The Abdus Salam International Centre for Theoretical Physics

United Nations Educational, Scientific and Cultural Organization

IAEA International Atomic Energy Agency

1879-1

# Nuclear Power Plant Simulators for Education

*29 October - 9 November, 2007*

## BWR Material

W.K. Lam
*Cassiopeia Technologies Inc., Toronto, Canada*

# *ICTP Workshop - BWR NPP & Simulator Overview*

Wilson Lam (wilson@cti-simulation.com)

CTI Simulation International Corp.

www.cti-simulation.com

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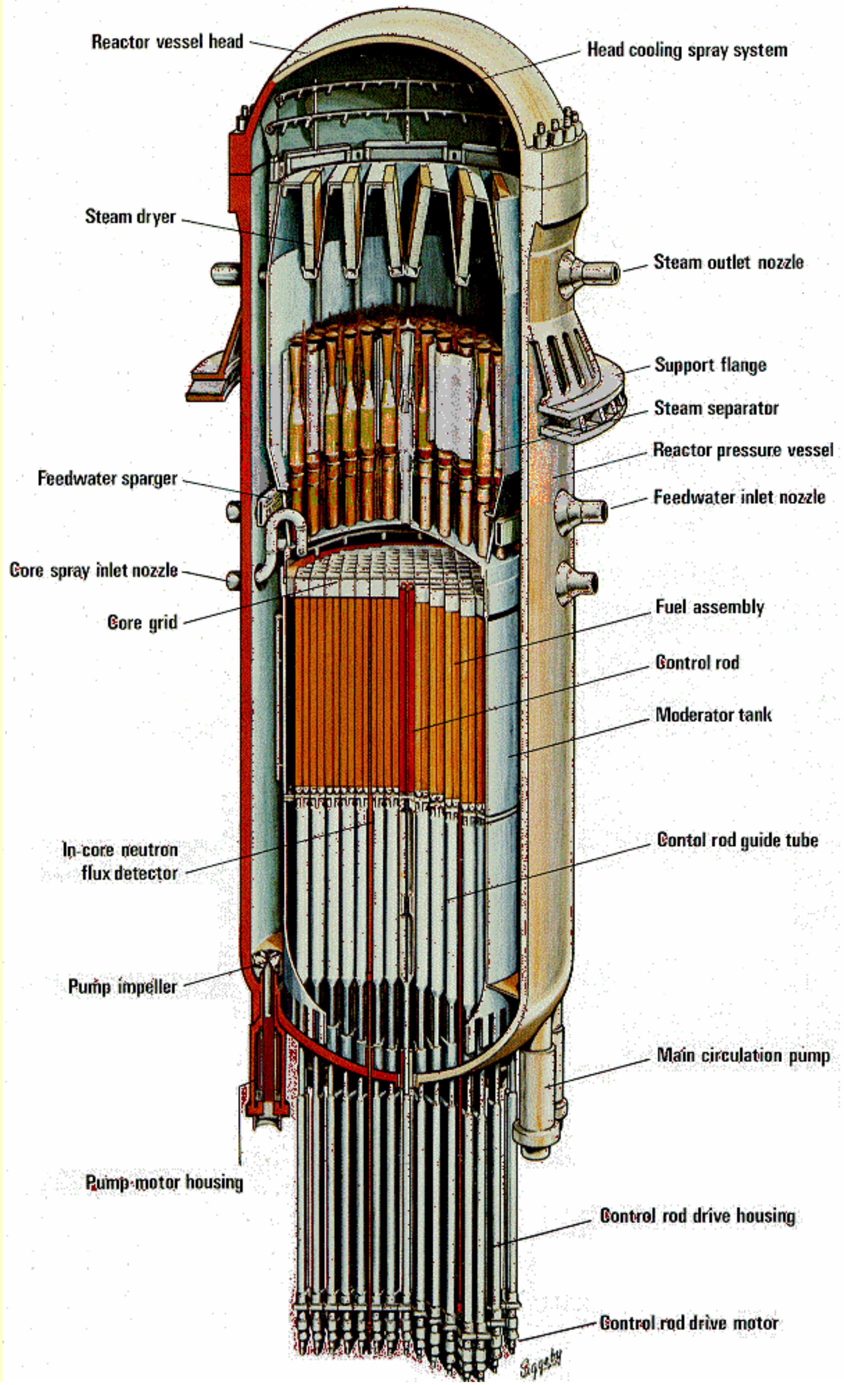
# *Advanced BWR Plant - generic features*

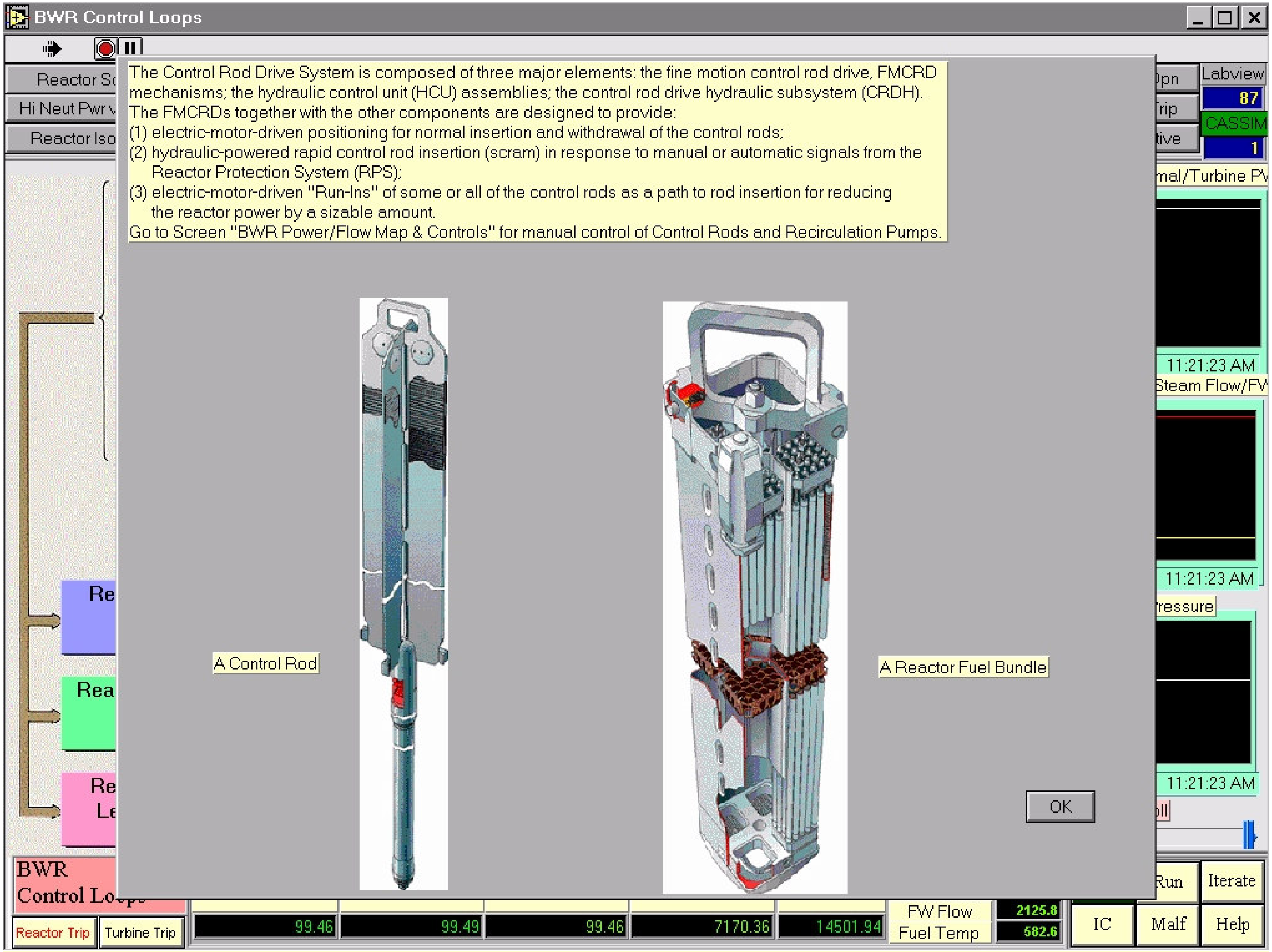
- Direct Cycle - heat generated in reactor core is directly utilized for steam generation inside the reactor vessel.
- Steam develops as small “bubbles” (void) entrained in core coolant. It is separated in the coolant flow from “Steam Separators”, and dried in “Steam Dryer” arrangement - minimize water carry-over & more long-lived radioactive products from the reactor water.

# BWR- ABB


Reactor vessel head
Head cooling spray system
Steam dryer
Steam outlet nozzle
Support flange
Steam separator
Reactor pressure vessel
Feedwater sparger
Feedwater inlet nozzle
Core spray inlet nozzle
Core grid
Fuel assembly
Control rod
Moderator tank
In-core neutron
flux detector
Contol rod guide tube
Pump impeller
Main circulation pump
Pump-motor housing
Control rod drive housing
Control rod drive motor

# *Reactor Core & Fuel Design*

- BWR core consists of a number of fuel bundles (assemblies).
- Each fuel bundle (assembly) consists of a number of fuel rods arranged in n x n square lattice (slightly enriched uranium fuel - typical enrichment 2 % to 5 % U-235 by weight). Average core power density ~ 60 % PWR.
- Number of control rods enter the core from the bottom, through guide tubes in the fuel assemblies.


BWR Control Loops
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.
Go to Screen "BWR Power/Flow Map & Controls" for manual control of Control Rods and Recirculation Pumps.
A Control Rod
A Reactor Fuel Bundle
OK
Labview
87
CASSIM
1
11:21:23 AM
11:21:23 AM
11:21:23 AM
BWR
Reactor Trip
Turbine Trip
99.46
99.49
99.46
7170.36
14501.94
FW Flow
Fuel Temp
2125.8
582.6
Iterate
IC
Malf
Help

# *Reactor Core & Fuel Design*

- BWR allows bulk boiling of water. Operating temperature ~ 288 C; steam pressure 7 MPa
- Reactor power control consists of control rods and recirculation flow control
- Control rods (neutron absorbing materials) maintain a constant desired power level by a adjusting their positions ~ 2 % per sec.
- Recirculation flow control also controls reactor power by altering the density of water used as moderator. The flow rate is adjusted by a variable speed pump. Power changes ~ 30 % per minute

# *Question #1*

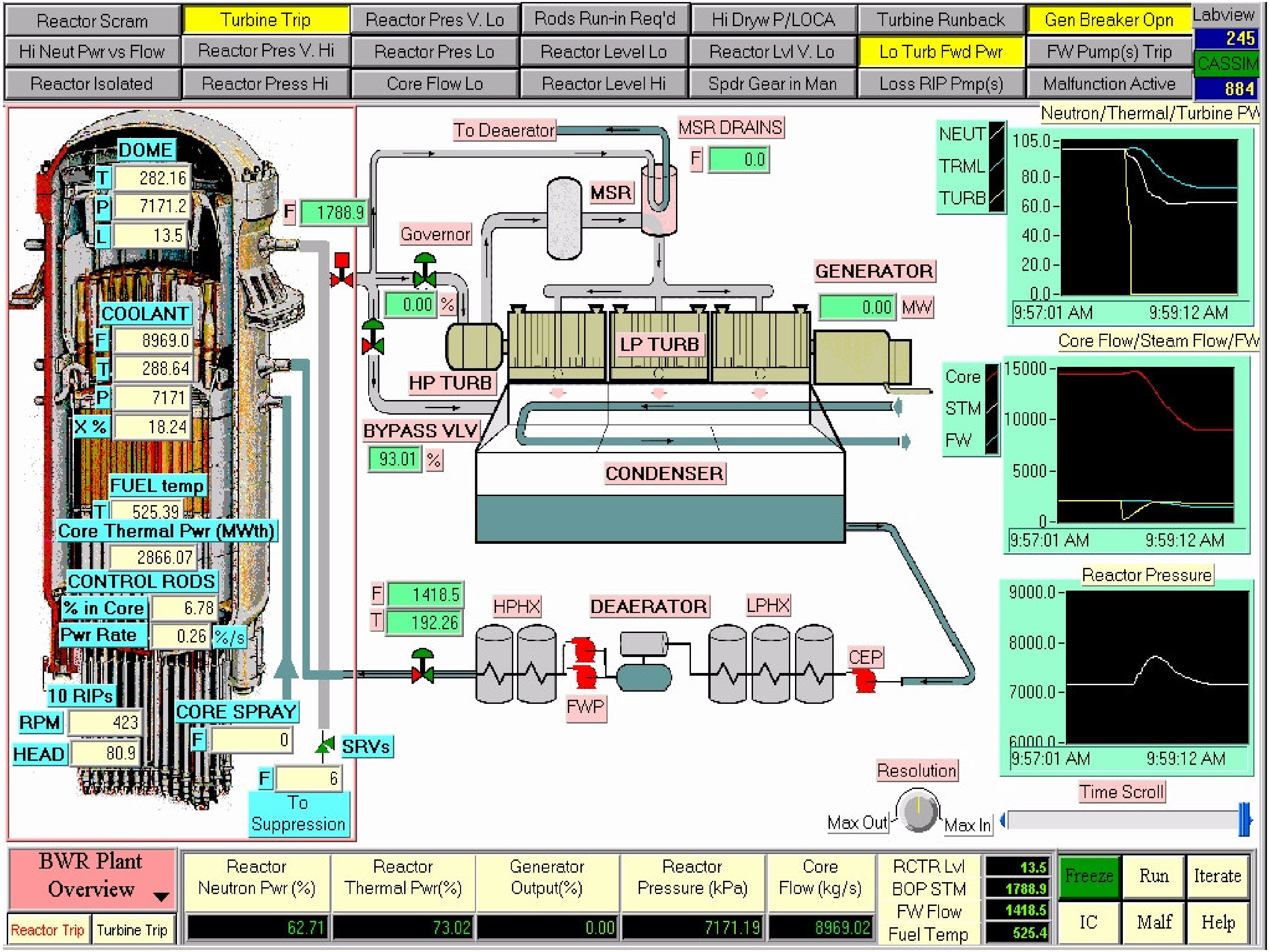
- Why adjusting recirculation flow rate in BWR will control reactor power ?

# *Question #2*

- Why Control Rods enter from the bottom of the core, as opposed to entering from the top of the core, like in PWR or PHWR ?

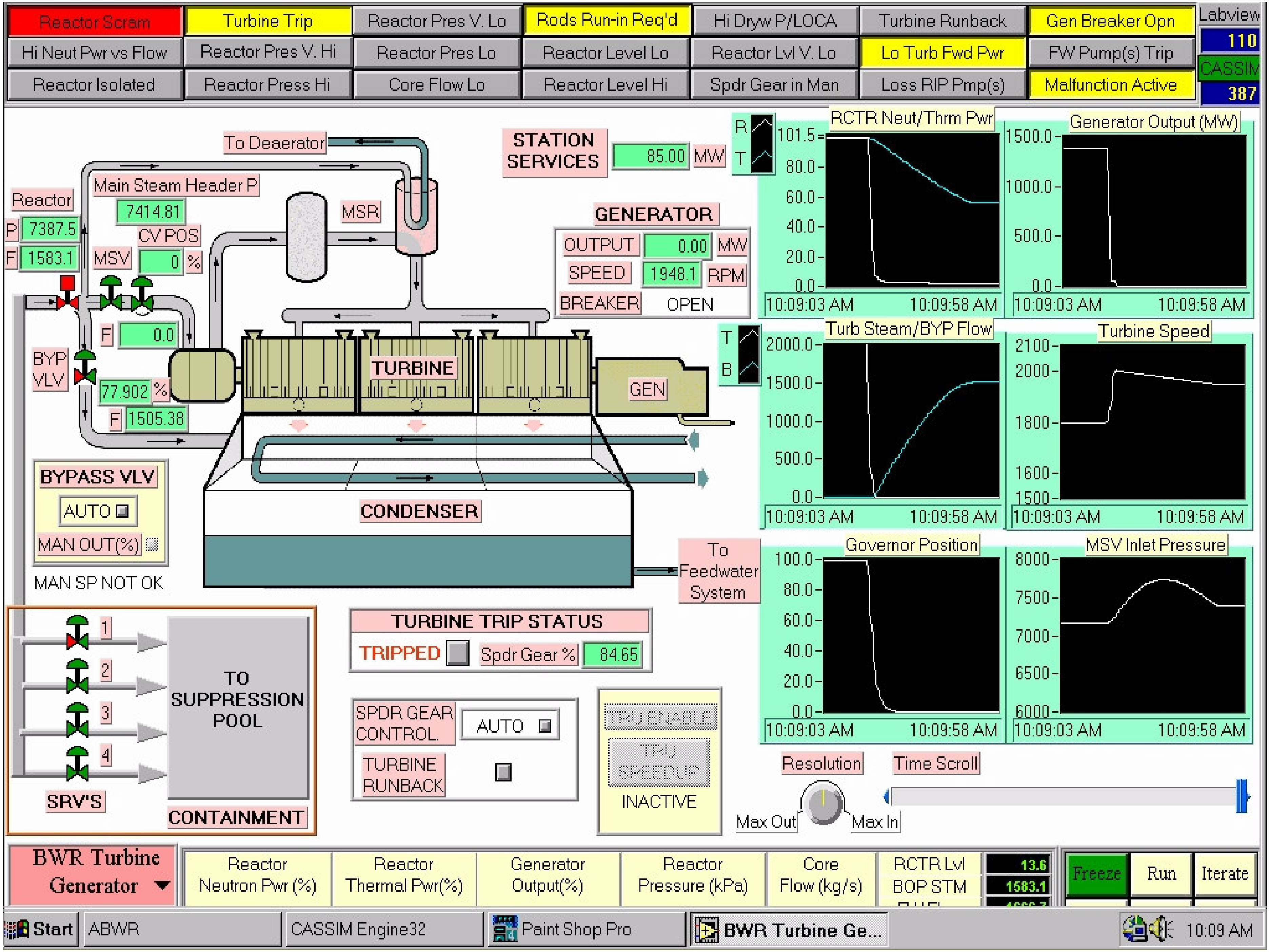
# *Main Steam System*

- "Dried steam" from Reactor Pressure Vessel (RPV) to the turbine plant through four steam lines connected to nozzles equipped with "flow limiters".
- Limit the coolant blowdown rate from the RPV = or < 200 % rated steam flow at 7.07 MPa upstream pressure in the event of steam line break occurs anywhere downstream the nozzle.
- Isolation valves inside and outside of containment wall.
- Safety Relief Valves (~16) connected to the four steam lines to prevent RPV overpressure, with blow down pipe to Suppression Pool.


Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
245
CASSIM
884
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
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
DOME
T 282.16
P 7171.2
L 13.5
COOLANT
F 8969.0
T 288.64
P 7171
X % 18.24
FUEL temp
T 525.39
Core Thermal Pwr (MWth)
2866.07
CONTROL RODS
% in Core 6.78
Pwr Rate 0.26 %/s
10 RIPs
RPM 423
HEAD 80.9
CORE SPRAY
F 0
SRVs
F 6
To Suppression
F 1788.9
To Deaerator
MSR DRAINS
F 0.0
MSR
Governor
0.00 %
GENERATOR
0.00 MW
HP TURB
LP TURB
BYPASS VLV
93.01 %
CONDENSER
F 1418.5
T 192.26
HPHX
DEAERATOR
LPHX
CEP
FWP
Neutron/Thermal/Turbine PW
NEUT
TRML
TURB
105.0
80.0
60.0
40.0
20.0
0.0
9:57:01 AM
9:59:12 AM
Core Flow/Steam Flow/FW
Core
STM
FW
15000
10000
5000
0
9:57:01 AM
9:59:12 AM
Reactor Pressure
9000.0
8000.0
7000.0
6000.0
9:57:01 AM
9:59:12 AM
Resolution
Max Out
Max In
Time Scroll
BWR Plant Overview
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 62.71
Reactor Thermal Pwr(%) 73.02
Generator Output(%) 0.00
Reactor Pressure (kPa) 7171.19
Core Flow (kg/s) 8969.02
RCTR Lvl 13.5
BOP STM 1788.9
FW Flow 1418.5
Fuel Temp 525.4
Freeze
Run
Iterate
IC
Malf
Help

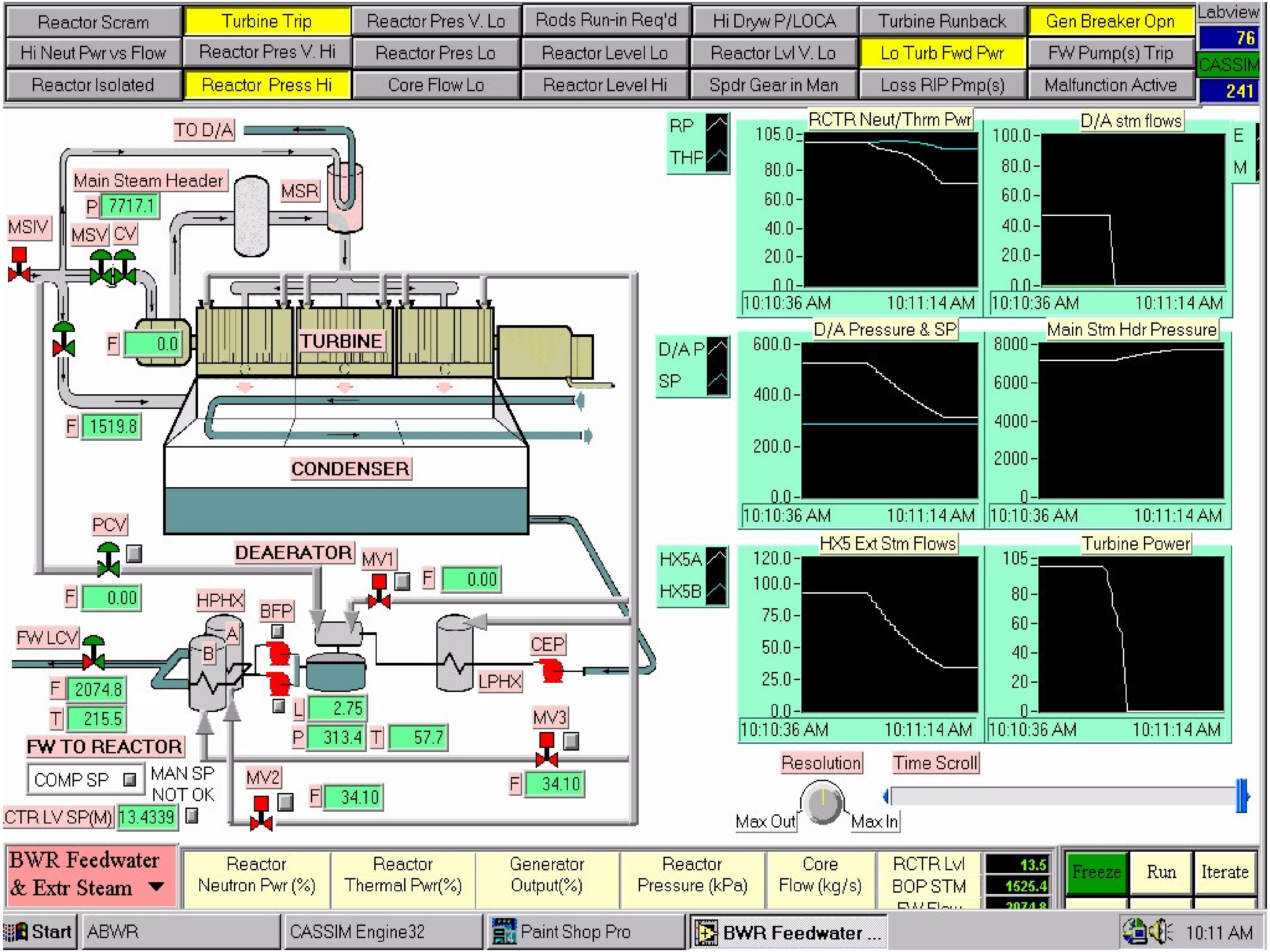
# *Turbine & Steam Bypass Systems*

- Saturated steam from RPV main steam lines admitted to turbine HP cylinder via the governor valves. After HP section, steam passes through MSR to LP turbine cylinders.
- A special Steam Bypass line prior to the turbine governor valves, enables dumping the full nominal steam flow directy to condenser in the event of plant upset (e.g. turbine trip), in order to avoid severe pressure surges and corresponding power peaks in reactor.


Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
110
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
387
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
To Deaerator
STATION SERVICES
85.00 MW
Main Steam Header P
7414.81
Reactor
P 7387.5
F 1583.1
CV POS
MSV
0 %
MSR
GENERATOR
OUTPUT 0.00 MW
SPEED 1948.1 RPM
BREAKER OPEN
F 0.0
BYP VLV
77.902 %
F 1505.38
TURBINE
GEN
CONDENSER
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
To Feedwater System
TURBINE TRIP STATUS
TRIPPED
Spdr Gear % 84.65
SPDR GEAR CONTROL
AUTO
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
1
2
3
4
TO SUPPRESSION POOL
SRV'S
CONTAINMENT
RCTR Neut/Thrm Pwr
R
T
101.5
80.0
60.0
40.0
20.0
0.0
10:09:03 AM
10:09:58 AM
Generator Output (MW)
1500.0
1000.0
500.0
0.0
Turb Steam/BYP Flow
T
B
2000.0
1500.0
1000.0
500.0
0.0
Turbine Speed
2100
2000
1800
1600
1500
Governor Position
100.0
80.0
60.0
40.0
20.0
0.0
MSV Inlet Pressure
8000
7500
7000
6500
6000
Resolution
Time Scroll
Max Out
Max In
BWR Turbine Generator
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
RCTR Lvl
BOP STM
13.6
1583.1
Freeze
Run
Iterate
Start
ABWR
CASSIM Engine32
Paint Shop Pro
BWR Turbine Ge...
10:09 AM

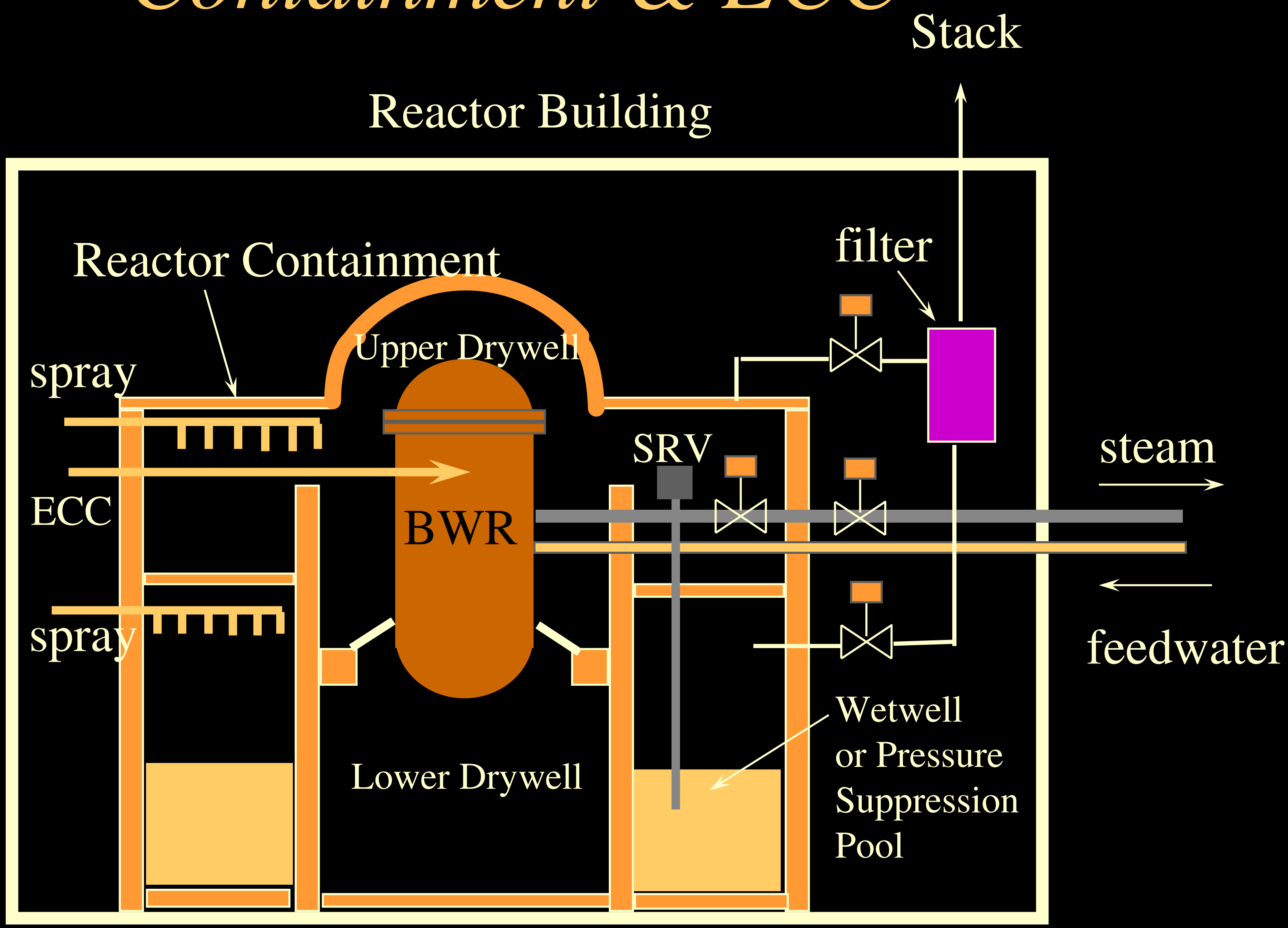
# *BOP & Feedwater System*

- Typical BOP systems - condenser; condensate pumps; deaerator; feedwater heaters; Reactor Feed Pumps (RFP); Reactor Level Control Valves.



Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
76
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
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
241
TO D/A
Main Steam Header
P 7717.1
MSR
MSIV
MSV CV
F 0.0
TURBINE
F 1519.8
CONDENSER
PCV
DEAERATOR
MV1
F 0.00
F 0.00
HPHX
BFP
A
B
FW LCV
CEP
LPHX
F 2074.8
T 215.5
L 2.75
P 313.4 T 57.7
FW TO REACTOR
MV3
COMP SP
MAN SP
NOT OK
MV2
F 34.10
F 34.10
CTR LV SP(M) 13.4339
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
10:10:36 AM
10:11:14 AM
Resolution
Time Scroll
Max Out
Max In
BWR Feedwater & Extr Steam
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
RCTR Lvl
BOP STM
13.5
1525.4
Freeze
Run
Iterate
Start
ABWR
CASSIM Engine32
Paint Shop Pro
BWR Feedwater ...
10:11 AM

# *Containment & ECC*



# *Containment*

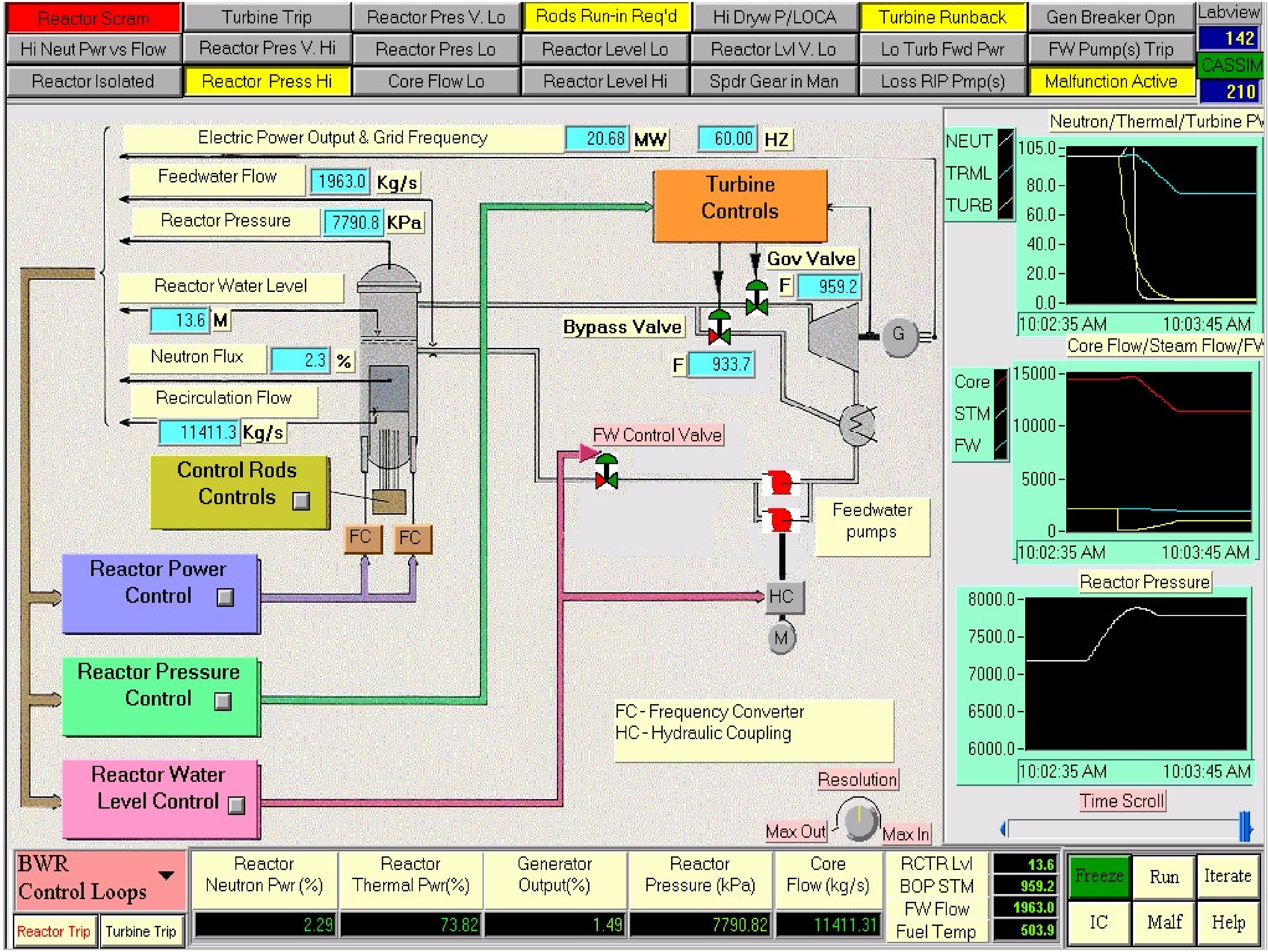
- Containment - cylindrical prestressed concrete structure with embedded steel liner - encloses reactor, reactor coolant pressure boundary & important ancillary systems.
- Pressure-suppression type with drywell and wetwell.
- Wetwell separated from drywell by partition floor. The wetwell's lower portion is filled with water - condensation pool. Upper portion serves a gas compression chamber.

# *Containment (cont'd)*

- Drywell pressurization (LOCA) - drywell atmosphere & steam pushed into the wetwell via a passage through the partition wall. Steam condensed in suppression pool. Non-condensables collected in the gas compression chamber
- Pressure suppression further supported by water spray system to gas compression chamber and the upper drywell.
- Containment vessel can also be vented manually or via rupture disk, to the stack through filter system

# *BWR Control Systems*

- Reactor Power Control
- Reactor Pressure Control
- Reactor Water Level Control
- Turbine Control
- Turbine Steam Bypass Control



Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
142
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
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
210
Electric Power Output & Grid Frequency
20.68 MW
60.00 HZ
Feedwater Flow
1963.0 Kg/s
Reactor Pressure
7790.8 KPa
Turbine Controls
Gov Valve
F 959.2
Reactor Water Level
13.6 M
Bypass Valve
F 933.7
G
Neutron Flux
2.3 %
Recirculation Flow
11411.3 Kg/s
FW Control Valve
Control Rods Controls
FC
FC
Feedwater pumps
Reactor Power Control
HC
M
Reactor Pressure Control
FC - Frequency Converter
HC - Hydraulic Coupling
Reactor Water Level Control
Resolution
Max Out
Max In
Neutron/Thermal/Turbine P
NEUT
TRML
TURB
105.0
80.0
60.0
40.0
20.0
0.0
10:02:35 AM
10:03:45 AM
Core Flow/Steam Flow/F
Core
STM
FW
15000
10000
5000
0
10:02:35 AM
10:03:45 AM
Reactor Pressure
8000.0
7500.0
7000.0
6500.0
6000.0
10:02:35 AM
10:03:45 AM
Time Scroll
BWR Control Loops
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
2.29
Reactor Thermal Pwr(%)
73.82
Generator Output(%)
1.49
Reactor Pressure (kPa)
7790.82
Core Flow (kg/s)
11411.31
RCTR Lvl
BOP STM
FW Flow
Fuel Temp
13.6
959.2
1963.0
503.9
Freeze
Run
Iterate
IC
Malf
Help

# *Reactor Power Control*

- 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.

# *Rod Control System*

- The Control Rod Drive System is composed of three major elements using fine position digital motor drive & hydraulic drive:
- (1) the fine motion control rod drive, FMCRD mechanisms;
- (2) the hydraulic control unit (HCU) assemblies;
- (3) the control rod drive hydraulic subsystem (CRDH).

# *Rod Control System (cont'd)*

The FMCRDs together with the other components are designed to provide:

- electric-motor-driven positioning for normal insertion and withdrawal of the control rods;
- hydraulic-powered rapid control rod insertion (scram) in response to manual or automatic signals from the Reactor Protection System (RPS);
- 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.

# *Rod Control System - Simulator*

- For the BWR Simulator, there are approximately 208 FMCRDs in total, they are positioned and calibrated with reactivity worth of -100 mk when all of them are 100% in core, and +70 mk when all of them are 100 % out of core; 0 mk when they are ~ 41 % in core.
- The rods are grouped in 8 banks, so each bank of rods have + 8.75 mk when fully out of core; and -12.5 mk when fully in core.
- The FMCRDs will be fully inserted into the core in the event of **a reactor scram**. In such case, the fast insertion speed is typically 3 sec. for 100 % insertion.

# *Rod Control System - Simulator*

- The full speed travel time for the rod movement during power maneuvering is typically 60 sec., or considering for the total FMCRDs in Auto mode, where all the rods move together, the reactivity change rate is ~ 2.8 mk per sec.
- Considering moving the banks of rods individually under Manual Mode, then the reactivity change rate for each bank under manual mode control is ~ 0.36 mk per sec.

# *Reactor Power Control*

- The recirculation flow 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 while keeping the power distribution in the core constant.

# *Reactor Pressure Control*

- In normal operation, the reactor pressure is automatically controlled to be **constant**.
- A pressure controller 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*

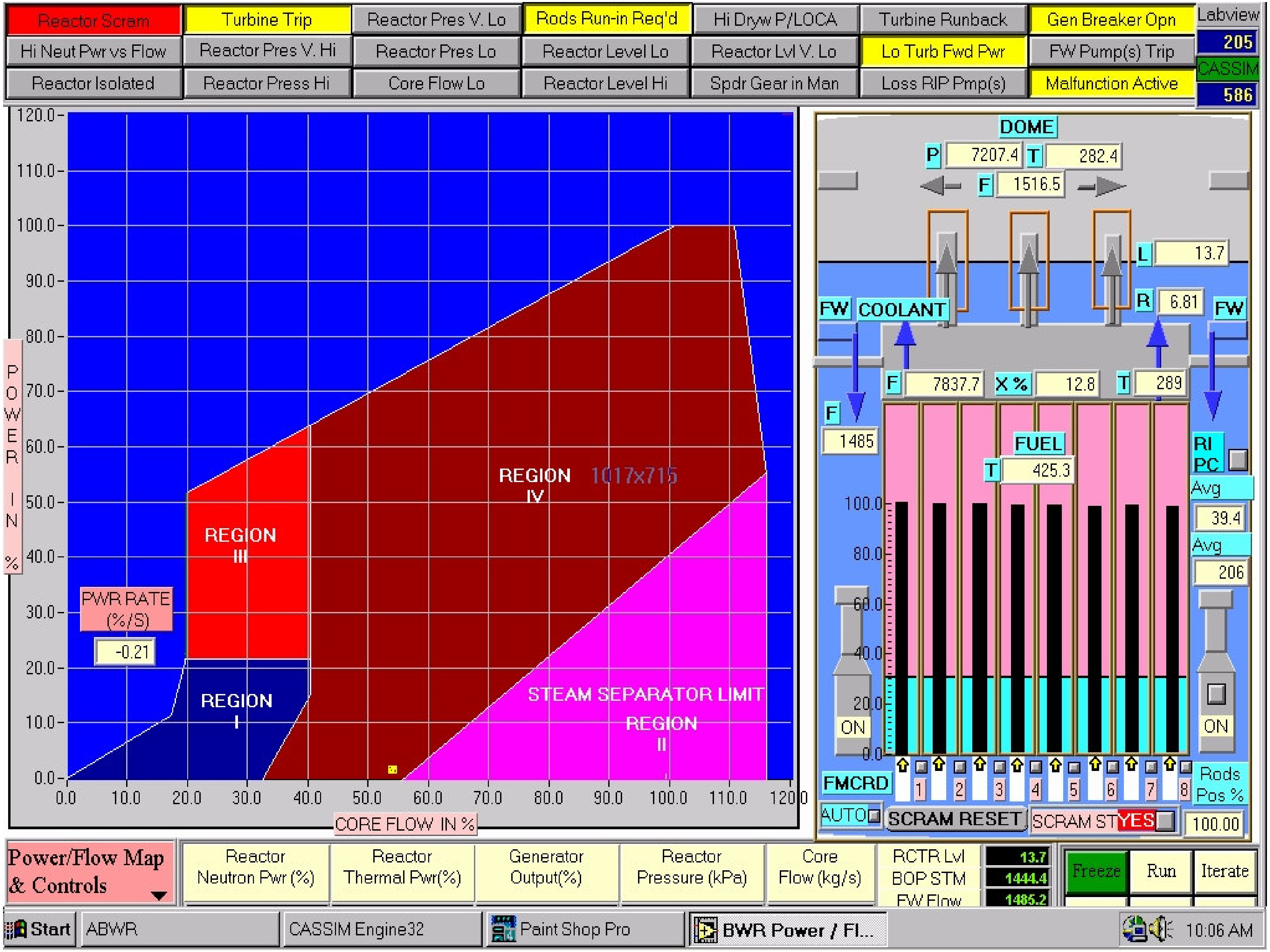
- The flow of feedwater is automatically controlled to maintain the specified water level by a "three elements" control scheme: steam flow, feedwater flow, water level.
- The valve opening of the feedwater control valve provided at the outlet of the feedwater pumps is regulated by  the control signal as result of this "three-element" control scheme.

# *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 "speeder gear" 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.

# *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 % 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.


Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
205
CASSIM
586
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
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
POWER IN %
CORE FLOW IN %
REGION I
REGION III
REGION IV
1017x715
STEAM SEPARATOR LIMIT
REGION II
PWR RATE (%/S)
-0.21
DOME
P 7207.4
T 282.4
F 1516.5
L 13.7
R 6.81
FW
COOLANT
F 7837.7
X % 12.8
T 289
F 1485
FUEL
T 425.3
RI PC
Avg 39.4
Avg 206
ON
FMCRD
AUTO
SCRAM RESET
SCRAM ST YES
Rods Pos %
100.00
Power/Flow Map & Controls
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
RCTR Lvl 13.7
BOP STM 1444.4
FW Flow 1485.2
Freeze
Run
Iterate
Start
ABWR
CASSIM Engine32
Paint Shop Pro
BWR Power / Fl...
10:06 AM

# *Power/Flow Map*

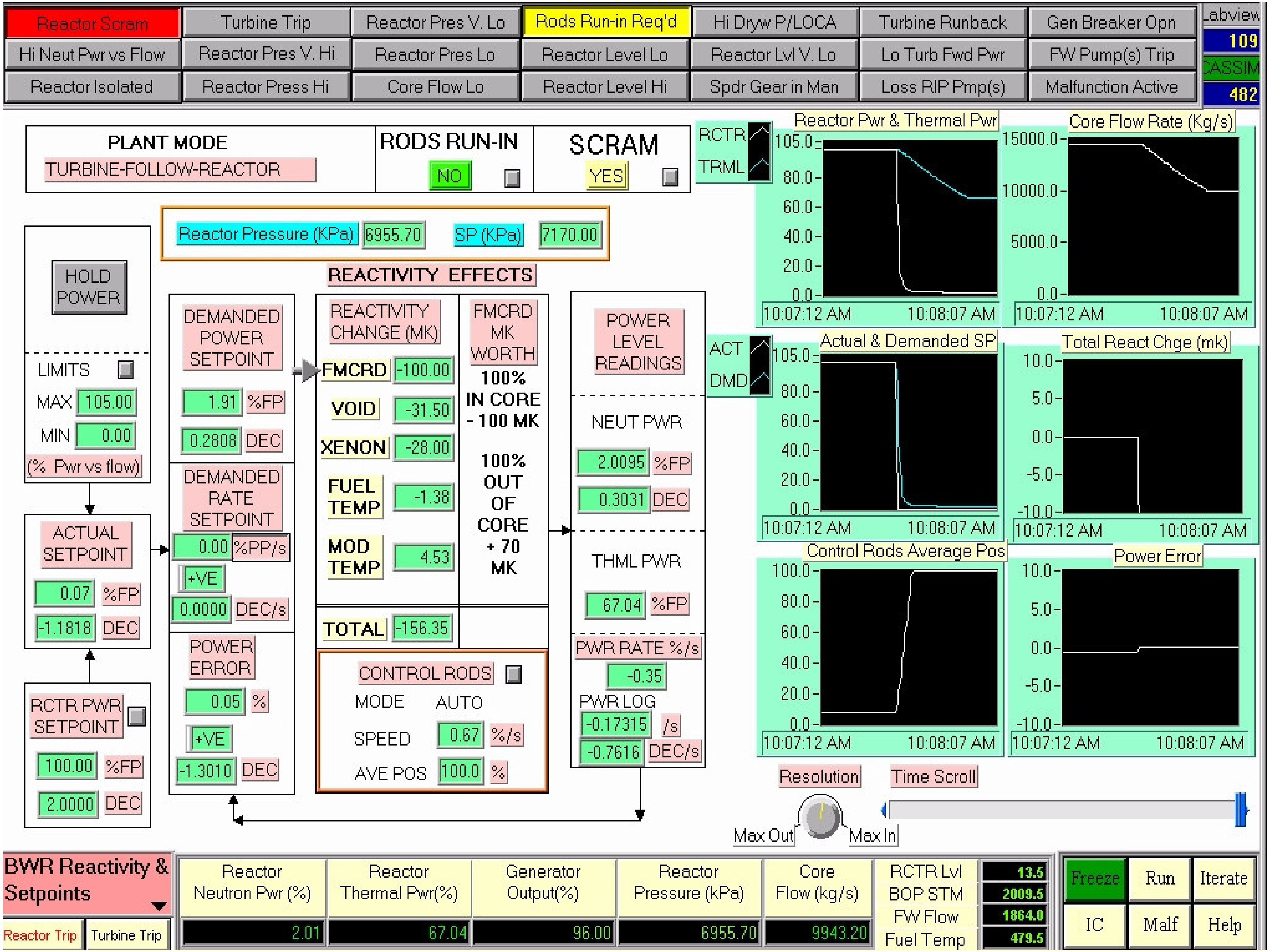
- Under normal plant start-up, load maneuvering, and shutdown, the operation path through **REGION IV** is recommended.
- In fact, the line which borders between Region I & IV, Region III & IV, the Blue region and Region IV is the "maximum power-flow" path to be followed for power increase and decreases and usually operation of the plant is "below" this "**maximum**" power-flow line.

# *Power /Flow Map*

- Limits are imposed to prevent operation in certain areas of the Power - Flow Map to maintain core thermal limits and to avoid operation above licensed power level - there are three measures to prevent that:
- Control Rods Withdrawal “Blocked” (if $>$ 105% ) ; Control Rods “Run-in” (if $>$ 110%); Scram ( if $>$ 113%).

# *Reactor Regulating System - Simulator*

- Power Error = Actual Power - Demanded Power
- If current power < 65 %, control rods moves "in" (+ve error) or "out" (-ve error) until power error = 0.
- If current power > 65 %. The new incremental demanded power setpoint signal is sent to the flow rate scheduler (flow = f (power)) 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.


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
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
109
CASSIM
482
PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN
NO
SCRAM
YES
Reactor Pressure (KPa)
6955.70
SP (KPa)
7170.00
REACTIVITY EFFECTS
HOLD POWER
LIMITS
MAX 105.00
MIN 0.00
(% Pwr vs flow)
ACTUAL SETPOINT
0.07 %FP
-1.1818 DEC
RCTR PWR SETPOINT
100.00 %FP
2.0000 DEC
DEMANDED POWER SETPOINT
1.91 %FP
0.2808 DEC
DEMANDED RATE SETPOINT
0.00 %PP/s
+VE
0.0000 DEC/s
POWER ERROR
0.05 %
+VE
-1.3010 DEC
REACTIVITY CHANGE (MK)
FMCRD -100.00
VOID -31.50
XENON -28.00
FUEL TEMP -1.38
MOD TEMP 4.53
TOTAL -156.35
FMCRD MK WORTH
100% IN CORE - 100 MK
100% OUT OF CORE + 70 MK
CONTROL RODS
MODE AUTO
SPEED 0.67 %/s
AVE POS 100.0 %
POWER LEVEL READINGS
NEUT PWR
2.0095 %FP
0.3031 DEC
THML PWR
67.04 %FP
PWR RATE %/s
-0.35
PWR LOG
-0.17315 /s
-0.7616 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
10:07:12 AM
10:08:07 AM
Resolution
Time Scroll
Max Out
Max In
BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.01
Reactor Thermal Pwr(%) 67.04
Generator Output(%) 96.00
Reactor Pressure (kPa) 6955.70
Core Flow (kg/s) 9943.20
RCTR Lvl 13.5
BOP STM 2009.5
FW Flow 1864.0
Fuel Temp 479.5
Freeze
Run
Iterate
IC
Malf
Help

# *Basic BWR Operation*

Plant Startup (cold start < 25 hours; hot start < 5 hours):

- Control Rods withdrawn to bring the reactor critical
- RPV heat-up & pressurization by further control rod withdrawals
- Initial power increase by continued rod withdrawals to a level where main turbine is synchronized
- Continued power increase using the control rod motion until the automatic flow control range is reached ~ 65 % FP.
- Reactor power is increased by increasing recirculation flow rate (65% - 100 % FP).
- Always operates in Turbine-Follow-Reactor Mode

# *Basic BWR Operation (cont'd)*

- Plant Shutdown: follow the reverse sequence of plant startup
- Reactor Shutdown Cooling - cool-down and decay heat removal is accomplished by bypassing steam to main condenser, and by the Residual Heat Removal System

# *BWR Load Following Capabilities*

- Load Regulation - 65 % to 100 % FP by automatic flow control; below 65 % FP by control rods motion.
- Frequency control - 1 to 10 % power change by automatic flow control.
- Load Shedding - automatic opening of turbine bypass valves, automatic flow reduction and control rod insertion.

# *Automatic Responses to Design Basis Events Accidents*

- Reactor Protection
- Containment Isolation
- ECCS actuation - detection of LOCA
- Suppression pool cooling and reactor scram on high pool temperature to mitigate inadvertent SRV opening event
- Other events - boron injection; feedwater flow runback, redundant actuation of scram; FMCRD run-in.

# *BWR Emergency Plant Operation*

- RPV Control - protection against extreme conditions on reactor water level, pressure, and power.
- Primary Containment Control - drywell temperature, pressure, hydrogen concentration.
- Secondary Containment Control - wetwell water level; temperature and radioactivity.
- Radioactivity Release Control - offsite radioactivity release controls.

# *Answer #1*

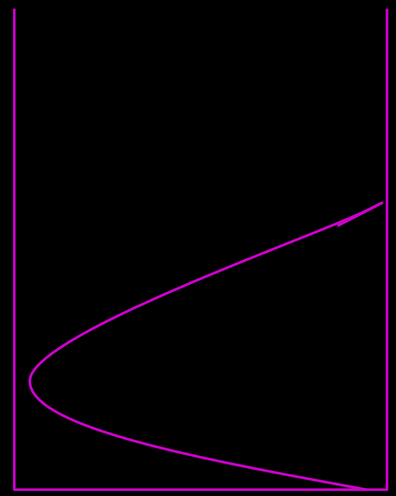
- Because in BWR, boiling core has steam bubbles entrained in light water coolant, which is also a moderator.
- Void in coolant has negative reactivity feedback - more steam bubbles, more void, more negative reactivity.
- Hence at high power, increasing the recirculation flow rate will reduce void density, thus less negative reactivity.

# *Answer #2*

- Neutron flux distribution in BWR core is a function of void fractions in core.
- Since voids are abound in the upper part of the core, the moderating power is highest in the non-boiling section of the core (lower part).
- This causes the peak neutron flux (power density) for a boiling core to shift from the center position towards the bottom of the core. Control rods entering rom the bottom can partially correct the skewed axial flux distribution

# Answer #2

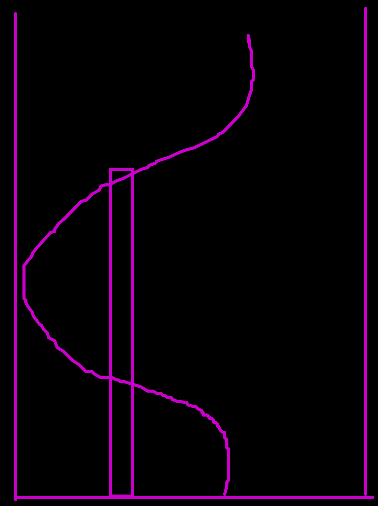
Core Height



Flux Shape

Before Control Rods entry from bottom

Core Height

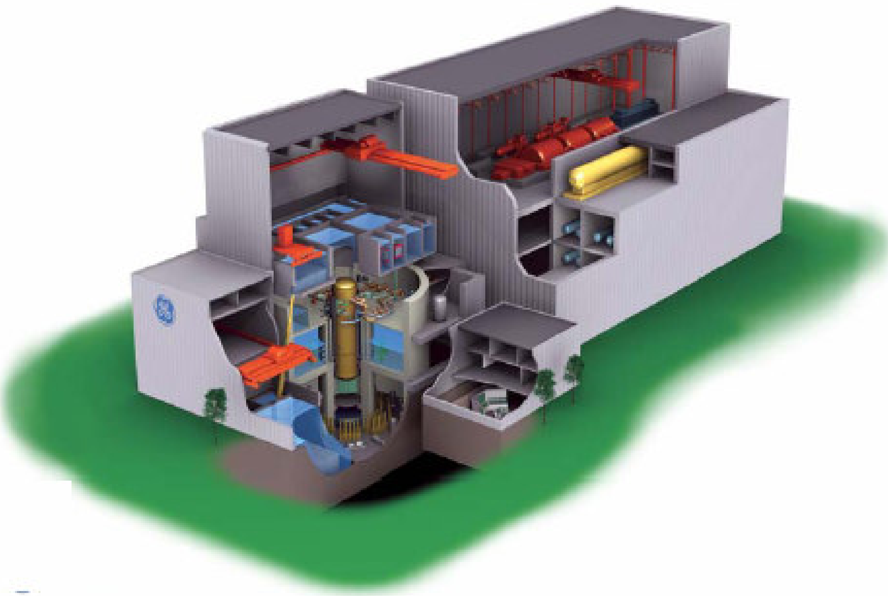


Flux Shape

After Control Rods entry from bottom

# GE's Generation III+…ESBWR

- **Design Highlights**
  - 1,550 MWe Boiling Water Reactor
  - Passive safety
  - Natural circulation
- **Key Benefits**
  - Reduced capital cost
  - Shorter construction period
  - Improved safety & security
  - Improved O&M costs



- **Status**
  - DOE 2010 awards completed May 2005
  - NRC design certification submission complete
  - NuStart, Entergy and Dominion select ESBWR

# *ICTP Workshop - BWR Modeling - Steady State*

Wilson Lam (wilson@cti-simulation.com)
CTI Simulation International Corp.
www.cti-simulation.com

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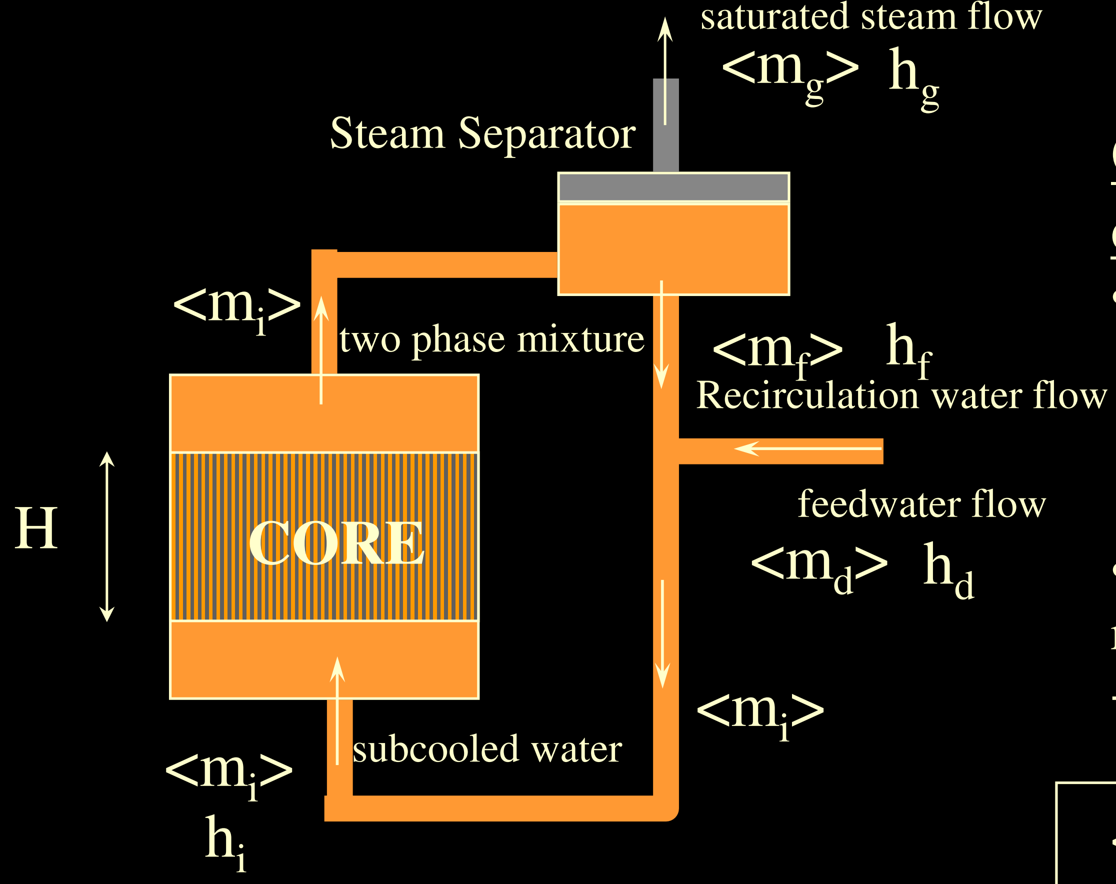
# *Boiling Reactor Mass & Heat Balance*




$\langle m \rangle = dm/dt$ = mass flow rate

$h$ = specific enthalpy

Overall mass in reactor core:

- steam flow = feedwater flow

$$\boxed{\langle m_g \rangle = \langle m_d \rangle \;\ldots(1)}$$

- subcooled water flow at reactor inlet = feedwater flow + recirculation flow

$$\langle m_i \rangle = \langle m_f \rangle + \langle m_d \rangle$$

or

$$\langle m_i \rangle = \langle m_f \rangle + \langle m_g \rangle$$

.....(2)

# *Boiling Reactor Mass & Energy Balance*

- The average exit quality by definition:

$$X = \langle m_g \rangle / (\langle m_g \rangle + \langle m_f \rangle) \quad \ldots.(3)$$

steam rate  recir water rate

FW rate

$$\text{or} \quad X = \langle m_d \rangle / (\langle m_d \rangle + \langle m_f \rangle)$$
$$= \langle m_d \rangle / \langle m_i \rangle = \langle m_g \rangle / \langle m_i \rangle \quad \ldots.(4)$$

steam rate  core flow rate

Recirculation Ratio = recirculation water / steam vapor produced

$$= \langle m_f \rangle / \langle m_g \rangle = (1 - X)/X \quad \ldots.(5)$$

$$\langle m_i \rangle = \langle m_f \rangle + \langle m_g \rangle = \langle m_f \rangle + (X /( 1 - X)) \langle m_f \rangle$$
$$= \langle m_f \rangle/(1 - X) \quad \ldots.(6)$$

# *Boiling Reactor Mass & Energy Balance*

- Energy Balance at reactor inlet:

$$\langle m_i \rangle .\, h_i = \langle m_f \rangle .h_f + \langle m_d \rangle .\, h_d$$

$$h_i = (1 - X) \,.\, h_f + X.\, h_d \qquad \ldots\ldots(7)$$

$$\text{or } X = (h_f - h_i) \,/\, (h_f - h_d)$$

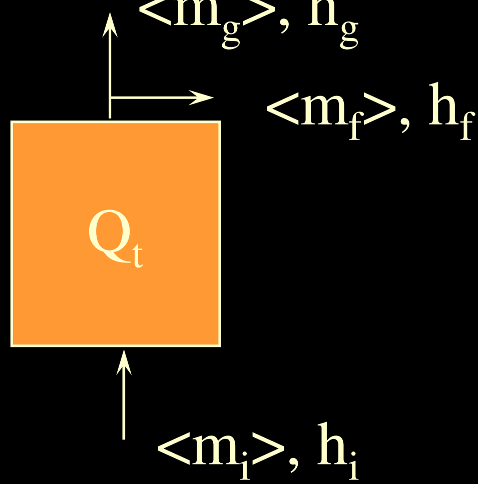
- Energy Balance at the core

$$\langle m_g \rangle .h_g + \langle m_f \rangle .h_f = Q_t + \langle m_i \rangle .h_i$$

$$\boxed{Q_t = \langle m_i \rangle .\, [(h_f + X.\, h_{fg}) - h_i] \;\ldots\ldots\; (8)}$$

where $h_{fg} = h_g - h_f$

$$\text{or } \boxed{Q_t = \langle m_g \rangle \,(h_g - h_d) \;\ldots\ldots\ldots(9)}$$



# *Exercises -BWR Modeling*

- Derive Equation (8) & (9)

# *BWR Spreadsheet Model*

- Given the following data:
  - Technical Data for US version of ABWR (see BWR Simulator Manual)
  - Technical Data for ABWR Power Flow Map (see Binder - Miscellaneous Section)
  - Technical Data for Available Energy for condensing turbine (see BWR Simulator Manual)

# *BWR Spreadsheet Model*

- Create an EXCEL spreadsheet (BWR)
- Column A's name is % FP - put in numbers 100%, 90%, 80%, 70%, 60%, 50%, 40%, 30%, 20%, 10%, 5%, 3%, 0%
- Column B's name is MW - first cell is MW(gross) at 100% FP (from AWBR Spec. sheet) = 1385 MW. Compute the rest of the cells in Column B using % numbers in Column A.
- Column C's name is KBTU/hr - to convert MW to KBTU/hr, multiply the cells in Column B by 3413. Compute all the cells in Column C using this energy conversion.

# *BWR Spreadsheet*

- Column D's name is Steam Flow (Kg/s) - to compute steam flow for the BWR plant in column D cells -
  - first find the available energy BTU per lb of steam from technical data chart for condensing turbine given. Note inlet steam pressure 6.8 Mpa = 1000 Psia; backpressure of 11.75 Hg = 3 in backpressure; inlet steam temp 284 deg. C = 543 deg. F
  - Multiply this number - BTU per lb of steam, by the efficiency of the turbine ~ assume 74 %, to get the "actual" BTU/lb for this turbine.
  - Divide the C Column's number (KBTU/hr) by actual BTU/lb to get Klb /hr.
  - Multiply this number by 0.126 to convert Klb/hr to Kg/s
  - To check your result, according to ABWR data spec., the 100 FP steam flow is 2122 Kg/s. You may have to adjust turb. eff.

# *BWR Spreadsheet*

- Column E's name is Core Flow (Kg/s).
- Enter the first cell = 14502 (from ABWR Spec.)
- Column F's name is Core Flow (%) - enter the % numbers to match ABWR Power/Flow Map (given data), following the typical startup path - e.g. 100%FP - 100 % coreflow; 90 %FP - 80 % coreflow; 70 % FP - 65% coreflow, …. etc.
- After all % numbers are entered for all cells in Column F, compute the coreflow (Kg/s) in all the remaining cells in Column E.
- Column G's name is Quality X - calculate X using other columns' cells values.
- Column H's name is Recirculation Flow (Kg/s) - calculate recirculation flow using other columns' cells values.

# *BWR Spreadsheet*

- Plot a curve for the Quality X versus Power (%)
- Comment on the Quality values as power increases.
- If you are to design a reactor power control system, using Control Rods, and other means, how would you do it ?

# *BWR Spreadsheet (cont'd)*

- Enter cell A23 name = Reactor Pressure; enter B23 value = 7.17 (as per ABWR spec.)
- Enter cell A24 name = Sat. Coolant Enthalpy hf (KJ/Kg)
- Enter cell B24 formula = 373.7665*POWER(B23,0.4235532)+415
- Enter cell A25 name = Sat. Vapor Enthalpy hg (KJ/Kg)
- Enter cell B25 formula = -0.9219176*POWER((B23-9), 2) -16.38835*(B23-9)+2742.03
- Now, enter Column I's name as Reactor Thermal Power (MWt). Compute Reactor Thermal Power values in Column I, using values in cell B24 - hf; cell B25 - hg; and other column values. Use Feedwater Enthalpy at given temp = 932.007 kJ/kg
- Verify your calculation for 100 % reactor thermal power using data in the ABWR Spec.(3926 $MW_{th}$)

# *Solutions: derivation of equation (8) & (9)*

- $<m_g>.h_g + <m_f>.h_f = Q_t + <m_i>.h_i$
- $Qt = <m_g>.h_g + <m_f>.h_f - <m_i>.h_i$

  $= <m_i>[(<m_g>/<m_i>).h_g + (<m_f>/<m_i>).h_f - h_i]$

  $= <m_i>[X.\ h_g + (1\text{-}X)\ .\ h_f - h_i]$ (using equation (4) & (6))

  $= <mi>[h_f + X\ .\ (h_g - h_f) - h_i]$, hence equation (8)
- $Qt = <m_g>.h_g + <m_f>.h_f - <m_i>.h_i$

  $= <m_g>.h_g + <m_f>.\ h_f - <m_f>.h_f - <m_d>.\ h_d$ (using (7))

  $= <m_g>\ (h_g - h_d)$, hence equation (9)

# *ICTP Workshop - BWR Modeling - Dynamic*

Wilson Lam (wilson@cti-simulation.com)
CTI Simulation International Corp.
www.cti-simulation.com

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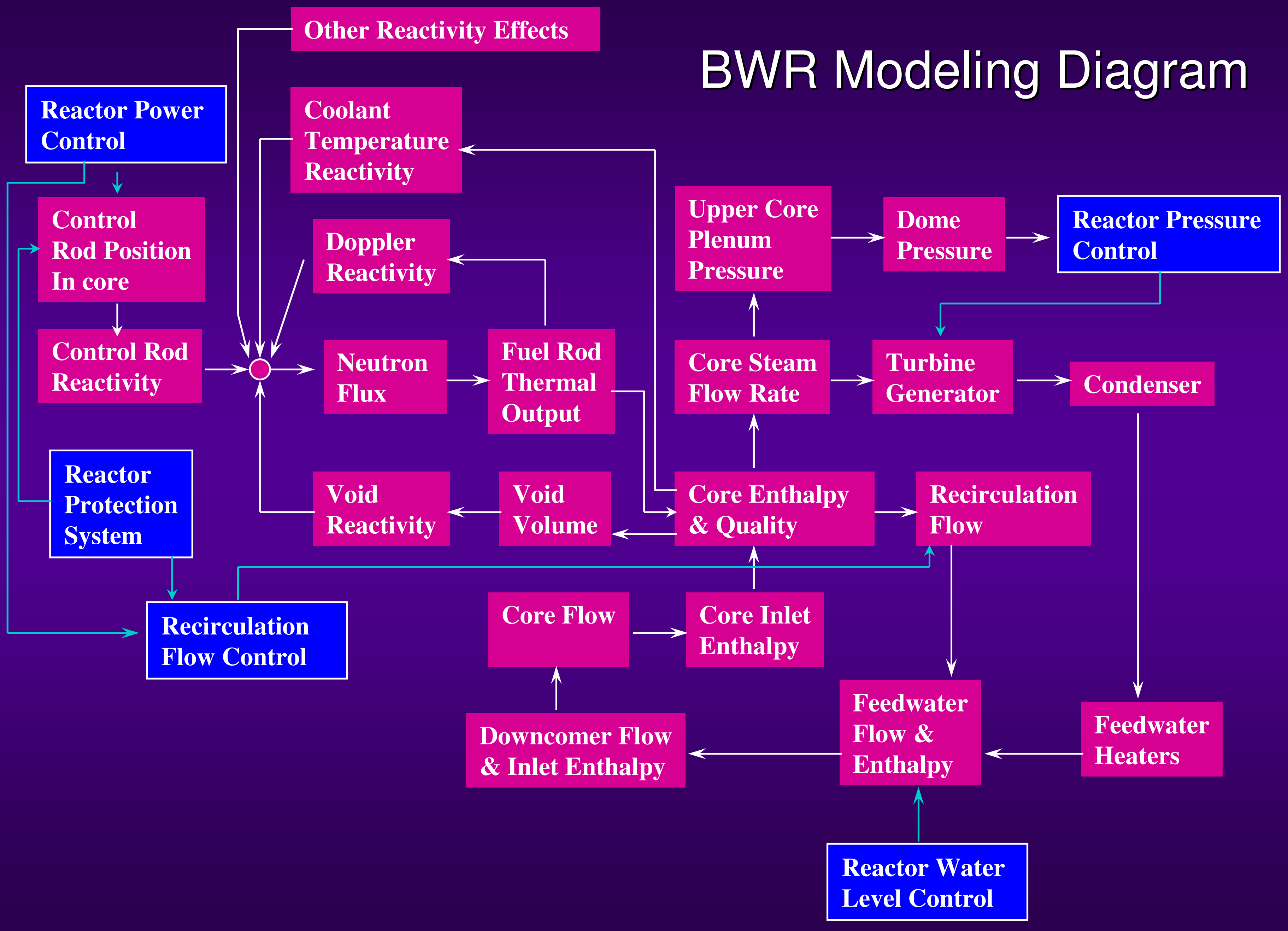
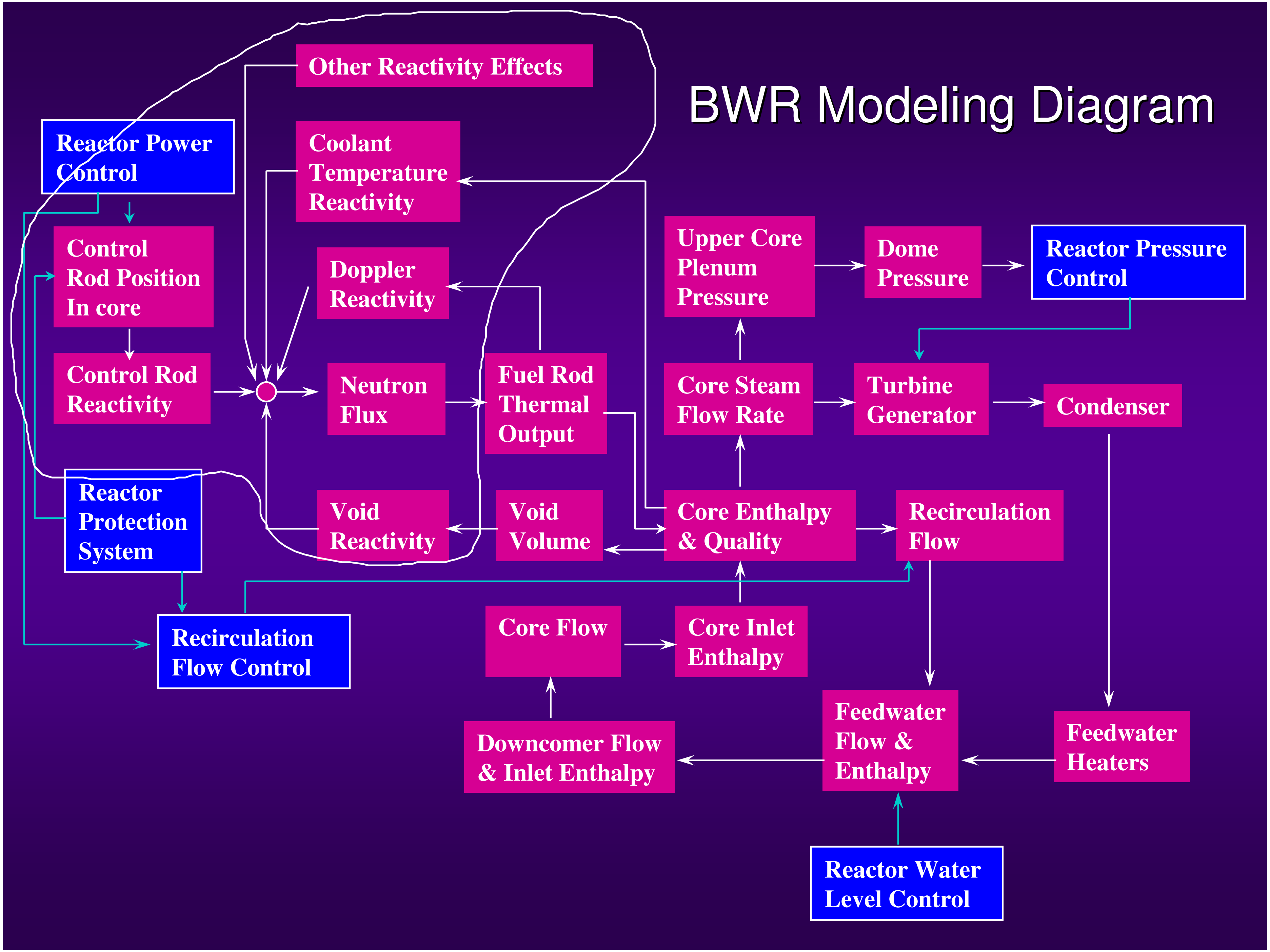
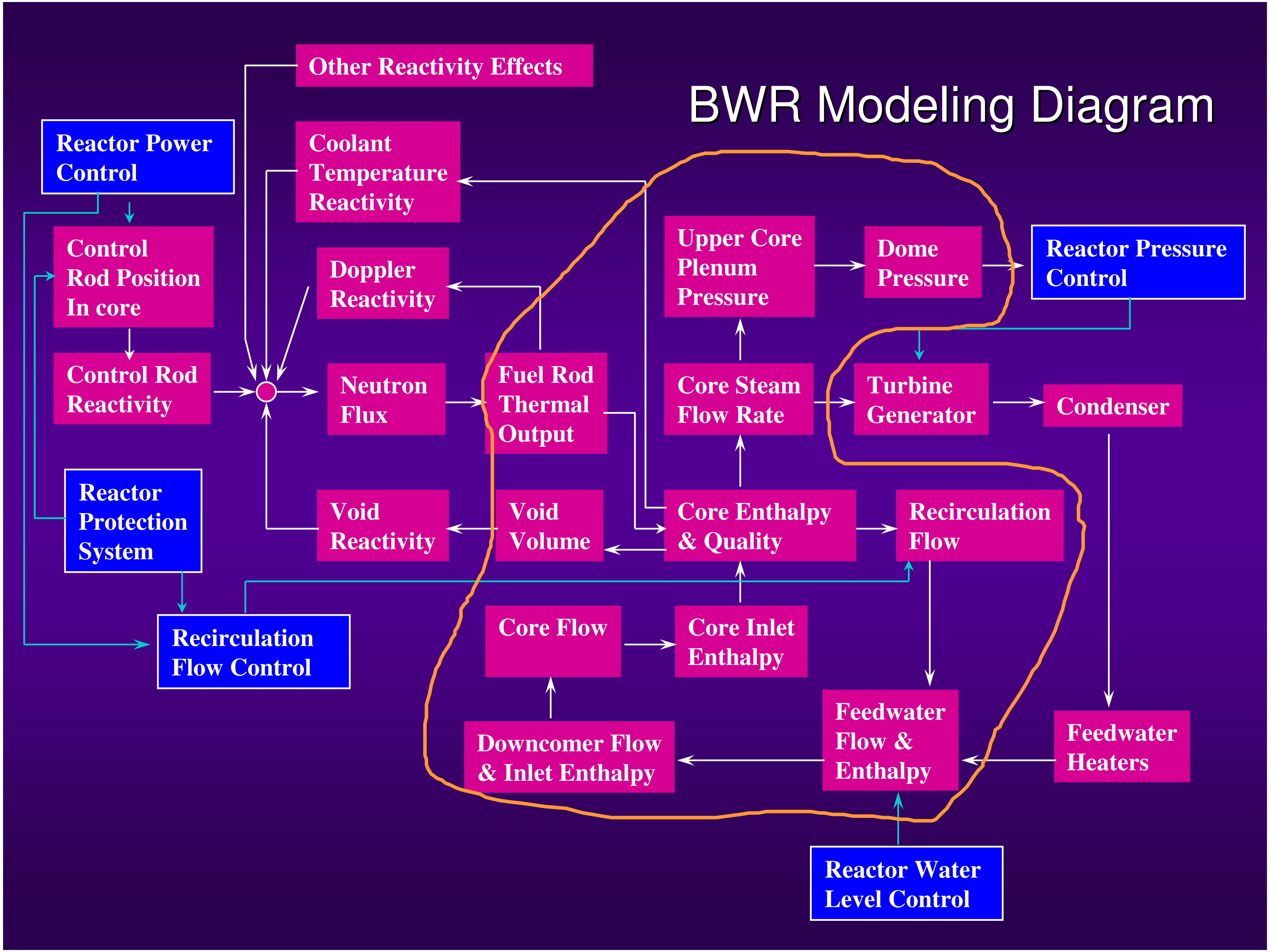
# BWR Modeling Diagram




BWR Modeling Diagram
Other Reactivity Effects
Reactor Power Control
Coolant Temperature Reactivity
Control Rod Position In core
Doppler Reactivity
Upper Core Plenum Pressure
Dome Pressure
Reactor Pressure Control
Control Rod Reactivity
Neutron Flux
Fuel Rod Thermal Output
Core Steam Flow Rate
Turbine Generator
Condenser
Reactor Protection System
Void Reactivity
Void Volume
Core Enthalpy & Quality
Recirculation Flow
Recirculation Flow Control
Core Flow
Core Inlet Enthalpy
Downcomer Flow & Inlet Enthalpy
Feedwater Flow & Enthalpy
Feedwater Heaters
Reactor Water Level Control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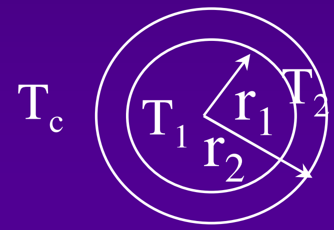
Feedwater Flow & Enthalpy
Feedwater Heaters
Downcomer Flow & Inlet Enthalpy
Reactor Water Level Control

# *Fuel heat Transfer*

*Lumped Parameter Technique for heat transfer from UO2 fuel rods :*

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

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




where

$Q'_n$ = nuclear heating of fuel rod (BTU/sec.ft)

$C_1$ = thermal capacity for fuel pellet (BTU/deg. F.ft) = $\pi r_1^2 c_{p1} \rho_1$

$C_2$ = thermal capacity of clad (BTU/deg. F.ft) = $2\pi r_2(\Delta r) c_{p2}\rho_2$

$R_1$ = resistance of $UO_2$ and gap (sec ft deg. F/BTU) = $\dfrac{1}{4\pi k_1} + \dfrac{1}{2\pi r_1 h_g}$

$k_1$ is UO2 thermal conductivity;

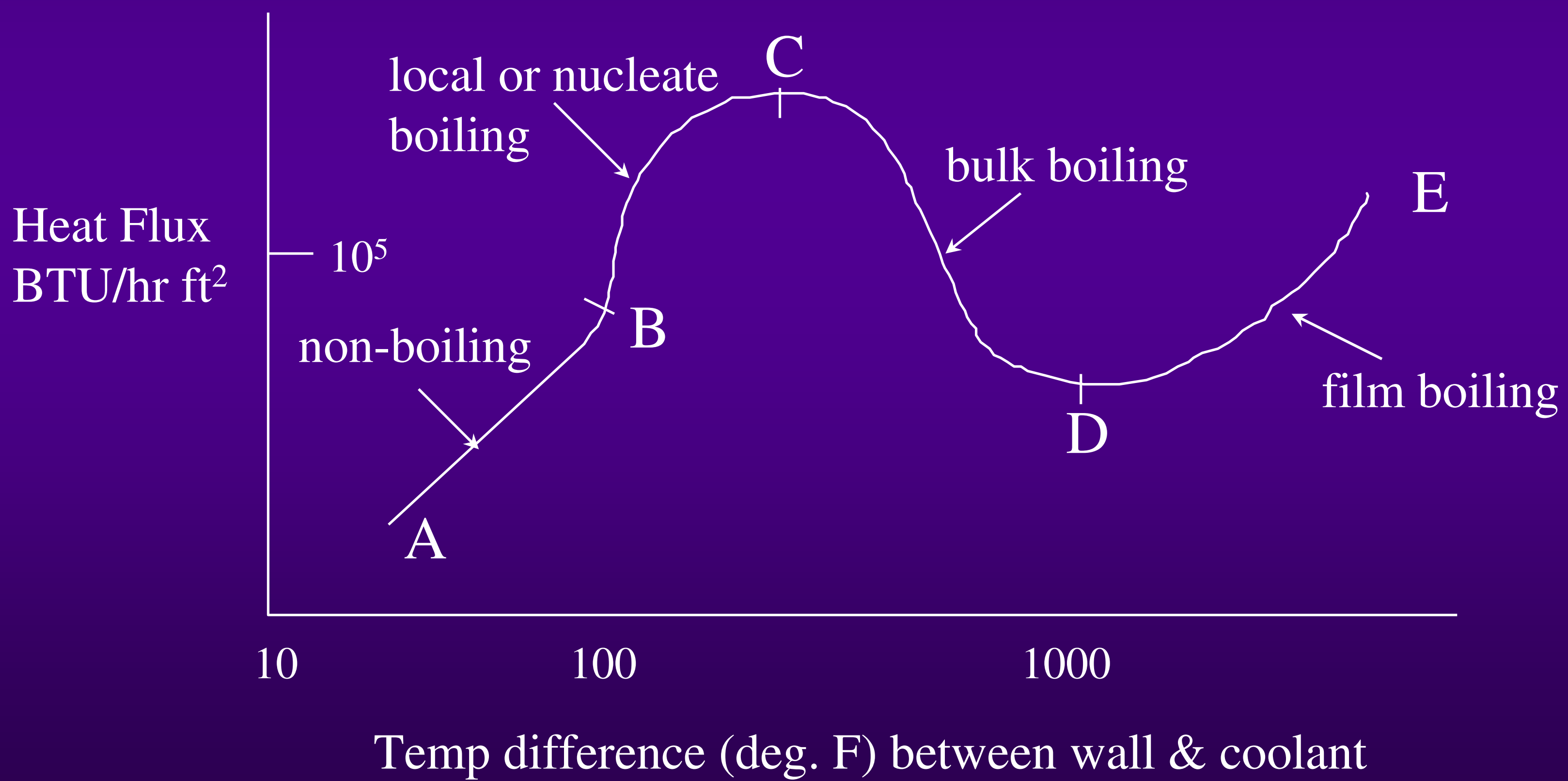
$h_g$ is gap conductance

$T_1$ = average pellet temp (deg. F) ; $T_2$ =average clad temp (deg. F)

$T_c$ = bulk coolant temp (deg. F)

# Heat Transfer to Coolant

- Boiling Heat Transfer Characteristics



# *Heat Transferred to Coolant*

- A-B: non-boiling; heat transfer by single phase convection.
- B-C: local or nucleate boiling; heated surface temp. exceeds sat. temp by few degrees; bubbles formed; large increase in heat flux due to mixing of liquid by bubbles.
- C-D: bulk boiling; heated surface blanketed by unstable, irregular film in violent motion. Heat transfer by conduction and radiation - hence heat flux decreases substantially.
- D-E: film boiling or burnt-out. At D, film becomes stable, and heat transfer improves as the surface gets hotter. However, very high temperature reached with high heat flux in this region, usually resulting in the destruction of the fuel or sheath - BURNT-OUT

***BWR operates in B-C nucleate boiling region, away from C.***

# *Quiz*

- The fuel element temperature in direct cycle BWR is (lower/higher) for the SAME steam conditions in indirect cycle (e.g. PWR) - why ?
- The direct cycle BWR can be operated at a much (higher/lower) pressure than that required to prevent boiling in the indirect cycle NPP using water as a heat transport fluid. What implications ?

# *Nucleate Boiling Heat Transfer*

- Thom's nucleate boiling heat transfer at pressures from 750 to 2000 psia:

$$(T_w - T_{sat}) = 0.7123 \frac{\sqrt{q''}}{e^{\left(\frac{P}{8690}\right)}} \quad \text{.....(3)}$$

where $T_w$ = fuel wall temperature (deg. C)
$T_{sat}$ = saturation temperature (deg. C)
$q''$ = heat flux (MW/m2)
P = pressure (Kpa)

# Average Fuel Energy Equation

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

where

$\rho_f$ = volume average fuel density

$V_f$ = fuel volume in one zone

$C_f$ = average fuel specific heat capacity

$T_f$ = average fuel temperature

$T_c$ = average coolant temperature

P = reactor power

U = overall heat transfer coefficient (Thom's nucleate boiling)

A = overall heat transfer area for fuel channel

# Average Core Coolant Energy Equation

The average core coolant energy equation is given by:

$$\rho_c V_c \frac{dh_o}{dt} = W_i h_i - W_o h_o + X.h_{fg} + UA(T_f - T_c) \quad \ldots\ldots\ldots\ldots\ldots (6.4\text{-}3)$$

where

$\rho_c$ = volume average coolant density

$V_c$ = coolant volume in one zone

$h_i$ = average coolant specific enthalpy at inlet of the core

$h_o$ = average coolant specific enthalpy at outlet of the core

A = overall heat transfer area for fuel channel

U = overall heat transfer coefficient. In the non-boiling region, the Dittus-Boetler correlation for forced convection is used, which is proportional to the (coolant flow)$^{0.8}$. In the boiling region, the heat transfer coefficient correlation is derived from Thom's nucleate boiling (equation 6.4-1)."

$T_f$ = average fuel 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

X = quality of coolant

$h_{fg}$ = latent heat of vaporization = hg – hf

# Boiling Core Dynamics

P - Dome Pressure
$H_{dc}$ - Enthalpy of fluid at downcomer
$W_{dc}$ - Downcomer flow

Sat. liq. enthalpy $H_f = f(P)$ ...(4)

Sat. Steam enthalpy $H_g = f(P)$ ...(5)

Latent heat of vap. $H_{fg} = H_g - H_f$ ....(6)

Sat. liq. density $\rho_{sat} = f(P)$ ....(7)

2 phase core exit enthalpy $H_{core} = H_{dc} + \dfrac{Q_t}{W_{dc}}$ .....(8)

Quality $X = \dfrac{H_{core} - H_f}{H_{fg}}$ ....(9)

Void Fraction $\alpha = \dfrac{1}{1 + (\frac{1-X}{X})\psi}$ where $\psi = \dfrac{\rho_g}{\rho_f} S, \; S = slip\ ratio$ ...(10)

Heat Generated from the core: $Q_t = W_{dc}(H_f + X.H_{fg} - H_{dc})$ ....(11)

Mass balance at dome: $$\frac{dV_w}{dt} = \frac{1}{\rho_f}((1-X).W_r - W_{dc} + W_{RH} + W_{FW})$$ …(12)

$V_w$= fluid vol. in dome
$W_r$= core flow
$W_{dc}$ = downcomer flow
$W_{RH}$= reheater drains flow
$W_{FW}$= feedwater flow

Dome water level: $$L_d = f(V_w)$$ …(13)

Energy balance at dome:

$$\frac{dH_d}{dt} = \frac{1}{\rho_f.V_w}[(1-X).W_r.(H_f - H_d) + W_{RH}(H_{RH} - H_d) + W_{FW}(H_{FW} - H_d)]$$ …(14)

$H_d$= fluid enthalpy at dome after mixing with feedwater
$H_{fw}$= feedwater enthalpy

$W_{dc}$ = downcomer flow
$W_{RH}$= reheater drains flow
$W_{FW}$= feedwater flow

Calculation of sat. steam density:

$$\frac{d\rho_g}{dt} = \frac{X.W_r - W_s + \rho_g.\frac{dV_w}{dt}}{V_D - V_w + V_{SM} + V_r.\alpha}$$

…(15)

$\rho_g$ = sat. steam enthalpy
$W_s$= steam flow from dome
$V_D$= volume of dome
$V_{SM}$=volume of steam main
$V_r$ = liq. vol. of core
$\alpha$ = void fraction in core

Calculation of Dome Pressure:

$$P = f(\rho_g)$$ ….(16)

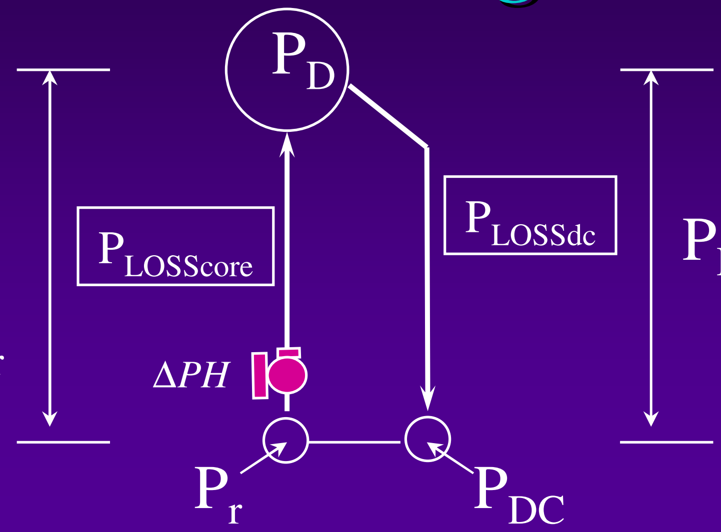
# *Driving Pressure in Boiling Core*

In steady state,

$$P_{DC} = P_D + P_{ELdc} - P_{LOSSdc} \quad \ldots(17)$$

$$P_D = P_r - P_{LOSScore} - P_{ELr} + \Delta PH$$

.......(18)




re-writing (18),

$$P_r = P_D + P_{LOSScore} + P_{ELr} - \Delta PH \quad \ldots\ldots(19)$$

Equating (17) & (19),

$$\underbrace{(P_{ELdc} - P_{ELr})}_{\Delta P_{EL}} + \Delta PH = (P_{LOSSdc} + P_{LOSScore}) \quad \ldots..(20)$$

Note: $\Delta P_{EL} = g.Z_{EL}.(\rho_{dc} - \rho_r)$ ....(21) $\rho_{dc}$, $\rho_r$ = mean fluid density at downcomer & core

where g = 0.00981 KPa/(Kg/m$^2$) - conversion constant from Kg/m$^2$ to Kpa;

$Z_{EL}$= elevation (m) of dome from bottom of reactor pressure vessel (RPV).

*Observation : if sum of pressure losses < pressure difference due to fluid densities in core & downcomer, natural circulation can be sustained, without circulation pump. Otherwise forced circulation is required with circulation pump.*

# Recirculation Flow & Pressure Losses

Applying Navier Stokes Equations of motion for an incompressible fluid,

$$\frac{dW_{dc}}{dt} = \frac{g_c.A_{dc}}{g.Z_{EL}}(\Delta PH + \Delta P_{EL} - P_{LOSSdc} - P_{LOSScore}) \quad \ldots..(22)$$

where $g_c$ = gravitational constant, 9.81 m/s$^2$

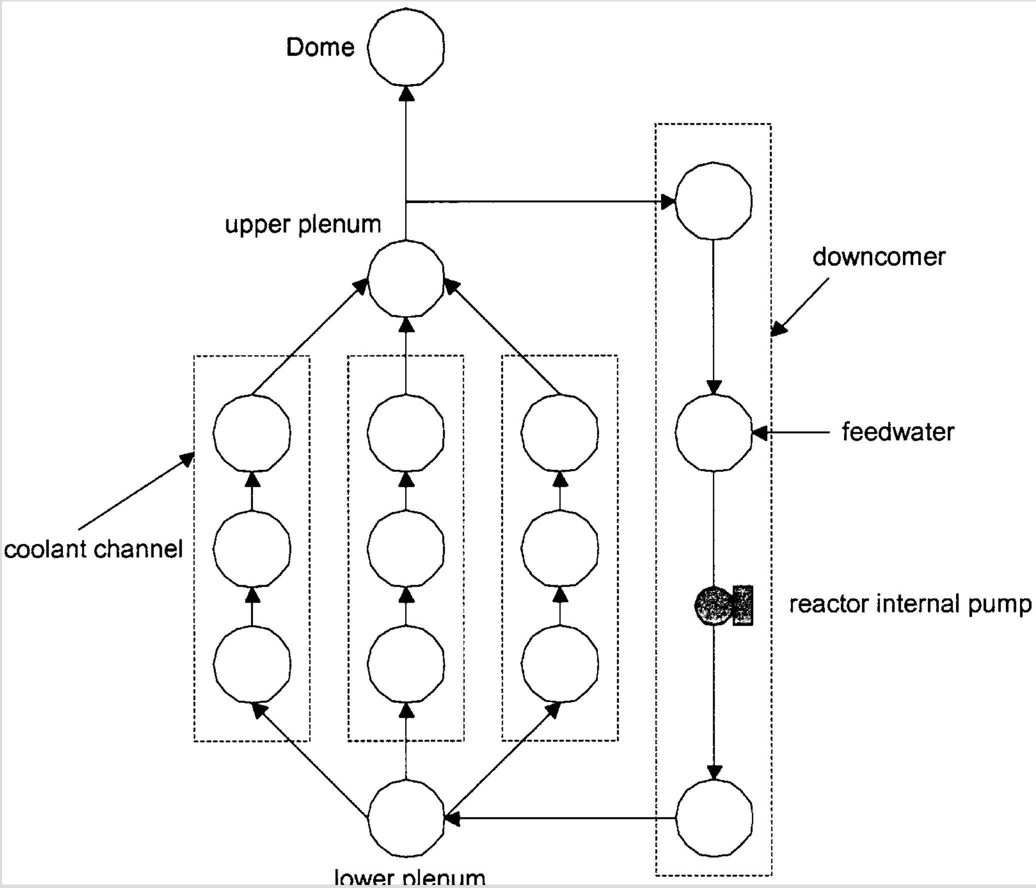
$A_{dc}$= cross-sectional area of downcomer section (m$^2$)

Pressure Losses calculation - important for reactor design

- Sum of frictional pressure losses in core and downcomer all computed in the flow direction.
- Sum of acceleration pressure losses
- Sum of pressure losses due to area contractions and expansions
- Consider all single phase & two phase flow losses in the calculations

*Reference: Nuclear Heat Transport - El. Wakil, ISBN 0-7002-2309-6*

# *Flow Network for Core Hydraulics (one phase & two Phase flow)*



# Boiling Boundary

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

# *Model Summary*

- Divide core into number of lumped channels
- Each lumped channel divided vertically into nodes (or zones) - the nodalization fineness depends on application.
- Each coolant channel node is assumed to have its own coolant flow, its own lumped fuel element

# *Model Summary - continued*

- Fuel heat transfer to coolant calculations start with lowest nodes, with nodes coolant inlet temperatures derived from the core lower plenum temperatures, and with coolant flows derived from hydraulic flow network at the lower plenum

## *Model Summary - continued*

- After obtaining the lowest node coolant outlet temperatures and average fuel temperatures, the calculations proceed to the next higher nodes, and so forth...
- A program check is performed in each node to see if coolant outlet enthalpy exceeds saturated coolant enthalpy at the prevailing pressure. If so, 2 phase flow techniques will be used.

# *Model Summary - continued*

- At the core exit upper plenum, the coolant temperatures from all lumped channels are mixed by the flow turbulence to determine the average coolant mixing temperatures at the upper plenum.

# ICTP Workshop - BWR Simulator Exercises

Wilson Lam (wilson@cti-simulation.com)
CTI Simulation International Corp.
www.cti-simulation.com

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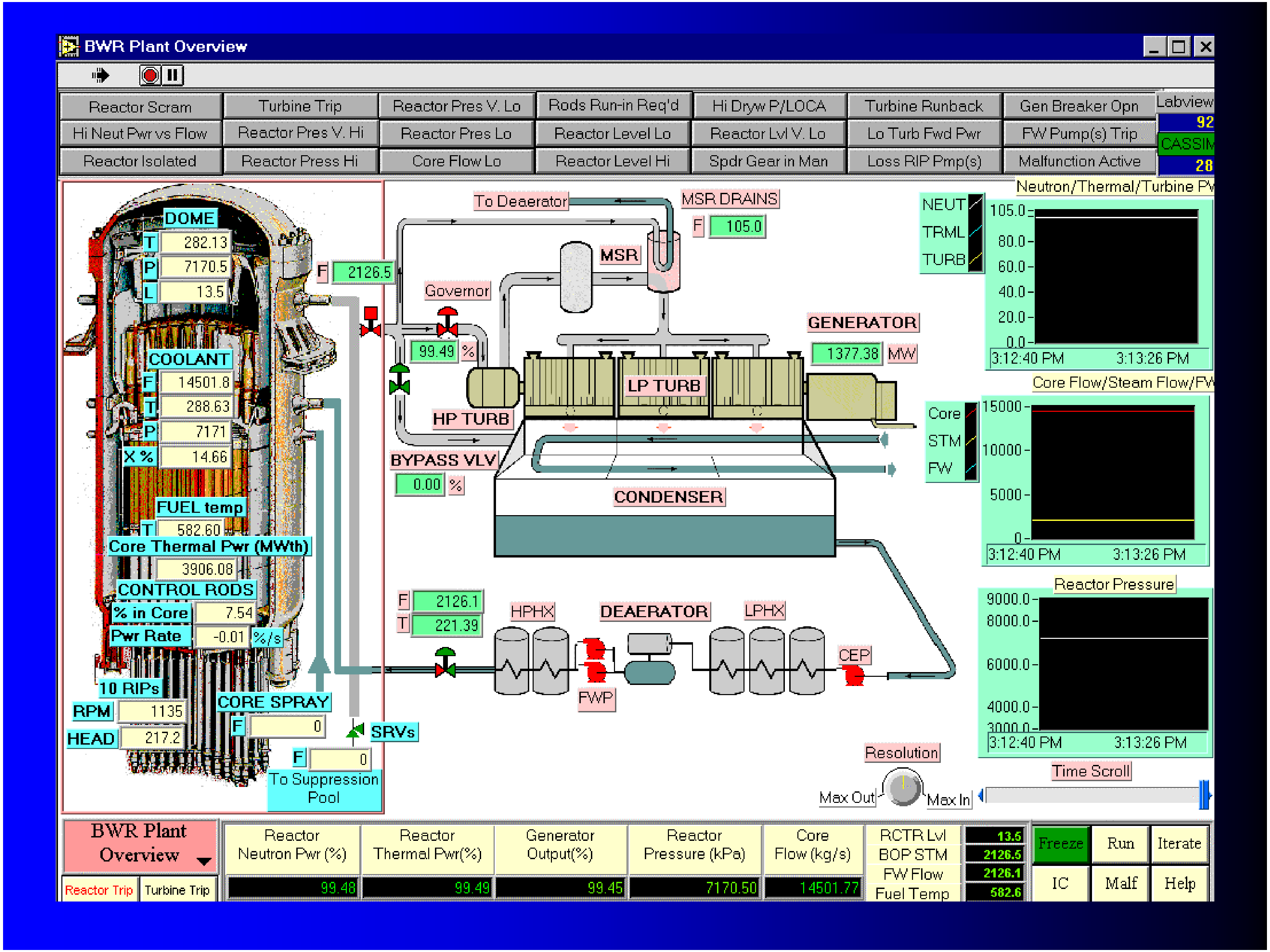
# BWR Simulator Familiarization

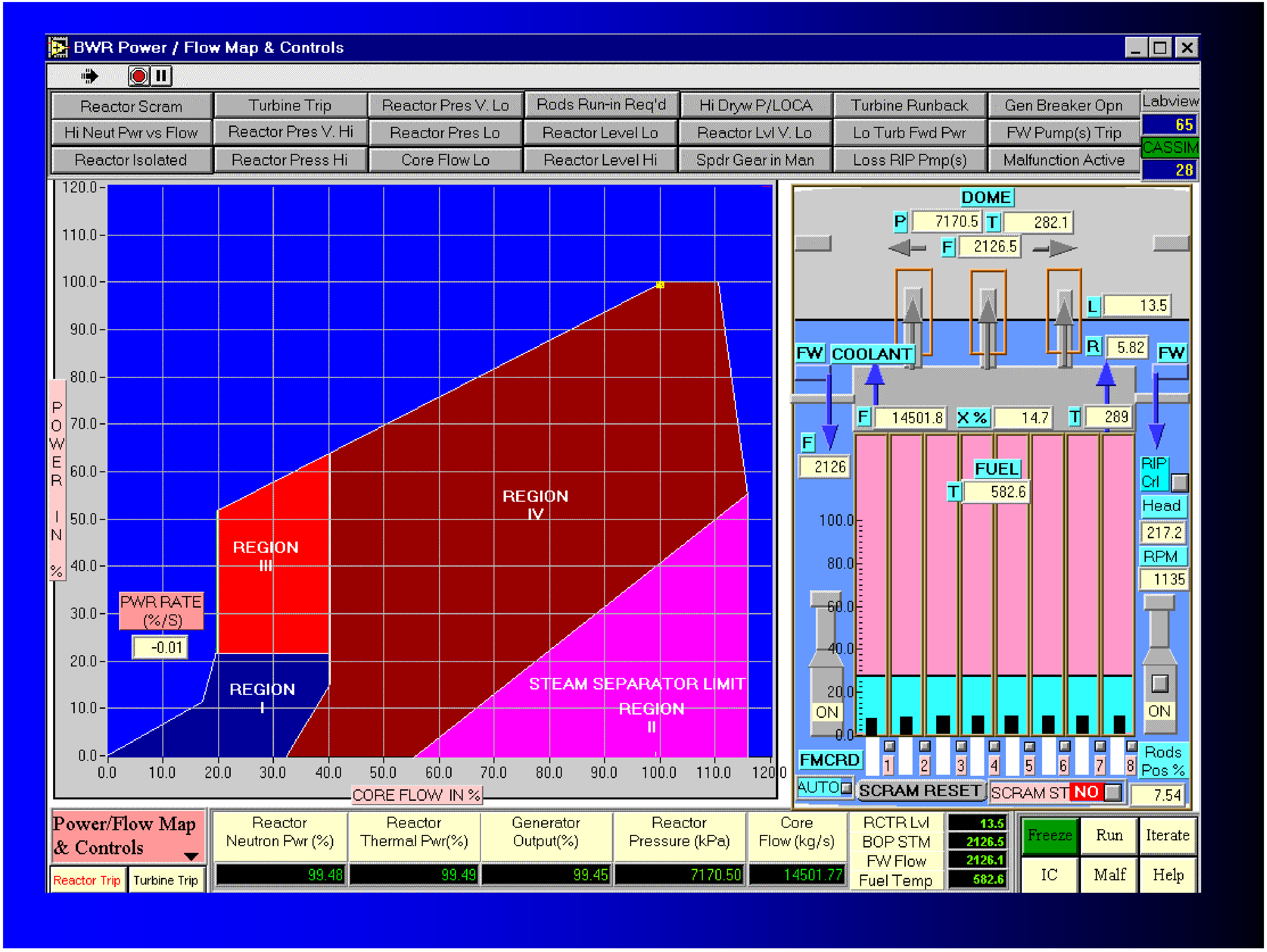
- BWR Simulator Manual.
- Practice BWR Simulator Startup, Initialization.
- Review BWR screens. Note the “hot” buttons on the screens, which bring up control pop-ups for user’s interactions.



BWR Plant Overview
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
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
DOME
T 282.13
P 7170.5
L 13.5
COOLANT
F 14501.8
T 288.63
P 7171
X % 14.66
FUEL temp
T 582.60
Core Thermal Pwr (MWth)
3906.08
CONTROL RODS
% in Core 7.54
Pwr Rate -0.01 %/s
10 RIPs
RPM 1135
HEAD 217.2
CORE SPRAY
F 0
SRVs
F 0
To Suppression Pool
F 2126.5
Governor
99.49 %
HP TURB
BYPASS VLV
0.00 %
To Deaerator
MSR DRAINS
F 105.0
MSR
GENERATOR
1377.38 MW
LP TURB
CONDENSER
F 2126.1
T 221.39
HPHX
DEAERATOR
LPHX
CEP
FWP
NEUT
TRML
TURB
Neutron/Thermal/Turbine PW
105.0
80.0
60.0
40.0
20.0
0.0
3:12:40 PM
3:13:26 PM
Core Flow/Steam Flow/FW
Core
STM
FW
15000
10000
5000
0
3:12:40 PM
3:13:26 PM
Reactor Pressure
9000.0
8000.0
6000.0
4000.0
3000.0
3:12:40 PM
3:13:26 PM
Resolution
Max Out
Max In
Time Scroll
BWR Plant Overview
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 99.48
Reactor Thermal Pwr(%) 99.49
Generator Output(%) 99.45
Reactor Pressure (kPa) 7170.50
Core Flow (kg/s) 14501.77
RCTR Lvl 13.5
BOP STM 2126.5
FW Flow 2126.1
Fuel Temp 582.6
Freeze
Run
Iterate
IC
Malf
Help

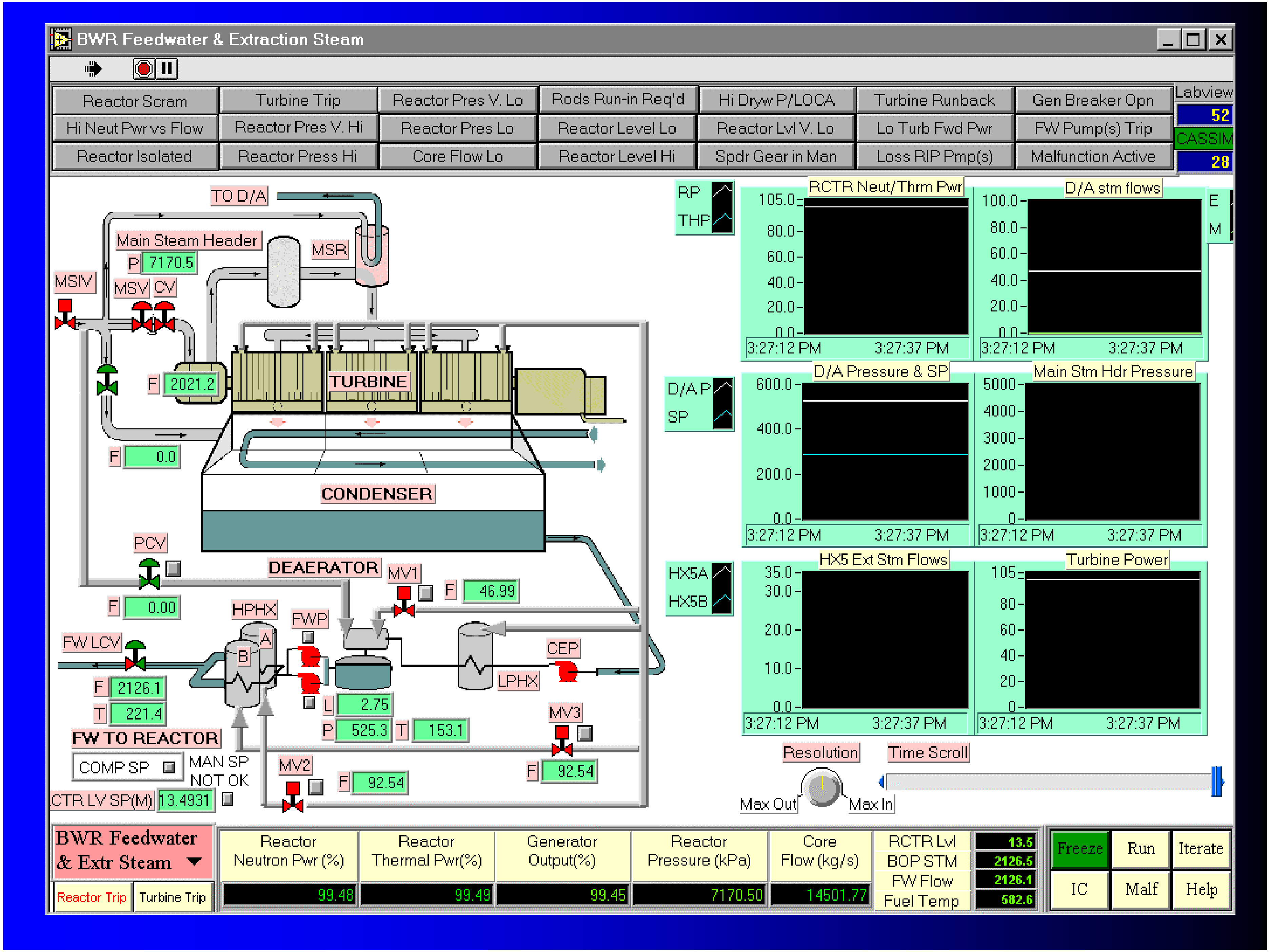
# Reactor Core

| 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<br>• Decay heat (3 groups)<br>• Reactivity feedback effects - void, xenon, fuel temperature, moderator temperature<br>• 2 phase flow & heat transfer<br>• Reactivity control rods<br>• Essential control loops - Reactor Pressure Control; Core Recirculation Flow Control; Reactor Power Regulation; Reactor Water Level Control; Turbine Load/Frequency Control | • Plant Overview<br>• BWR Reactivity & Setpoints<br>• BWR Power /Flow Map & Controls | • Reactor power and rate of change (input to control computer)<br>• Manual control of control rods<br>• Reactor scram<br>• Manual Control Rods “run-in”<br>• Manual control of core recirculation flow rate<br>• Manual adjustment of reactor water control level setpoint | • Increasing and decreasing core flow due to Flow Control malfunctions<br>• Inadvertent withdrawal of one bank of control rods<br>• Inadvertent insertion of one bank of control rods<br>• Inadvertent reactor isolation<br>• Power loss to 3 Reactor Internal Pumps (RIPs)<br>• Reactor bottom break |


BWR Power / Flow Map & Controls
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
65
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
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
28
POWER IN %
CORE FLOW IN %
REGION I
REGION II
REGION III
REGION IV
STEAM SEPARATOR LIMIT
PWR RATE (%/S)
-0.01
DOME
P 7170.5
T 282.1
F 2126.5
L 13.5
R 5.82
FW
COOLANT
F 14501.8
X % 14.7
T 289
F 2126
FUEL
T 582.6
RIP Crl
Head 217.2
RPM 1135
ON
FMCRD
Rods Pos %
AUTO
SCRAM RESET
SCRAM ST NO
7.54
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 99.48
Reactor Thermal Pwr(%) 99.49
Generator Output(%) 99.45
Reactor Pressure (kPa) 7170.50
Core Flow (kg/s) 14501.77
RCTR Lvl 13.5
BOP STM 2126.5
FW Flow 2126.1
Fuel Temp 582.6
Freeze
Run
Iterate
IC
Malf
Help

# Steam & Feedwater

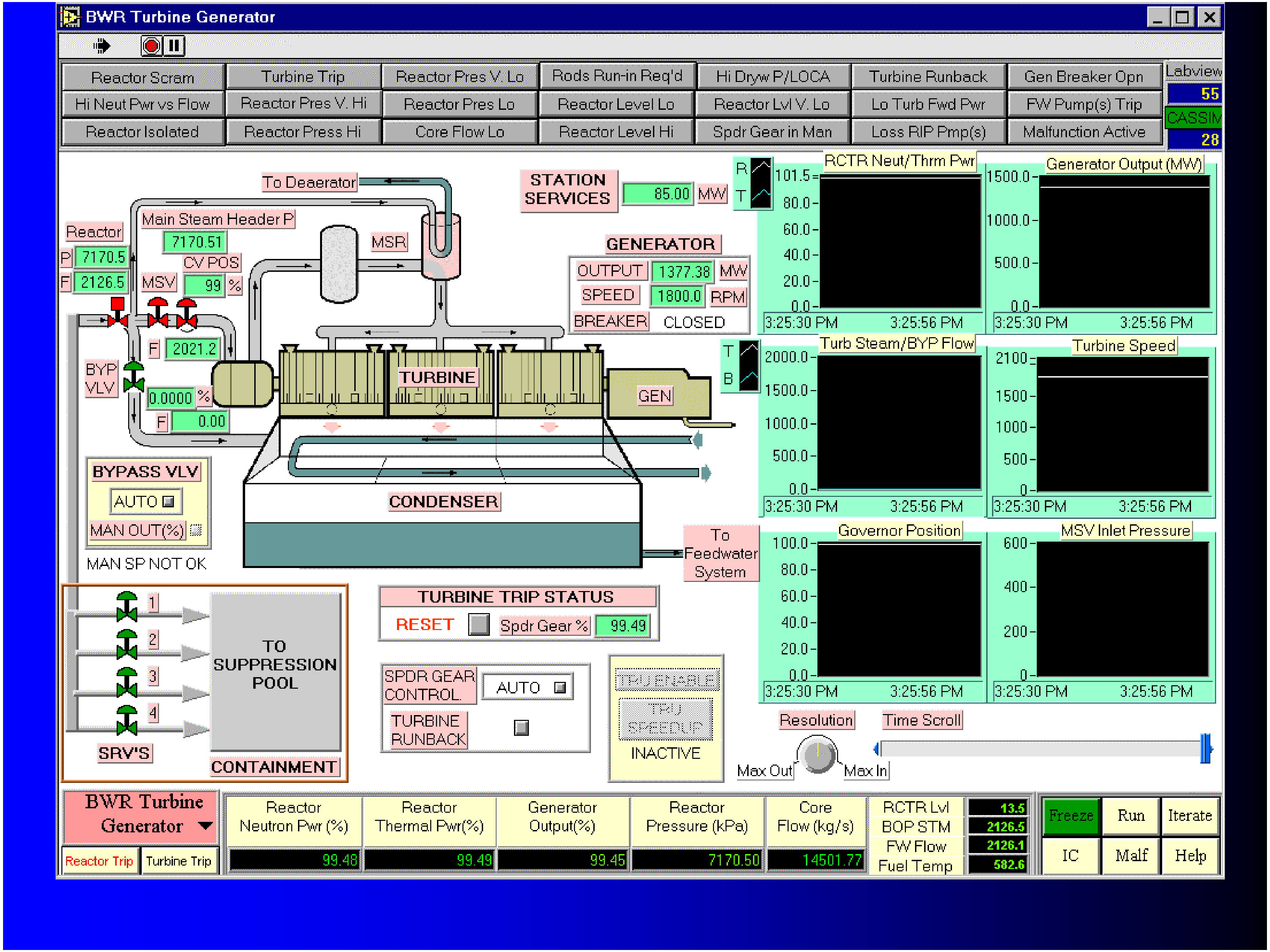
| SYSTEM | SIMULATION SCOPE | DISPLAY PAGES | OPERATOR CONTROLS | MALFUNCTIONS |
|---|---|---|---|---|
| STEAM & FEED-WATER | • Steam supply to turbine and reheater<br>• Main Steam Isolation Valve<br>• Turbine Bypass to condenser<br>• Steam Relief Valves to Suppression Pool in containment<br>• Extraction steam to feed heating<br>• Feedwater system | • BWR Feedwater and Extraction Steam | • Reactor water level setpoint changes: computer or manual<br>• Extraction steam to feedwater heating isolating valves controls<br>• Deaerator main steam extraction pressure control<br>• Feed pump on/off controls | • Loss of both feedwater pumps<br>• Loss of feedwater heating<br>• Reactor feedwater level control valve fails open<br>• Safety valves on one main steam line fail open<br>• Steam line break inside Drywell<br>• Feedwater line break inside Drywell |



BWR Feedwater & Extraction Steam
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
52
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
28
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
TO D/A
Main Steam Header
P 7170.5
MSR
MSIV
MSV
CV
F 2021.2
TURBINE
F 0.0
CONDENSER
PCV
DEAERATOR
MV1
F 46.99
F 0.00
HPHX
FWP
FW LCV
CEP
LPHX
F 2126.1
T 221.4
L 2.75
P 525.3
T 153.1
MV3
FW TO REACTOR
COMP SP
MAN SP NOT OK
MV2
F 92.54
F 92.54
CTR LV SP(M) 13.4931
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
3:27:12 PM
3:27:37 PM
Resolution
Time Scroll
Max Out
Max In
BWR Feedwater & Extr Steam
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 99.48
Reactor Thermal Pwr(%) 99.49
Generator Output(%) 99.45
Reactor Pressure (kPa) 7170.50
Core Flow (kg/s) 14501.77
RCTR Lvl 13.5
BOP STM 2126.5
FW Flow 2126.1
Fuel Temp 582.6
Freeze
Run
Iterate
IC
Malf
Help

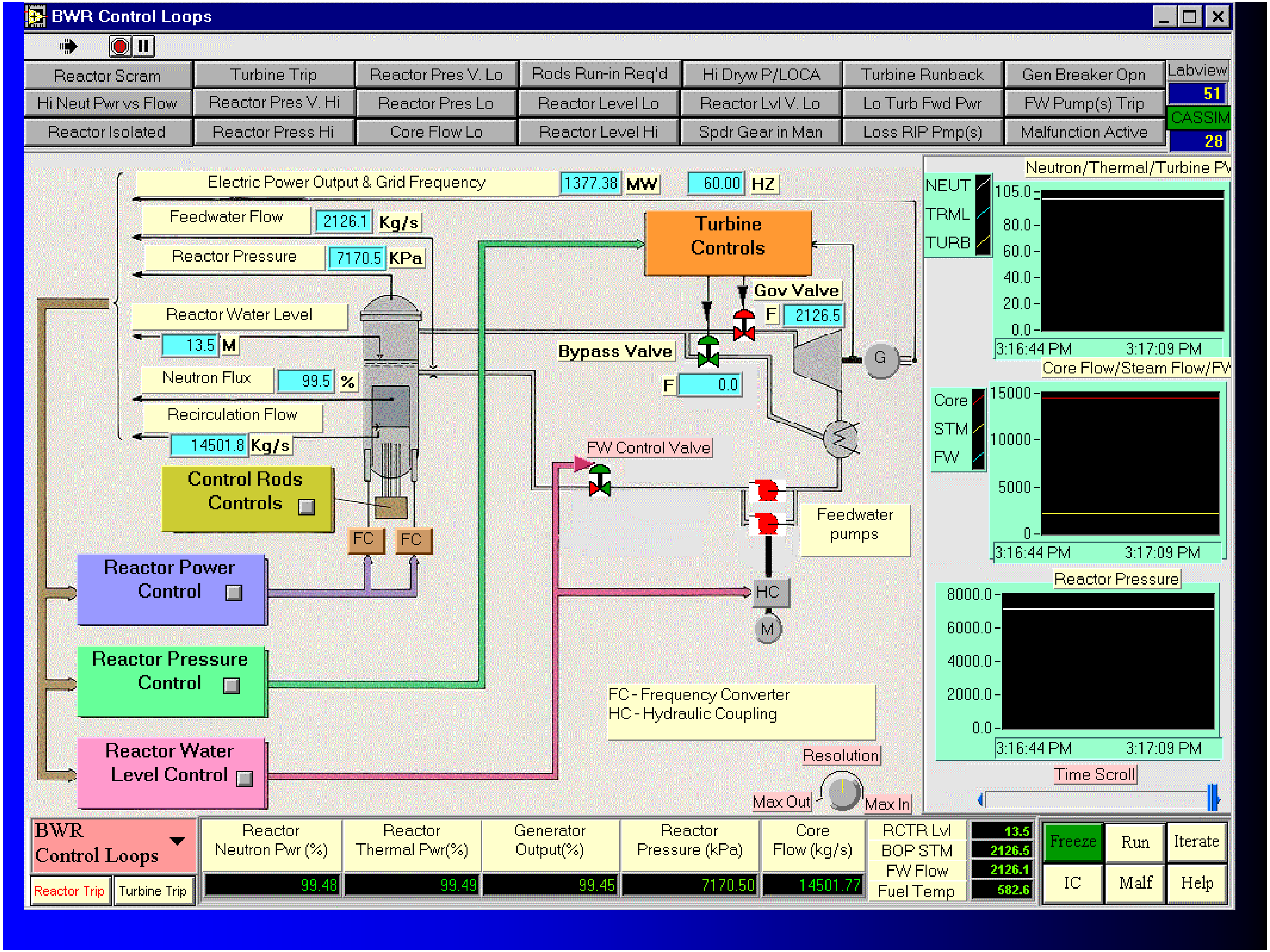
# Turbine Generator

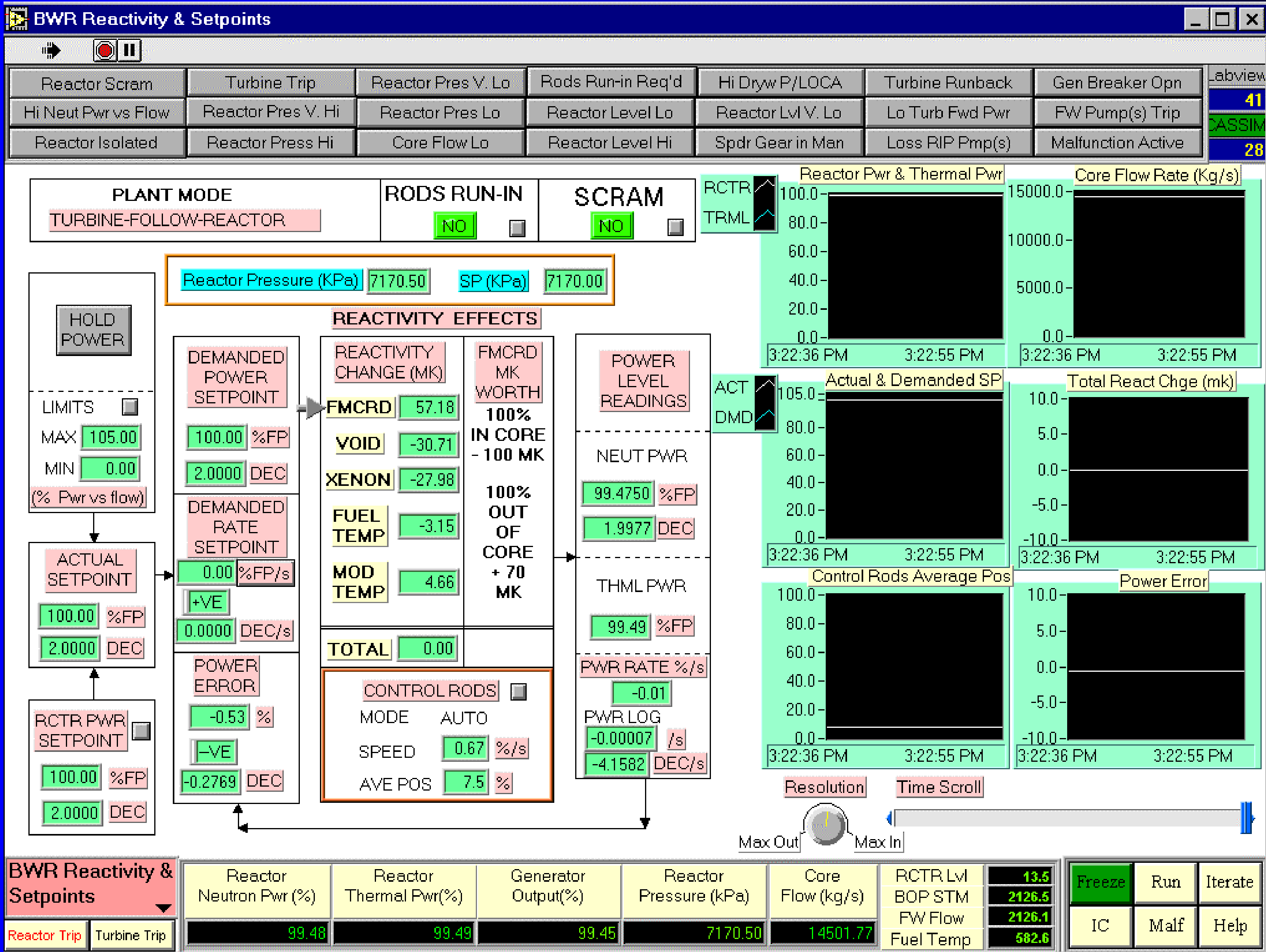
| SYSTEM | SIMULATION SCOPE | DISPLAY PAGES | OPERATOR CONTROLS | MALFUNCTIONS |
|---|---|---|---|---|
| TURBINE-GENERATOR | • Simple turbine model<br>• Mechanical power and generator output are proportional to steam flow<br>• Speeder gear and governor valve allow synchronized and non-synchronized operation | • BWR Turbine-Generator | • Turbine trip<br>• Turbine run-back<br>• Turbine run-up and synchronization<br>• Turbine Speeder Gear control: manual or computer control<br>• Steam Bypass Valve Computer or Manual Control | • Turbine throttle pressure transmitter fails low<br>• Turbine trip with Bypass Valve failed closed<br>• Increasing and decreasing steam flow due to Pressure Control System failures |


BWR Turbine Generator
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
55
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
28
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
To Deaerator
STATION SERVICES
85.00
MW
Main Steam Header P
7170.51
Reactor
P 7170.5
F 2126.5
CV POS
MSV
99
%
MSR
GENERATOR
OUTPUT
1377.38
MW
SPEED
1800.0
RPM
BREAKER
CLOSED
F 2021.2
BYP VLV
0.0000
%
F 0.00
TURBINE
GEN
CONDENSER
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
To Feedwater System
1
2
3
4
TO SUPPRESSION POOL
SRV'S
CONTAINMENT
TURBINE TRIP STATUS
RESET
Spdr Gear %
99.49
SPDR GEAR CONTROL
AUTO
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
RCTR Neut/Thrm Pwr
R
T
101.5
80.0
60.0
40.0
20.0
0.0
3:25:30 PM
3:25:56 PM
Generator Output (MW)
1500.0
1000.0
500.0
0.0
Turb Steam/BYP Flow
B
2000.0
1500.0
1000.0
500.0
Turbine Speed
2100
1500
1000
500
0
Governor Position
100.0
MSV Inlet Pressure
600
400
200
Resolution
Time Scroll
Max Out
Max In
BWR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
99.48
Reactor Thermal Pwr(%)
99.49
Generator Output(%)
99.45
Reactor Pressure (kPa)
7170.50
Core Flow (kg/s)
14501.77
RCTR Lvl
13.5
BOP STM
2126.5
FW Flow
2126.1
Fuel Temp
582.6
Freeze
Run
Iterate
IC
Malf
Help

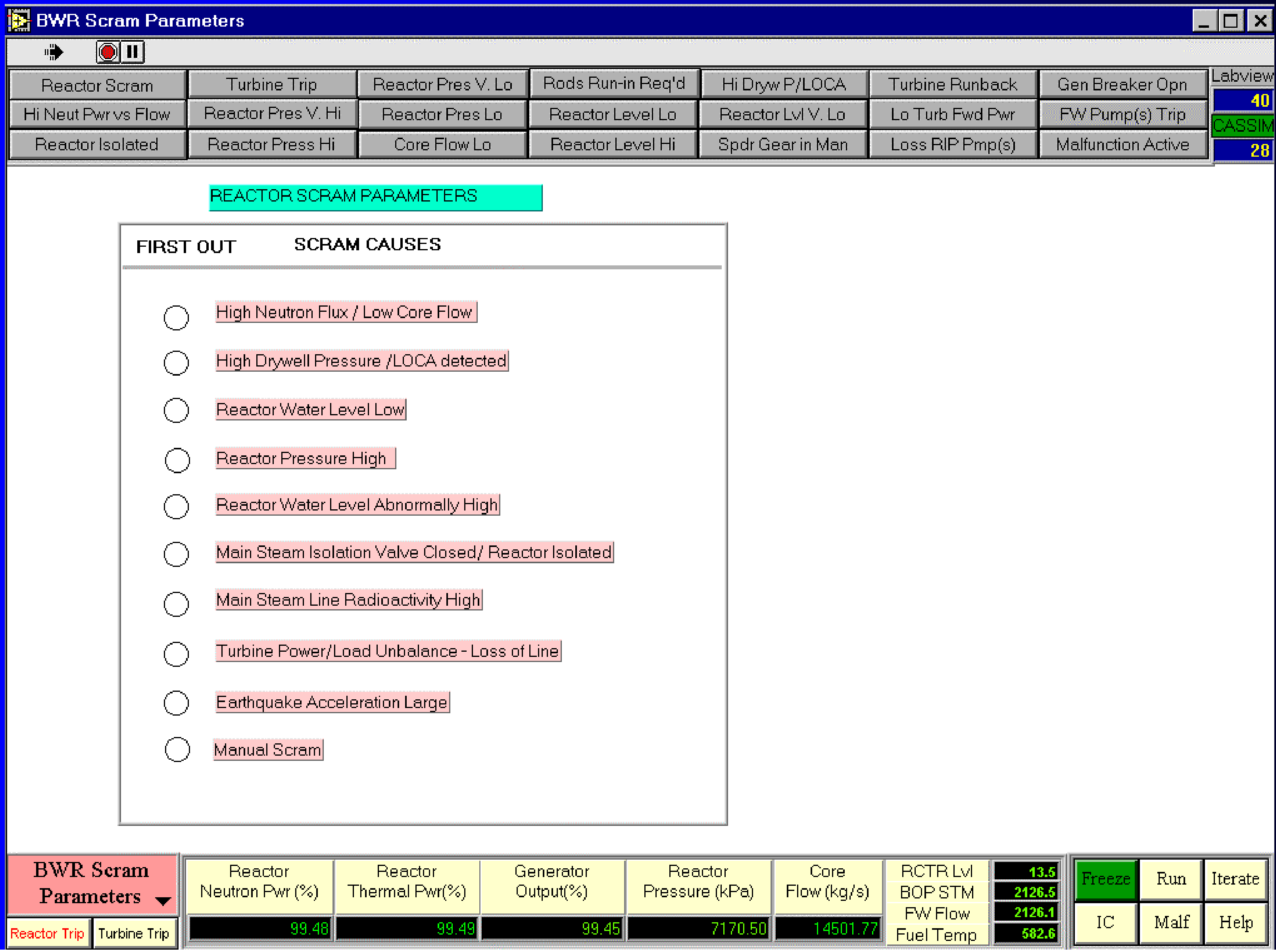
# Overall Unit

| SYSTEM | SIMULATION SCOPE | DISPLAY PAGES | OPERATOR CONTROLS | MALFUNCTIONS |
|---|---|---|---|---|
| OVERALL UNIT | • Fully dynamic interaction between all simulated systems<br>• Turbine-Following-Reactor load maneuvering<br>• Unit annunciation<br>• Major control loops | • BWR Plant Overview<br>• BWR Reactivity & Setpoints | | |

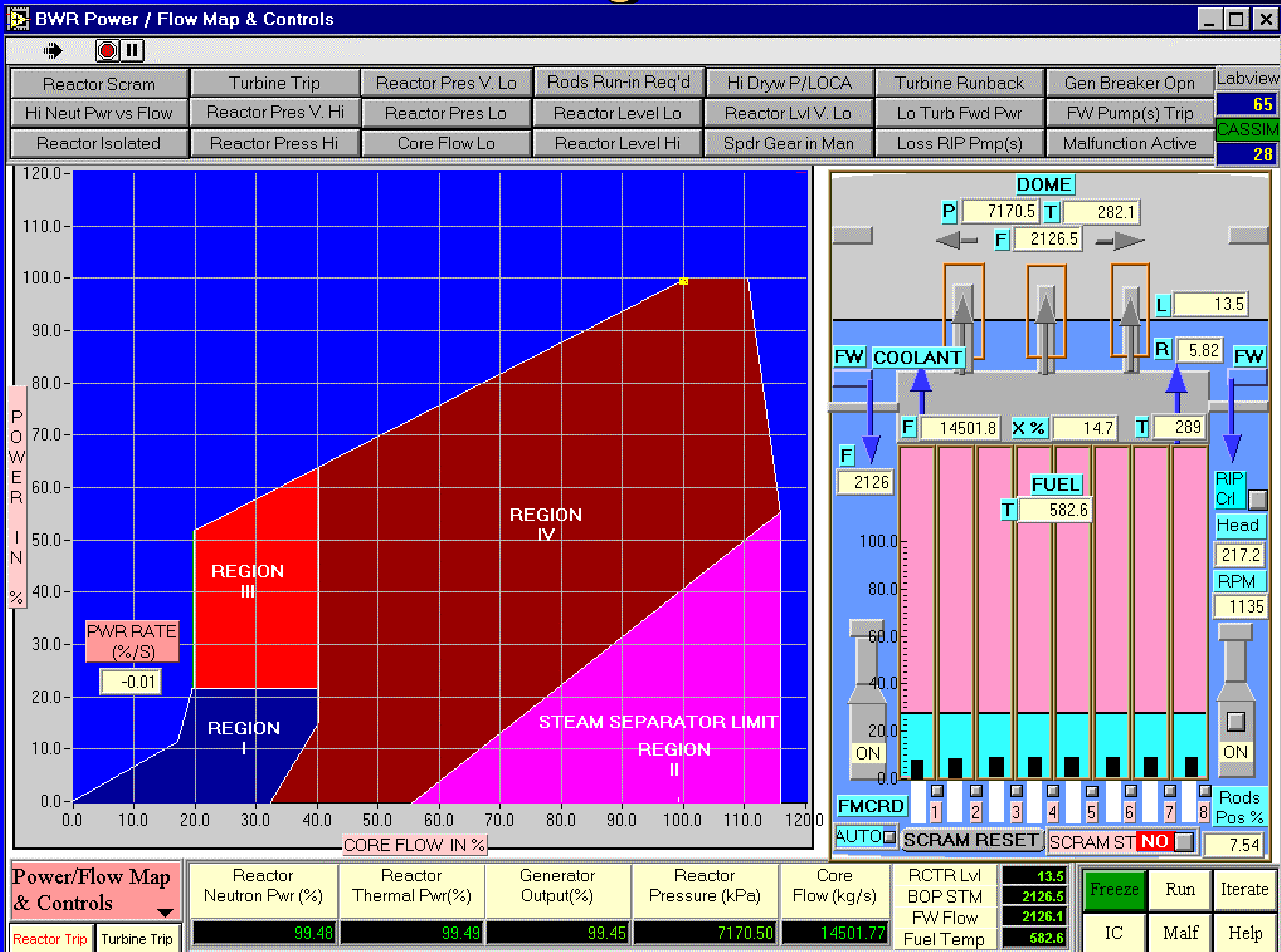

BWR Control Loops
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
51
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
28
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
Electric Power Output & Grid Frequency
1377.38 MW
60.00 HZ
Feedwater Flow
2126.1 Kg/s
Reactor Pressure
7170.5 KPa
Reactor Water Level
13.5 M
Neutron Flux
99.5 %
Recirculation Flow
14501.8 Kg/s
Turbine Controls
Gov Valve
F 2126.5
Bypass Valve
F 0.0
G
FW Control Valve
Control Rods Controls
FC
FC
Feedwater pumps
Reactor Power Control
HC
M
Reactor Pressure Control
FC - Frequency Converter
HC - Hydraulic Coupling
Reactor Water Level Control
Resolution
Max Out
Max In
Neutron/Thermal/Turbine Pw
NEUT
TRML
TURB
105.0
80.0
60.0
40.0
20.0
0.0
3:16:44 PM
3:17:09 PM
Core Flow/Steam Flow/FW
Core
STM
FW
15000
10000
5000
0
3:16:44 PM
3:17:09 PM
Reactor Pressure
8000.0
6000.0
4000.0
2000.0
0.0
3:16:44 PM
3:17:09 PM
Time Scroll
BWR Control Loops
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
99.48
Reactor Thermal Pwr(%)
99.49
Generator Output(%)
99.45
Reactor Pressure (kPa)
7170.50
Core Flow (kg/s)
14501.77
RCTR Lvl
13.5
BOP STM
2126.5
FW Flow
2126.1
Fuel Temp
582.6
Freeze
Run
Iterate
IC
Malf
Help


BWR Reactivity & Setpoints
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
Spdr Gear in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
41
CASSIM
28
PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN
NO
SCRAM
NO
Reactor Pressure (KPa) 7170.50
SP (KPa) 7170.00
REACTIVITY EFFECTS
HOLD POWER
LIMITS
MAX 105.00
MIN 0.00
(% Pwr vs flow)
ACTUAL SETPOINT
100.00 %FP
2.0000 DEC
RCTR PWR SETPOINT
100.00 %FP
2.0000 DEC
DEMANDED POWER SETPOINT
100.00 %FP
2.0000 DEC
DEMANDED RATE SETPOINT
0.00 %FP/s
+VE
0.0000 DEC/s
POWER ERROR
-0.53 %
-VE
-0.2769 DEC
REACTIVITY CHANGE (MK)
FMCRD 57.18
VOID -30.71
XENON -27.98
FUEL TEMP -3.15
MOD TEMP 4.66
TOTAL 0.00
FMCRD MK WORTH
100% IN CORE - 100 MK
100% OUT OF CORE + 70 MK
CONTROL RODS
MODE AUTO
SPEED 0.67 %/s
AVE POS 7.5 %
POWER LEVEL READINGS
NEUT PWR
99.4750 %FP
1.9977 DEC
THML PWR
99.49 %FP
PWR RATE %/s
-0.01
PWR LOG
-0.00007 /s
-4.1582 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
3:22:36 PM
3:22:55 PM
Resolution
Time Scroll
Max Out
Max In
BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 99.48
Reactor Thermal Pwr(%) 99.49
Generator Output(%) 99.45
Reactor Pressure (kPa) 7170.50
Core Flow (kg/s) 14501.77
RCTR Lvl 13.5
BOP STM 2126.5
FW Flow 2126.1
Fuel Temp 582.6
Freeze
Run
Iterate
IC
Malf
Help

# Scram Causes



# Resetting Scram



# BWR Simulator Manual Exercise 4.1.1 - Power Reduction

- POWER MANEUVER: 10% Power Reduction and Return to Full Power.
- Record (1) Control Rods position (2) Recirculation Flow (3) Quality (4) Void reactivity feedbacks during this maneuver.
- Explain how reactor power is controlled during this maneuver.

# ICTP Workshop - BWR Simulator Exercises

Wilson Lam (wilson@cti-simulation.com)
CTI Simulation International Corp.
www.cti-simulation.com

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# BWR Simulator Manual
# Exercise 4.1.3, 4.2.9 - Turbine Trip

- Practice Turbine Trip & Recovery.
- Record Reactor Pressure during this transient.
- Does Reactor Pressure resume to the setpoint of 7170 Kpa after the transient settles down ? If so, explain how reactor pressure is being controlled now.
- Re-initialize the Simulator. This time insert the malfunction “Turbine Trip with Bypass Valves Failed Closed”. Explain what happens. See 4.2.9

# Malfunction 4.2.1 - Loss of FW

- Load 100 % FP IC. Open BWR Feedwater & Extraction Steam Screen
- Insert Malfunction “Loss of FW - Both FW Pumps Trips”
- Follow text on P.26, and answer questions.

# Malfunction 4.2.3 - Decreasing Core Flow

- Follow Text on P. 29 to practice Malfunction" Decreasing Core Flow due to Flow Control Failure"
- As coolant flow decreases, core quality increases. Why ? What happens to reactor power ?
- Explain the responses of the Reactor Power Control System

# Malfunction Exercise 4.2.4 - Decreasing Steam Flow

- Re-initialize the Simulator to 100% FP.
- Go to Power/Flow Map Screen.
- Insert the Malfunction "Decreasing Steam Flow from Dome due to Pressure Control Function." (see P.30)
- What happens to Reactor Pressure ?
- What happens to  the Reactor Power ?
- Can you explain the Reactor Power transient responses ?

# Malfunction 4.2.12 - Inadvertent Reactor Isolation

- Practice Malfunction 4.2.12 “ Inadvertent Reactor Isolation”.
- Follow Text on P.38 of BWR Simulator Manual. Record parameters.
- What happens to reactor power ?
- What is the cause for reactor scram ?

# Malfunction 4.2.13 - Loss of FW Heating

- Practice Malfunction 4.2.13 “ Loss of FW Heating”.
- Follow Text on P.39 of BWR Simulator Manual. Record parameters.
- Explain the changes in reactor power and other BOP parameters.

# Malfunction 4.2.15 - Steam Line Break

- Practice Malfunction 4.2.15 “ Steam Line Break inside Drywell”
- Follow Text on P. 42 of BWR Simulator Manual. Record parameters.
- Explain changes to reactor power and other BOP parameters.
- Explain the actions of ECC.

# Malfunction 4.2.16 - Feedwater Line Break

- Practice Malfunction 4.1.16 “ FW line Break inside Drywell”
- Follow Text on P.43 of BWR Simulator Manual. Record parameters.
- Explain changes to reactor power and other BOP parameters.
- Explain the actions of ECC.