic torque, β, to normalized flow, ν, and speed, α, by tabulating:

$$\frac{h}{\nu^2}, \frac{\beta}{\nu^2} \quad vs \quad \frac{\alpha}{\nu} \quad for \quad 0 < \left|\frac{\alpha}{\nu}\right| < 1$$

$$\frac{h}{\alpha^2}, \frac{\beta}{\alpha^2} \quad vs \quad \frac{\nu}{\alpha} \quad for \quad 0 < \left|\frac{\nu}{\alpha}\right| < 1$$

These curves are fitted with a high-order polynomial function of *(α/ν)*, and *(ν/α)* respectively, and are used by the model to compute pump head and torque. The pump head so determined is used as an input to the primary hydraulic model (Chapter 6.5). The pump torque is used as input to the torque balance equation (Equation 6.12-1).

The pump curve should be degraded for two-phase flow conditions under accident conditions (LOCA).

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[9] Stepanoff, A.J., *Centrifugal and Axial Flow Pumps: Theory, Design and Application*, Wiley, NY, 1957.

6.13 FEEDWATER FLOW

The feedwater flow is determined from the control valve position, and the pressure difference between the upper plenum of the reactor pressure vessel and the feedwater/condensate system:

$$\frac{dw_{fw}}{dt} = (P_c + \Delta P_{fw} + \Delta P_c - P_D) - \rho_c g \Delta Z_c - \rho_{fw} g \Delta Z_{fw} - \rho_c g \Delta Z_{cc} - (K_c + K_{fw} + K_{fwv}) W_{fw}^2 \qquad (6.13\text{-}1)$$

Where

P_c = condenser pressure

ΔP_{fw} = feedwater pump head

ΔP_c = condensate pump head

P_D = Reactor upper plenum pressure

K_c = loss coefficients of condensate flow

K_{fw} = loss coefficients of feedwater flow

K_{fwv} = loss coefficients of feedwater control valves

ρ_c = density of condensate

ρ_{fw} = density of feedwater

ΔZ_c = elevation head of feedwater heater above condensate heater

ΔZ_{fw} = elevation head of steam generator above feedwater heaters

ΔZ_{cc} = elevation head of condenser

The feedwater enthalpy is obtained from the time lag between the feedwater heater and steam generator

$$\frac{dh_{fw}}{dt} = \frac{h_{fwh} - h_{fw}}{\tau} \qquad \text{............(6.13-2)}$$

Where

h_{fw} = feedwater enthalpy at steam generator

h_{fwh} = feedwater enthalpy at feedwater heater, which is obtained from the heat balance between extraction steam from turbine for feedwater heating, and the feedwater.

6.14 MAIN STEAM SYSTEM

The main steam system model includes the main steam piping from the steam drum of the steam generator, the main steam isolation valve (MSV), the turbine stop valves, the turbine control valves and the condenser steam dump valves.

The thermodynamic state of the main steam system is governed by conservation of energy and mass,

$$\frac{dM_h}{dt} = W_{DOME} - (W_T + W_D + W_B) \quad \text{..........................(6.14-1)}$$

$$\frac{dU_h}{dt} = W_{DOME} h_{DOME} - (W_T + W_D + W_B) h_h \quad \text{.....................(6.14-2)}$$

Where

M_h = total steam vapor mass in the system

W_{DOME} = steam flows from reactor steam dome to steam header

W_T = turbine control valve flow rate

W_D = steam dump valve flow rate

W_B = steam line break flow rate

The specific volume and specific internal energy are given by:

$$\nu_h = \frac{V_h}{M_h} \quad \text{..............(6.14-3)}$$

$$u_h = \frac{U_h}{M_h} \quad \text{.............(6.14-4)}$$

The main steam pressure is determined from the equation of state (i.e. steam table look-up) :

$$P = f(\nu_h,\ u_h) \quad \text{.......(6.14-5)}$$

The flow between the reactor dome and the main steam system has the following quasi-steady-state approximation:

$$P_{DOME} - P_h = K_V \frac{1}{2} \frac{W|W|}{\rho_h A_V^2} + K_{NZ} \frac{1}{2} \frac{W|W|}{\rho_h A_{NZ}^2} \quad \text{..............(6.14-6)}$$

Where

P_{DOME} = reactor steam dome pressure

P_h = main steam line pressure

K_V = main steam isolating valve loss coefficient

K_{NZ} = flow restrictor loss coefficients

W = steam flow rate

A_V = total isolation valve flow area

A_{NZ} = flow restriction throat area

ρ_h = steam density

The steam flow rate determined by equation (6.14-6) should not exceed choke flow conditions. Steam flow rate through the turbine valves and steam dump valves and the steam line break flow, are all assumed to be choked flow.

6.15 CONTROL AND PROTECTION SYSTEMS

The control systems available in this simulator include the systems described in section 3.2 "BWR Control Loops". In this section, brief model descriptions are provided for the following systems:

(1) Control rods control
(2) Recirculation flow control
(3) Reactor pressure control
(4) Reactor water level control
(5) Turbine power control
(6) Turbine steam bypass system
(7) Protection system

6.15.1 Control rods control system

The control rods control system and the reactor power control system are illustrated in the following simplified block diagram.

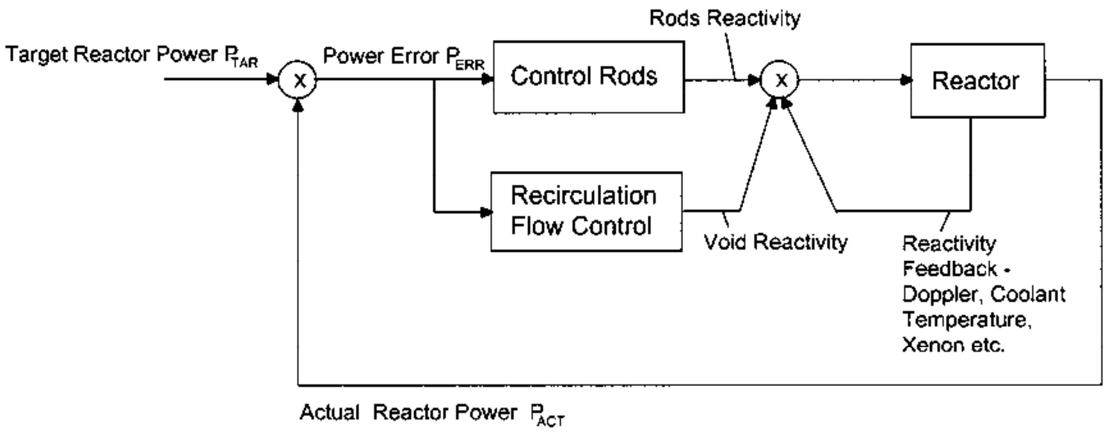

FIG. 7. Simplified block diagram for reactor power control.

- The target reactor power P_{TAR} selected by operator input is compared with the current reactor power, obtained by flux detectors and core temperature measurements. The power error signal, $P_{ERR} = P_{ACT} - P_{TAR}$ is sent to the control rods controller, and the recirculation flow controller (see Figure 8).
- In response to the power error signal, the rod drive system will adjust the position of the rods inside the core according to the Power/Flow Map (see Section 3.3). The simplified explanation of the rods control logic is:

  (a) For reactor power < 65%,

  - If $P_{ERR} < 0$, move the rods "out" to increase reactor power.
  - If $P_{ERR} > 0$, move the rods "in" to decrease reactor power.
  - If $P_{ERR} = 0$, no movement of control rods.

(b) For reactor power ≥ 65%,

- Recirculation flow controller is in control of reactor power (see below).
- Control rods will be moved "in" or "out" automatically to assist the recirculation flow controller in controlling reactor power, ONLY IF the absolute value of power error, ABS(P_{ERR}), is greater than a certain deadband. Under normal circumstances, ABS(P_{ERR}) will not exceed this deadband, because the recirculation flow controller is controlling reactor power within this deadband.

6.15.2 Recirculation flow control

- In response to power error signal, the recirculation flow control derives a "core flow setpoint" according to the Power/Flow Map (see Section 3.3).
- The core flow controller compares this setpoint versus the current core flow, and generates a control signal for the speed controller for the reactor internal pumps (RIP).
- The speed controller in turn will change the frequency of the induction motor that drives the pump to the speed required to achieve the core flow setpoint.
- Different pump speed will give rise to different pump dynamic head in the core circulation flow path, resulting in different core flow. The changes in core flow will alter the void density of the two-phase water in the core, a moderating material used to slow the fission neutrons. The end result is a change in reactivity due to changes in void density.

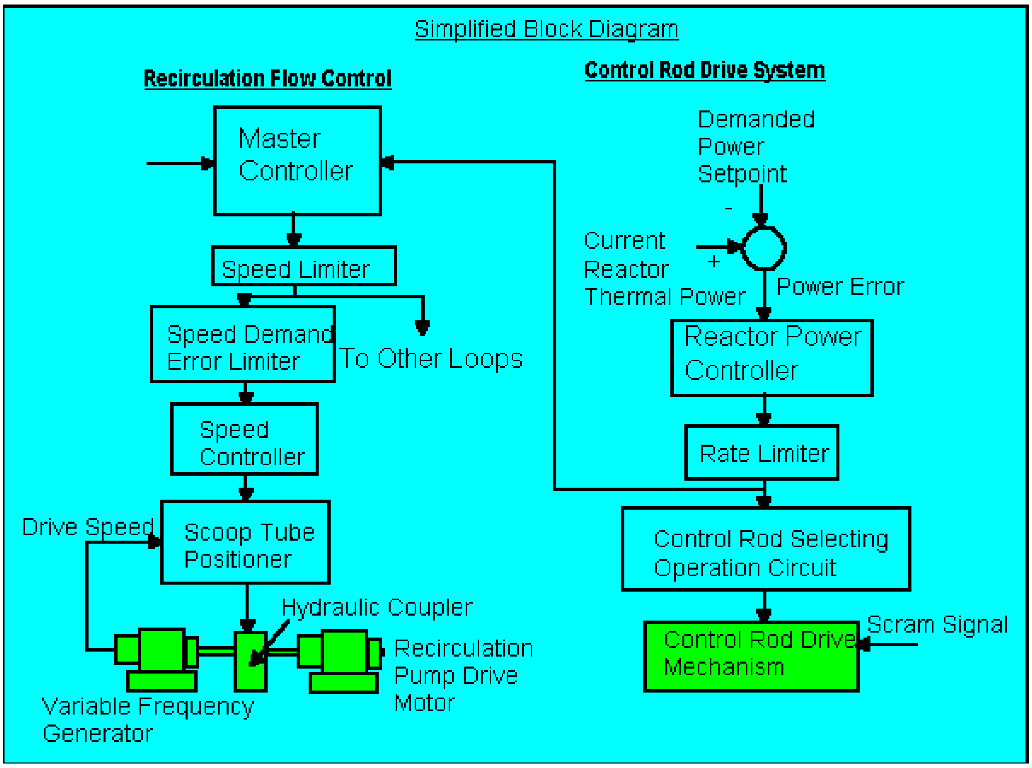

FIG. 8. Simplified control block diagram for control rod drive system and recirculation flow control.

6.15.3 Reactor pressure control system

The reactor pressure is automatically controlled to be constant. See detailed description in section 3.2 " BWR Control Loops".

For that purpose, a reactor pressure controller (RPC) is provided and is used to regulate the turbine inlet steam pressure by opening and closing the turbine governor control valve and the turbine bypass (or called steam bypass) valve.

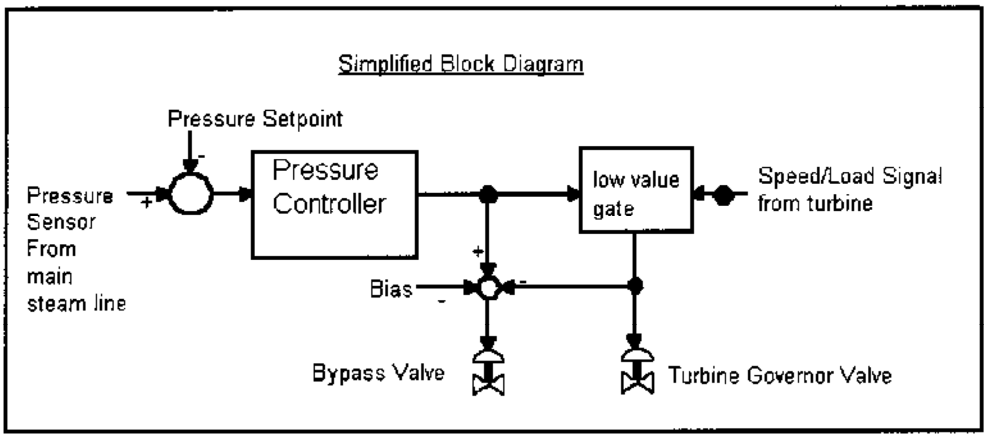

FIG. 9. Simplified reactor pressure controller block diagram.

Currently, the steam generator pressure setpoint is set at plant design pressure of 7170 kPa.

6.15.4 Reactor water level control system

Reactor water level control is achieved through the use of the three-element controller. The level controller is a PI reset controller adjusted to provide mostly integrating action and very little proportional signal to trim the feedwater flow. This controller has the following equation formulation:

$$M_L = K_{CL} [e_L + (1/\tau) \int e_L \, dt] \quad \text{.............(6.15.4-1)}$$

Where

M_L = reactor level controller signal to control valve

K_{CL} = proportional gain

e_L = reactor level error

τ = reset time constant

Feedwater flow/steam flow controller is also a PI controller adjusted to provide mostly proportional action.

$$M_{FS} = K_{CF} [e_{FS} + (1/\tau) \int e_{FS} \, dt] \quad \text{.............(6.15.4-2)}$$

Where

M_{FS} = reactor flow controller signal to control valve

K_{CF} = proportional gain

e_{FS} = flow error = steam flow - feedwater flow

τ = reset time constant

After comparing steam flow with feedwater flow and correcting for level, the three element controller generates a total control signal $M = M_L + M_{FS}$ to manipulate feedwater control valve position, which will eventually provide the adjusted feedwater flow rate to the steam generators.

6.15.5 Turbine power control system

- The turbine power control system involves a turbine governor control system which will regulate the steam flow through the turbine to meet target load by controlling the opening of the turbine governor valve.
- Under normal operation, the reactor pressure control (RPC) keeps the inlet pressure of the turbine constant, by adjusting the opening of the "turbine speeder" gear which controls the opening of the governor valve.
- Should the generator speed increase due to sudden load rejection of the generator, the speed control unit of the turbine governor system will take control over the reactor pressure control (RPC) and will close the turbine governor valve. Similar override situations apply for abnormal conditions in turbine such as turbine run-back, and turbine trip.

6.15.6 Turbine steam bypass control system

As described in (6.15.3), reactor pressure is maintained at an equilibrium constant value determined by the heat balance between the input to the reactor core, and the turbine steam consumption. In the event of a sudden turbine load reduction, such as load rejection, or turbine trip, an automatic steam bypass system is provided to dump the steam to the condenser, if the reactor pressure exceeds a predetermined setpoint. See more details in Section 3.2.

6.15.7 Protection system

(1) Reactor scrams:

- High neutron flux/low core flow
  If at any time, the current reactor power exceeds 113% of the power designed for the current flow rate in accordance with the Power/Flow Map, the reactor will be scrammed.
- High drywell pressure/LOCA detected
- Reactor water level low
  Low level scram SP < 11 m (normal level = 13.4 m)
- Reactor pressure high—
  Scram SP > 7870 kPa
- Reactor water level high
  Scram SP > 14.5 m (normal level = 13.4 m)
- Main steam isolation valve closed/reactor isolated

- Main steam line radioactivity high
- Turbine power/load unbalance — loss of line
- Earth acceleration high
- Manual scram

(2) Control rods “Blocked” — as mentioned in Section 3.3, if at any time the current power exceeds 105% of the power designed for the current flow rate (in accordance with the maximum power-flow line in the Power/Flow Map), the control rods withdrawal will be “blocked” until the power drops to 5% less than the current value.

(3) Control rods “Run-in” — as mentioned in Section 3.3, if at any time the current power exceeds 110% of the power designed for the current flow rate (in accordance with the maximum power-flow line in the Power/Flow Map), the control rods will be inserted into the core to reduce power quickly, and the “Control Rods Run-in” will be stopped until the power has been reduced to 10% less than the current value.

(4) Emergency core cooling injection — as mentioned in Section 3.1, in the unlikely event of major accidents inside the drywell, such as the feedwater line break, steam line break, and reactor bottom break, these breaks will cause high pressure in drywell, which in turn will trigger the LOCA (loss of coolant) signal. As a result, ECC will be started, and reactor will be scrammed, and isolated.

6.15.8 Automatic responses to design basis events accidents

The safety systems for BWR involve the following layers of defense against design basis events accidents:

(1) Reactor protection — control rods blocked; control rods run-in, boron injection, and reactor scram.
(2) Containment isolation — in the event of steam line break, feedwater line break, LOCA
(3) ECCS actuation — core spray for decay heat removal.
(4) Safety relief valve opening to suppression pool — to crash cool and relieve high pressure in reactor pressure vessel.
(5) Suppression pool cooling – to provide long term cooling of the isolated containment in case of LOCA.

6.16 Containment Model

(A) Dalton Model for Suppression Pool

The steam from a pipe break in the drywell would enter the wetwell through the vents where it would condense in the suppression pool. This mechanism is designed to reduce the pressure in the containment structure such that the design pressures are not exceeded. The water in the suppression pool is then pumped back to the reactor vessel with the Emergency Core Cooling System (ECCS).

The mass flow rate of steam from the RPV to the drywell through the steam line break is driven by the pressure difference between the reactor vessel and the drywell. Likewise the mass flow of steam entering the wetwell from the drywell is also proportional to the pressure difference between the drywell and wetwell.

The model described below presents a simple model of the Wetwell Suppression Pool. The Dalton model is used in conjunction with the conservation equations of the condensable and non-condensable phases to derive an explicit relation for the rate of change of pressure in terms of the known quantities. Back substitution is used to determine the rate of enthalpy of each phase. Subsequent integration of the finite difference equations results in the determination of pressure, enthalpy, and the phasic mass in each time step.

In this model representing the Wetwell of the ABWR, individual control volumes representing steam and non-condensable gas in the air-space region must be combined to represent steam and gas at thermal equilibrium. The steam and gas masses can be calculated by their individual continuity equations. However, only one energy equation can be written for the combined air-space control volume.

The Nomenclature used in the equations in this section is:

α: mass flow rates into or out of a region
β: energy flow rates into or out of a region.
ρ: density
ν: kinematic viscosity
μ: dynamic viscosity
g: gravitational acceleration
h: specific enthalpy
j: mass flux
L: distance for mass diffusion from pool surface to bulk
m: mass
p: pressure
R: Universal gas constant
t: time
T: temperature
v: specific volume
V: volume of the region

(a) Continuity equations for water, steam and gas in the Wetwell:

$$\frac{dm_i}{dt} = \alpha_i. \quad where \quad i = 1,2,3 \quad ....(1,2,3)$$

Subscript 1 is assigned to the water in the pool region; 2 to steam region; 3 to non-condensable gas region.

(a.1) For the pool water region, the source term α_1 is the rate of condensation of steam into the pool.

$$\alpha_1 = f_1 .* W_{out-DW} \ldots\ldots\ldots\ldots\ldots (a.1)$$

where

f_1 = steam fraction of the gas+air+steam mixture going from Drywell to Wetwell.

$W_{out\text{-}DW}$ = total flow rate of air + non condensable gas + steam to Wetwell from Drywell.

(a.2) For the steam region, assume there is a complete condensation of steam inflow into the pool. Given the uniform temperature distribution, mass transfer to (evaporation) or from (condensation) the air-space from the pool takes place due to the existing gradient between water molecules near the water surface and that in the bulk of the air-space. Such mass transfer takes place by diffusion, governed by humidity. With this assumption, the pool surface heat transfer is calculated considering natural convection turbulent flow from the horizontal surface. The Nusselt number for mass transfer for a heated horizontal plate facing upward is given as:

$$Nu_{L.m} = 0.12(\mathrm{Gr_L Sc})^{1/3} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (a.2.1)$$

Where

Sc = Schmidt number = ratio of kinematic viscosity to the diffusion coefficient

$$\mathrm{Gr}_L = \frac{g(\frac{\Delta\rho}{\rho_{avg}})}{v_{kin}} L^3 \ldots\ldots\ldots\ldots\ldots\ldots (a.2.2)$$

where
g = gravitational acceleration
$\Delta\rho$ = difference between mixture density at the surface and that at the bulk of the pool.

ρ_{avg} = average of the mixture density at the surface of the pool and the density at the bulk of the pool.
L = diffusion length from the pool.
v_{kin} = Kinematic viscosity

The mass transfer at the surface of the pool, α_2, is given by:

$$\alpha_2 = \frac{\mathrm{Nu}_{L,m}}{\mathrm{Sc}} \frac{\mu}{L}(Y_s - Y_b) \ldots\ldots\ldots\ldots.(a.2.3)$$

where

μ = dynamic viscosity
Y_s = ratio of steam density to the mixture of steam and gas density at surface of pool
Y_b = ratio of steam density to the mixture of steam and gas density at the bulk of atmosphere.

(a.3) For the gas region, the source term α_3 is given by the following equation after the bubble grows to a size that it can float to the surface of the pool

$$\alpha_3 = (1 - f_1) W_{out-ww} \ldots\ldots\ldots\ldots\ldots\ldots\ldots.(a.3)$$

(b) Energy equation for the pool:

$$\frac{dm_1}{dt} * h_1 + m_1 * \frac{dh_1}{dt} = \beta_1 + V_1 * \frac{dp_1}{dt} \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots(4)$$

where

$$\beta_1 = \alpha_1 * h_{vap}$$

(c) Energy equation for the atmosphere in the gas chamber above the pool:

$$\left(\frac{dm_2}{dt} h_2 + m_2 \frac{dh_2}{dt}\right) + \left(\frac{dm_3}{dt} h_3 + m_3 \frac{dh_3}{dt}\right) = \beta_{23} + V_2 \frac{dp_1}{dt} \ldots\ldots\ldots\ldots\ldots(5)$$

where

$$\beta_{23} = \alpha_2 h_{vap} + (1 - f_1) W_{out-ww} C_v T_1$$

The combined subscript of 23 is assigned to the steam and non-condensable gas mixture in the atmosphere above the pool.

(d) The rate of volume change constraints for water and steam:

$$\frac{dV_1}{dt} + \frac{dV_2}{dt} = 0 \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots(6)$$

Using the specific volume v and functional differentials of v with respect to h, and p,

$$\frac{dm_1}{dt} v_1 + m_1\left(\frac{\partial v_1}{\partial h_1}\frac{dh_1}{dt} + \frac{\partial v_1}{\partial p_1}\frac{dp_1}{dt}\right) + \frac{dm_2}{dt} v_2 + m_2\left(\frac{\partial v_2}{\partial h_2}\frac{dh_2}{dt} + \frac{\partial v_2}{\partial p_2}\frac{dp_2}{dt}\right) = 0 \ldots\ldots(7)$$

(e) The Dalton relation regarding sharing of the same atmospheric (air-space) volume by steam and non-condensable gas:

$$\frac{dV_2}{dt} - \frac{dV_3}{dt} = 0 \quad ....(8)$$

Using again the specific volume *v* and functional differentials of *v* with respect to *h*, and *p*,

$$\frac{dm_2}{dt} v_2 + m_2 \left(\frac{\partial v_2}{\partial h_2}\frac{dh_2}{dt} + \frac{\partial v_2}{\partial p_2} * \frac{dp_2}{dt}\right) = \frac{dm_3}{dt} v_3 + m_3 \left(\frac{\partial v_3}{\partial h_3}\frac{dh_3}{dt} + \frac{dv_3}{dp_3}\frac{dp_3}{dt}\right) ......(9)$$

(f) The Dalton relation regarding thermal equilibrium of steam and gas:

$$\frac{dT_2}{dt} = \frac{dT_3}{dt} \quad ....(10)$$

Using functional differentials of *T* with respect to *h*, and *p*,

$$\frac{\partial T_2}{\partial h_2}\frac{dh_2}{dt} + \frac{\partial T_2}{\partial p_2}\frac{dp_2}{dt} = \frac{\partial T_3}{\partial h_3}\frac{dh_3}{dt} + \frac{\partial T_3}{\partial p_3}\frac{dp_3}{dt} \quad ....(11)$$

(g) The Dalton relation regarding partial pressures:

$$\frac{dp_1}{dt} - \frac{dp_2}{dt} - \frac{dp_3}{dt} = 0 \quad ....(12)$$

The above set of equations can be linearized by developing the Jacobian matrix. By eliminating the mass derivatives, the set can now be arranged in the form of matrix differential equations:

$$[A]\underline{X} = \underline{C} \quad ....(13)$$

Provided that all physical properties are evaluated in the previous time step, the vector X contains:

$$\underline{X} = \left(\frac{dh_1}{dt}, \frac{dh_2}{dt}, \frac{dh_3}{dt}, \frac{dp_1}{dt}, \frac{dp_2}{dt}, \frac{dp_3}{dt}\right)^T \quad ....(14)$$

The constant vector **C** contains:

$$\underline{C} = ((\beta_1 - \alpha_1 h_1), (\beta_{23} - \alpha_2 h_2 - \alpha_3 h_3), -(\alpha_1 v_1 + \alpha_2 v_2), -(\alpha_2 v_2 - \alpha_3 v_3), 0, 0)^T \quad ....(15)$$

The elements of the coefficient matrix [A] consist of mass, volume, and derivatives of temperature and specific volume with respect to enthalpy and pressure. This set of six equations and six unknowns can be solved by the Gauss-Seidel method. Then, the current time step values can be calculated by the integration of the derivatives to obtain the enthalpies and pressures of the respective regions (water, steam, non-condensable gas) during the transients.

The Dalton Model for the Suppression Pool is simulated to cover the SRV Transients. However, for pipe break transients, a special model described below is required for modeling the hydrodynamics of "vent clearing" and "bubble growth" at the vent pipe exit.

(B) **"Vent Clearing" Model**

The figure below shows the geometrical configuration used to simulate the process of transient injection of steam and gas mixture into the suppression pool from the Drywell during pipe break. The Drywell is depicted as a reservoir in the diagram below. An orifice between the reservoir (Drywell) and the pipe line connecting the reservoir (Drywell) is used to model the port area between hotwell and wetwell. In the diagram below, the vent pipe line is shown to be vertical. However in the ABWR, the vent pipes are horizontal. It is assumed that the vent pipe orientation has little significance on modeling.

The steam/gas mixture is injected into the pool. The pressure inside the pipeline begins to increase. A pressure disturbance initiated at the orifice moves inside the pipe with sonic speed until it reaches the surface of the water. When the pressure signal reaches the upper surface of the water slug, the vent clearing stage begins. During the vent clearing, the pressure inside the pipe continues to increase

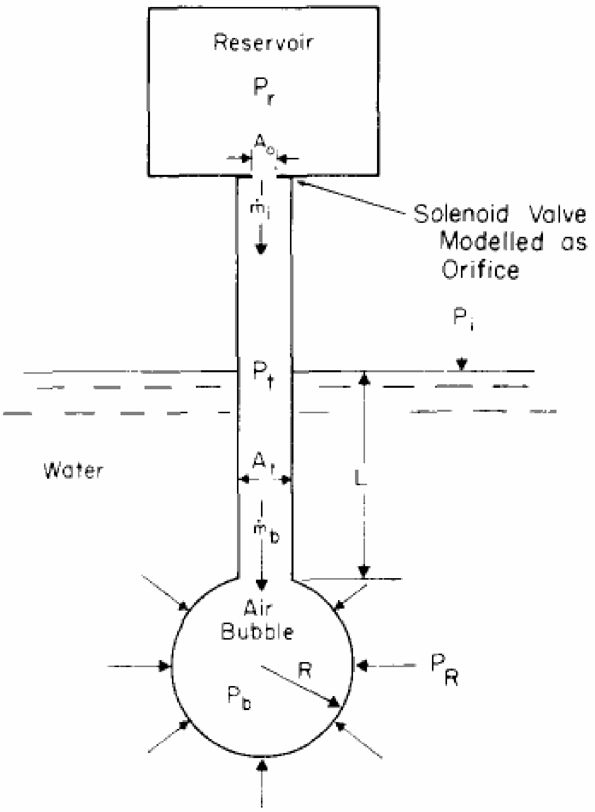

Figure 2 below shows the physical model for vent clearing. During the vent clearing stage, the water slug is assumed to behave as a rigid body. Friction at the pipe wall is neglected. The time dependent momentum equation for the water slug, represented by control volume *abcd* in figure 2, can be written as

$$\frac{d}{dt}(m_t u) + \rho_L u^2 A_t = [P_t(t) - (P_i + \rho_L g X)]A_t \quad \text{.............(16)}$$

where

m_t is the total mass of the liquid being moved with a velocity, u.
A_t is the cross-sectional area of the vent pipe.
P_t is the pressure in the vent pipe.
P_i is the pressure above the free surface of water in the pool.
g is the gravitational acceleration.
X is the distance measured from the original position of the free surface of the water slug.
ρ_L is liquid density of the water slug.

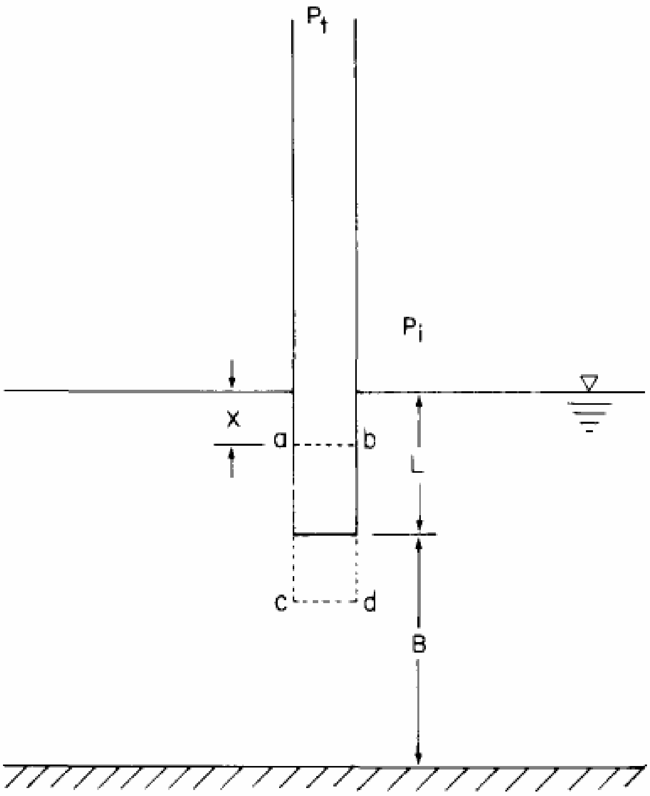

Figure 2 – Physical model for vent clearing

For a submergence depth, L, of the vent pipe, and writing $u = dX/dt$, equation (16) can be transformed into the following equation for X (Reference 1),

$$L\frac{d^2X}{dt} + (\frac{dX}{dt})^2 + gX = \frac{P_t(t) - P_i}{\rho_L} \quad \text{..................(17)}$$

Where

P_t is the pressure in the vent pipe.
P_i is the pressure above the free surface of water in the pool.
g is the gravitational acceleration.

The pressure inside the pipeline is obtained by making a mass balance on the pipe volume between the orifice and the water slug:

$$\frac{d}{dt}[\frac{P_t(t)V_t(t)}{R_G T_t}] = \frac{dm_b}{dt} - \frac{dm_i}{dt} \quad \text{............................(18)}$$

Where

P_t is the pressure in the vent pipe.
V_t is the vent volume between the orifice and the free surface of the water slug. During the time the interface does not move, V_t is constant. When the water slug moves, V_t depends on the length of the pipe. After the slug has been ejected completely, the volume of the pipe line V_t remains constant.
R_G is the gas constant for air
T_t is the stagnation temperature of air in the pipe
m_b is the mass flow rate for air into the bubble. The value is zero until the vent is cleared. When the vent is cleared, the value depends on the bubble growth.
m_i is the mass flow rate through the orifice into the vent pipe.

The mass flow rate through the orifice can be written as

$$\frac{dm_i(t)}{dt} = C_0 A_0 \sqrt{\frac{\gamma}{R_G}} \frac{P_r}{\sqrt{T_r}} \frac{M_0}{(1 + \frac{\gamma - 1}{2} M_0{}^2)^{\frac{(\gamma+1)}{2(\gamma-1)}}} \quad \text{...........................(19)}$$

C_0 is the discharge coefficient of the orifice

A_0 is the cross-sectional area of the orifice

γ is the coefficient of isentropic expansion of air.

P_r is the pressure in the reservoir upstream of the orifice= Drywell pressure.

T_r is the temperature of the reservoir

M_0 is the Mach number of the flow through the orifice.

Equation [19] is written in terms of the Mach number of the orifice as it facilitates evaluation of the mass flow rate under both non-choked and choked conditions.

For air, choking of the orifice will occur when ratio of pressures upstream and downstream of the orifice exceeds 1.89. In equation (19), the coefficient of discharge of the orifice is a function of both the reservoir pressure and the stagnation pressure in the pipe. The stagnation pressures in the pipe and the reservoir are related through the Mach number, M_o, at the orifice as

$$\frac{P_r}{P_t} = \left(\frac{1+\frac{\gamma-1}{2}M_0^{\,2}}{1+\frac{\gamma-1}{2}C_0 M_0^{\,2}}\right)^{\frac{\gamma}{\gamma-1}} \quad \text{..............................(20)}$$

The gas flow rate out of the pipe and into the bubble can be found in terms of the pressure in the pipe. From an equation similar to equation (19), the mass flow rate into the bubble, m_b, can be obtained as,

$$\frac{dm_b(t)}{dt} = C_t A_t \sqrt{\frac{\gamma}{R_G}}\frac{P_t}{\sqrt{T_t}}\frac{M_t}{(1+\frac{\gamma-1}{2}M_t^{\,2})^{\frac{(\gamma+1)}{2(\gamma-1)}}} \quad \text{...............(21)}$$

C_t is the pressure loss coefficient which depends on the pipe diameter, the pressure difference between the pipe and the bubble, and the size of the bubble.

A_t is the cross-sectional area of the pipe

γ is the coefficient of isentropic expansion of air.

P_t is the pressure in the pipe

T_t is the temperature of air in the pipe

M_t is the Mach number of the flow in the pipe.

The pressure in the pipe P_t and the bubble P_b are related through the Mach number, M_t, at the pipe exit as

$$\frac{P_t}{P_b} = [\frac{1+\frac{\gamma-2}{2}M_t^{\,2}}{1+\frac{\gamma-1}{2}C_t M_t^{\,2}}]^{\frac{\gamma}{(\gamma-1)}} \quad \text{........................(22)}$$

(C) Model for bubble growth

When the water slug is ejected completely, the air bubble begins to form at the exit of the vent pipe. The growth rate of the bubble is determined by the surface tension, the liquid inertia, the mass flow rate, and the driving pressure.

The following assumptions are used to simplify the modeling:

(1) The surface tension of water is so small that the pressure drop across the bubble wall is negligible.
(2) The bubble is assumed to grow as a sphere.
(3) The pressure inside the bubble is uniform.
(4) The temperature, density, and pressure within the bubble are related through isentropic relations and the ideal gas law.
(5) Liquid and gas viscosities are negligible.
(6) The distance between the bubble center and the exit of the vent pipe is equal to the bubble radius R.
(7) The center of the air bubble moves with velocity *dR/dt*

Employing the above assumptions, the equation describing the bubble growth can be written as

$$P_b(t) = \rho_L[\frac{5}{4}(\frac{dR}{dt})^2 + R(t)\frac{d^2R}{dt} + g(L+R)] + P_i \quad \text{..................(23)}$$

Where

P_b is the pressure inside the bubble
ρ_L is the density of the liquid in the pool
R is the radius of the bubble
g is the gravitational acceleration
L is the emergence length of the pipe.
P_i is the pressure of the atmosphere above the pool.

The pressure inside the bubble can be obtained by combining the gas law and the mass conservation equation for the bubble as

$$\frac{d}{dt}[\frac{P_b(t)V_b(t)}{R_G T_b(t)}] = \frac{dm_b}{dt} \quad \text{....................(24)}$$

Where

P_b is the pressure inside the bubble
T_b is the air temperature in the bubble.
V_b is the bubble volume
R_G is the gas constant for air.
m_b is the mass flow rate for air into the bubble.

The volume of the bubble is related to the bubble radius as

$$V_b = \frac{4}{3}\pi R^3 \quad \text{...........(25)}$$

The bubble temperature can be determined by making an energy balance for the bubble.

Assuming that initially the temperature of the bubble is the temperature of the reservoir (Drywell) and the bubble expands adiabatically, the rate of change of temperature of the bubble with reference to the reservoir, the vent pipe or the pool temperature can be written as

$$\frac{dT_b}{dt} = \frac{[P_b - P_i - (L+R)\rho_L g]4\pi R^2 \frac{dR}{dt}}{\int_0^t C_v \frac{dm_b}{dt}dt} \quad \text{..........(26)}$$

Where

T_b is the air temperature in the bubble.
P_b is the pressure inside the bubble
P_i is the pressure of the atmosphere above the pool.
R is the radius of the bubble
g is the gravitational acceleration
L is the emergence length of the pipe.
C_v is the specific heat of air at constant volume.

(D) Solution Approach for Vent Clearing and Bubble Growth

Assuming the temperature, T_t, in the vent pipe is equal to the temperature, T_r of the reservoir, and the initial vent volume between the orifice and the water slug is known, equations (17) to (26) represent ten relations in ten unknowns *(X, P_t, dm_i/dt, dm_b/dt, M_o, M_t, P_b, V_b, R, T_b).*

These equations can be solved numerically, if the reservoir pressure, air temperature and initial pressure in the vent line are known.

Prior to complete vent clearing, m_b is zero and only equations (17)—(20) are solved. It must be mentioned that during vent clearing, the vent volume, V_t, can be written as a sum of initial vent volume, V_{ti}, and the volume of the vent emptied by the water slug.

$$V_t = V_{ti} + A_t X \quad ....(27)$$

After vent clearing, the vent volume remains constant

$$V_t = V_{ti} + A_t L \quad ....(28)$$

and equation (17) need not be solved.

During the bubble growth period, equations (18)-(26) are solved simultaneously.

(E) Interfacing between the Drywell Model, the Vent Clearing and Bubble Growth Models and the Dalton Suppression Pool Model.

(1) In the event of pipe break, the steam ingress from the pipe will be released in the Drywell containing non-condensable gas. The Drywell model will calculate the Drywell pressure P_{dw} and the source terms α_i and β_i for mass and energy flows for the various regions in the Wetwell – gas, steam and water (Dalton Suppression Pool model).
(2) For the "Vent Clearing" and "Bubble Growth" Model, the Drywell pressure P_r at the Vent inlet is equated to the Drywell pressure. The drywell temperature T_r is obtained from the enthalpy of the steam and gas mixture calculated in the Drywell model. Assuming that the temperature, T_t, in the vent pipe is equal to the temperature, T_r of the reservoir, and given the initial vent volume between the orifice and the water slug, equations (17) to (26) can be used to calculate the ten unknowns *(X, P_t, dm_i/dt, dm_b/dt, M_o, M_t, P_b, V_b, R, T_b).*
(3) Given the calculated value of dm_i/dt, W_{out_ww} can be computed.. As a result, new values for source terms for the water, non-condensable gas, and steam regions in the Suppression pool can be recomputed, as well as the pressure and enthalpy of the non-condensable gas, steam and water regions in the wetwell,

(E) Drywell Model

The Model for the Drywell is the same as that for the Wetwell (Suppression Pool), with different source terms. The differences are highlighted as follows:

- **Continuity equations (1,2,3) for water, steam, and air in the Drywell.**
  (1) The source terms for the water region in the Drywell =
    (a) LOCA break flow's saturated water component *$(1-X_q)W_{break}$* +
    (b) ECC spray flow in the Drywell
    (c) Condensation rate of steam due to ECC Spray
  (2) Source term for the saturated-steam region in the Drywell =

(a) LOCA Break flow's saturated steam component, $X_q W_{break}$ **-- eowr**

(b) Condensation rate of steam due to ECC spray -

(c) Fraction of steam vapor vented to Wetwell.

(3) Source term for air region in Drywell = non-condensable gas release rate from LOCA – fraction of air vented to Wetwell.

- **Equation (4) for the energy of the saturated liquid accumulated in the Drywell**.
  (1) Source term for Liquid-region energy = $(1-X_q)\ W_{break}\ h_{liq}$ + ECC Spray flow energy content + energy content of the amount of steam condensation due to ECC Spray
- **Equation (5) for the energy of the steam and air regions in the Drywell**.
  (1) Source term for Air and Steam Regions in Drywell = $X_q\ W_{break}\ h_{vap}$ + non-condensable gas release heat content **–**
  (2) Condensation rate of steam due to ECC spray *times* latent heat –
  (3) Energy of the fraction of steam vented to Wetwell –
  (4) Energy of the fraction of air content vented to Wetwell.
- Equations (6) to (15) are applicable to the Drywell.
- Enthalpies and pressures of the respective region (water, steam, air) can be found by solving the matrix equations (just like for the Wetwell). The Drywell pressure can be calculated by the sum of partial pressures of steam and air. The Drywell air + steam temperature can be derived by h_2 and h_3.

Reference:

1. Analysis of BWR pressure suppression pool dynamics, by McCauley, E.W. et al, 1976 August 30. DOE Report # UCRL-78694; CONF-760954-2.
2. Meier M., Andreani M., Smith B.L., and Yadigaroglu G. "Numerical and experimental study of large steam-air bubbles condensing in water." In Procs. of the Third International Conference on Multiphase Flow, ICMF'98, 1998.

Appendix

BWR TECHNICAL DATA

General plant data

Power plant output, gross	1385	MW(e)
Power plant output, net	1300	MW(e)
Reactor thermal output	3926	MWth
Power plant efficiency, net	33.1	%
Cooling water temperature	≈ 28.0	^oC
Nuclear steam supply system		
Number of coolant loops	1	
Steam flow rate at nominal conditions	2122	kg/s
Feedwater flow rate at nominal conditions	2118	kg/s
Reactor coolant system		
Primary coolant flow rate	14502	kg/s
Reactor operating pressure	7.07	MPa
Steam temperature/pressure	287.8/7.07	^oC/MPa
Feedwater temperature	215.6	^oC
Core coolant inlet temperature	278	^oC
Core coolant outlet temperature	288	^oC
Mean temperature rise across core	10	^oC
Reactor core		
Active core height	3.710	m
Equivalent core diameter	5.164	m
Heat transfer surface in the core	9254	m^2
Fuel weight	159	t U
Average fuel power density	24.7	kW/kg U
Average core power density	50.6	kW/l
Thermal heat flux, F_q	424	kW/m^2
Enthalpy rise, F_H	273	
Fuel material	Sintered UO_2	
Fuel (assembly) rod total length	4470	mm
Rod array	10x10, square lattice	
Number of fuel assemblies	872	
Number of fuel rods/assembly	92	
Number of spacers	8	
Enrichment (range) of first core, average	2.0 (appr.)	Wt%
Enrichment of reload fuel at equilibrium core	3 to 4	Wt%
Operating cycle length (fuel cycle length)	24	months
Average discharge burnup of fuel [capability]	>50,000	MWd/*t*
Cladding tube material	annealed, recrystallised Zr 2	
Cladding tube wall thickness	0.66	mm
Outer diameter of fuel rods	10.3	mm
Fuel channel/box; material	Zr-2	
Overall weight of assembly, including box	300	kg

Uranium weight/assembly	181	kg
Active length of fuel rods	3.810	mm
Burnable absorber, strategy/material	axial and radial grading/ Gd_2O_3 mixed with fuel	
Number of control rods	205	
Absorber material	B_4C and Hafnium	
Drive mechanism	electro-mechanical/hydraulic	
Positioning rate	30	mm/s
Soluble neutron absorber	Boron	

Reactor pressure vessel

Inner diameter of cylindrical shell	7 100	mm
Wall thickness of cylindrical shell	190	mm
Total height, inside	21 000	mm
Base material:	cylindrical shell	low-alloy carbon steel
RPV head	[to ASTM A533, grade B, ASTM A508, class 3, or equiv.]	
lining	stainless steel	
Design pressure/temperature	8.62/301.7	MPa/°C
Transport weight (lower part w/rigging)	1164	t
RPV head	≈100	t

Reactor recirculation pump

Type	variable speed, wet motor, single stage, vertical internal pump	
Number	10	
Design pressure/temperature	same as for RPV	MPa/°C
Design mass flow rate (at operating conditions)	1453 (each)	kg/s
Pump head	0.287	MPa
Rated power of pump motor (nominal flow rate)	≈800	kW
Pump casing material	same as for RPV	
Pump speed (at rated conditions)	≤1500	rpm

Primary containment

Type	Pressure-suppression/ reinforced concrete	
Overall form (spherical/cyl.)	cylindrical	
Design pressure/temperature	310.3/171.1	kPa/°C
Design leakage rate	0.5	vol%/day
Is secondary containment provided?	Yes	

Reactor auxiliary systems

Reactor water cleanup,	capacity	42.36	kg/s	
	filter type	deep bed		
Residual heat removal,	at high pressure		kg/s	
	at low pressure (100 °C)		253.8	MW
Coolant injection,	at high pressure (HPCF)		36.3 kg/s	
	at low pressure (LPCF)	253.8	kg/s	

Power supply systems

Main transformer,	rated voltage	24/525	kV
	rated capacity	1660	MVA
Plant transformers,	rated voltage	24/4.16/13.8	kV
	rated capacity	50/15/35	MVA
Medium voltage busbars (6 kV or 10 kV)		13.8/4.16	kV
Standby diesel generating units: number		3	
	rated power	6.57	MW
Number of diesel-backed busbar systems		3	
Voltage level of these		4160	V AC

Turbine plant

Number of turbines per reactor	1		
Type of turbine(s)	six flow, tandem compound, single reheat		
Number of turbine sections per unit (e.g. HP/LP/LP)		1 HP/3 LP	
Turbine speed		1800	rpm
HP inlet pressure/temperature	6.792/283.7	MPa/°C	

Generator

Type	3-phase, turbo-generator		
Rated power	1620	MVA	
Active power		1385	MW
Voltage	24	24 kV	
Frequency	60	Hz	

Condenser

Type	shell type (3 shells)	
Number of tubes	1 tube pass/shell	
Heat transfer area	124,170	m^2
Cooling water flow rate	34.68	m^3/s
Condenser pressure (HP shell)	11.75	kPa

Condensate pumps

Number	4 x 50%	
Flow rate	≈ 435	kg/s
Pump head	3.82	MPa

Condensate cleanup system

Full flow/part flow	Full flow
Filter type (deep bed or rod type)	deep bed

Feedwater pumps

Number	3 × 65%	
Flow rate	≈ 1000	kg/s
Pump head	6	MPa
Feedwater temperature (final)	216	°C

Condensate and feedwater heaters

Number of heating stages,		low pressure	3 × 4
	high pressure	2 × 2	