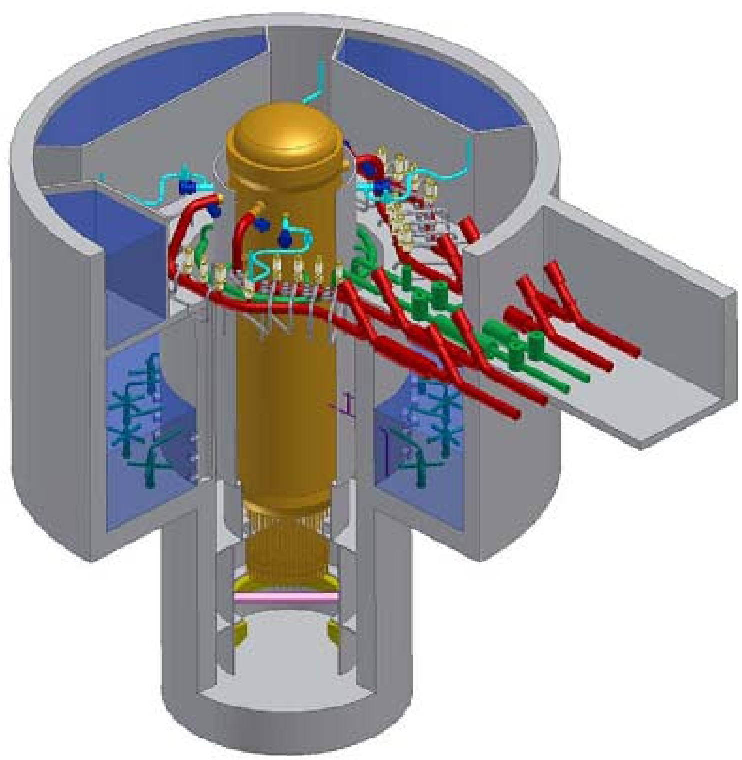
Boiling Water Reactor Simulator with Passive Safety Systems

User Manual

October 2009



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Foreword

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

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

The IAEA simulator collection currently includes the following simulators:

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

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

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

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

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

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

- The IAEA advanced BWR, which largely represents the GE ESBWR passive BWR design and was also created by CTI. The simulator development is a concerted effort from the developer, the Agency's staff, and from Dr. Bharat Shiralkar, a thermal-hydraulics expert.

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

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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) with passive safety system simulator (or interchangeably called passive 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 passive 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 passive boiling water reactor. Therefore no attempt has been made to provide detailed descriptions of each individual passive BWR subsystem. Those descriptions are commonly found in passive BWR nuclear power plant (NPP) general description (Reference 1). However, details are provided where necessary to describe the distinctive features of ESBWR, 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 (Anticipated Operational Occurrences (AOOs)) lead to reactor scram or turbine trip. Other serious malfunction events (Design Basis Accidents (DBAs), e.g. LOCA) lead to actuation of the passive emergency core cooling safety system.

It should be mentioned that the equipment and processes modeled in the simulator represent realistic passive 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 passive 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 INTRODUCTION

In a Boiling Water Reactor (BWR) Nuclear Supply Steam System (NSSS), the steam is directly produced in the reactor pressure vessel (RPV), is directed to steam turbine, hence the process is called “Direct Cycle”. In the Pressurized Water Reactors (PWRs), the steam is produced in steam generators that are connected to RPV, hence the process is called “Indirect Cycle”. Both the BWRs and PWRs are classified as Light water Reactors (LWRs) since they use light water (as opposed to heavy water, D_2O, as in CANDU) as coolant and moderator.

The objective of this chapter is to provide a description of the evolution of the BWR, including a brief outline of the Generation III Advanced Boiling Water Reactor (ABWR), and the Generation III + Economic Simplified Boiling Water Reactor (ESBWR).

The first section provides a general background of the development of the BWR. This includes the description of the reactor and reactor system design, safety system design and the containment design [1, 2]. The next section deals with the key features of the ESBWR including the natural circulation design, operating domain and passive safety features [3].

Acknowledgement: much of the historical descriptions of BWR evolutions (below) are derived from Chapter 58 of the Report (available in the public domain) for New Generation of BWRs, written by Hardayal S. Mehta and Daniel C. Pappone. Acknowledgement and credits should be given to these authors.

1.3 EVOLUTION OF BWR FROM BWR/1 THROUGH ESBWR

The BWR nuclear plant, like the Pressurized Water Reactor (PWR), has its origins in the technology developed in the 1950’s for the U.S. Navy’s nuclear submarine program. The first commercial BWR nuclear plant to be built was the 5 MWe Vallecitos plant (1957) located near San Jose, California. The Vallecitos plant confirmed the ability of the BWR concept to successfully and safely produce electricity for a grid. The first large-scale BWR, Dresden 1 (1960), then followed. The BWR design has subsequently undergone a series of evolutionary changes.

The BWR design has been simplified in two key areas – the reactor systems and the containment design. Table 1 chronicles the development of the BWR.

Table 1 Evolution of the BWR

BWR version	First Commercial Operation Date	Representative Plant/ Characteristics
BWR/1	1960	Dresden 1 Initial commercial-size BWR Dual cycle
BWR/2	1969	Oyster Creek Plants purchased solely on economics Large direct cycle Forced circulation Variable speed pumps for recirculation flow control
BWR/3	1971	Dresden 2 Internal jet pump application Improved ECCS: spray and flood capability
BWR/4	1972	Vermont Yankee Increased power density (20%)
BWR/5	1977	Tokai 2 Improved ECCS Valve flow control
BWR/6	1978	Cofrentes Compact control room Solid-state nuclear system protection system
ABWR	1996	Kashiwazaki-Kariva 6 Reactor internal pumps Fine-motion control rod drives Advanced control room, digital and fiber optic Technology Improved ECCS: high/low pressure flooders
ESBWR	Under review	Natural circulation Passive ECCS

1.3.1 General Progression of BWR Designs

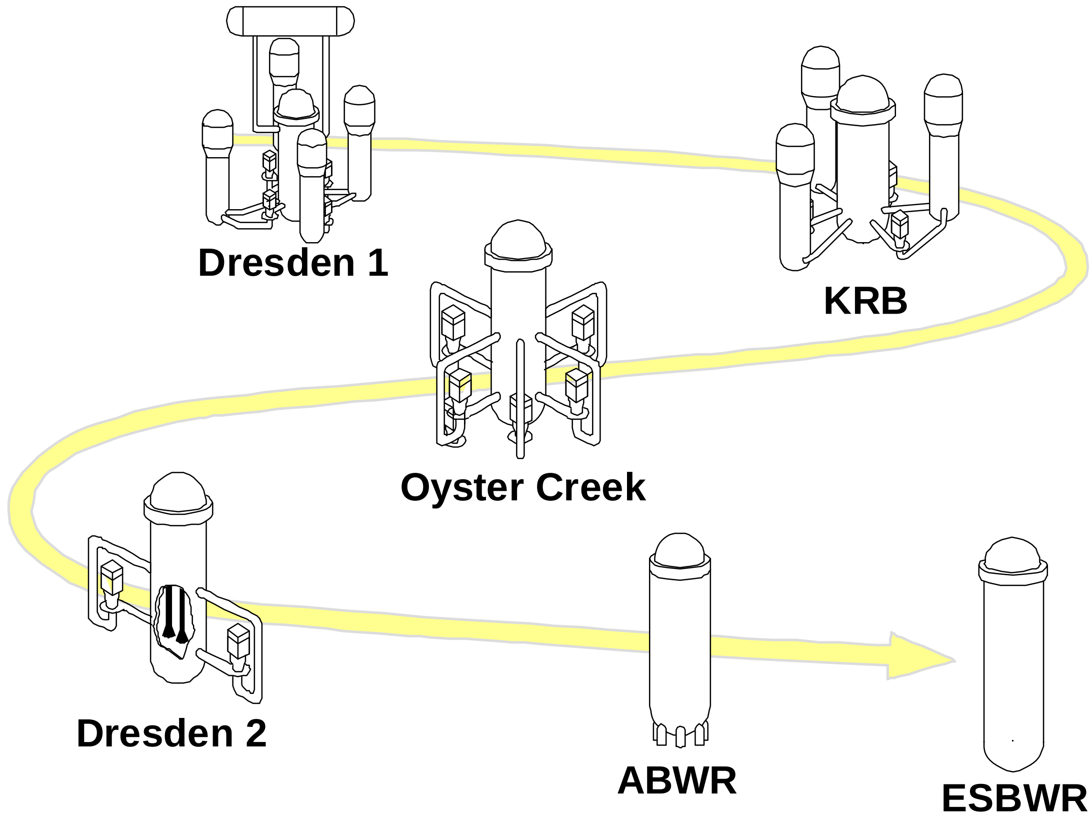


Figure 1 above illustrates the evolution of the reactor system design. Dresden 1 was based upon dual steam cycle, not the direct steam cycle that characterizes BWRs. Steam was generated in the reactor but then flowed to an elevated steam drum and a secondary steam generator before making its way to the turbine. The first step down the path of simplicity that ultimately led to the ESBWR was the elimination of the external steam drum by introducing two technical innovations – the internal steam separator and the steam dryer (KRB, 1962). The first large direct cycle BWRs (Oyster Creek and Nine Mile Point Unit 1) appeared in the mid-1960's and were characterized by the elimination of the steam generators and the use of five external recirculation loops to provide forced circulation flow through the core. Later, reactor systems were further simplified by the introduction of jet pumps inside the vessel for driving the core flow. The use of internal jet pumps allowed two external recirculation loops to drive the core flow, down from the five loops used in the large BWR/2 plants, thus reducing the associated piping, valves, pumps and large vessel nozzles. This change first appeared in the Dresden-2 BWR/3 plant.

The use of reactor internal pumps in the ABWR design took this process of simplification another step further. By using internal pumps attached directly to the vessel itself, the jet pumps and the external recirculation systems, with all the associated large pumps, valves, piping, and snubbers, have been eliminated altogether. The development of the ABWR took place during the 1980's under the sponsorship of Tokyo Electric Power Company (TEPCO). In 1988, TEPCO announced that the next Kashiwazaki-Kariva units (K-6&7) to be constructed would be ABWRs. The licensing activities for the K-6&7 were conducted with the Japanese regulatory agency, MITI (Ministry of International Trade and Industry) in parallel with the review of the ABWR in the U.S. by the Nuclear Regulatory Commission (NRC). K-6 entered commercial operation in 1996 and K-7 in 1997. Two more ABWRs entered commercial operation in Japan – Hamaoka-5 in 2005 and Shika -2 in 2006. In addition to these four units in Japan, two more ABWRs are being constructed for the Taiwan Power Company (TPC) at TPC's Lungmen site.

In the U.S., the ABWR First-of-a-Kind Engineering (FOAKE) program was completed in September 1996. FOAKE represented a major step toward one of the U.S. industry's goals – to have a certified design that is 90% engineered prior to the start of construction. The ABWR Design Certification was signed into law on May 2, 1997 by the NRC. In September 2007, NRG Energy submitted to the NRC a combined construction permit – operating license application (COLA) to build two ABWR units at the South Texas Project site in Texas.

The ESBWR, and its smaller predecessor, the simplified boiling water reactor (SBWR), took the process of simplification to its logical conclusion with the use of a taller vessel and a shorter core with chimney to achieve natural recirculation without the use of any pumps. Following the Three Mile Island accident in 1979, there was significant interest in developing a reactor with passive safety features and less dependence on operator actions. Utilities also took this opportunity to request a reactor that was simpler to operate, had fewer components and no dependence on diesel-generators for safety actions. GE began an internal study of a new BWR concept based on these principles and the SBWR was born in the early 1980s. This concept attracted development support from the U.S. Department of Energy (DOE), EPRI and a number of US utilities. Key new features (explained in sections below), such as the Gravity Driven Cooling System (GDCS), Depressurization Valves (DPV), and leak-tight wetwell/drywell vacuum breakers were tested. As interest grew, an international team was formed to complete the design, and additional separate effects, component and integrated system tests, particularly of the innovative new feature, the Passive Containment Cooling System (PCCS), were run in Europe and Japan. A Design Certification Program was begun the late 1980s with the objective of obtaining a standardized license, similar to that obtained for ABWR. However, as more of the design details became known, it became clear that, at 670 MWe, the SBWR was too small to generate the right economics for a new-build project. The certification program was stopped, but efforts continued to make an SBWR attractive for power generation. With European Utility support, the SBWR was uprated gradually to its current power level of approximately 1550 MWe. This was made possible by staying within the RPV size limit established by the ABWR, and by taking advantage of the modular approach to passive safety afforded by Isolation Condensers (IC) and PCCS.

The Design Certification application for the ESBWR was submitted to the US NRC in August 2005 and was formally accepted for docketing. The NRC's new plant review and licensing process has been improved, including allowance for parallel review of both the Design Certification and the COLA, with a focus on standardization, and reducing and eliminating re-reviews of the same open items. GE is working with a number of utilities who have selected the ESBWR technology, and is participating in the U.S. Department of Energy's Nuclear Power 2010 program, which was established by the DOE to act as a catalyst for new build nuclear energy in the U.S.

1.3.2 Reactor System Design

The first light water reactors were pressurized in order to prevent the water from boiling in the core or turning to steam. The conventional wisdom was that steam bubbles would somehow affect the behavior of the neutrons and cause the reactor to behave erratically and possibly overheat. Samuel Untermyer, a scientist at Argonne National Laboratory, postulated that if water bubbled or steamed in an overheating reactor core, the chain reaction would slow down [4]. This concept of a negative void reactivity coefficient proved to be fundamental to the successful design and operation of the boiling water reactors. Untermyer initiated the Boiling Water Reactor Experiment (BORAX) series of test reactors at the National Reactor Testing Station in Idaho. In 1953, the first test reactor in the series, BORAX-1 demonstrated two theories: (1) boiling water in a reactor was a stable condition and (2) the void coefficient can control the power increases. Argonne modified the BORAX reactor design several times while conducting experiments to develop an understanding of the parameters necessary for operating a boiling water reactor safely.

Argonne's ultimate goal was to evolve a reactor useful for electrical generation. The early experiments with the BORAX reactors demonstrated the inherent safety and stability of the BWR. Using a turbine generator from an old sawmill, the BORAX-III reactor lit the town of Arco, Idaho on July 17, 1955 with nuclear power for the first time in history. The first BWR power plant, EBWR (Experimental BWR), built by Argonne Laboratory, was designed for 20 MWe. The plant ran from 1956 to 1967, gradually increasing its power level and reliability to the point where it supplied electricity to the entire Argonne Lab.

The main characteristic of the BWR is the fact that bulk boiling takes place in the reactor core. In the direct cycle BWR, the simplest design, the steam is passed directly from the reactor vessel to the turbine. The BWR designs can be further classified by the means used to drive the coolant through the core. In a forced circulation design, the core flow is driven by pumps external or internal to the reactor vessel. In a natural circulation design, the difference in density between the single-phase fluid in the downcomer region and the two-phase mixture in the core provides the driving force [5].

The main characteristic of the BWR can be summarized as:

(1) The core normal operating conditions result in two phases: a subcooled liquid phase in the non-boiling region; and a saturated steam-water mixture in the boiling region.

(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 temperature is 288°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 $N \times N$ 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 2% full power per sec. The recirculation flow control also controls reactor power by causing the density of the water 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 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 then 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 drywell and wetwell.

The advantages of the BWR compared to the pressurized water reactor (PWR) are:

- The reactor vessel and associated components operate at a substantially lower pressure (about 1000 psig) compared to a PWR (about 2200 psig).
- There are no steam generators and no pressurizer vessel.
- Lower risk (probability) of a rupture causing loss of coolant compared to a PWR, and lower risk of a severe accident should such a rupture occur. This is due to fewer pipes, fewer large diameter pipes, fewer welds and no steam generator tubes.
- The distance between the core and the vessel wall is greater; therefore, the pressure vessel is subject to significantly less irradiation compared to a PWR, and so does not become as brittle with age.
- The fuel operates at a lower temperature.
- The single reactor vessel makes emergency conditions simpler to diagnose and emergency actions simpler to execute.
- A BWR may be designed to operate using only natural circulation so that recirculation pumps are eliminated entirely.

The disadvantages of a BWR are:

- The core nuclear and thermal-hydraulic calculations are more complex when considering two-phase flow.
- The reactor pressure vessel is much larger than that of a PWR of similar power. However, the overall cost is reduced because a modern BWR has no main steam generators or pressurizer.
- The primary coolant passes through the turbine, contaminating it with short-lived activation products. Therefore, radiation exposure must be managed in the vicinity of the steam piping and turbine.

Table 2 Comparison of Key Features of GE BWRs

Parameter	BWR/2	BWR/4	BWR/6	ABWR	ESBWR
Power (MWt/MWe)	1930/670	3293/1098	3900/1360	3926/1350	4500/1550
Vessel height/dia (m)	19.5/5.4	21.9/6.4	21.8/6.4	21.1/7.1	27.7/7.1
Fuel bundles, number	560	764	800	872	1132
Active fuel height (m)	3.7	3.7	3.7	3.7	3.0
Power density (kW/L)	40.5	50	54.2	51	54
Recirculation pumps	5 (external)	2 (external)	2 (external)	10 (internal)	0
Number/type of CRDs	137/LP	185/LP	193/LP	205/FM	269/FM
Safety system pumps	12	9	9	18	0
Safety diesel generators	2	2	3	3	0
Alternate shutdown	2 SLC pumps	2 SLC pumps	2 SLC pumps	2 SLC pumps	2 SLC accumulators
Control and instrumentation	Analog single channel	Analog single channel	Analog single channel	Digital multiple channel	Digital multiple channel
Core damage (freq/yr)	10^{-5}	10^{-5}	10^{-6}	$2x10^{-7}$	$3x10^{-8}$
Safety bldg vol (m^3/Mwe)	110	120	170	180	130

Table 2 (above) provides a comparison of the key features of the commercial BWRs. The first standardized boiling water reactor design was the BWR/2 (Oyster Creek and Nine Mile Point 1). The BWR/2 direct cycle design used the internal steam separator and steam dryer demonstrated at KRB. This simplified the reactor design by eliminating the external steam drum and steam generator pressure vessels. The steam separator and dryer assemblies are removed to allow access to the core during refueling. The control blades enter the core from the bottom of the vessel. Control rod entry from the bottom of the core provides the best axial flux shaping and resultant fuel economy for the boiling water reactor. The bottom-entry drives do not interfere with refueling operations. The piston drive and high pressure scram accumulators provides high scram forces and ensures rapid rod insertion.

The BWR/2 design also standardized the use of forced recirculation core flow. Oyster Creek and Nine Mile Point 1 passed the entire core flow through five external recirculation loops with pumps. Forced recirculation core flow provides a second means of controlling core power, in addition to the control rods. Forced recirculation allows the core power to be controlled by changing recirculation flow by exploiting the negative void reactivity characteristics of the BWR core design. An increase in the core flow increases the liquid in the core, which increases the neutron moderation and power generation. The core stabilizes at a higher power level. The ability to change power gradually by using core flow also reduces the thermal-mechanical duty on the fuel. The control rod drives move the control blades in finite increments, which may result in a significant local power increase near the tip of the control blade. Varying the core flow also varies the boiling boundary in the core. This allows spectral shift core management schemes to be employed. Early in the cycle, the core flow is kept at low, lowering the axial flux shape and boiling boundary in the core, which increases the voiding in the top of the core. This voiding hardens the neutron spectrum in the upper part of the core resulting in the breeding of plutonium. At the end of the cycle when the reactivity in the lower part of the core is exhausted, the core flow is increased, shifting the axial flux shape upward and burning the plutonium generated in the upper part of the core early in the cycle. Thus, the ability to control core power using recirculation flow ultimately results in better fuel cycle economics for the plant.

The next significant simplification introduced in the reactor system was the internal jet pump in the BWR/3 design. These pumps sufficiently boosted recirculation flow so that only two external recirculation loops were needed. This change was first incorporated in the Dresden 2 plant. The use of internal jet pumps resulted in the elimination of several of the recirculation lines and pumps used previously in the BWR/2 reactors. Savings in investment for recirculation lines and pumps were partially offset by the higher power requirements due to the lower efficiency of the jet pump. More notably, the use of jet pumps gives some safety advantages in that the number and size of major nozzle penetrations on the vessel is reduced and the reactor internal arrangement allows the capability of reflooding the vessel and maintaining coolant level in the core even if there is a complete severance of the recirculation line. Increased natural circulation capability was also gained with the jet pump system.

The basic jet pump BWR design was carried through most of the plants comprising today's BWR operating fleet. The BWR/4 design (Browns Ferry and Peach Bottom) introduced improved Emergency Core Cooling Systems (ECCS) and higher core power densities. The BWR/5 design (Tokai 2) introduced constant speed recirculation pumps and flow control valves used to control the reactor core flow and improved ECCS. The change from variable speed recirculation pumps to constant speed pumps with valve control allowed the plants to follow more rapid load variations and reduced the capital cost of the overall control system. The BWR/6 design (Cofrentes) incorporated a five-hole nozzle design in the jet pump that increased the pumping efficiency and allowed the use of smaller jet pumps. With smaller jet pumps, the size of the core could be increased, increasing the reactor output for the same size vessel.

The ABWR design (Kashiwazaki-Kariwa 6) incorporated reactor internal pumps to drive the core flow. By using pumps attached directly to the vessel itself, the jet pumps and the external recirculation systems, with all their pumps, valves, piping, and snubbers, have been eliminated altogether. With the elimination of the large external recirculation piping, there were no large pipes attached to the vessel below the top of the core. This allowed the ABWR to be designed to keep the core submerged and cooled during the break of any pipe forming the reactor coolant pressure boundary. Fine motion control rod drives permit small power changes, improved startup times, and improved power maneuvering.

The ESBWR design uses natural circulation to provide the driving force for the core flow, eliminating the recirculation pumps altogether. Building on the basic ABWR vessel design, the height of the vessel has been increased to provide the additional driving head necessary to achieve the core flows and power levels necessary for a modern reactor. Variable feedwater temperature control has been added to provide a second means of reactivity control in addition to the fine motion control rod drives. Figure 2 shows the direct cycle power conversion system used in most BWRs.

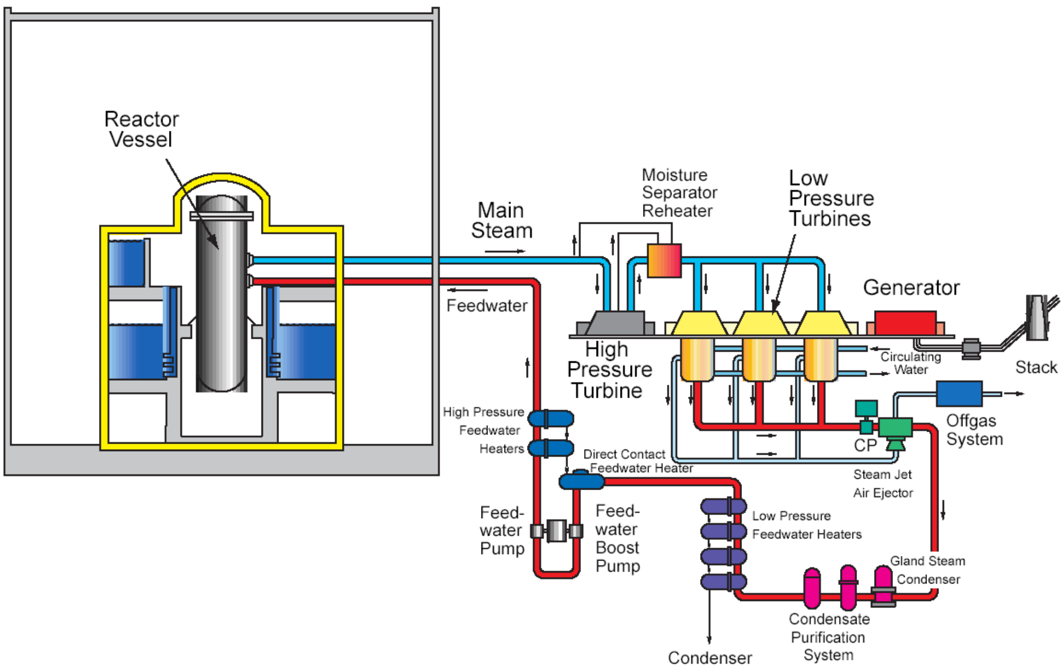


Figure 2 ESBWR Steam and Power Conversion System

1.3.3 Safety System Design

A Loss-of-Coolant Accident (LOCA) is defined as a break in the primary coolant system pressure boundary. The design basis accident was specified to be the instantaneous double-ended guillotine break of the largest pipe attached to the reactor pressure vessel.

However, the safety systems must also be designed to mitigate the consequences of the full spectrum of potential break sizes and locations anywhere in the primary coolant piping, as well as equipment failures such as a stuck open relief valve that result in a an uncontrolled loss of primary system coolant. The Emergency Core Cooling System (ECCS) is designed to provide cooling water to the core terminating any heatup of the core, and to provide long-term removal of the core decay heat. The containment provides the final barrier to prevent the release of fission products from the fuel. The suppression pool in the containment provides a heat sink and internal water source for core cooling. The containment heat removal systems transfer the decay heat to the ultimate heat sink. In addition to providing for core and containment cooling, the incorporation of the LOCA affected the structural design of the plant. The vessel and piping structures had to be designed to withstand the jet reaction and pipe whip forces from the broken pipe. Structures and systems had to be protected against the impingement of the jet from the broken pipe. Systems had to be designed to mitigate the hydrogen generated by the reaction of the zirconium fuel cladding with steam at high temperatures. The ECCS, containment, and supporting systems designed for mitigating the consequences of a LOCA have resulted in a network of safety system with the capacity and redundancy to handle a wide range of transient and accident events.

The first generation of BWRs were designed without emergency core cooling systems, but rather made use of highly reliable feedwater systems. With the increase in core power associated with the first commercial scale reactors (the BWR/2 design), concerns were raised with respect to providing adequate core cooling during a LOCA and preventing a core meltdown that would threaten containment integrity. In the late 1960's, the Report of the Advisory Task Force on Power Reactor Emergency Cooling led to the requirement that all nuclear plants must have emergency core cooling systems [6]. Some form of emergency core cooling, typically a core spray and high-pressure coolant injection, was retrofit to the BWR/1s. The BWR/2s, BWR/3s, and BWR/4s were under design and construction during this time frame and a separate ECCS was incorporated into those reactor designs.

The BWR/2 recirculation loop design took water from the downcomer region and pumped it into the lower plenum region of the vessel. The limiting pipe break LOCA for the BWR/2 was in the recirculation discharge pipe where it attached to the vessel. A break at this location would leave a large rupture in the bottom of the system that would prevent reflooding the vessel. Therefore, it was decided that the best means of cooling the core would be with a spray system that would wet and cool the fuel from above. The core spray consisted of ring spargers around the periphery of the core, just above the top of the fuel bundles. The ring spargers do not interfere with the core flow during normal operation or fuel bundle movement during refueling. Two completely separate systems (electrical power, pumps, valves, piping, etc) were provided for redundancy. Because the core spray system was designed for the large break LOCA, the core spray pumps are low pressure, high flow capacity pumps. These pumps cannot inject into the vessel at normal operating pressures. To address breaks that are small enough that the vessel will not depressurize through the break, logic was added to some of the vessel pressure relief valves. When the logic detected the LOCA conditions, low reactor water level and high containment drywell pressure, and sensed that the core spray pumps were running, the logic would open the relief valves and depressurize the vessel, effectively making the small break into a large break. This logic is known as the Automatic Depressurization System (ADS). For long-term recovery from the accident, provisions were made to flood the containment to a level above the top of the core. In addition to the ECCS network, the BWR/2 design includes isolation condensers to provide decay heat removal from the vessel in the event that the main steamline isolation valves are closed and the main condenser is unavailable as a heat sink. The isolation condensers are simply passive heat exchangers located in pools of water above the vessel and outside the containment. Steam from the reactor vessel is passed through the heat exchanger tubes, condensed, and the condensate is returned to the vessel. The water on the shell side of the heat exchange is boiled and vented to the atmosphere.

The BWR/2 plant design also marked the standardization on the pressure suppression containment design. In the pressure suppression containment design, steam released from the reactor vessel is directed into the suppression pool where it is condensed. The containment cooling system, which takes water from the suppression pool, passes it through a heat exchanger where it is cooled, then returns the cooled water to the containment, either back to the pool or through sprays in the drywell and suppression chamber airspace.

The ECCS network design used for the BWR/3 and BWR/4 are similar. The two core spray systems and the ADS system were carried forward from the BWR/2. The jet pump design introduced in the BWR/3 eliminated the large recirculation pipe connection to the bottom head of the vessel. It now became possible to reflood the core region of the vessel up to the top of the jet pumps. The containment cooling system described above grew in scope and became the Residual Heat Removal (RHR) system. The RHR system was connected to the reactor vessel to provide decay heat removal when the reactor is shut down. During the short term part of the accident, the flow from the four RHR pumps is directed into the recirculation discharge lines and injected through the jet pumps into the core shroud region of the vessel. This mode is called the Low Pressure Coolant Injection (LPCI) mode. Once the core is reflooded, the RHR pumps are realigned into the containment cooling mode. The two core spray systems are dedicated to core cooling. In addition, a turbine driven large capacity high pressure coolant injection (HPCI) pump was added to the ECCS network. This pump provides inventory makeup for small breaks with the vessel at operating pressure. The improved core cooling provided by the ECCS network allowed larger plants with higher power cores to be designed. With the higher core powers, though, the size of the isolation condenser became impractical. In place of the isolation condensers, the later BWR/3s and all the BWR/4s used a second non-safety turbine driven injection pump to provide inventory makeup when the vessel is isolated. The Reactor Core Isolation Cooling (RCIC) system is similar to the HPCI but smaller in capacity because it is sized to make up only the boiloff due to decay heat. The control valves and instrumentation for the HPCI and RCIC systems are powered from the station emergency batteries, allowing them to function in station blackout situations where no power is available from offsite or the station emergency diesel generators.

At the time the BWR/2, BWR/3 and BWR/4 ECCS were designed, there were no regulations defining the ECCS performance acceptance criteria. The design intent of the ECCS was to prevent the fuel cladding from melting and to keep the core in a coolable geometry during a LOCA. GE designed the ECCS for these plants to limit the peak cladding temperature below 2,700°F, which provided some margin to the melting point of approximately 3,400°F. In 1973, 10CFR50.46 was issued which defined the ECCS performance acceptance criteria. In these new regulations, the peak cladding temperature was limited to 2,200°F. More sophisticated analytical models, combined with a transition in fuel design from a 7x7 lattice to an 8x8 lattice with the fuel rods operating at lower powers allowed the plants to minimize the impact of the new regulations on the plant power output.

The BWR/5 was designed during the timeframe when the ECCS performance issues were being raised. The ECCS network incorporated in the BWR/5 improved the performance and reliability of the ECCS over the BWR/3-4 design. The network consists of three electrical divisions. Two divisions each contain two ECCS pumps, an RHR pump that can be used for LPCI injection or containment cooling and either a low pressure core spray system or a second LPCI pump dedicated to vessel injection. The third electrical division has a High Pressure Core Spray (HPCS) system powered by a dedicated diesel generator, which replaces the turbine driven HPCI. The HPCI turbine would trip off when the vessel depressurized below about 100 psi, whereas the electrically driven HPCS operates through the full vessel pressure range. Also, the HPCI injects through the feedwater line into the vessel downcomer region. With a large recirculation line break, the water injected by the HPCI would be lost out the break and not reach the core region. The HPCS injects directly into the core region through one of the two core spray spargers. The LPCI flow path was rerouted from the recirculation discharge line to dedicated lines allowing the LPCI to inject directly into the core region inside the core shroud. In the earlier BWR/3-4 design, a break of the recirculation discharge line also disabled the injection flow path for the LPCI injecting into that division. The RCIC was retained to provide makeup when the vessel is isolated and for station blackout events. The BWR/5 was carried forward into the BWR/6 with the capacities increased to accommodate the higher BW/6 core power levels. The ECCS network design was able to meet the requirements imposed by 10CFR50.46 without any impact on plant operations.

The internal recirculation pumps in the ABWR eliminated the large external recirculation loop piping. The ABWR was the first BWR designed after the adoption of the 10CFR50.46 limits. With the elimination of the large pipes attached to the lower section of the vessel, it was now possible to design the ECCS so that the core would remain covered during the design basis LOCA, thus assuring adequate core cooling throughout the event. It was possible to significantly downsize ECCS equipment as a result of eliminating large vessel nozzles below the top of the core. Capacity requirements are sized based on operating requirements - transient response and shutdown cooling – rather than on the need for large reflood capability. Inside the reactor vessel, core spray spargers were eliminated, since no postulated LOCA would lead to core uncovery. For transient response, the initiation water levels for RCIC and the high pressure core flooder were separated so that there is reduced duty on the equipment relative to earlier BWRs. There are three complete shutdown cooling loops, including dedicated vessel nozzles. Complex operating modes of the Residual Heat Removal (RHR) Systems, such as steam condensing, were eliminated. Finally, heat removal, in addition to core injection, was automated so that the operator no longer needs to choose which mode to perform during transients and accidents.

Like the ABWR, the ESBWR is designed to keep the core covered and cooled during a LOCA. However, the ESBWR core and containment cooling systems represent a radical departure from those of the earlier BWR product lines in that the cooling systems are passive and do not rely on electrically driven pumps. The ESBWR passive safety systems are discussed in more detail in later Section.

1.3.4 Containment Design

The containment is essentially a pressure vessel housing the reactor that provides a safety barrier to protect personnel and the public in the event of a reactor accident. Figure 3 illustrates the evolution of the BWR containment from the earliest versions to today's ESBWR design. Most of the initial BWR/1 plants used spherical dry containments. The containment vessel was sized to be able to hold the sensible energy content of the steam and hot water in the reactor coolant system and limit the pressure rise to within the design value of the vessel. Larger reactor sizes could be accommodated by increasing the volume of the containment vessel and by increasing the pressure rating of the vessel. However, because of the higher energy content in the BWR vessel as compared to a similar PWR, dry containments could not be designed which were economically competitive with PWRs. Therefore, GE selected the pressure suppression containment for the standard BWR plants [7].

Figure 3 Evolution of the BWR Containment Design

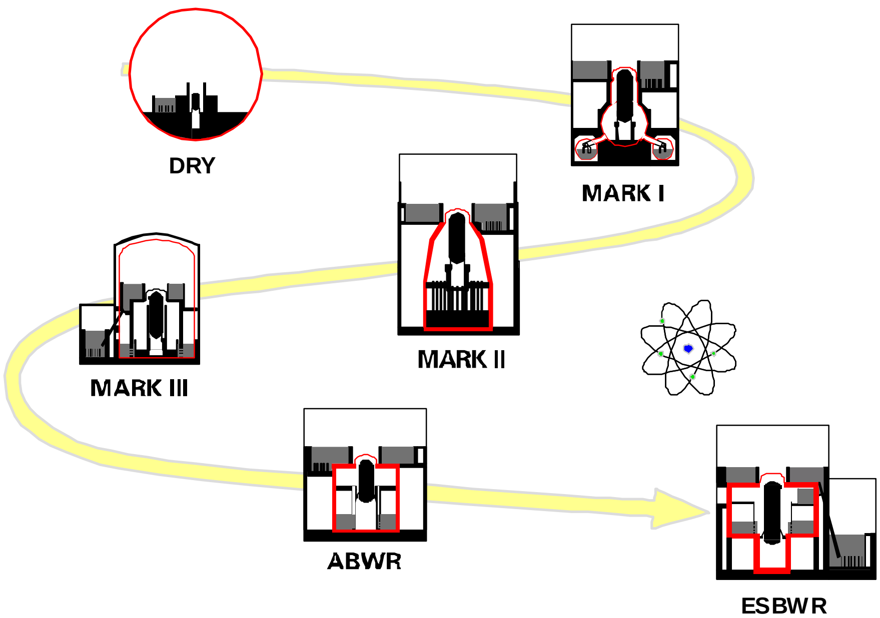

DRY
MARK I
MARK II
MARK III
ABWR
ESBWR

Table 3 Comparison of Key Features of GE BWR Containments

Parameter	Mark I	Mark II	Mark III	ABWR	ESBWR
Power level, MWe	1100	1100	1220	1371	1600
Reactor pressure vessel inside diameter, m	6.4	6.4	6.0	7.1	7.1
Drywell Free volume, m^3 Design pressure, MPa	 4672 0.43	 7872 0.31	 7929 0.17	 7350 0.31	 7206 0.31
Wetwell Free volume, m^3 Water volume, m^3 Design pressure MPa	 4834 3480 0.43	 5318 3268 0.31	 32904 4332 0.10	 5960 3580 0.31	 5467 4383 0.31
Vents Orientation Size, m Number	 Vertical 0.6 76	 Vertical 0.6 70	 Horizontal 0.7 117	 Horizontal 0.7 30	 Horizontal 0.7 30

Table 3 provides a comparison of the key features of the pressure suppression containments used in the BWR plants. In a pressure suppression containment, the reactor and primary system piping are contained in one chamber, the drywell, and the suppression pool is contained in a separate chamber, the wetwell. A system of vents provide a flow path from the drywell to the wetwell, with the vents discharging below the surface of the suppression pool. In the event of a LOCA, the steam and water from the vessel, as well as the air in the drywell, are driven through the vents into the suppression pool where the water in the pool quenches the steam. The advantages of a pressure suppression containment are:

- High heat capacity
- Lower design pressure
- Superior ability to accommodate rapid depressurization
- The pool has the ability to filter and retain fission products
- A large source of readily available makeup water in the case of accidents
- Simplified, compact design

The Mark I containment was the first of the pressure suppression containment designs. The Mark I design has a characteristic light bulb configuration for the drywell, surrounded by a steel torus that houses a large water pressure suppression pool. The light bulb design came from the need to have a removable closure at the top for reactor servicing and refueling and sufficient room at the bottom for the recirculation piping. The torus design provided a large surface area for the vents. The structural analyses of the drywell and torus was also simplified because closed-form solutions were available for these simple shapes (the Mark I design was developed in an era of limited computing power). The drawbacks of the Mark I design was the difficulty of its construction and the construction of the reactor building around it, and the limited room inside the drywell for construction of the primary system piping and maintenance of components. The Mark I containment was used with the BWR/2, BWR/3, and BWR/4.

The Mark II over-under configuration was designed to address the construction issues associated with the Mark I design. The main advantages of the Mark II design were 1) more volume in the drywell to allow better access to the steam and ECCS piping, 2) simpler vent configuration using straight pipes, 3) the potential to use different construction materials, and 4) a smaller reactor building. The Mark II containment design was introduced with the BWR/5 but was also used for a couple of the late BWR/4 plants.

The Mark III containment design, introduced with the BWR/6, represented a major improvement in simplicity. Its containment structure is a right circular cylinder that is easy to construct, and provides access to equipment and ample space for maintenance activities. The large volume allowed a reduction in the design pressure. Other features of the Mark III include horizontal vents to reduce overall loss-of-coolant (LOCA) dynamic loads and a freestanding all-steel structure to ensure leak tightness.

The ABWR containment is significantly smaller than the Mark III containment because the elimination of the recirculation loops translates into a significantly more compact containment and reactor building. The structure itself is made of reinforced concrete with a steel liner from which it derives its name – RCCV, or reinforced concrete containment vessel. The ESBWR containment is similar in construction to the ABWR, but slightly larger to accommodate the passive ECCS systems.

2. ESBWR MAJOR SYSTEMS

2.1 NATURAL CIRCULATION DESIGN

Natural circulation in BWRs is a proven technology [8, 9]. Some of the early GE BWRs employed natural circulation. These were small plants (e.g. Dodewaard at 183 MWt and Humboldt Bay at 165 MWt), but they demonstrated the feasibility of the natural circulation BWR and provided valuable operating data and experience. GE moved to forced circulation plants to achieve higher power ratings in a compact pressure vessel. Pressure vessel fabrication capability at the time was a factor in this decision. Now, after several decades, GE is returning to natural circulation for the ESBWR.

The ESBWR uses natural circulation to provide core flow. Natural circulation in the ESBWR is established due to the density differences between the water in the vessel annulus (outside the shroud and chimney) and the steam/water mixture inside the shroud and chimney. The colder, higher density water in the annulus creates a higher pressure or a driving head when compared to the hotter, lower density fluid (steam/water) in the core and chimney. The energy produced in the core of the reactor heats the water entering at the bottom of the core, and begins converting it to a steam/water mixture. In the core the subcooled water is first heated to the saturation temperature, and then as more heat is added boiling of the core coolant starts. As the coolant travels upward through the core the percent of saturated steam increases until at the exit of the core the average percent of saturated steam is approximately 18 weight %. This steam/water mixture travels upward through the chimney to the steam separators where centrifugal force separates the steam from the water. The separated, saturated water returns to the volume around the separators while the slightly "wet" steam travels upward to the steam dryer and eventually out the main steamline nozzles and piping to the turbine.

Cooler feedwater re-enters the vessel at the top of the annulus, where it mixes with the saturated water around the separators and subcools this water. The resulting mixture is subcooled only a few degrees below the saturation temperature. The cooler mixture then travels downward through the annulus to re-enter the core. The water therefore forms a recirculation loop within the vessel. The mass of steam leaving the vessel is matched by the mass of feedwater entering.

The chimney adds height to this density difference, in effect providing additional driving head to the circulation process. A forced circulation BWR acts in the same basic manner but uses the internal or external pumps to add driving head to this recirculation flow instead of the elevation head provided by the chimney. A pump has entrance and exit losses associated with it and the pump must overcome these losses as well as produce the driving head to overcome these losses.

Natural circulation provides major simplification by removal of the recirculation pumps and associated piping, heat exchangers and controls. It is also synergistic with two other requirements that GE considered to be important in the design of a new Generation III+ reactor:

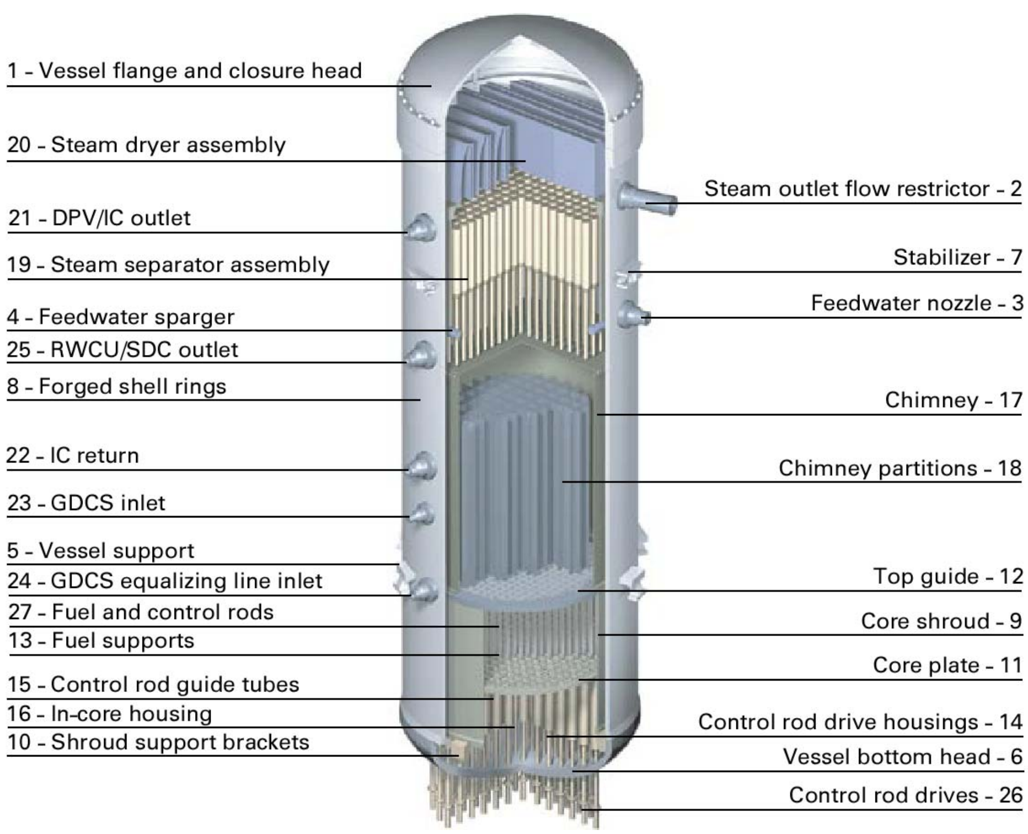
(1) Large safety margins with a very reliable passive Emergency Core Cooling System (ECCS), and

(2) Avoidance of safety/relief valve (SRV) opening for pressurization transients such as turbine trips or main steam line isolation events.

Both of these features need a tall pressure vessel with large water volumes. The tall vessel leads to enhanced natural circulation flow, so the natural circulation capability comes with no additional cost.

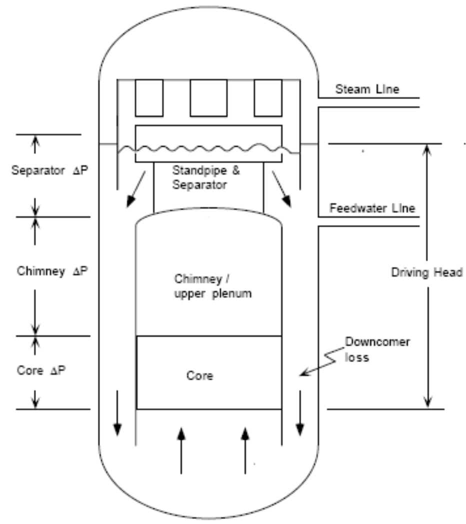
The ESBWR builds on the design features of operating BWRs. Figure 4 shows a cutaway of the ESBWR reactor pressure vessel. Most components in the ESBWR are standard BWR components that have been operating in the field for years (steam separators, control rods and guide tubes, core support structure, etc.). The core consists of conventional BWR fuel bundles, shortened from 12 ft. to 10 ft. to improve pressure drop and stability characteristics. The absence of hardware in the downcomer (jet pumps or internal pumps) reduces flow losses and further enhances natural circulation. The main difference is the taller reactor vessel with the addition of a partitioned chimney above the core and a correspondingly taller downcomer annulus. The fluid in the taller downcomer provides the additional driving head for natural circulation flow through the core, as well as a large water inventory for a Loss of Coolant Accident (LOCA). Steam in the chimney also provides a cushion to dampen void collapse in the core during pressurization transients, leading to a softer response with no SRV discharges.

Figure 4 ESBWR Reactor Assembly



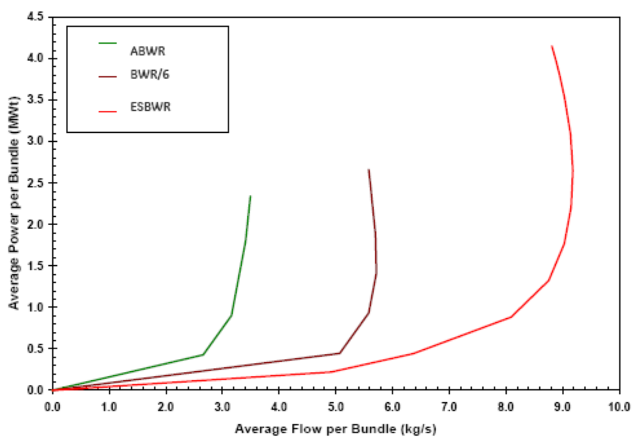
Natural Circulation Flow

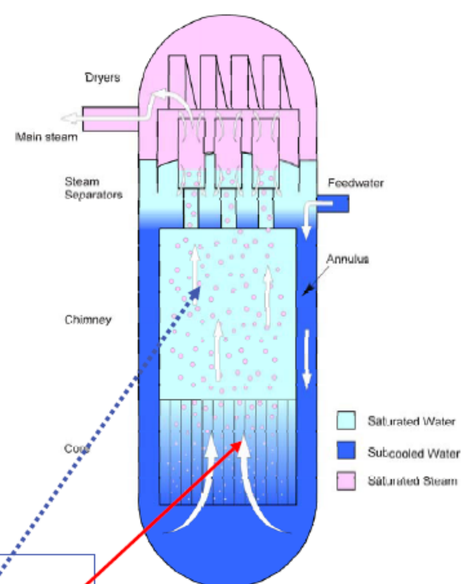
- Core flow depends on
  - > driving head
  - > losses through the loop
- Driving head
  - > proportional to core + chimney height
    - – **Void Fraction**
- Loop losses
  - > downcomer
    - – **Single-phase Δp, handbook loss coefficient**
  - > core (fuel bundle)
    - – **Two-phase Δp, test data/correlation**
  - > chimney ~ small
  - > separator
    - – **Two-phase Δp, test data/correlation**



Schematic of Flow and Pressure Drops in a Reactor

Enhanced Natural Circulation





- Higher driving head
  - Chimney/taller vessel
- Reduced flow restrictions
  - Shorter core
  - Increase downcomer area

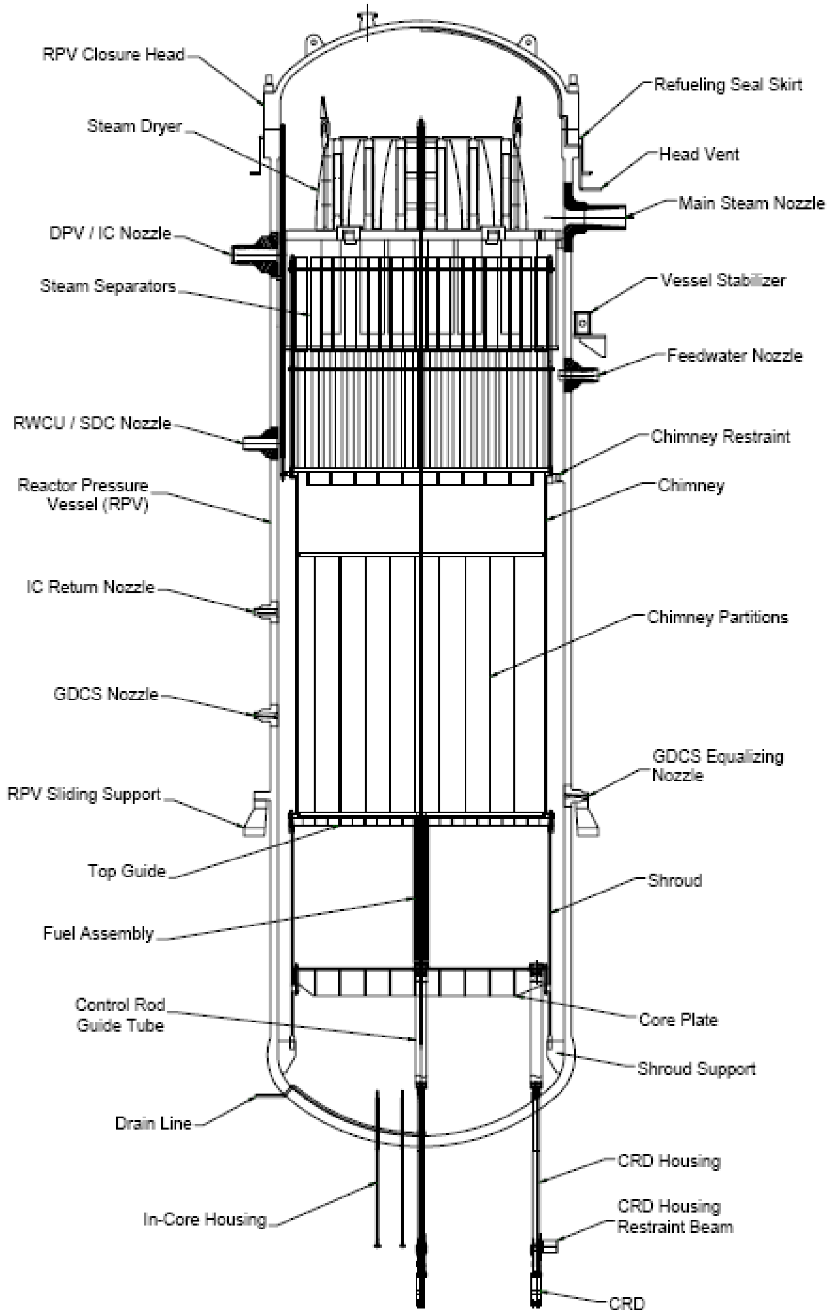
2.2 ESBWR PRESSURE VESSEL AND INTERNALS

The Reactor Pressure Vessel (RPV) assembly for ESBWR consists of the pressure vessel and its appurtenances, supports and insulation, and the reactor internals enclosed by the vessel (excluding the core, in-core nuclear instrumentation, neutron sources, control rods, and control rod drives). The Reactor Pressure Vessel (RPV) system key features are shown in Figure 5.

The RPV consists of a vertical, cylindrical pressure vessel of welded construction, with a removable top head, and head flanges, seals and bolting. The vessel also includes penetrations, nozzles, shroud support, and venturi shaped flow restrictors in the steam outlet nozzles. The shroud support carries the weight of peripheral fuel assemblies, neutron sources, core plate, top guide, shroud, chimney and chimney head with steam separators, and it laterally supports the fuel assemblies. Sliding block type supports near the bottom of the vessel support and anchor the vessel on the RPV support structure in the containment.

The overall RPV height permits natural circulation driving forces to produce abundant core coolant flow. An increased internal flow-path length relative to most prior BWRs is provided by a long “chimney” in the space, which extends from the top of the core to the entrance to the steam separator assembly. This chimney feature existed in the Humboldt Bay and Dodewaard natural circulation BWRs (mentioned above). The chimney and steam separator assembly are supported by a shroud assembly, which extends to the top of the core. The large RPV volume provides a large reserve of water above the core, which translates directly into a much longer period of time (compared to prior BWRs) before core uncovery can occur as a result of feedwater flow interruption or a LOCA. This gives an extended period of time during which automatic systems or plant operators can re-establish reactor inventory control using any of several normal, nonsafety-related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety-related equipment. The large RPV volume also reduces the reactor pressurization rates that develop and can eventually lead to actuation of the safety-relief valves when the reactor is suddenly isolated from the normal heat sink.

The FMCRDs are mounted into permanently attached CRD housings. The CRD housings extend through, and are welded to CRD penetrations (stub tubes) formed in the RPV bottom head. The RPV insulation is supported from the shield wall surrounding the vessel. A steel frame that is independent of the vessel and piping supports insulation for the upper head and flange. Insulation access panels and insulation around penetrations are designed for ease of installation and removal for vessel in-service inspection and maintenance operations.



Reactor Pressure Vessel System Key Features

Figure 5

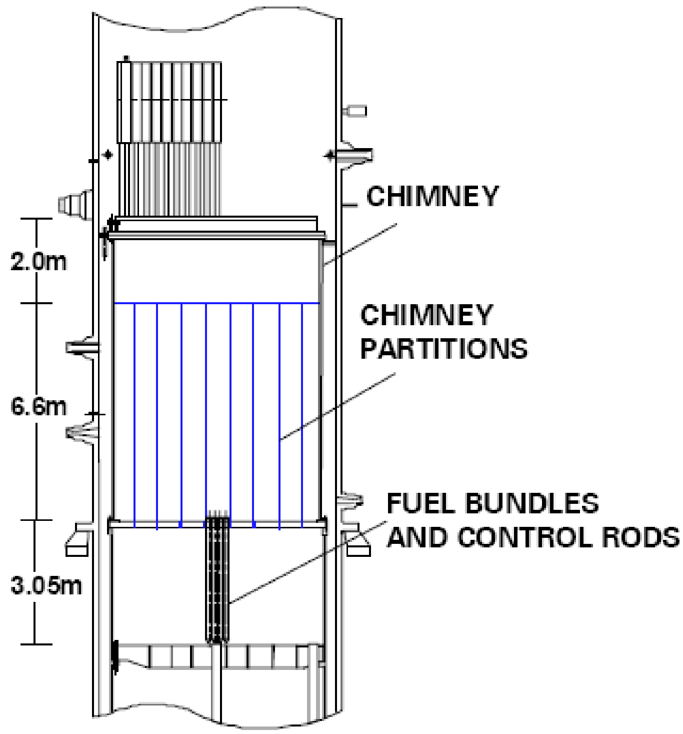


Figure 6 - Dimensions for chimney, chimney partitions and core

2.3 ESBWR CORE

The reactor core is an upright cylinder located within the reactor vessel. The major components of the reactor core are the fuel assemblies, control rods, and incore detectors. The water flowing past the fuel picks up the heat of fission being produced in the fuel. This process provides cooling for the fuel and, as a result, transforms the water into steam.

It is important that the steam leaving the reactor vessel be as free of water as possible in order to prevent carryover into the steam piping and resulting damage to the turbine blades in the feed pump turbines and in the main turbine generator. To accomplish this, the steam and water
mixture exiting the reactor core is directed into steam separators via the upper plenum, where most of the liquid water is removed. The steam passes through the steam dryer assembly, where the remaining water is removed. Steam then exits the reactor vessel through the main steamline. Water removed in the steam separators and dryer assembly mixes with the incoming feedwater in the downcomer annulus.

2.3.1 Nuclear Fuel Assemblies

The nuclear fuel is the source of fission energy in a nuclear reactor. In the ESBWR, the fuel consists primarily of natural uranium (uranium-238) slightly "enriched" with the isotope uranium-235 (initial core: 1.7 % to 3.2 %; reload core: 4.2 %).

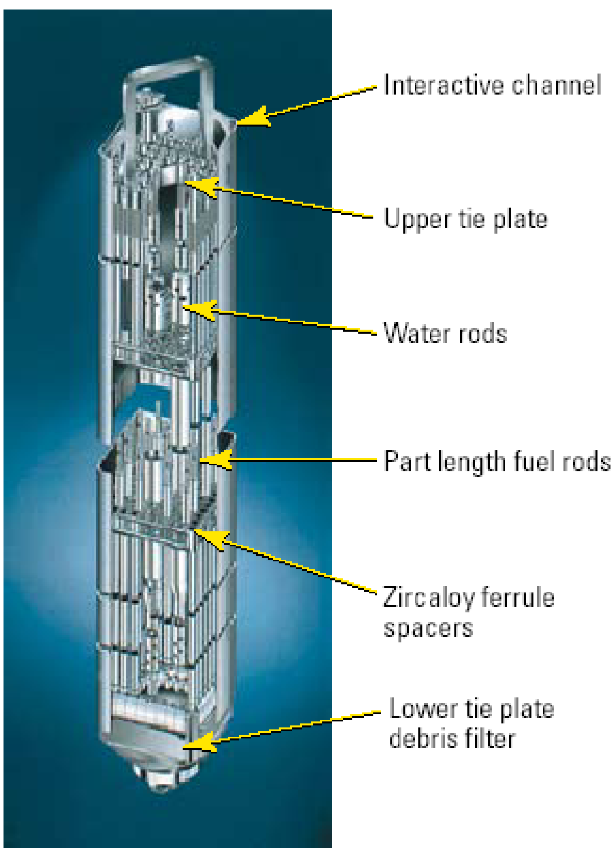


Figure 7 - GE14 Fuel Assembly.

Each of the 1132 removable fuel assemblies (GE14 – see Figure 7 above) consists of a bundle of rods approximately 12 feet long. Each assembly consists of 92 rods in a 10-by-10 array; 90 fuel rods containing the nuclear fuel, and two water rods containing water only. Each of the rods is approximately one half-inch in diameter. The fuel is in the form of uranium oxide pellets stacked inside, and sealed by, the fuel rods.

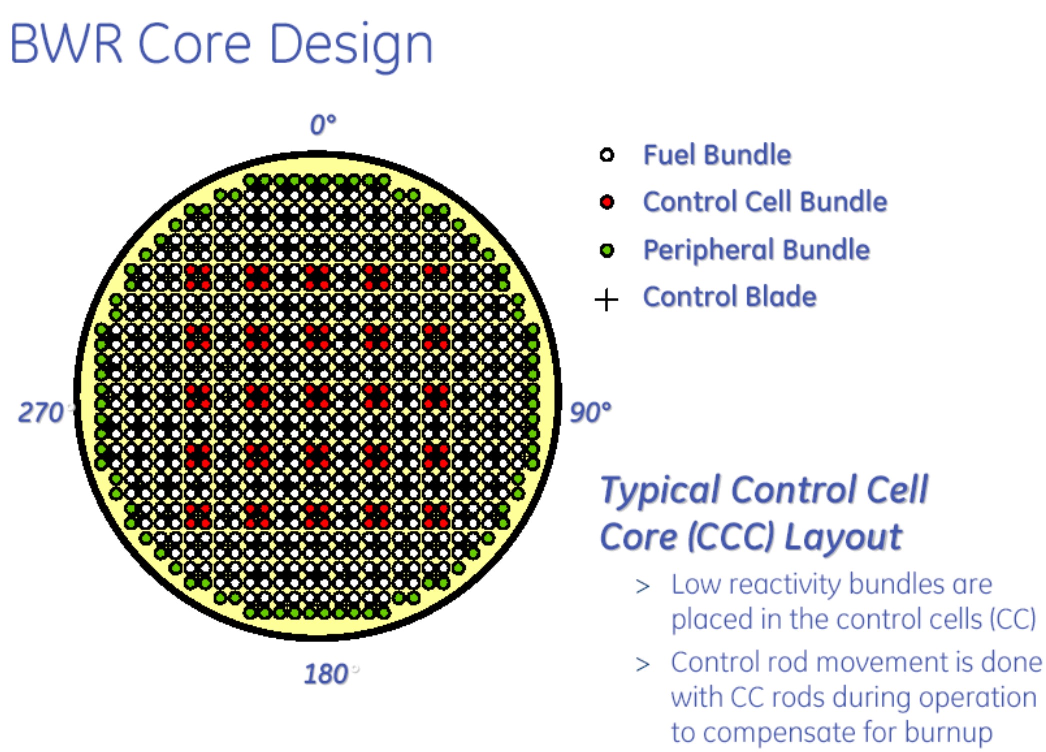
The fission reaction liberates energy in the form of heat and, if enough fission reactions are taking place, enough heat will be generated to heat both the fuel and the water flowing through and around the fuel assemblies.
Control Rods

2.3.2 Control Rod Drive System

There are 269 control rods which extend up from the bottom of the reactor core within the channel formed by the four fuel assemblies in each control cell. The control "rods" are in fact cruciform shaped blades. The control rod blades are approximately 12 feet in length and are driven vertically, up and down, by control rod drive mechanisms attached to the bottom head of the reactor vessel. Thin tubes within the blades of the control rods contain the element boron which add negative reactivity by absorbing the neutrons which otherwise would be available for sustaining the fission process in the core.

The 1132 fuel assemblies are arranged in the reactor core (Figure 8 below) in such a manner that, with the exception of fuel assemblies at the outside edges of the core, each of the 269 control rods is surrounded by four fuel assemblies. This arrangement of a control rod and four fuel assemblies is called a control cell (Figure 9 below). There are 64 Local Power Range Monitoring (LPRM) assemblies evenly distributed throughout the reactor core.

Figure 8 – ESBWR Core Design



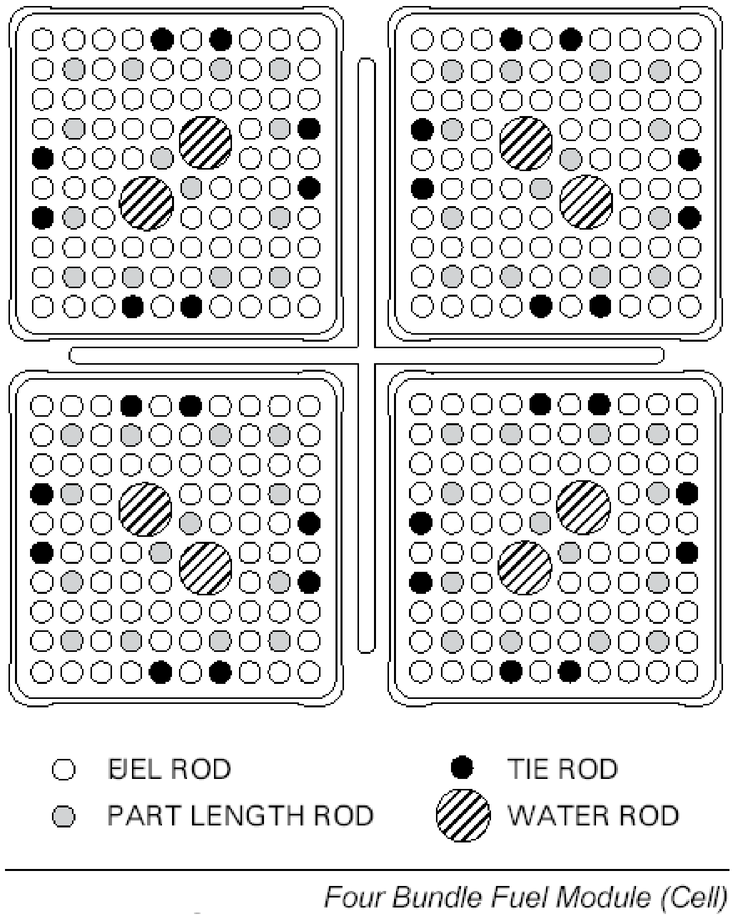
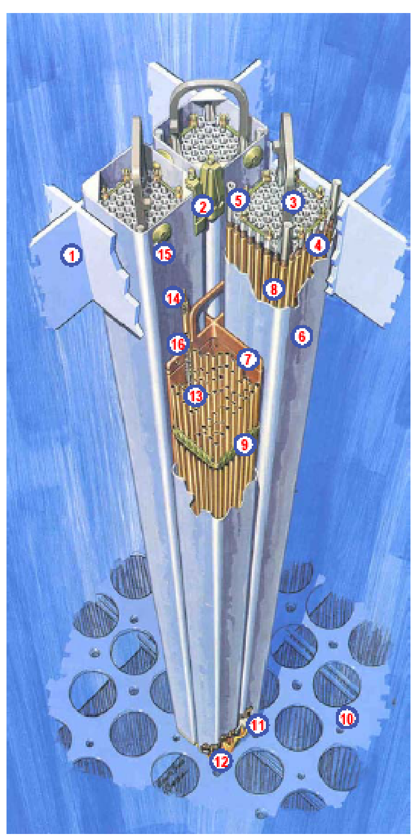


Figure 9 - The arrangement of a control rod and four fuel assemblies - a control cell.

BWR Fuel Assemblies and Control Rod



1 Top Fuel Guide
2 Channel Fastener
3 Upper Tie Plate
4 Expansion Spring
5 Locking Tab
6 Channel
7 Control Rod
8 Fuel Rod
9 Spacer
10 Core Plate Assembly
11 Lower Tie Plate
12 Fuel Support Piece
13 Fuel Pellets
14 End Plug
15 Channel Spacer
16 Plenum Spring

When starting up the reactor, the control rods are slowly withdrawn one at a time downward from the core allowing the fission process to proceed and, therefore, reactor power to be raised
in a controlled manner. When the control rods are fully inserted upwards into the core, a sufficient number of neutrons are absorbed to prevent any power production.

Each control rod can be individually selected for insertion or withdrawal. The Rod Sequence Control System enforces interlocks to limit control rod "worth." Control Rod Worth is defined as the amount of “reactivity” added as a result of an increment of rod motion.

The Control Rod Drive (CRD) system is composed of three major elements:

- The Fine Motion Control Rod Drive (FMCRD) mechanisms;
- The hydraulic control unit (HCU) assemblies, and;
- The Control Rod Drive Hydraulic (CRDH) subsystem.

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) for abnormal operating conditions. Simultaneous with scram, the FMCRDs also provide electric-motor-driven run-in of all control rods as a path to rod insertion that is diverse from the hydraulic-powered scram. The hydraulic power required for scram is provided by high-pressure water stored in the individual HCUs. Each HCU is designed to scram up to two FMCRDs. The HCUs also provide the flow path for purge water to the associated drives during normal operation. The CRDH subsystem supplies high pressure demineralized water, which is regulated and distributed to provide charging of the HCU scram accumulators, purge water flow to the FMCRDs, and backup makeup water to the RPV when the feedwater flow is not available.

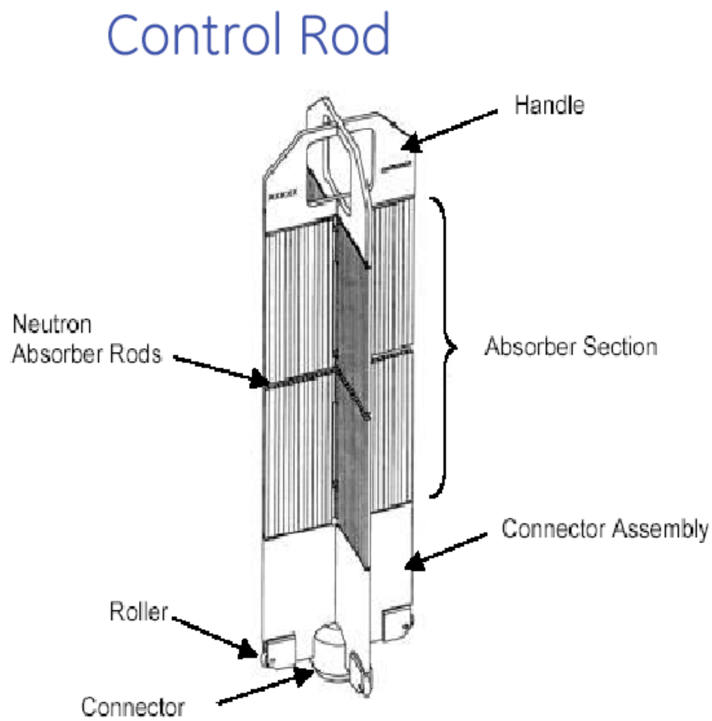


Figure 10 - ESBWR Control Rod

During power operation, the CRD system controls changes in core reactivity by movement and positioning of the neutron absorbing control rods within the core in fine increments via the FMCRD electric motors, which are operated in response to control signals from the RC&IS.

The CRD system provides rapid control rod insertion (scram) in response to manual or automatic signals from the Reactor Protection System (RPS), so that no fuel damage results from any plant transient.

The FMCRDs are mounted in housings welded into the RPV bottom head. The CRD system is separated both physically and electrically from the Standby Liquid Control (SLC) system.

The control rods are used as the primary mechanism for controlling the rate at which fissions take place in the nuclear fuel and, therefore, the primary mechanism for controlling the power level in the reactor.

2.3.3 Rod Sequence Control System (RSCS)

The RSCS provides control rod selection and movement interlocks to enforce a mandatory sequence of movements when withdrawing or inserting control rods. A mandatory sequence is
required to maintain relative positions between adjacent control rods within predetermined limits in order to limit the reactivity worth of any given amount of control rod motion.

Note that, as more rods are withdrawn to maintain a heatup from cold to hot conditions, overall rod worths (the worth of all notch positions) decrease as a result of the geometry effects of a "larger" core. At higher power levels, rod worths decrease even more drastically as core voids increase.

The Rod Sequence Checkoff Procedure is an operating procedure to be used by ESWBR plant operator. It shows the required sequence of rod movements enforced by the RSCS, and is used by the operator to document each rod movement step. The rod movement sequence is divided into groups denoted by the "GROUP #" designation on the Rod Sequence Procedure Checkoff Sheets.

The steps in each group indicate the order in which the rods assigned to the group are to be moved, and the positions in the reactor core they are moved between.

The RSCS allows rods within an RSCS Group to be inserted or withdrawn in any order, but requires that RSCS groups be moved in sequence as follows:

- When withdrawing rods, all rods in the present group must be withdrawn to their Withdraw Limit prior to selecting a rod in the next higher group.
- Conversely, when inserting rods, all rods in the group must be inserted to their Insert Limit prior to proceeding to steps in the next lower group.
- If an attempt is made to select a rod for which the above criteria is not met, the selection is prevented and a red SELECT ERROR display lights illuminates on the Rod Select Panel.

2.3.3.1 Limitations on Control Rod Insertion and Withdrawal:

The RSCS establishes maximum rod withdrawal and insertion limits for the control rods in each group as follows:

- When withdrawing rods within a group, withdraw motion is blocked (stopped) if a rod within the group is inadvertently withdrawn past the group's Withdraw Limit.
- Conversely, when inserting rods, the sequence logic prevents further insertion if a rod is inserted past its Insert Limit.

2.3.4 Neutron Monitoring System

The Neutron Monitoring System consists of electronic monitoring devices which process signals proportional to reactor power from detectors located within the reactor core. These detectors provide the operator with an indication of the rate of neutron production in the fuel region.

There are three types of in-core detectors used in the ESBWR. Each type of detector operates over a certain range of detection.

- At very low power, a highly sensitive detector is needed to "count" the relatively small number of neutrons existing adjacent to the detector.
- At higher power levels, the increased neutron population in the core can "overwhelm" these detectors such that they become unresponsive to further increases in power. For this reason, a less sensitive, intermediate range detector is provided.
- At even higher power levels, up to full reactor power, a third type of detector is used to provide reactor power indication, as the intermediate range detectors reach their indicating limits.

The Neutron Monitoring System is divided into three monitors or sub-systems corresponding to the three types of detectors; the Source Range Monitor (lowest range), the Intermediate Range Monitor (mid-range) and the Average Power Range Monitor (high range).

2.3.4.1 Source Range Monitor (SRM):

The SRM provides a display of the relative neutron population in the reactor core in units of counts per second (cps). The SRM provides reactor power level (neutron production) indication over the lowest range of monitoring sensitivity, termed the source range. The source range provides indication of changes in reactor power level as the fission process is first increased to a self-sustaining level (criticality) and will overlap with the next higher range of nuclear instrumentation - the Intermediate Range Monitor (IRM). The SRM provides rod block signals to inhibit control rod withdrawal whenever the SRM indication is not within normal limits.

2.3.4.2 Intermediate Range Monitor (IRM):

The IRM provides a display of relative neutron population in the reactor core in units of percent of scale (not percent of full reactor power). The IRM provides indication of reactor power between, and overlaps with, the Source Range Monitor (SRM) at the lower end of its indication and the Average Power Range Monitor (APRM) at its upper end. An IRM Range Switch is used to keep the IRM indication "on scale" as power is raised and lowered in the intermediate range. The IRM provides rod block and scram signals.

2.3.4.3 Average Power Range Monitor (APRM):

The APRM provides a display of reactor power in units of percent of full power. The APRM averages the inputs from a large number of detectors distributed within the reactor core. Although the APRM measures neutron flux (distribution), it is calibrated to read in percent of the maximum reactor heating power, termed rated thermal power. At 100 percent reactor power, the fuel will generate over 3 billion watts of heat energy (3,292 MW thermal). The flow of water past the fuel (core flow) removes this heat energy as the water is transformed into steam.

2.3.5 Reactor Protection System (RPS)

The RPS is an instrumentation and logic system which senses a number of plant parameters and initiates a rapid reactor shutdown, or "scram," if pre-established limits or setpoints are exceeded.

A manual or automatic scram signal will cause valves (called scram valves) to open, admitting
high pressure water to the lower side of a drive piston on each of the 269 control rod drive hydraulic control unit (HCU) assemblies. The resulting force rapidly inserts the attached control rods into the core. The control rods contain the element boron which has a strong affinity for neutron absorption. The large amount of negative reactivity added from the scram transient rapidly reduces power to shutdown levels.

2.3.5.1 RPS Logic System:

The RPS logic system consists of two identical and independent trip systems, Trip System A and Trip System B, consisting of sensor instruments and logic relays. Typically, there are four sensor instrument channels associated with each process parameter (pressure, level, etc.).

The outputs from the sensor channels are connected to the trip systems such that two, designated Channel A and C, are connected to Trip System A, and the remaining two, designated Channel B and D, are connected to Trip System B. During normal operation, the output of each individual instrument channel is energized, maintaining the output of the associated trip system energized. The scram valves associated with each of the 269 control rods are therefore held closed, allowing normal control rod operation. If, for instance, only the sensor associated with sensor Channel A deenergizes ("trips"), the output of the associated trip
system, Trip System A only, will deenergize (called a "half scram"). The scram valves are held closed, however, and no scram results since Trip System B remains energized. If then, either sensor Channel B or D trips, Trip System B will also deenergize, the scram valves will all open, and a full scram results. This type of logic is termed "one-out-of-two-once taken twice," meaning that either one of two sensors can trip a trip system resulting in half of a scram ("one-out-of-two-once"), however, the other trip system must also trip (the "taken twice" part) to produce an actual, full scram. A consequence of this logic scheme is that a single, spurious failure producing a trip within one of the four sensor channels will not result in a "false scram" with the resulting, inadvertent loss of power generation. Also, since the scheme is designed such that any single failure within a sensor channel will not prevent a full scram, the design is termed "single failure proof."

The RPS is also termed "fail safe" or "deenergize to trip" since a scram is initiated if sensor or logic system power is lost.

2.3.5.2 Scram Parameters:

A scram is initiated by the RPS in one of two ways:

- Manually when the operator desires to shutdown the reactor should an unsafe condition be observed. To initiate a manual scram, the Manual Scram Pushbutton is depressed.
- Automatically when sensed parameters exceed the scram setpoints.

Although the actual ESBWR plant has additional scram sensors, the ESBWR simulator simulates automatic scrams actuated by any one of six different sensed parameters as follows:

- High neutron power as detected by IRM Upscale, or APRM Upscale
- High Reactor Vessel Pressure
- Low Reactor Vessel Water Level
- High Primary Containment (Drywell) Pressure
- Closure of the Main Steam Isolation Valve (MSIV)

2.3.5.3 Control Rod Blocks

Circuitry within the Neutron Monitoring Systems and the Rod Sequence Control System provide interlock signals to the Control Rod System to inhibit control rod insertion or withdrawal to prevent unsafe conditions from developing. These control rod movement interlocks are termed "rod blocks."

Rod Block Signals from the Neutron Monitoring Systems –
When changing reactor power, the Neutron Monitoring System will provide control rod withdrawal blocks as a result of upscale or downscale conditions. Upscale rod blocks are initiated at setpoints below the upscale scram setpoints to prevent further increases in reactor power by rod withdrawal until conditions are corrected. Downscale withdrawal rod blocks prevent control rod withdrawal until the operator can take corrective action to bring the indication back on-scale.

Rod Block Signals from the Rod Sequence Control System (RSCS) –
The RSCS provides control rod selection and movement interlocks (rod blocks) to enforce a mandatory sequence of movements when withdrawing or inserting control rods. The sequence is required to maintain relative positions between adjacent rods within predetermined limits in order to limit the control rod worth.

- When withdrawing control rods, withdraw motion is blocked if a rod within the group is inadvertently withdrawn past the group's Withdraw Limit.
- When inserting rods within an RSCS Group, insert motion is blocked if a rod is inserted past the group's Insert Limit.

2.3.6 Core Nuclear reactivity Characteristics

In a boiling water reactor, two reactivity coefficients are of primary importance: the **fuel Doppler coefficient** and the **moderator density reactivity coefficient**. The moderator density reactivity coefficient may be broken into two components: that due to temperature and that due to steam voids.

(a) Fuel Doppler Reactivity Coefficient: As in all light water moderated and low enrichment reactors, the fuel Doppler reactivity coefficient is negative and prompt in its effect, opposing reactor power transients. When reactor power increases, the UO2 temperature increases with minimum time delay and results in higher neutron absorption by resonance capture in the U-238.

(b) Moderator Density Reactivity Coefficient: During normal plant operations, the steam void component of the moderator density reactivity coefficient is of prime importance. The steam void component is large and negative at all power levels. This steam void effect results in the following operating advantages:

> –Xenon Override Capability: Since the steam void reactivity effect is large compared with xenon reactivity, the ESBWR core has the capability of overriding the negative reactivity introduced by the buildup of xenon following a power decrease.
>
> –Xenon Stability: The steam void reactivity is the primary factor in providing the high resistance to spatial xenon oscillations in a boiling water reactor. Xenon instability is an oscillatory phenomenon of xenon concentration throughout the reactor that is theoretically possible in any type of reactor. These spatial xenon oscillations give rise to local power oscillations which can make it difficult to maintain the reactor within its thermal operating limits. Since these oscillations can be initiated by reactor power level changes, a reactor which
> is susceptible to xenon oscillations may be restricted in its load-following capability.

The inherent resistance of the ESBWR to xenon instability permits significant flexibility in load-following capability.

> –Load Changing by Control Rod Movement:

The ESBWR is capable of daily load following between 100% and 50% power by adjusting control rod density within the core.

2.3.7 Reactivity Control

Reactor shutdown control in BWRs is assured through the combined use of the control rods and burnable poison in the fuel. Only a few materials have nuclear cross sections that are suitable for burnable poisons. An ideal burnable poison must be essentially depleted in one operating cycle so that no residual poison exists to penalize the cycle length. It is also desirable that the positive reactivity from poison burnup matches the almost linear decrease in fuel reactivity from fission product buildup and U-235 depletion. A self-shielded burnable poison consisting of digadolinia trioxide (Gd2O3), called gadolinia, dispersed in selected fuel rods in each fuel assembly provides the desired characteristics. The gadolinia concentration is selected such that the poison is essentially depleted during the operating cycle. Gadolinia has been used in GE BWRs since the early 1970's, and has proven to be an effective and efficient burnable poison. In addition to its use for reactivity control, gadolinia is also used to improve axial power distributions by axial zoning of the burnable poison concentration.

The core is designed so that adequate shutdown capability is available at all times. To permit margin for credible reactivity changes, the combination of control rods and burnable poison has the capability to shut down the core with the maximum worth control rod pair fully withdrawn at any time during the fuel cycle. This capacity is experimentally demonstrated when reactivity alternations are made to the reactor core, such as during the initial core startup, and during each startup after a refueling outage.

2.3.8 Fuel Management

The flexibility of the ESBWR core design permits significant variation of the intervals between refueling. The first shutdown for refueling can occur anywhere from one to two years after commencement of initial power operation. Thereafter, the cycle length can be varied up to 24 months with GE14 fuel. The desired cycle length can be obtained by adjusting both the refueling batch size and the average enrichment of the reload bundles. A wide range of batch average discharge exposures can be supported depending up licensing limits and uranium supply considerations. While GE can recommend operating margins that have been proven adequate, utility specifications on operating margins can be readily introduced into the ESBWR core.

The average bundle enrichments and batch sizes are a function of the desired cycle length. The initial ESBWR core has an average enrichment ranging from approximately 1.7 wt% U-235 to approximately 3.2 wt% U-235 for cycle lengths ranging from one to two years. For ESBWR reload cores using GE14 fuel, the average bundle enrichment is roughly 4.2 wt% U-235 with a reload batch fraction of 35% for a two year cycle.

2.4 ESBWR OPERATION IN NATURAL CIRCULATION

The ESBWR will adjust reactor power output using control rods, but with electrically driven control rod drives that move slower and have finer positioning capability than the locking piston design for conventional BWRs. Because of its size, ESBWR will not normally be operated in a load follow mode. The FMCRDs can accommodate a duty corresponding to daily load following cycles for 10 years. Variable feedwater temperature control has been added to provide a second means of reactivity control in addition to the fine motion control rod drives.

Present-day BWRs can operate at about 50% of rated power in natural circulation; however, stability considerations prevent steady operation in this region. There have been recirculation pump trip events in operating plants, which led to a natural circulation state at around 50% of rated power. The operating conditions in present day BWRs (power, flow, power distribution) in natural circulation following the pump trip were well predicted by the simulation analysis models used for ESBWR performance analysis.

The operating parameters for the ESBWR, such as the power density, steam quality, void fraction and void coefficient are within the range of operating plant data. Figure 11 shows a comparison of the ESBWR power–flow operating map with those of operating BWRs. This figure is based on the power per bundle and flow per bundle so that a meaningful comparison can be made. The power per bundle and flow per bundle for the ESBWR are both lower than for a modern jet pump plant at rated operating conditions, but the ratio of power to flow is similar to that for an uprated BWR at Maximum Extended Load Line Limit Analysis -Plus (MELLLA+) conditions. This means that the core exit steam quality (ratio of steam flow to core flow) is also similar.

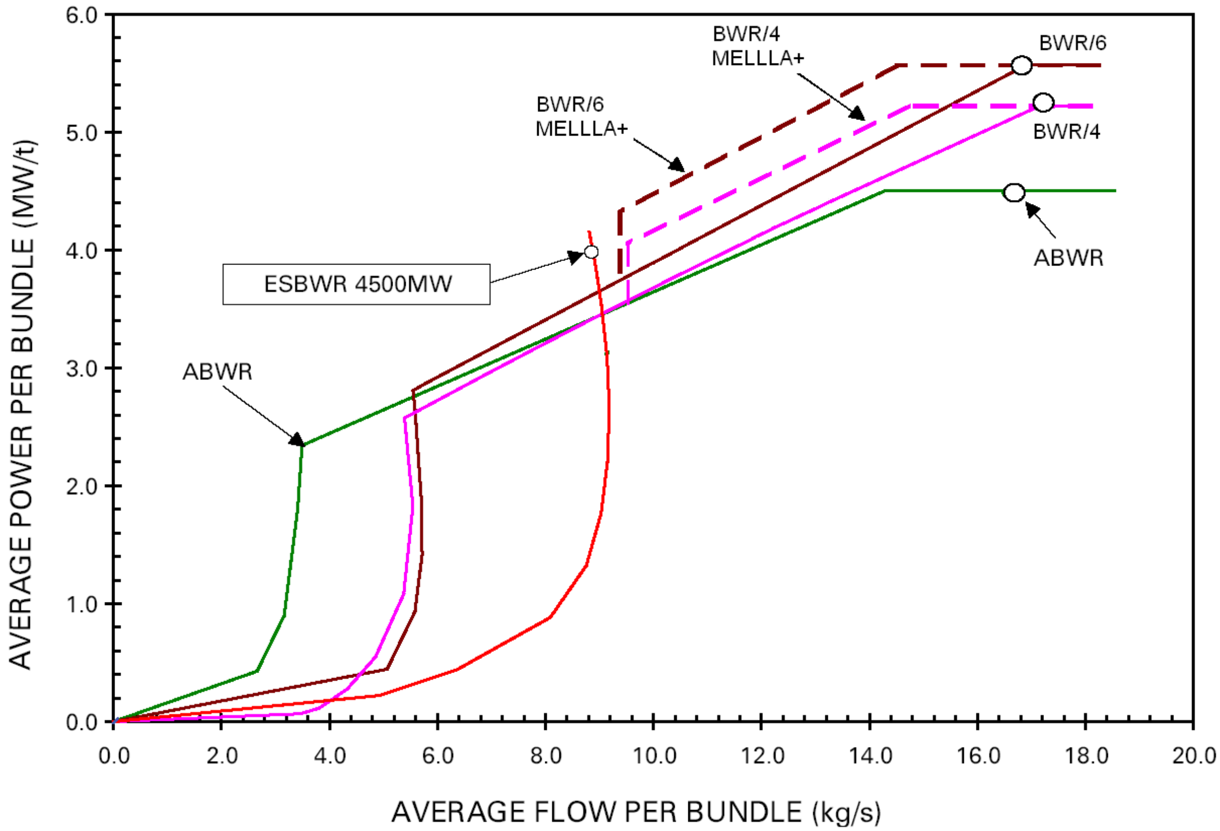


Figure 11 Comparison of ESBWR Operating Power/Flow Map with Operating BWRs

2.5 ESBWR NUCLEAR BOILER SYSTEM

The primary functions of the Nuclear Boiler System (NBS) are:

- To deliver steam from the RPV to the turbine main steam system (TMSS);
- To deliver feedwater from the Condensate and Feedwater System (C&FS) to the RPV;
- To provide overpressure protection of the Reactor Coolant Pressure Boundary (RCPB);
- To provide automatic depressurization of the RPV in the event of a LOCA where the RPV does not depressurize rapidly;
- With the exception of monitoring the neutron flux, to provide the instrumentation necessary for monitoring conditions in the RPV such as RPV pressure, metal temperature, and water level instrumentation.

The main steam lines (MSLs) are designed to direct steam from the RPV to the TMSS; the feedwater lines (FWLs) to direct feedwater from the C&FS to the RPV; the RPV instrumentation to monitor the conditions within the RPV over the full range of reactor power operation.

The NBS contains the valves necessary for isolation of the MSLs, FW lines, and their drain lines at the containment boundary. The NBS contains the safety-relief valve discharge lines, including the steam quencher located in the suppression pool at the end of each discharge line.

The NBS also contains the RPV head vent line and non-condensable gas removal line.

2.6 MAIN STEAM LINE ISOLATION VALVES (MSIV)

Each MSIV assembly consists of a main steam line isolation valve, a pneumatic accumulator, connecting piping and associated controls.

There are two MSIVs welded into each of the four MSLs. On each MSL there is one MSIV inside the containment and one MSIV outside the containment. Each set of two MSIVs isolate their respective MSL upon receipt of isolation signal and close on loss of pneumatic pressure to the valve.

The MSIV has a fast-closing time greater than or equal to the value used in the MSIV closure (non-accident) events analysis and less than or equal to the value used in the MSLB accident analysis. During MSIV fast closure, N_2 or air pressure is admitted to the upper piston compartment. Admitting N_2 or air to both the upper and lower piston compartments tests the valve with a slow closing speed, which is based upon approximately 45-60 seconds for full stroke of the valve.

When all the MSIVs are closed, the combined leakage through the MSIVs for all four MSLs is less than or equal to the value used in the LOCA inside containment radiological analysis.

2.7 SAFETY-RELIEF VALVES

The nuclear pressure relief system consists of safety-relief valves (SRVs) located on the MSLs
between the RPV and the inboard MSIV. The SRVs provide two main protection functions:

- Over pressure Safety Operation: The SRVs function as safety valves and open to prevent nuclear system over pressurization. They are self-actuating by inlet steam pressure. The safety mode of operation is initiated when direct and increasing static inlet steam pressure overcomes the restraining spring and frictional forces acting against the inlet steam pressure at the valve disc. This moves the disc in the opening direction. The condition at which this actuation is initiated corresponds to the set-pressure value stamped on the nameplate of the valves. The rated capacity of the SRVs is sufficient to prevent a rise in pressure within the RPV to more than 120% of the design pressure during Anticipated Transients Without Scram (ATWS) events.

- Automatic Depressurization Operation: Ten of the SRVs open automatically during a LOCA to depressurize the reactor vessel.

Each SRV has one dedicated, independent pneumatic accumulator, which provides the safety-related, ensured nitrogen supply for opening the valve. The SRVs are flange mounted onto forged outlet fittings located on the top of the main steamline piping in the drywell. The SRVs discharge through lines routed to quenchers in the suppression pool.

2.8 AUTOMATIC DEPRESSURIZATION SYSTEM

The Automatic Depressurization System (ADS) function of the NBS depressurizes the RPV in
sufficient time for the Gravity-Driven Cooling System (GDCS) injection flow to replenish core coolant to maintain core temperature below design limits in the event of a LOCA. It also maintains the reactor depressurized for continued operation of GDCS after an accident without need for power.

The ADS consists of SRVs and depressurization valves (DPVs) and their associated instrumentation and controls. The SRVs and DPVs are actuated in groups of valves at staggered times by delay timers as the reactor undergoes a relatively slow depressurization. This minimizes reactor level swell during the depressurization, thereby enhancing the passive re-supply of coolant by the GDCS.

The use of a combination of SRVs and DPVs to accomplish the ADS function improves ADS reliability against hypothetical common-mode failures of otherwise non-diverse ADS components. It also minimizes components and maintenance as compared to using only SRVs or only DPVs for this function. By using the SRVs for two different purposes, the number of DPVs required is minimized. By using DPVs for the additional depressurization capability needed beyond what the SRVs can provide, the total number of SRVs, SRV discharge lines, and quenchers in the suppression pool is minimized. The need for SRV maintenance, periodic calibration and testing, and the potential for simmering are minimized with this arrangement. The ADS automatically actuates on a low RPV water level signal that persists for a preset time.

When a coincident high drywell pressure signal is present, ADS actuates earlier and at a higher RPV water level. Two-out-of-four logic is used to activate the SRVs and DPVs. The persistence requirement for the low RPV water level signal ensures that momentary system perturbations do not actuate ADS when it is not required. The two-out-of-four logic ensures that a single failure does not cause spurious system actuation while also ensuring that a single failure cannot prevent initiation. The ADS may also be manually initiated from the main control room.

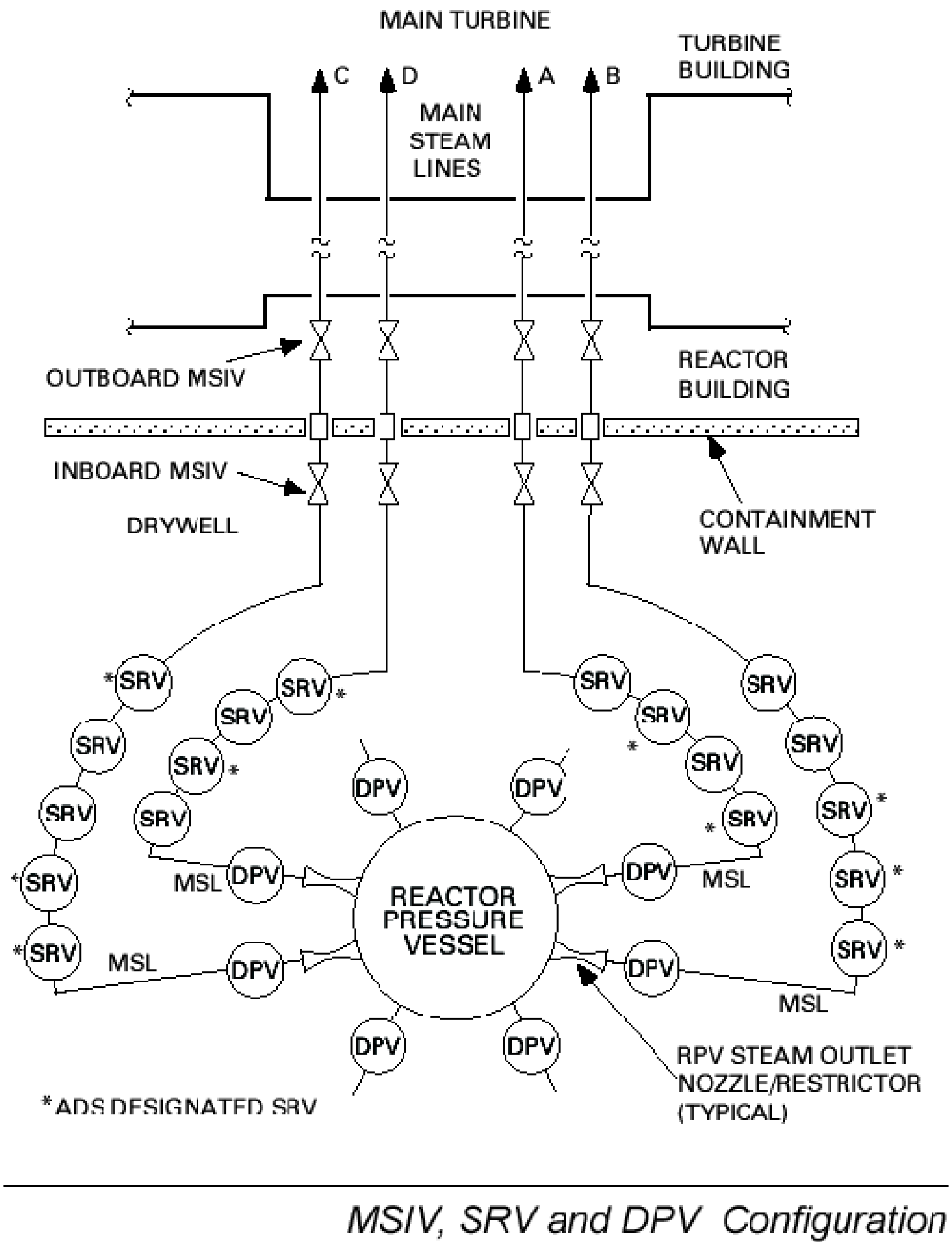


Figure 12 Safety-Relief Valves, Depressurization Valves and Steamline Diagram

2.9 MAIN TURBINE AND GENERATOR

Steam produced in the reactor vessel is used to spin the shaft of the main turbine. The turbine shaft is directly connected to the main generator. When the main generator is rotated, electrical power is produced.

The main turbine shaft rotates as a result of steam admitted onto large steel blades attached to the turbine shaft. The steam is admitted to the turbine blades through four turbine control valves used to regulate the flow of steam being admitted and, therefore, the amount of electricity produced by the main generator. When the plant is operating at its full power level, and the flow of steam from the reactor is at its maximum value, the generator will produce electrical power.

A turning gear motor rotates the turbine shaft at approximately 3 rpm whenever the turbine is shutdown to prevent bowing of the shaft due to uneven heating. On a trip of the turbine, speed will coast down, and the turning gear is automatically engaged.

2.9.1 Main Condenser and Turbine Bypass Valves

The steam flowing through the main turbine is exhausted into the main condenser where the steam is transformed into water. The main condenser also receives steam directly from the main steamline through **the turbine bypass valves**. These valves control the amount of steam which bypasses the main turbine whenever the reactor is producing more steam than the main turbine is using. The system is designed to bypass at least 110% of the rated main steam flow directly to the condenser. The Turbine Bypass System, in combination with the reactor systems, provides the capability to shed 100% of the T-G rated load without reactor trip and without the operation of SRVs.

Steam entering the main condenser is directed over the outside of thousands of tubes located inside of the condenser. The tubes have cold water flowing through them so that the steam on the outside cools and condenses into water. The water so produced is then returned to the reactor vessel by the Reactor Feedwater System.

2.9.2 Circulating Water System

The Circulating Water System recirculates cold through the inside of tubes in the Main Condenser in order to condense the steam flowing around the outside of the tubes. The large flow rate needed to condense the steam requires the plant to be located near a large source of water such as the ocean or a river. The Circulating Water System utilizes large pumps to pump water through the condenser tubes where the water picks up heat. The heated water is then directed to the cooling towers. In the cooling towers, the heat is transferred to the air by evaporation. The evaporation process sends large clouds or "plumes" of clean water vapor into the air.

2.10 REACTOR FEEDWATER SYSTEM

The water formed by condensation of steam in the main condenser is returned to the reactor vessel, through the feedwater heaters, by the Reactor Feedwater System. Water is pumped through demineralizers where any impurities are removed.

The Reactor Feedwater System consists of four 33-37% capacity condensate pumps (three normally operating and one on automatic standby), four 33-45% capacity reactor FW/FW Booster pumps (three normally in operation and one on automatic standby), four stages of low-pressure closed FW heaters, a direct contact FW heater (feedwater tank) and two stages of high-pressure FW heaters, piping, valves, and instrumentation. The condensate pumps take suction from the condenser hotwell and discharge the deaerated condensate into one common header, which feeds the Condensate Purification System (CPS). Downstream of the CPS, the condensate is taken by a single header, through the auxiliary condenser/coolers, (one gland steam exhauster condenser and two sets of SJAE condensers and offgas recombiner condenser (coolers). The condensate then branches into three parallel strings of low-pressure FW heaters.

Each string contains four stages of low-pressure FW heaters. The strings join together at a common header which is routed to the feedwater tank, which supplies heated feedwater to the suction of the Reactor Feedwater (FW) pumps. Each reactor FW/FW Booster pump is driven by an adjustable speed electrical motor.

The Reactor Feed Pumps, raise the pressure of the water so it will flow to the reactor vessel. The feedwater heaters use steam extracted from the main turbine to preheat the feedwater returning to the reactor vessel. By raising the temperature of the feedwater, overall plant efficiency is increased since less nuclear heating is required to transform the water into steam again.

2.10.1 Feedwater Lines (FWLs)

The feedwater piping consists of two FWLs connecting to a feedwater supply header. Two containment isolation valves consisting of a simple check valve inside the drywell and a positive acting check valve outside the containment accomplish isolation of each FWL.

2.10.2 Feedwater Control System

The Feedwater Control System (FWCS) provides logic for controlling the supply of feedwater flow to the reactor vessel in response to automatic or operator manual control signals. This control maintains reactor water level within predetermined limits for all operating conditions including startup. A fault-tolerant, triplicated, digital controller uses water level, steam flow and feedwater flow signals to form a three-element control strategy to accomplish this function.

Single-element control based only on reactor water level is used when steam flow or feedwater flow signals are not available. During very low steam flow conditions during plant startup FWCS regulates the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system overboard flow to maintain reactor water level and to minimize feedwater temperature oscillations.

2.11 PASSIVE SAFETY FEATURES

The ESBWR Safety Systems design incorporates four redundant and independent divisions of the passive Gravity Driven Core Cooling System, the Automatic Depressurization System (ADS) and a Passive Containment Cooling System (PCCS). These systems are shown in Figure 13

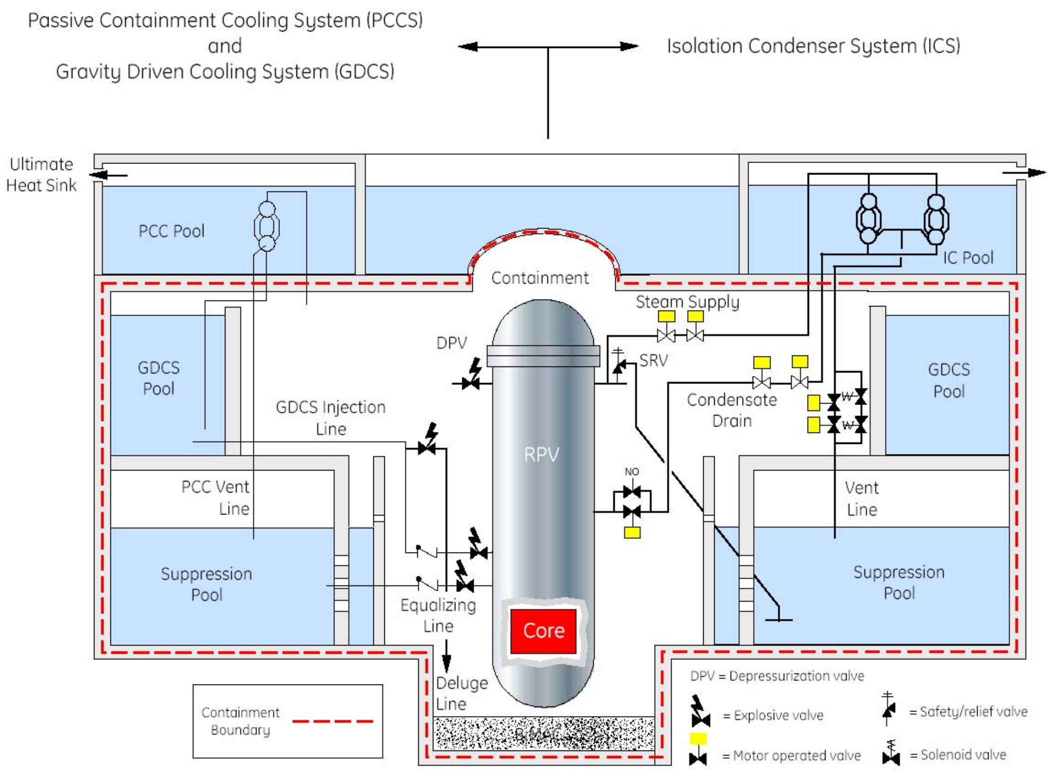


Figure 13 ESBWR Key Safety Systems

Emergency core cooling is provided by the Gravity-Driven Cooling System (GDCS) in conjunction with the ADS in case of a LOCA. When an initiation signal is received, the ADS depressurizes the reactor vessel and the GDCS injects sufficient cooling water to maintain the fuel cladding temperatures below temperature limits defined in US NRC Regulation 10 CFR 50.46.

Heat removal and inventory addition are also provided by the Isolation Condenser System (ICS) and the Standby Liquid Control System (SLCS). The RPV has no external recirculation loops or large pipe nozzles below the top of the core region. This, together with a high capacity ADS allowed the incorporation of an ECCS driven solely by gravity, not needing any pumps. The water source needed for the ECCS function is stored in the containment upper drywell, with sufficient water to insure core coverage to 1 meter above the top of active fuel as well as flooding the lower drywell.

The PCCS heat exchangers are located above and immediately outside of containment. There is sufficient water in the external pools to remove decay heat for at least 72 hours following a postulated design basis accident, and provisions exist for external makeup beyond that, if necessary.

As a result of these, simplifications in the ESBWR safety systems, there is an increase in the calculated safety performance margin of the ESBWR over earlier BWRs. This has been confirmed by a Probabilistic Risk Assessment (PRA) for the ESBWR, which shows that the ESBWR is a calculated factor of about 5 lower than ABWR and 50 better than BWR/6 in avoiding possible core damage from degraded events.

2.11.1 Gravity Driven Core Cooling System

The GDCS injects water into the downcomer annulus region of the reactor after a LOCA and reactor vessel depressurization. It provides short-term gravity-driven water makeup from three
separate water pools located within the upper drywell at an elevation above the active core region. The system also provides long-term post-LOCA makeup from the suppression pool to meet long-term core decay heat boil-off requirements. Following any initiating event that progresses to severe accident conditions, the system floods the lower drywell region with water if the core melts through the RPV.

The GDCS is completely automatic in actuation and operation. A backup to automatic actuation is the ability to actuate by operator action.

The sequence of operations following a LOCA is as follows:

- A confirmed low RPV water level signal actuates the ADS to reduce RPV pressure.
- When a coincident high drywell pressure signal is present, ADS initiates earlier and at a higher RPV water level.
- Simultaneously, short-term and long-term system timers in the GDCS logic start, which, after time-out and satisfying permissive conditions, actuate squib valves providing an open flow path from the respective water sources (GDCS pools and suppression pool, respectively) to the vessel.
- The short-term system supplies gravity-driven flow to eight separate nozzles on the vessel with suction flow from the three separate GDCS pools. The long-term system supplies gravity-driven flow to four other nozzles with suction flow from the suppression pool through equalizing lines.

Both the short-term and long-term systems are designed to ensure that adequate reactor vessel inventory is provided assuming a LOCA in one division and failure of one squib valve to actuate in the second division.

The GDCS is composed of four divisions. A single division of the GDCS consists of three independent subsystems: a short-term cooling (injection) system, a long-term cooling (equalizing) system, and a deluge line. The short-term and long-term systems provide cooling water under force of gravity to replace RPV water inventory lost during a postulated LOCA and subsequent decay heat boil-off. The deluge line connects the GDCS pool to the lower drywell. A schematic of the GDCS is shown in Figure 14.

Each division of the GDCS injection system consists of one 200 mm pipe exiting from the GDCS pool. A 100 mm deluge line branches off and is terminated with three 50 mm squib valves and deluge line tailpipes to flood the lower drywell. The injection line continues after the deluge line connection from the upper drywell region through the drywell annulus where the line branches into two 150 mm branch lines each containing a biased-open check valve and a squib valve.

Each division of the long-term system consists of one 150 mm equalizing line with a check valve and a squib valve, routed between the suppression pool and the RPV. All piping is stainless steel and rated for reactor pressure and temperature. The RPV injection line nozzles and the equalizing line nozzles all contain integral flow limiters.

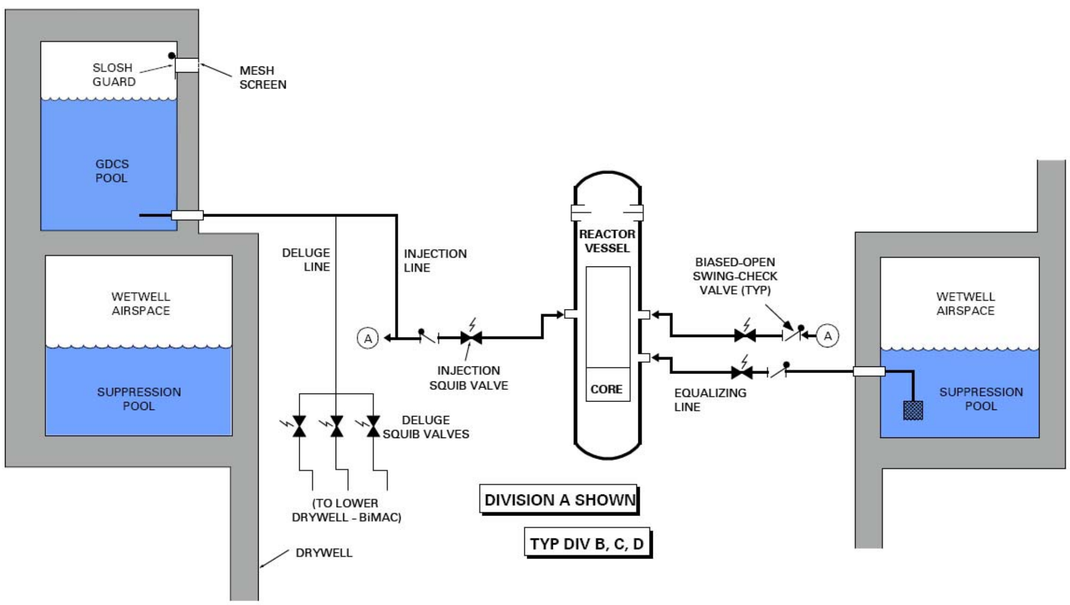


Figure 14 ESBWR Gravity Driven Cooling System

In the injection lines and the equalizing lines there exists a biased-open check valve located upstream of the squib-actuated valve. The GDCS squib valves are gas propellant type shear valves that are normally closed and which open when a pyrotechnic booster charge is ignited. During normal reactor operation, the squib valve is designed to provide zero leakage. Once the squib valve is actuated it provides a permanent open flow path to the vessel.

The check valves mitigate the consequences of spurious GDCS squib valve operation and minimize the loss of RPV inventory after the squib valves are actuated and the vessel pressure is still higher than the GDCS pool pressure plus its gravity head. Once the vessel has depressurized below GDCS pool surface pressure plus its gravity head, the differential pressure opens the check valve and allow water to begin flowing into the vessel.

The GDCS deluge lines provide a means of flooding the lower drywell region with GDCS pool water in the event of a postulated core melt sequence which causes failure of the lower vessel head and allows the molten fuel to reach the lower drywell floor. A core melt sequence would result from a common mode failure of the short-term and long term systems, which prevents them from performing their intended function. Deluge line flow is initiated by thermocouples, which sense high lower drywell region basemat temperature indicative of molten fuel on the lower drywell floor. Squib-type valves in the deluge lines are actuated upon detection of high basemat temperatures. The deluge lines do not require the actuation of squib-actuated valves on the injection lines of the GDCS piping to perform their function.

The deluge valves are opened based on very high temperatures in the lower drywell, indicative of a severe accident. Once the deluge valve is actuated it provides a permanent open flow path from the GDCS pools to the lower drywell region. Flow then drains to the lower drywell via permanently open drywell lines. This supports the BiMAC core catcher function.

The GDCS check valves remain partially open when zero differential pressure exists across the valve. This is to minimize the potential for sticking in the closed position during long periods of non-use.

Suppression pool equalization lines have an intake strainer to prevent the entry of debris material into the system that might be carried into the pool during a large break LOCA. The GDCS pool airspace opening to DW is covered by a mesh screen or equivalent to prevent debris from entering pool and potentially blocking the coolant flow through the fuel. A slosh guard is added to the opening to minimize any sloshing of GDCS pool water into the drywell following dynamic events.

The GDCS equalizing lines perform the RPV inventory control function in the long term. By closing the loop between suppression pool and RPV, inventory which is transferred to the suppression pool either by PCCS condensation shortfall, or by steam condensation in the drywell (which eventually spills back to the suppression pool) can be added back to the RPV.

GDCS pool level is the only essential system parameter that must be monitored in the main control room to verify system readiness and its proper function following initiation. Low level alarm instrumentation is included as part of GDCS.

2.11.2 Passive Containment Cooling System

The PCCS maintains the containment within its pressure limits for Design Basis Accidents (DBAs). The system is designed as a passive system with no components that must actively function, and it is also designed for conditions that equal or exceed the upper limits of containment severe accident capability. The PCCS consists of six, low-pressure, totally independent loops, each containing a steam condenser (Passive Containment Cooling Condenser), as shown in Figure 15. Each PCCS condenser loop is designed for 11 MWt capacity and is made of two identical modules. Together with the pressure suppression containment, the PCCS condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without makeup to the IC/PCC pool, and beyond 72 hours with pool makeup. The PCCS condensers are located in a large pool (IC/PCC pool) positioned above, and outside, the ESBWR containment (DW). The IC/PC Pool is vented to atmosphere.

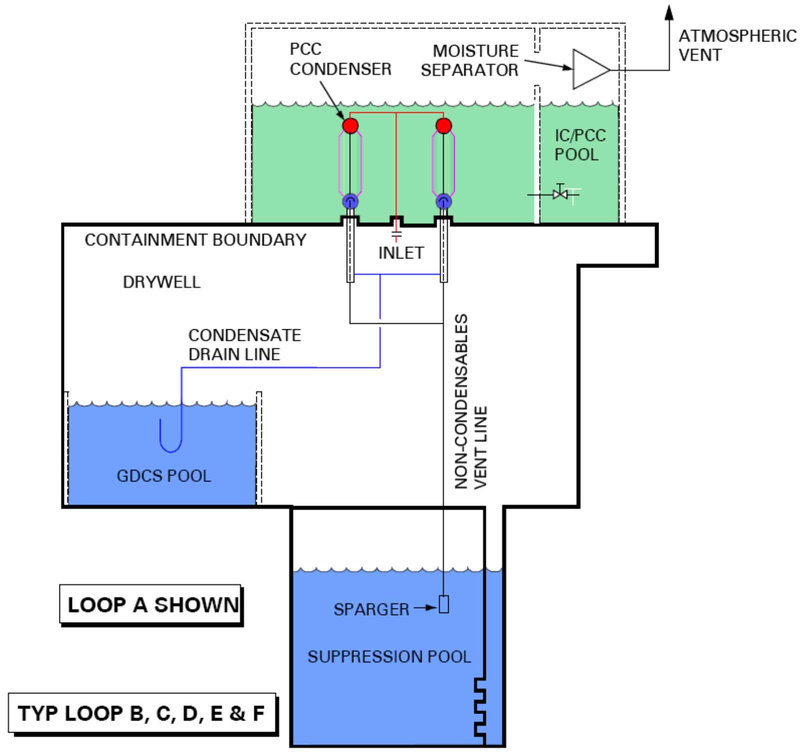


Figure 15 ESBWR Passive Containment Cooling System

Each PCCS condenser loop is configured as follows. A central steam supply pipe is provided which is open to the containment at its lower end, and it feeds two horizontal headers through two branch pipes at its upper end. Steam is condensed inside vertical tubes and the condensate is collected in two lower headers. The vent and drain lines from each lower header are routed to the DW through a single containment penetration per condenser module as shown on the diagram. The condensate drains into an annular duct around the vent pipe and then flows in a line that connects to a large common drain line, which also receives flow from the other header, ending in a GDCS pool.

The non-condensable vent line is the pathway by which drywell noncondensables are transferred to the wetwell. This ensures a low noncondensable concentration in the steam in the condenser, necessary for good heat transfer. During periods in which PCCS heat removal is less than decay heat, excess steam also flows to the suppression pool via this pathway.

The PCCS loops receive a steam-gas mixture supply directly from the DW. The PCCS loops are initially driven by the pressure between the containment DW and the suppression pool during a LOCA and then by gravity drainage of steam condensed in the tubes, so they require no sensing, control, logic or power-actuated devices to function. The PCCS loops are an extension of the safety-related containment and do not have isolation valves.

Located on the drain line and submerged in the GDCS pool, just upstream of the discharge point, is a loop seal. It prevents backflow of steam and gas mixture from the DW to the vent line, which would otherwise short circuit the flow through the PCCS condenser to the vent line. It also provides long-term operational assurance that the PCCS condenser is fed via the steam supply line.

Each PCCS condenser is located in a subcompartment of the IC/PCC pool, and all pool subcompartments communicate at their lower ends to enable full use of the collective water inventory independent of the operational status of any given IC/PCCS sub-loop. A valve is provided at the bottom of each PCC subcompartment that can be closed so the subcompartment can be emptied of water to allow PCCS condenser maintenance.

Pool water can heat up to about 101°C (214°F); steam formed, being non-radioactive and having a slight positive pressure relative to station ambient, vents from the steam space above each PCCS condenser where it is released to the atmosphere through large-diameter discharge vents. A moisture separator is installed at the entrance to the discharge vent lines to preclude excessive moisture carryover and loss of IC/PCC pool water. IC/PCC pool makeup clean water supply for replenishing level is normally provided from the Makeup Water System.

Level control is accomplished by using an air-operated valve in the makeup water supply line. The valve opening and closing is controlled by water level signal sent by a level transmitter sensing water level in the IC/PCC pool. Cooling and cleanup of IC/PCC pool water is performed by the Fuel and Auxiliary Pools Cooling System (FAPCS). The FAPCS provides safety-related dedicated makeup piping, independent of any other piping, which provides an attachment connection at grade elevation in the station yard outside the reactor building, whereby a post-LOCA water supply can be connected. There have been extensive qualification tests of the PCCS, including full-scale component tests and full height scaled integral tests.

The PCC condensers are closed-loop extensions of the containment pressure boundary. Therefore, there are no containment isolation valves and they are always in “ready standby”. The PCCS can be periodically pressure-tested as part of overall containment pressure testing.

The PCC loops can be isolated for individual pressure testing during maintenance. During refueling outages, in-service inspection (ISI) of PCC condensers can be performed, if necessary, because ultrasonic testing of tube-to-heater welds and eddy current testing of tubes can be done with PCC condensers in place. The PCC condensers are located in the IC/PCC pools.

The essential monitored parameters for the IC/PCC pools are pool water level and pool radiation.

IC/PCC pool water level monitoring is a function of the FAPCS. IC/PCC pool radiation monitoring is a function of the PRMS.

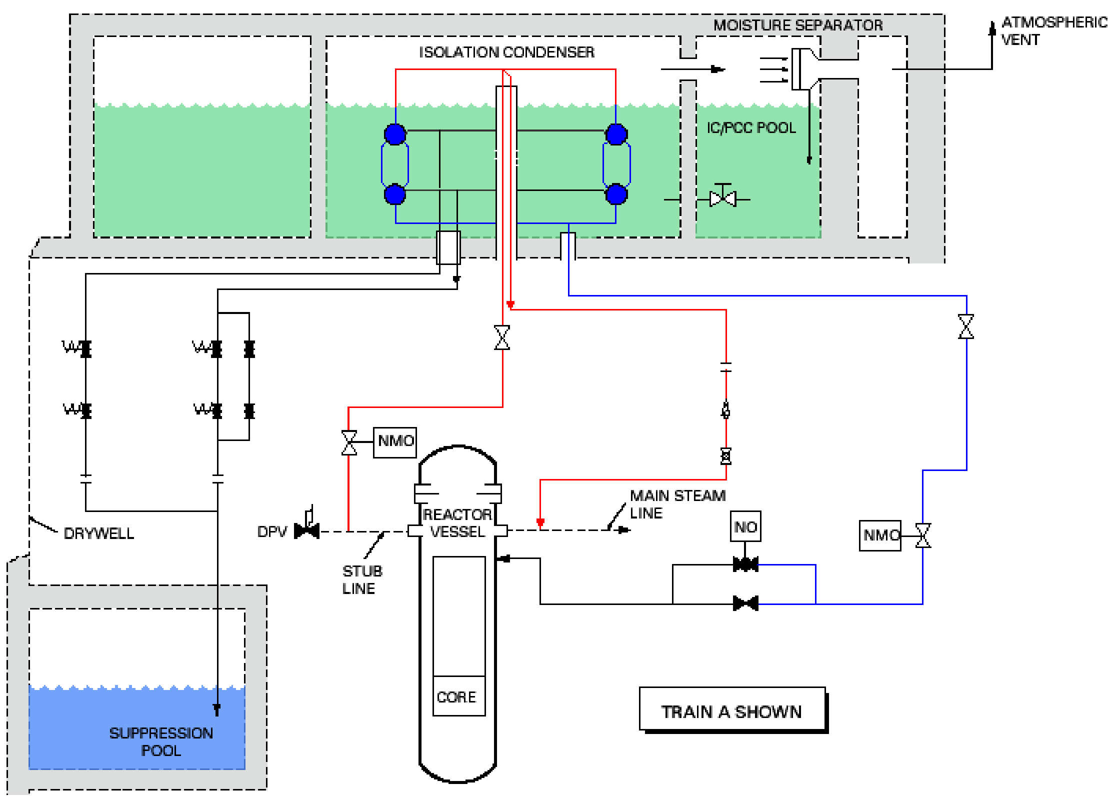
2.11.3 Isolation Condenser System

The primary function of the Isolation Condenser System (ICS) is to limit reactor pressure and prevent Safety Relief Valve (SRV) operation following an isolation of the main steam lines. The ICS, together with the water stored in the RPV, conserves sufficient reactor coolant volumes to avoid automatic depressurization caused by low reactor water level. The ICS removes excess sensible and core decay heat from the reactor, in a passive way and with minimal loss of coolant inventory from the reactor, when the normal heat removal system is unavailable. The ICS is designed as a safety-related system to remove reactor decay heat following reactor shutdown and isolation. It also prevents unnecessary reactor depressurization and operation of ECCS, which can also perform this function.

The ICS is initiated automatically on a high reactor pressure, MSIV closure or a low water Level 2 signal. To start an IC into operation, a condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The ICS can also be initiated manually from the MCR.

The ICS is isolated automatically when either a high radiation level or excess flow is detected in the steam supply line or condensate return line.

The ICS consists of four totally independent trains, each containing an isolation condenser (IC) that condenses steam on the tube side and transfers heat to a large IC/PCCS pool positioned immediately outside the containment, which is vented to the atmosphere. The IC/PCC pool is divided into subcompartments that are interconnected at their lower ends to provide full use of the water inventory for heat removal by any IC.



Isolation Condenser System (Standby Mode)

The IC, connected by piping to the reactor pressure vessel, is placed at an elevation above the source of steam (vessel) and, when the steam is condensed, the condensate is returned to the vessel via a condensate return pipe. The steam side connection between the vessel and the IC is normally open and the condensate line is normally closed. This allows the IC and drain piping to fill with condensate, which is maintained at a subcooled temperature by the pool water during normal reactor operation. The IC is started into operation by opening condensate return valves and draining the condensate to the reactor, thus causing steam from the reactor to fill the tubes which transfer heat to the cooler pool water. Each IC is made of two identical modules.

A vent line is provided for both upper and lower headers to remove the noncondensable gases away from the unit, during IC operation. The vent lines are routed to the containment through a single penetration.

A purge line is provided to assure that, during normal plant operation (IC system standby conditions), the excess of hydrogen (from hydrogen water chemistry control additions) or air from the feedwater does not accumulate in the IC steam supply line, thus assuring that the IC tubes are not be blanketed with noncondensables when the system is first started. The purge line penetrates the containment roof slab.

Containment isolation valves are provided on the steam supply piping and the condensate return piping.

The Fuel and Auxiliary Pools Cooling System (FAPCS) performs cooling and cleanup of IC/PCC pool water. During IC operation, IC/PCC pool water can boil, and the steam produced is vented to the atmosphere.

This boil-off action of non-radioactive water is a safe means for removing and rejecting all reactor decay heat. The IC/PCC pool has an installed capacity that provides at least 72 hours of reactor decay heat removal capability. The heat rejection process can be continued indefinitely by replenishing the IC/PCC pool inventory. A safety-related FAPCS makeup line is provided to convey emergency makeup water into the IC/PCC pool from a water supply outside of the reactor building. The flow path for this makeup can be established independent of FAPCS operation, simply by manually opening the isolation valve on the FAPCS makeup line located at grade level in the yard area external to the reactor building.

The ICS passively removes sensible and core decay heat from the reactor (i.e., heat transfer from the IC tubes to the surrounding IC/PCC pool water is accomplished by natural convection, and no forced circulation equipment is required) when the normal heat removal system is unavailable following any of the following events:

- Sudden reactor isolation at power operating conditions;

- During station blackout (i.e., unavailability of all AC power), and;

- Anticipated Transient Without Scram (ATWS).

The ICs are sized to remove post-reactor isolation decay heat with 3 of 4 ICs operating and to reduce reactor pressure and temperature to safe shutdown conditions, with occasional venting noncondensible gases to the suppression pool. The heat exchangers (ICs) are independent of station AC power and function whenever normal heat removal systems are unavailable to maintain reactor pressure and temperature below limits.

The control room operators can perform periodic surveillance testing of the ICS valves via remote manual switches that actuate the isolation valves and the condensate return valves. Status lights on the valves verify the opening and closure of the valves.

The essential monitored parameters for the IC/PCC pools are pool water level and pool radiation. IC/PCC pool water level monitoring is a function of the FAPCS. IC/PCC pool radiation monitoring is a function of the PRMS.

2.12 STANDBY LIQUID CONTROL SYSTEM

The Standby Liquid Control (SLC) system provides an alternate method of reactor shutdown (i.e. without control rods) from full power to cold subcritical by the injection of a neutron absorbing solution into the RPV. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The SLCS is sized to counteract the positive reactivity effect of shutting down from rated power to cold shutdown condition. It also adds additional inventory to the RPV after confirmation of a LOCA.

The SLCS is automatically initiated in case of signals indicative of LOCA or ATWS. It can also be manually initiated from the main control room to inject the neutron absorbing solution into the reactor.

The SLCS is a two-division passive system using pressurized accumulators to inject borated water rapidly and directly into the bypass area of the core. Each division is 50% capacity. Injection will take place after either of two squib valves in each division fires upon actuation signal from the SSLC. Figure 16 illustrates the SLCS configuration.

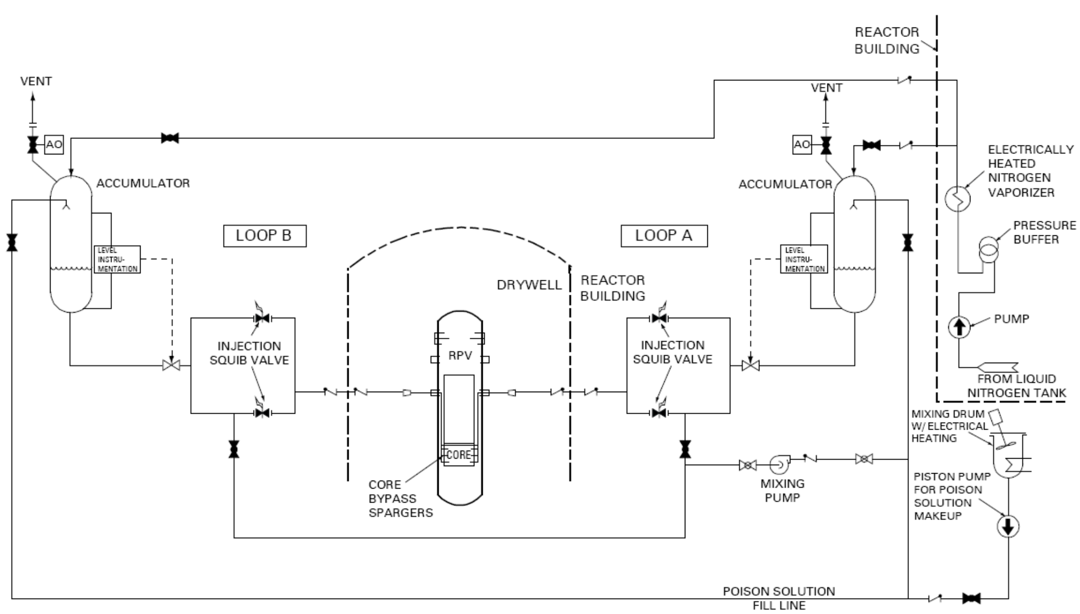


Figure 16 ESBWR Standby Liquid Control System

In addition to the accumulators and injection valves, supporting non-safety grade equipment includes a high pressure nitrogen charging system for pressurization and to make up for losses, and a mixing and boron solution makeup system. The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the fuel. The specified neutron absorber solution is sodium pentaborate using 94% of the isotope B10 at a concentration of 12.5%. This combination not only minimizes the quantity of liquid to be injected, but also assures no auxiliary heating is needed to prevent precipitation of the sodium pentaborate out of solution in the accumulator and piping. At all times, when it is possible to make the reactor critical, the SLCS will be able to deliver enough sodium pentaborate solution into the reactor to assure reactor shutdown.

Upon completion of injecting the boron solution, redundant accumulator level measurement instrumentation using 2 out of 4 logic closes the injection line shut-off valve in each SLCS division. Closure of these valves prevents injection of nitrogen from the accumulator into the reactor vessel that could interfere with Isolation Condenser System operation, or cause additional containment pressurization. As a backup, the accumulator vent valves are also opened at the same time.

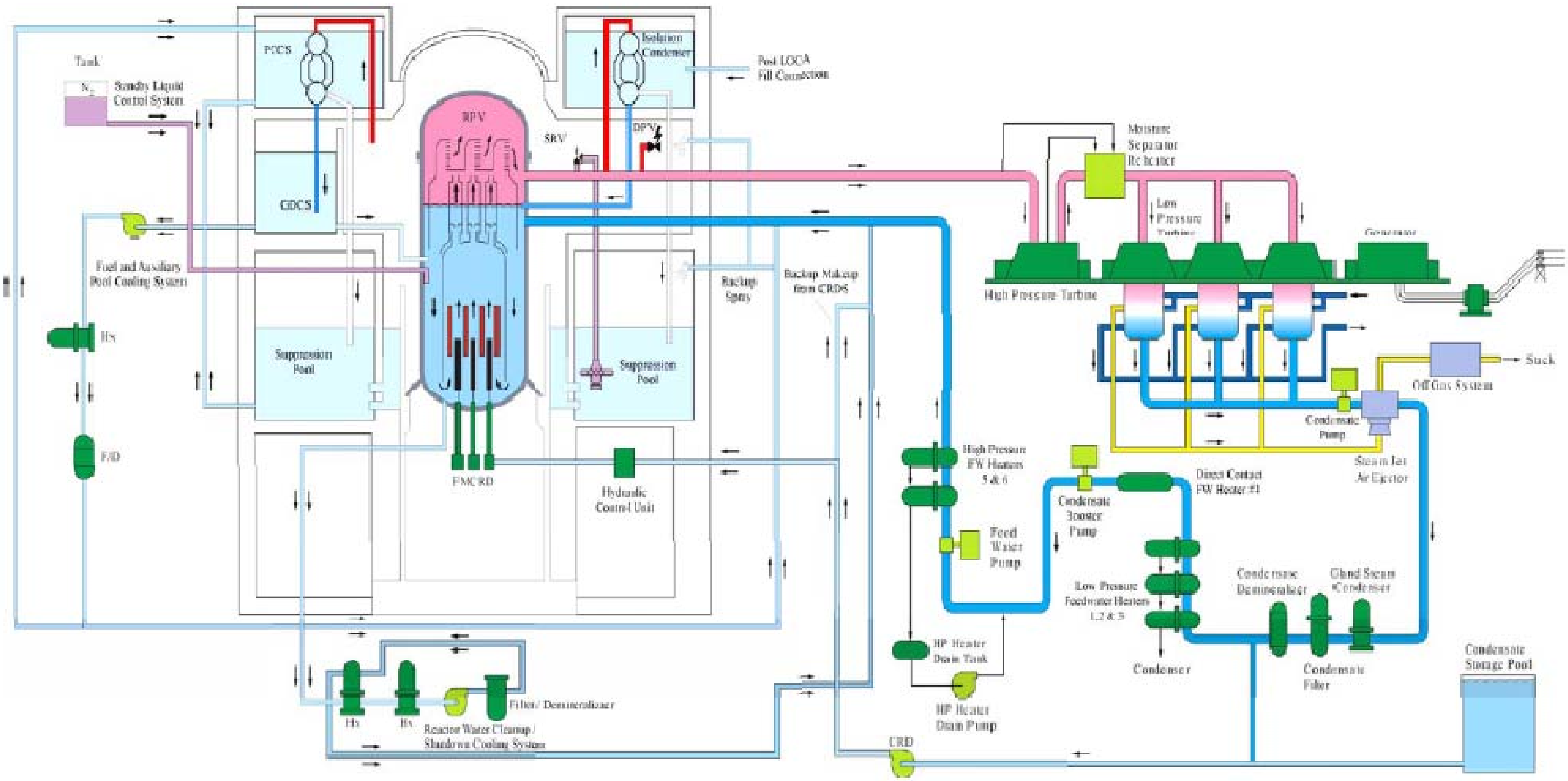


Fig. 17 ESBWR System Overview

2.13 REACTOR WATER CLEANUP/SHUTDOWN COOLING SYSTEM

The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system is one of the Reactor Auxiliary Systems. RWCU/SDC system has the following primary functions:

- Purifies reactor coolant during normal operation and shutdown;
- Transfers sensible and core decay heat produced when the reactor is being shutdown or is in the shutdown condition;
- Provides decay heat removal and high pressure cooling of the primary coolant during periods of reactor isolation (hot standby);
- Implements the overboarding of excess reactor coolant during startup and hot standby;
- Maintains coolant flow from the reactor vessel bottom head to reduce thermal stratification;
- Warms the reactor coolant prior to startup and vessel hydro testing;

The system consists of two independent trains. Each train includes:

- One non-regenerative heat exchanger (NRHX);
- One regenerative heat exchanger (RHX);
- One low capacity cleanup (function) pump;
- One high capacity SDC pump;
- One demineralizer, and;
- Associated valves and pipes.

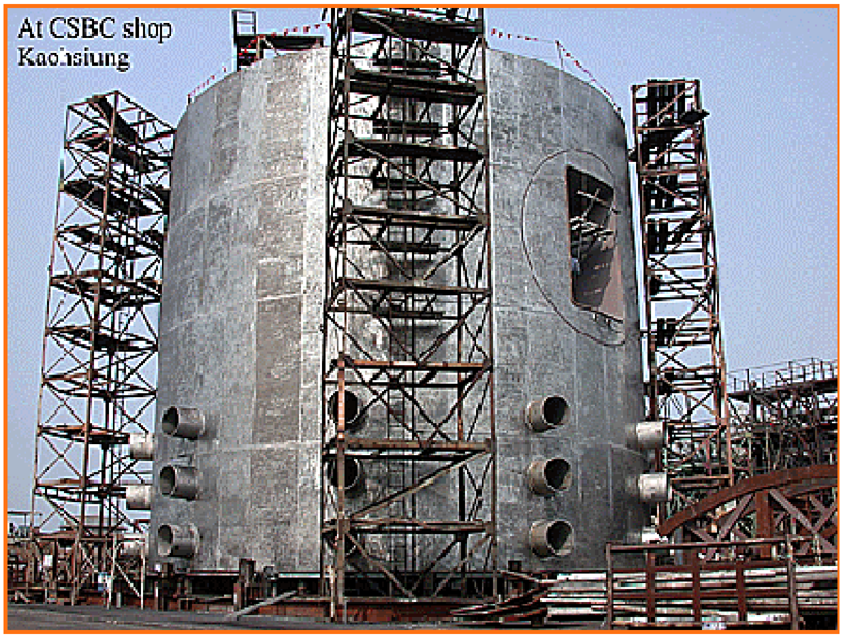
The RWCU/SDC system maintains the temperature difference between the reactor dome and the bottom head drain to preclude excessive thermal stratification.

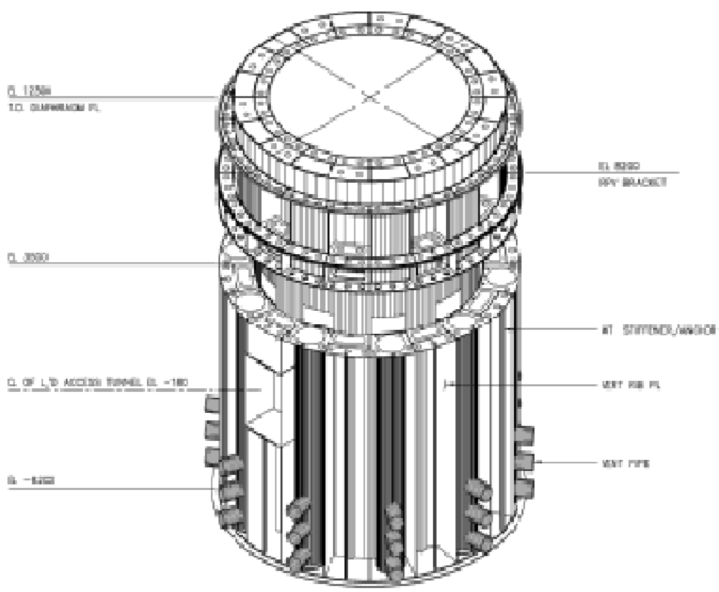
Flow rate, pressure, temperature and conductivity are measured, recorded or indicated, and alarmed if appropriate, in the MCR. Pumps are provided with interlocks for the automatic operation and with switch and status indication for manual operation from the MCR. Motor-operated isolation valves are automatically and manually actuated.

2.14 CONTAINMENT SYSTEM

The ESBWR containment, centrally located in the Reactor Building, features the same basic pressure suppression design concept previously applied in over three decades of BWR power generating reactor plants. The containment structure is a steel-lined, reinforced right circular cylindrical concrete vessel (RCCV). The RCCV supports the upper pools whose walls are integrated into the top slab of the containment to provide structural capability for LOCA and testing pressures. It fulfils its design basis as a fission product barrier at the pressure conditions associated with a postulated pipe rupture.

Main features include the upper and lower drywell surrounding the RPV and a wetwell containing the suppression pool that serves as a heat sink during abnormal operations and accidents.





The containment is constructed as a right circular cylinder set on the reinforced concrete basemat of the reactor building. The drywell comprises two volumes: an upper drywell volume surrounding the upper portion of the RPV and housing the steam and feedwater piping, the SRVs, GDCS pools, main steam drain piping and upper drywell coolers; and a lower drywell volume surrounding the lower portion of the RPV, housing the FMCRDs, neutron monitoring system, equipment platform, lower drywell coolers and two drywell sumps. The drywell top opening is enclosed with a steel head removable for refueling operations.

The gas space above the suppression pool serves as the LOCA blowdown reservoir for the upper and lower drywell nitrogen and non-condensable gases that pass through the twelve drywell-to-wetwell vertical vents, each with three horizontal vents located below the suppression pool surface (see above). The suppression pool water serves as the heat sink to condense steam released into the drywell during a LOCA or steam from SRV actuations. Access into the upper and lower drywells is provided through a double sealed personnel lock and an equipment hatch. The equipment hatch is removable only during refueling or maintenance outages. A hatch located in the Reactor Building provides access into the wetwell.

During plant startup, the Containment Inerting System, in conjunction with the containment purge system and the drywell cooling fans, is utilized to establish an inert gas environment in the containment with nitrogen to limit the oxygen concentration. This precludes combustion of any hydrogen that might be released subsequent to a LOCA. After the containment is inerted and sealed for plant power operation, small flows of nitrogen gas are added to the drywell and the wetwell as necessary to keep oxygen concentrations below 4% and to maintain a positive pressure for preventing air in-leakage. High-pressure nitrogen is also used for pneumatic controls inside the containment to preclude adding air to the inert atmosphere.

The containment structure has the capability to maintain its functional integrity at the pressures and temperatures that could follow a LOCA pipe break postulated to occur simultaneously with loss of off-site power. The containment structure is designed to accommodate the full range of loading conditions associated with normal and abnormal operations including LOCA-related design loads in and above the suppression pool (including negative differential pressure between the drywell, wetwell and the remainder of the Reactor Building), and safe shutdown earthquake (SSE) loads.

The containment structure is protected from, or designed to withstand, fluid jet forces associated with outflow from the postulated rupture of any pipe within the containment. The containment design considers and utilizes leak-before-break (LBB) applicability only in regard to protection against dynamic effects associated with a postulation of rupture in high energy piping.

The containment structure has design features to accommodate flooding to sufficient depth above the top of active fuel to permit safe removal of fuel assemblies from the reactor core after a postulated design basis accident (DBA).

The containment structure is configured to channel flow from postulated pipe ruptures in the drywell to the suppression pool through vents submerged in the suppression pool, which are designed to accommodate the energy of the blowdown fluid.

The containment structure and penetration isolation system, with concurrent operation of other
accident mitigation systems, are designed to limit fission product leakage during and following a postulated DBA to values well below leakage calculated for allowable off-site doses.

The design value for a maximum steam bypass leakage between the drywell and the wetwell through the diaphragm floor including any leakage through the wetwell-to-drywell vacuum breakers is limited. Satisfying this limit is confirmed by initial preoperational tests as well as by periodic tests conducted during refueling outages. These tests are conducted at differential pressure conditions between the drywell and wetwell that do not clear the drywell-to-wetwell horizontal vents.

A watertight barrier is provided between the open reactor and the drywell during refueling. This enables the reactor well to be flooded prior to removal of the reactor steam separator, dryer assembly and to facilitate underwater fuel handling operations. Piping, cooling air ducts and return air vent openings in the reactor well platform must be removed, vents closed and sealed watertight before filling the reactor well with water. The refueling bellows assembly is provided to accommodate the movement of the vessel caused by operating temperature variations and seismic activity.

Containment isolation is accomplished with inboard and outboard isolation valves on each piping penetration that are signaled to close on predefined plant parameters. Systems performing a post-LOCA function are capable of having their isolation valves reopened as needed. Drywell coolers are provided to remove heat released into the drywell atmosphere during normal reactor operations.

2.14.1 Containment Inerting System

The Containment Inerting System is designed to establish and maintain an inert atmosphere within the containment during all plant operating modes, except during plant shutdown for refueling or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power. The objective of the system is to establish conditions that help preclude combustion of hydrogen and thereby prevent damage to safety-related equipment and structures.

The Containment Inerting System does not perform any safety-related function except for its containment isolation function. Failure of the Containment Inerting System does not compromise any safety-related system or component nor does it prevent a safe shutdown of the plant. The containment inerting process is a nonsafety-related readiness function, which is not used after the initiation of an accident, and thus, the Containment Inerting System is not a safety related system.

The Containment Inerting System establishes an inert atmosphere (i.e., a very low oxygen concentration by volume) throughout the containment following an outage (or other occasions when the containment has become filled with air) and maintains it inert during normal conditions. The system maintains a slight positive pressure in the containment to prevent air (oxygen) in-leakage.

The Containment Inerting System is comprised of a pressurized liquid nitrogen storage tank, a steam-heated main vaporizer for large nitrogen flow, electric heater for vaporizing makeup flow, injection and exhaust lines, a bleed line, associated valves, controls, and instrumentation. All Containment Inerting System components are located inside the reactor building except the liquid nitrogen storage tank and the steam-heated main vaporizer, which are located in the yard.

2.14.2 Containment Monitoring System

The Containment Monitoring System (CMS) shall provide the following functions:

- Drywell and Wetwell – Hydrogen, Oxygen concentrations and Gamma radiation levels Monitoring;
- Drywell and Wetwell Pressure Monitoring;
- Drywell/Wetwell Differential Pressure Monitoring;
- Upper Drywell Level Monitoring;
- Suppression Pool Water Level Monitoring;
- Suppression Pool Temperature Monitoring;
- Transmission of signals from dewpoint sensors that are used in Integrated Leak Rate Tests (ILRT);
- Post-Accident Sampling
- Lower Drywell (Post-LOCA) Pool Level Monitoring.

3. PASSIVE BWR NPP SIMULATOR – GENERAL

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 a passive BWR plant responses during normal operations and accident situations. It also has a user-machine interface that mimics the control panel instrumentation, including the plant display system. More importantly, it allows user interaction with the simulator during the operation of the simulated passive BWR plant.

The minimum hardware configuration for the simulator consists of a Pentium PC or equivalent (minimum 1.7GHz CPU speed), minimum of 512Mbytes RAM , at least 30Gbytes 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 passive BWR.

TABLE I. SUMMARY OF PASSIVE 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 • 3-D spatial reactor neutronic model with axial and radial flux mapping. • 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; Reactor Power Regulation; Reactor Water Level Control; Turbine Load/Frequency Control	• Passive Plant Overview • Passive BWR Reactivity & Setpoints • Passive 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 adjustment of reactor water control level setpoint	• Inadvertent withdrawal of one bank of control rods • Inadvertent insertion of one bank of control rods • Inadvertent reactor isolation
STEAM & FEEDWATER	• 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	• Passive BWR Feedwater and Extraction Steam	• Reactor water level setpoint changes: computer or manual • 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	• Passive 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 • Manual Open/Close of MSIV	• Turbine throttle pressure transmitter fails low • Turbine trip with Bypass Valve failed closed • Inadvertent Opening of Bypass Valve • Increasing and decreasing steam flow due to Pressure Control System failures • Loss of Condenser Vacuum • Load Rejection
OVERALL UNIT	• Fully dynamic interaction between all simulated systems • Turbine-Following-Reactor load maneuvering • Unit annunciation • Major control loops	• Passive BWR Plant Overview • Passive BWR Reactivity & Setpoints		

SYSTEM	SIMULATION SCOPE	DISPLAY PAGES	OPERATOR CONTROLS	MALFUNCTIONS
REACTOR CLEANUP /SHUTDOWN COOLING	The system consists of two independent trains. Each train being modeled includes: • One non-regenerative heat exchanger (NRHX); • One regenerative heat exchanger (RHX); • One low capacity cleanup (function) pump; • One high capacity SDC pump; • One demineralizer, and; • Associated valves and pipes. It is simple model for RWCU /SDC system to maintain the temperature difference between the reactor dome and the bottom head drain for the various plant operations: Cleanup; Cooldown; Warmup.	• Passive BWR RWCU/SDC	• Manual controls for : CLEANUP, COOLDOWN , WARMUP	• None
CONTAINMENT & PASSIVE COOLING SYSTEMS	• Passive Components outside containment: IC Pool and Condenser; PCC Pool and Condenser. • Inside containment: GDCS Pool; GDCS Injection Valve; Deluge Valve; Equalizing Valve. • Pressure and temperature responses of the drywell due to the LOCA 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. Increased vapor pressure corresponding to the suppression pool surface temperature. • Note: wetwell pressurization should be largely due to	• Passive BWR Containment	• Manual Open/Close Isolation Condenser Discharge Valve	• Inadvertent Isolation Condenser Initiation • LOCA Break flow ~ 1660 kg/s

	transfer of noncondensible (air or nitrogen) from the drywell. But due to the current assumptions made in model development, air or nitrogen was not modeled in the current containment model.			

3.1 SIMULATOR STARTUP

- Select program ‘PBWR’ for execution - the executable file is PBWR.exe
- Click anywhere on ‘Passive 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 Passive 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.

3.2 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:\PBWR_Simulator\FP_100.IC’
- Click ‘YES’ to 'Load C:\PBWR_Simulator\FP_100.IC’
- Click ‘Return’
- Start the simulator operating by selecting ‘Run’.

3.3 LIST OF BWR SIMULATOR DISPLAY SCREENS

(1) Passive BWR Plant Overview
(2) Passive BWR Control Loops

(3) Passive BWR Power/Flow Map & Controls
(4) Passive BWR Reactivity & Setpoints
(5) Passive BWR Scram Parameters
(6) Passive BWR Turbine Generator
(7) Passive BWR Feedwater & Extraction Steam
(8) Passive BWR Containment
(9) Passive BWR Cleanup/ Shutdown Cooling System
(10) Passive BWR Trends

3.4 PASSIVE BWR SIMULATOR DISPLAY COMMON FEATURES

The Passive BWR simulator has 9 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 (Deg. 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:

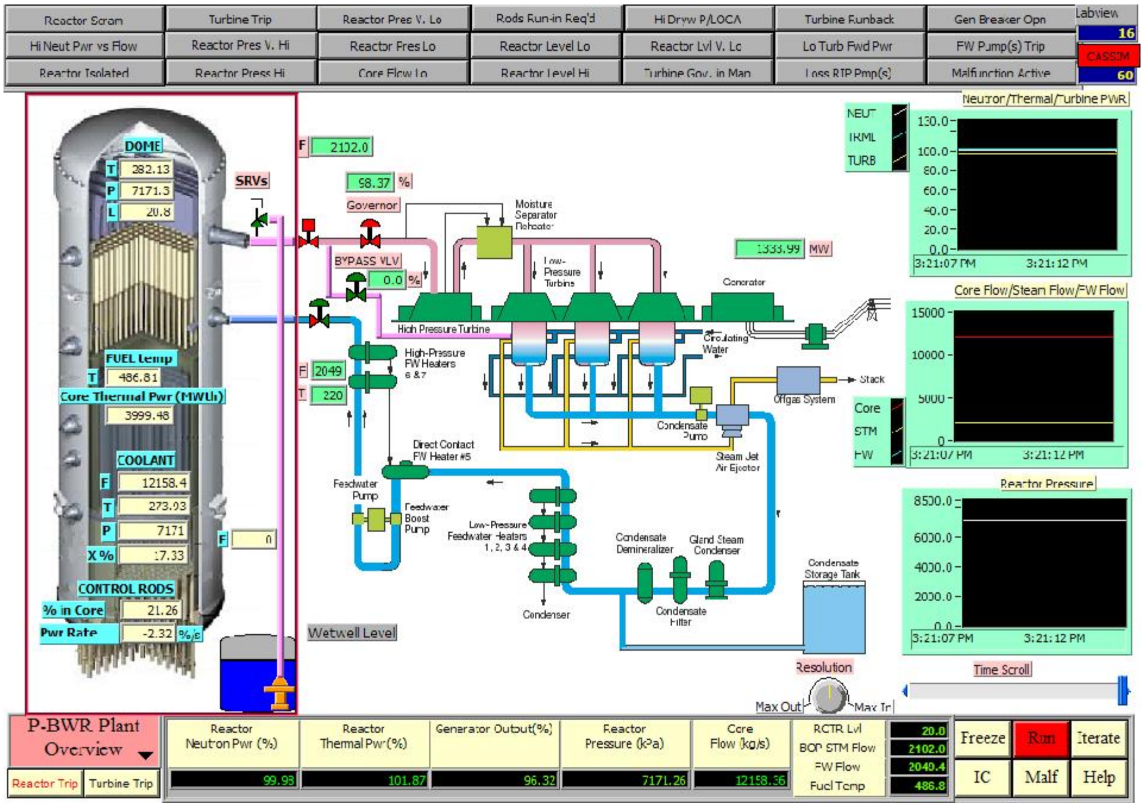
- All pressure values are designated as "P" next to the display box, and have units of KPa;
- All temperature values are designated as "T" next to the display box and have values of deg. C;
- All flow values are designated as "F" next to the display box and have values of Kg/s;
- 2 phase qualities are 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.

4. PASSIVE BWR SIMULATOR DISPLAY SCREENS

4.1 PASSIVE 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 3-D spatial 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 reactivity control rods — FMCRD, fine motion control rods, and reactivity feedback effects due to Xenon, 2-phase voiding in coolant channels; fuel temperature (Doppler) and moderator (light water) temperature.

- The reactor parameters displayed are:

  Reactor dome section

  ⇒ Dome steam temperature (ºC)

  ⇒ Dome pressure (p)

  ⇒ Steam flow from core (Kg/s)

  ⇒ Reactor water level (m)

  Reactor core section

  ⇒ Neutron power rate (%/ second)

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

  ⇒ Average fuel temperature (ºC)

  ⇒ Coolant flow rate in core (Kg/s)

- ⇒ Coolant pressure at core exit (p)
- ⇒ Coolant temperature at core exit (ºC)
- ⇒ Coolant quality at core exit (X%)
- ⇒ Control rods position in core (% of total length in core). Note control rods reactivity worth is as follows: 100% in core - negative 170 milli-K; 100% out-of-core - positive 120 milli-K.

- 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 containment. Here all the SRVs are shown in "one equivalent valve" symbol; in fact there are a series of SRVs associated with each main steam line; and there are four separate main steam lines. (See Figure 12 on P. 44). 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 various injection sources), in case of loss of coolant accident.

  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 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 Passive 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/sec)

  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/sec)
  - ⇒ 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:

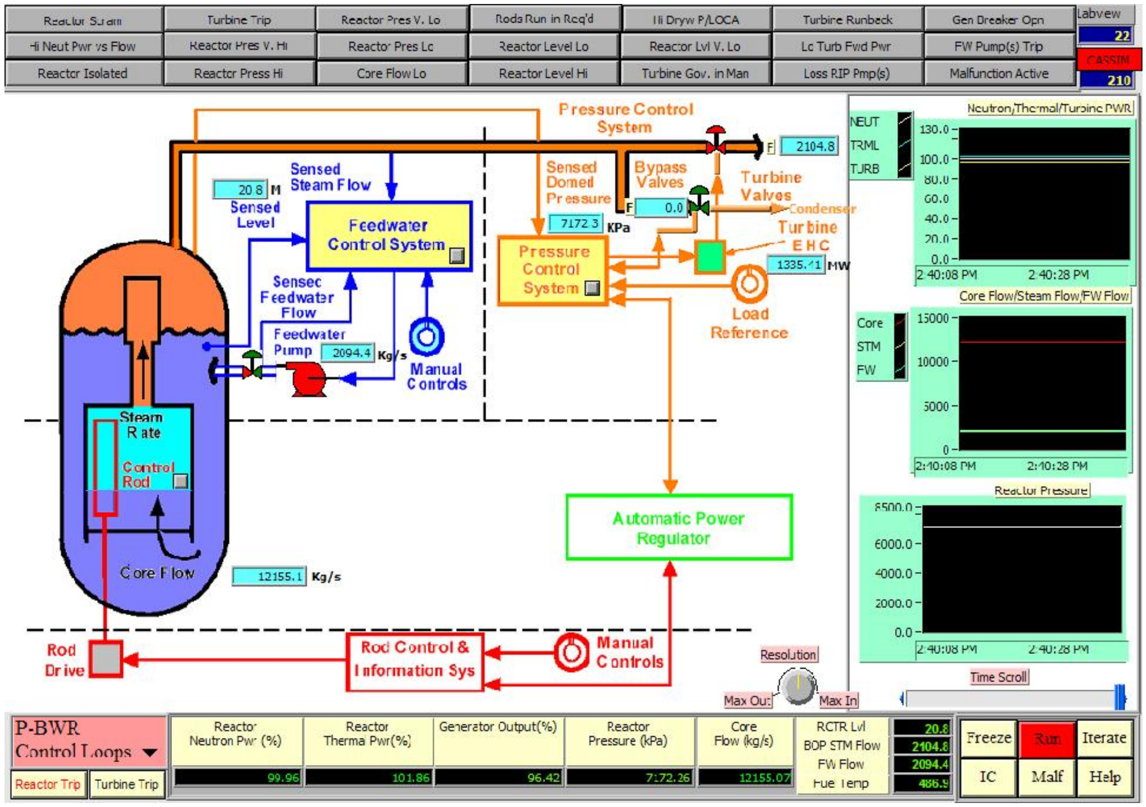
- Reactor neutron power, reactor thermal power and turbine power (0-100%)

- 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.

4.2 PASSIVE BWR CONTROL LOOPS SCREEN



This screen shows all the essential control loops for the passive 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
- Main steam flow
- Bypass steam 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, go to Screen Passive BWR Power/Flow Map & Controls".

- **Reactor power control** — The reactor power output control system consists of control rods, rod drive system and the reactor power regulation 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 core flow depends on the driving head and the loop losses. The difference in density between the single phase fluid in the downcomer region and the two-phase mixture in the core provides the driving head. The loop losses include the sum of ΔP losses in downcomer, core, chimney and separator respectively. See details on natural circulation in Section 2.1.

- **Reactor pressure control** — press the button to display a pop-up window, which describe 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 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 110% 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, and 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.

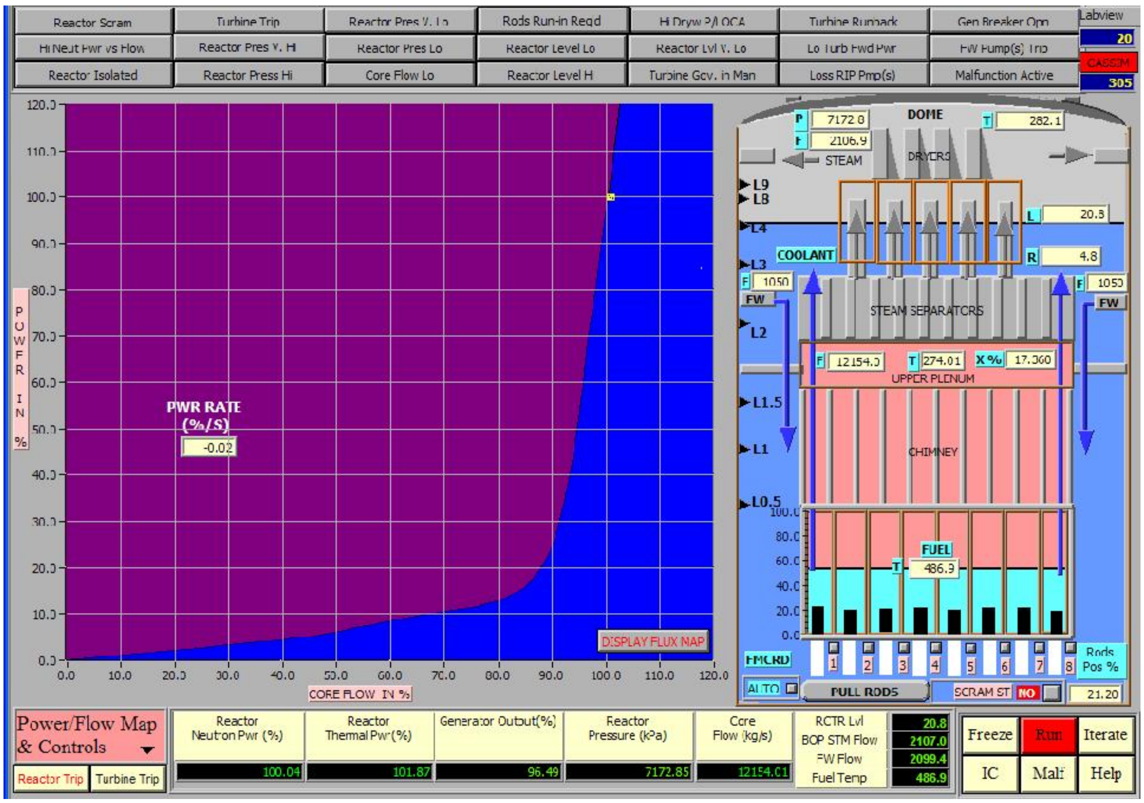
4.3 PASSIVE BWR POWER/FLOW MAP & CONTROLS

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)



POWER/FLOW MAP

- The power flow map is a representation of reactor power vs. core 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 can be traced on this map.
- The core power-flow map is only a single line and there is <u>no active control of the core flow at a given power level</u>.
- To provide additional operational flexibility, a core power – feedwater temperature operating map is being envisioned by the reactor control designer. The system hardware required to develop and implement such an operating map is being developed by the reactor vendor. Note that the feedwater operating map is <u>not included</u> in the simulation

model scope. However, the maximum feedwater temperature is limited to less than or equal to 215.6°C (420°F) at all power levels by use of administrative controls – e.g. rods run-in, equipment design, or a combination of both

- 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:

  (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 Passive 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.

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 conditions “low flow - high power” region. Research has shown that such oscillations are characterized by “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 most limiting stability condition in the passive BWR normal operating region is at the rated power/flow condition. Therefore, the passive BWR is designed so that the core remains stable throughout the whole operating region, including plant startup. In order to establish a high degree of confidence that oscillations will not occur, conservative design criteria were imposed on the channel, core wide and regional decay ratios under all conditions of normal operation and anticipated transients. The passive BWR licensing basis for stability is satisfied by determining a stability criteria map of core decay ratio vs. channel decay ratio to establish margins to stability.

Because oscillations in power and flow are precluded by design, the requirements of US NRC GDC 10 are met through the analysis for AOOs, and are automatically satisfied with respect to stability. In addition, the passive BWR will implement a Detect and

Suppress solution as a defense-in-depth system. As reactor stability evaluation is an advanced topic, for further technical details, refer to the thermal hydraulic stability discussed in detail in Appendix 4D of Reference #10.

REACTOR CORE GRAPHICS

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

- **FMCRD auto/manual button** — this button when pressed will allow the user to switch the control rods to be 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 rods 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 control rods move together as controlled by the reactor power regulating system.

  Note:

  - There are approximately 269 FMCRDs in total, they are positioned and calibrated with reactivity worth of -170 mk when all of them are 100% in core, and +119 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 + 14.875 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 sec.
  - 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 sec. for 100% insertion.
  - The average rods position in core is shown on the right hand bottom corner.

- **SCRAM status indication, manual scram/reset button, PULL RODS button —** when the reactor is scrammed, and if scram conditions still exist, there will be a "YES" sign next to the "SCRAM ST" indicator. As well, the alarm "Reactor Scram" is in "Red" color. Assuming the scram conditions have already disappeared, and 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 "PULL RODS" 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.

- **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 lower plenum of the core. The coolant enters the core lower plenum generally with subcooled temperature. As the coolant enters the core fuel channel, the subcooled water receives heat from the fuel bundles, and becomes two-phase fluid that exits the core with quality. The steam vapor from the coolant escapes the fluid and becomes saturated steam; the remaining water content of the fluid is recirculated back to the downcomer after mixing with incoming feedwater.
- The boiling height – 2-phase boiling region of the core is shown in a “pink” color. It is animated, so as the boiling height changes as the core conditions change, the “pink” section boundary 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 as a” 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, where the flow, pressure and temperature are shown.

FLUX MAPPING

The 3-D reactor core flux axial and radial maps can be displayed when pressing the button: "DISPLAY FLUX MAP" on the bottom right of the Power/Flow Map.

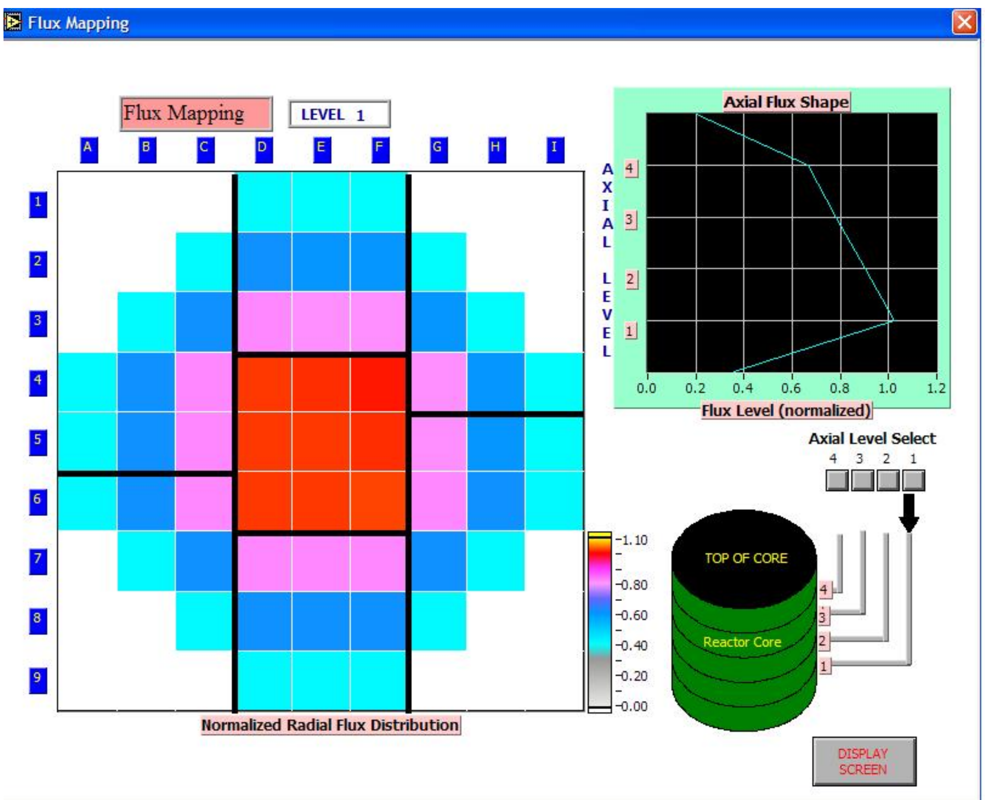
Note: in order to see the Flux Map Screen from the Power/Flow Map Screen, the user has to cancel all existing Window programs running in the background (e.g. Window Explorer, etc.).

The flux mapping display shows :

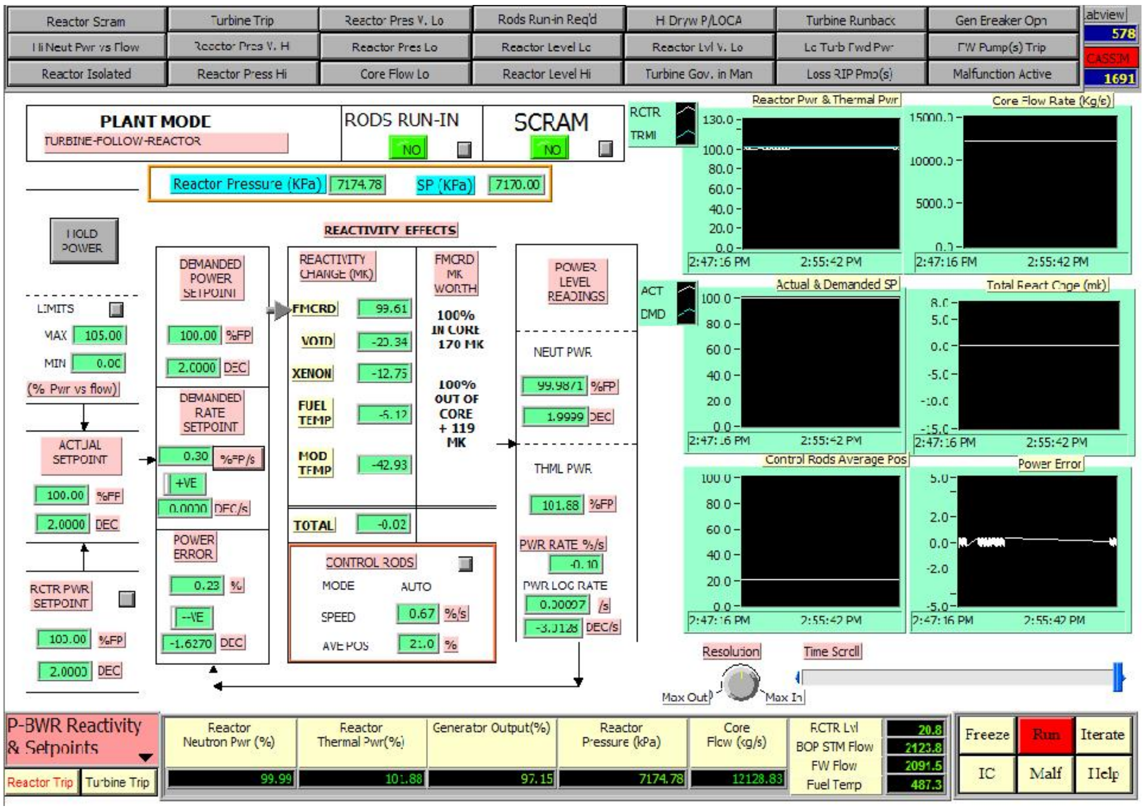
(a) The axial flux shape for four evenly distributed axial levels 1, 2, 3, 4. Level 1 is near the bottom of core, and level 4 is near the top of fuel.

(b) For each axial level, the normalized radial flux distribution is shown for 81 cells. The color shown for each cell corresponds to the flux intensity normalized level, as represented by the color chart provided. Press the respective "Axial Level Select" button to show the corresponding radial flux map for that axial level.

To go back to the Power/Flow Screen, press the "DISPLAY SCREEN" button.



4.4 PASSIVE 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 power 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 sec. (if current power is > 20% FP), or % present power per sec. (if the current power is < 20% FP). The purpose is to allow higher power rate change only at higher power.

  The suggested nominal rate is:

  Below 20 % FP, use 100% present power per sec.

  Above 20 % FP, use 0.5 % per sec. Lower rate should be used, particularly if one observes fluctuations in level, and power during load changes.

  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.

In addition, 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

- 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.

  Note: in the real plant, the eight banks of rods will move in accordance with the Rod Sequence Control System (RSCS) as described in previous Section 2.3.3. However, for this educational simulator, simplification has been taken such that all the 8 banks will move together for insertion or withdrawal.

- The display information on 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.

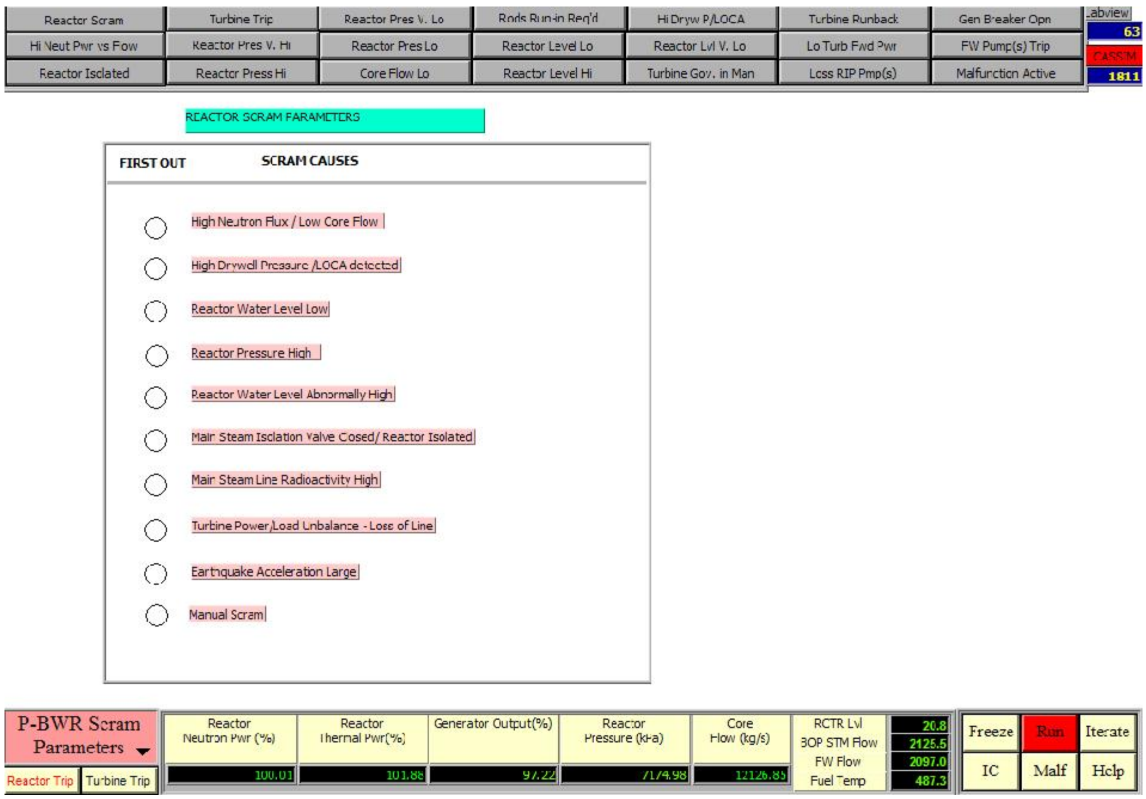
- Note that the passive BWR plant is always operating with turbine-following-reactor mode.

- The buttons at the top of the screen allow the user to perform a "manual" "rods run-in", as well as "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 control rods "AUTO/MANUAL".

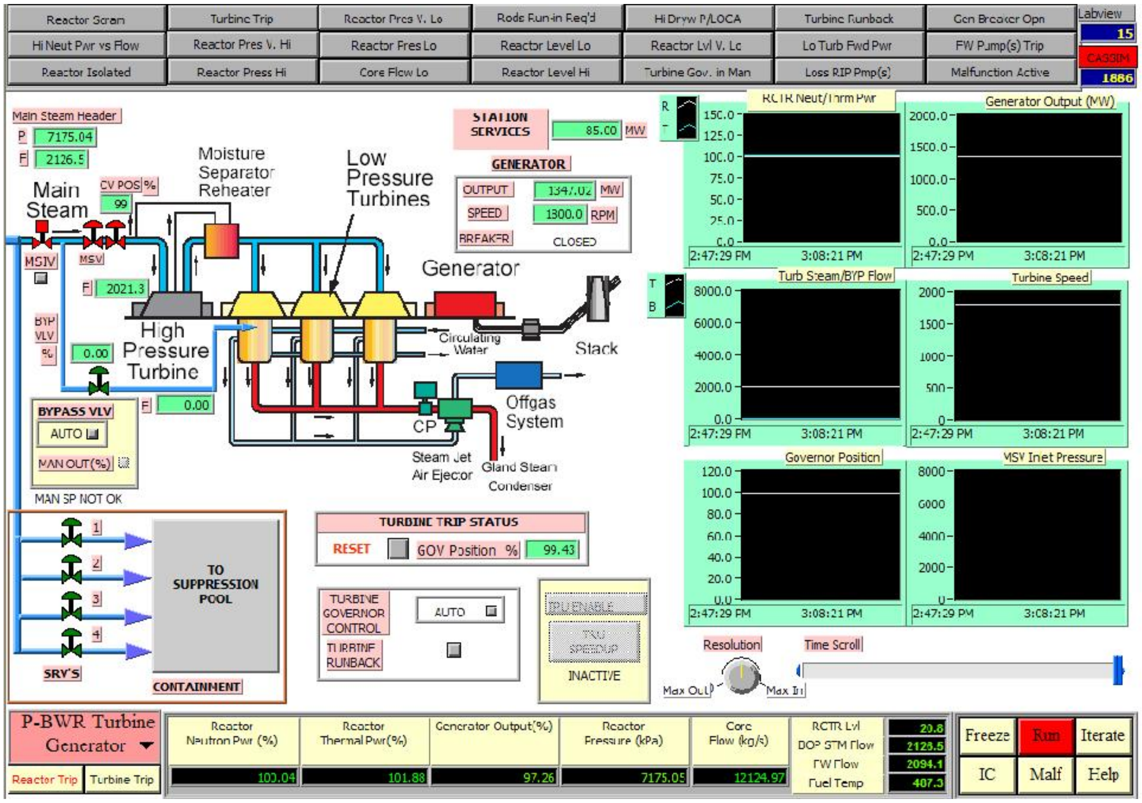
4.5 PASSIVE BWR SCRAM PARAMETERS SCREEN



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 level low — L3 the scram setpoint is 19 meters above reactor bottom. Normal level is 20.6 meters above reactor bottom.
- Reactor pressure high — the scram setpoint is 7870 KPa. Normal reactor pressure is 7170 KPa.
- Reactor level very high —L9 the scram setpoint is 22.39 meters above reactor bottom.
- Main steam isolation valve closed/reactor isolated.
- Turbine power/load unbalance or loss of line (load rejection).
- Manual scram.

4.6 PASSIVE BWR TURBINE GENERATOR SCREEN



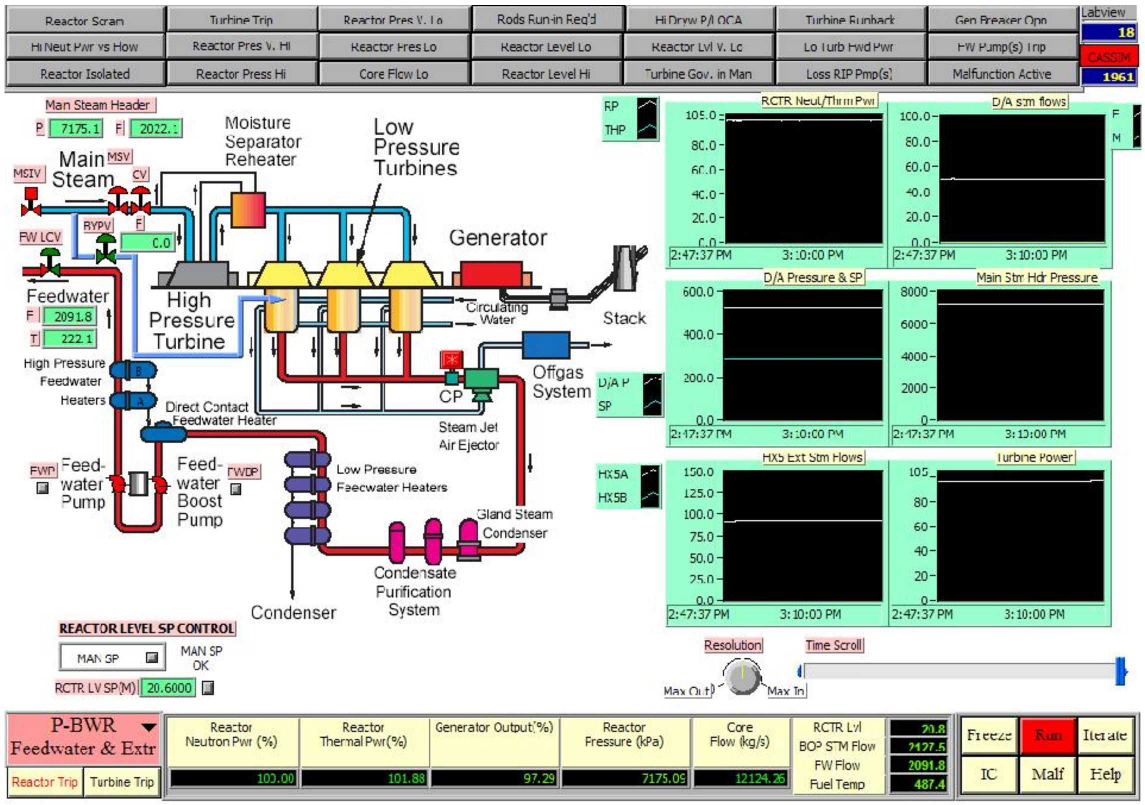
This screen shows the main parameters and controls associated with the turbine and the generator. The parameters displayed are:

- 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/sec)
- Governor control valve position (% open)
- Generator output (MW)
- Turbine/generator speed of rotation (rpm)
- Generator breaker trip status
- Turbine trip status
- Turbine control status
- All the trend displays have been covered elsewhere or are self explanatory

The following pop-up menus are provided:

- TURBINE RUNBACK — sets target (%) and rate (%/sec) of runback when 'Accept' is selected
- TURBINE TRIP STATUS — trip or reset
- Steam bypass valve 'AUTO/MANUAL' control — AUTO select allows transfer to MANUAL control, following which the manual position of the valve may be set.
- Computer 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 "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.
- Turbine runup/speedup controls
- MSIV manual open/close controls

4.7 PASSIVE BWR FEEDWATER AND EXTRACTION STEAM SCREEN



This screen shows the portion of the feedwater system that includes the Condensate Pumps (CP), Low Pressure Heaters, Direct Contact Feedwater Heater, the Reactor feedwater pumps, the high pressure heaters, with the output of the HP heaters going to the reactor water level control valves. The following parameters are displayed:

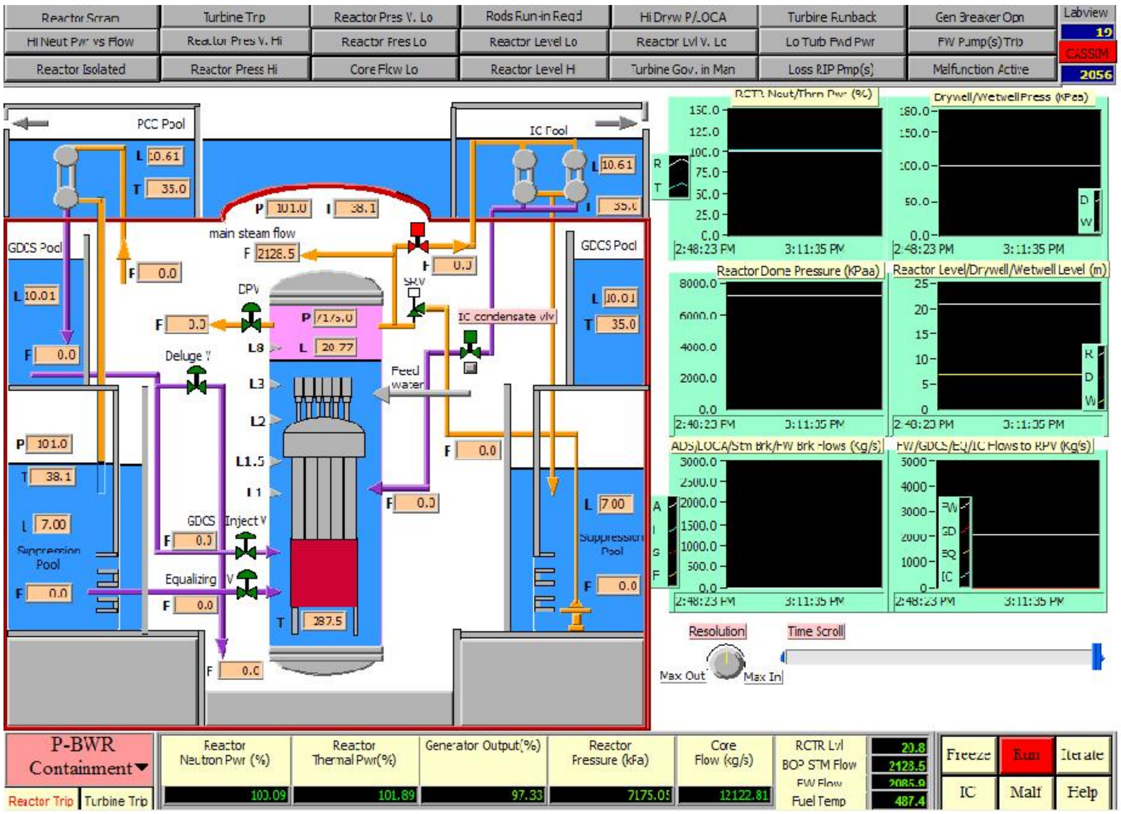
- Main steam header pressure after the main steam isolation valve, steam flow through the turbine governor valve and the bypass valve.
- Main feedwater pump and auxiliary feedwater pump status with associated pop-up menus for 'ON/OFF' controls.
- Flow rate at reactor level control valve outlet and feedwater temperature.

Assume Top of Fuel (TAF) = 9 m,; normal water level setpoint is 20.7 m, the respective Level Trip Setpoints are implemented as follows:

- L 9 = 22.39 m – action: Reactor scram
- L 8 = 21.89 m – action: Turbine Trip.
- L 4 = 20.60 m – normal Reactor Level.
- L 3 = 19.00 m – action: Reactor Scram.
- L 2 = 16.50 m – action : initiate Isolation Condenser System; 29 delay, close MSIV
- L 1.5 = 13.00 m - action: start ADS blow down; initiate ICS.
- L 1 = 11.00 m – action: start equalizing valve.

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

4.8 PASSIVE BWR CONTAINMENT SCREEN



The Passive BWR Containment Screen and accompany model attempt to present the Drywell, Wetwell, IC Pool & steam condensation functions, PCC Pool and steam condensation functions, GDCS Pool and ECC injection functions, ADS functions, system behavior and parameters as described in **Section 2.11 Passive Safety Features**:

The parameters shown on the screen are:

(a) IC Pool and Condenser: IC Pool level and temperature, steam supply isolation valve status (from RPV to IC Pool), IC Condensate valve status (from IC condenser to RPV), steam flow to IC Pool from RPV.

(b) PCC Pool and Condenser: PCC Pool level and temperature; steam flow from Drywell to PCC Pool, condensate flow from PC Pool to GDCS Pool.

(c) RPV: Dome pressure, reactor water level, average fuel temperature; DPV valve status, flow from DPV to Drywell; SRV status, steam relief flow from SRV to Suppression Pool.

(d) Drywell: Drywell pressure, temperature; water level (in case of LOCA)

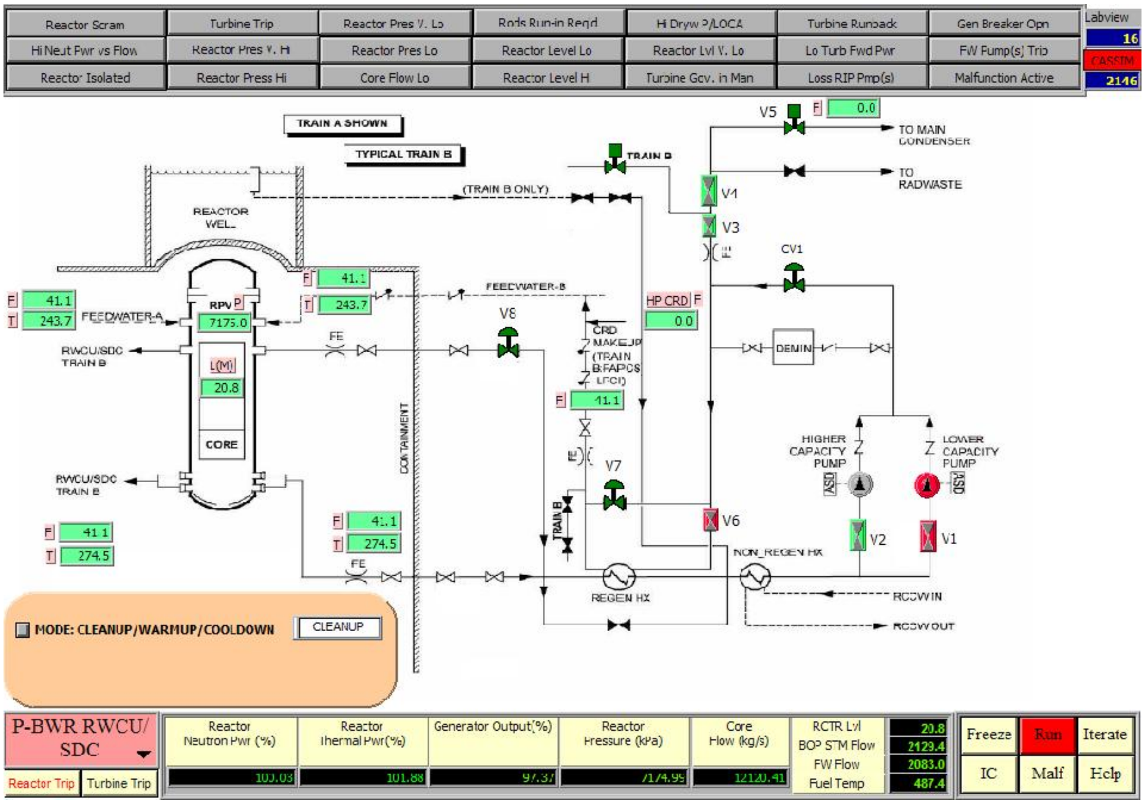
(e) GDCS Pool: GDCS pool level and temperature; deluge valve status and flow; GDCS injection valve status and flow.

(f) Wetwell (Suppression Pool): Wetwell pressure, pool temperature, level; equalizing valve status and flow; dynamic simulation of vent clearing in the event of LOCA.

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)
- Feedwater flow to RPV, GDCS flow to RPV, Wetwell equalizing flow to RPV, IC condensate flow to RPV (bottom right).

Pop-up includes:
- IC Condensate Valve “open/close”.

4.9 PASSIVE BWR CLEANUP/SHUTDOWN COOLING SCREEN



The passive BWR Cleanup/Shutdown Cooling Screen represents the Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) system as described in Section 2.13. It is a simple model comprising two independent trains. Each train includes:

- One non-regenerative heat exchanger (NRHX);
- One regenerative heat exchanger (RHX);
- One low capacity cleanup (function) pump;
- One high capacity SDC pump;
- One demineralizer, and;
- Associated valves and pipes.

The pop-up MODE control can be used to initiate respective operational mode: – CLEANUP, WARMUP, COOLDOWN"

(a) Power Normal Operation with RWCU/SDC System at CLEANUP Mode — During normal power operation, reactor water flows from the reactor vessel and is cooled while passing through the tube side of the RHXs and the tube side of the NRHXs. The RWCU/SDC pumps then pump the reactor water through the demineralizers, and back through the RHX shell side where the reactor water is reheated and is returned to the reactor vessel via the feedwater lines. See above graphic for the associated valving and pumping arrangement for CLEANUP mode.

(b) Plant Shutdown Cooling - The ESBWR is a passive plant and does not have the traditional RHR system. For normal shutdown and cooldown, residual and decay heat is removed via the main condenser first and then the RWCU/SDC system as discussed below.

When RWCU/SDC system is in the COOLDOWN Mode, V1 and V2 will be opened. Both the low and the high capacity speed-adjustable pumps will be turned on. CV1 will be opened to allow more flow in parallel to the demineralizer. V6 will be closed, and V7 will be opened to bypass the RHX. In this configuration, the hot coolant is cooled down via the tube side of NRHX. The shell side of NRHX has RCCW flow for cooling, whose flow rate and discharge temperature from NRHX is controlled by a controller.

Note: Should the RPV becomes isolated following a scram during power operations, the ICS provides cooling of the reactor. The ICS automatically removes residual and decay heat to limit reactor pressure within safety limits when the reactor isolation occurs.

(c) Plant Startup & Warmup — During plant startup from cold shutdown state with mode switch at "WARMUP", the feedwater and/or CRD water will be introduced to the reactor RPV. With this configuration, V1 and V2 will be opened, and both low and high capacity speed-adjustable pumps will be turned on. CV1 will be opened to allow more flow in parallel to the demineralizer. V6 is opened, V7 is closed, so that the coolant is heated by RHX, on its return to RPV.

During heatup, reactor water level will increase due to thermal expansion. If the water level is higher than L4, it will be dumped, or overboarded, to the main condenser to maintain reactor water level at L4. Overboarding of reactor water is accomplished by opening V3, V4, and V5., and closing V6 and V7.

After warmup the RPV pressure is brought to saturation pressure by opening the vessel to the main condenser through the turbine bypass lines to promote deaeration of the reactor water. The RWCU/SDC system continues to remove excess water by dumping, or overboarding, to the condenser hotwell, if the water level is higher than L4.

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5.0 PASSIVE BWR SIMULATOR EXERCISES

5.1 POWER MANEUVER, SHUTDOWN, STARTUP EXERCISES

5.1.1 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/sec
  - ⇒ '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/sec.

5.1.2 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/sec from 100 % to 65 %. **Monitor the Reactor Water Level at all times**
- From 65 % to 20 %, use the rate of 0.5 % FP/sec (**Note: choose a slower rate, e.g. 0.3 % per sec., if there are fluctuations in core flow, power, and level**).
- From 20 % to 0 %, use the rate 100 % present power (PP)/sec.
- Record the following values:

Parameter	Unit						Comments
Reactor Power	%	100%	75%	50%	25%	0%	
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	%						

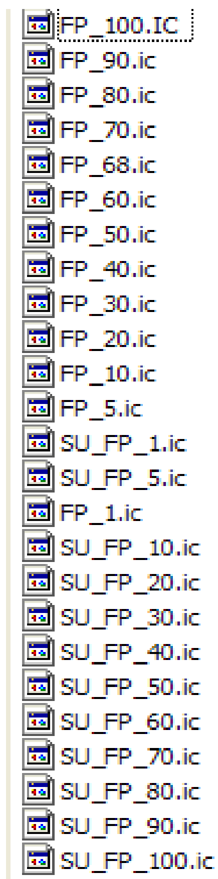
Under "Comments" please note type of parameter change as a function of reactor power 0% $\rightarrow$ 100% FP: constant, linear increase or decrease, non-linear increase or decrease. In particular, comment on the power — flow path during the power evolution.

- Increase reactor power back to 100% after ~ 0% is reached, using 25% steps
- From 0 % to 20 %, use 100 % present power/sec.
- From 20 % to 65 % , use the rate 0.5% per sec. (maximum rate is 1 % FP/sec.). 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/sec. or slower
- Repeat the above recordings and comments.

Note:

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. 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 consider functional 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 following IC stored points provide snapshots of the plant in various stages of unloading and reloading.



Note: due to some non-linearity in the modeling, there are some power conversion anomalies observed at low power – for example, the IC said “1 % FP “ neutron power, the electrical output may not be at the stated power in MWe. Henceforth, caution is advised for the simulator users regarding these anomalies.

5.1.3 Turbine trip and recovery

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Some examples are high velocity separator drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum and reactor high water level.

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the Steam Bypass & Pressure Control (SB&PC) system. The reactor power decreases when the SCRRI/SRI (Selected Control Rods Run-In) actuates.

The neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the pressure increase is limited by the initiation of the steam bypass operation. Peak simulated thermal power does not increase significantly. After the control system verifies that the bypass capacity is adequate, the system activates the SCRRI/SRI to reduce the power to 60% and later proceed to a possible restart or a controlled shutdown.

(a) Sequence of Events :

1. Turbine Trip initiates closure of turbine Main Stop Valves
2. Turbine Trip initiates Turbine Bypass operation.
3. The Turbine Bypass Valves start to open to regulate pressure.
4. Selected Control Rods Run In activated to decrease reactor power to 60 % FP.
5. FW temperature is decreasing due to reduced turbine steam flow.
6. New steady state is established.

(b) Operator Actions:

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

(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) Monitor control rod drive positions.

(c) Practice Transient in Simulator:

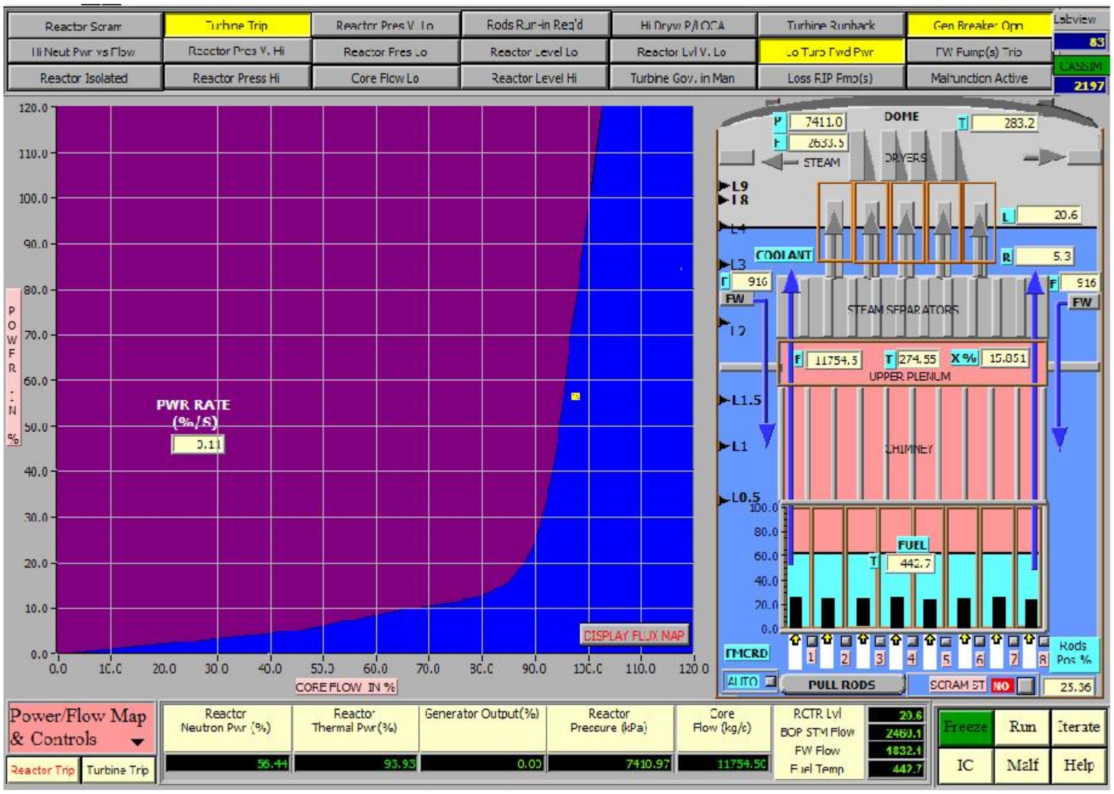
To observe the transients for Turbine Trip in the simulator, first load the simulator at Initial state of 100% full power, press the turbine trip button on the left hand bottom corner of the screen, and confirm turbine trip.

- 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 ? What is the steady state core 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?
- Monitor feedwater temperature. Does it have impact on the operation ?

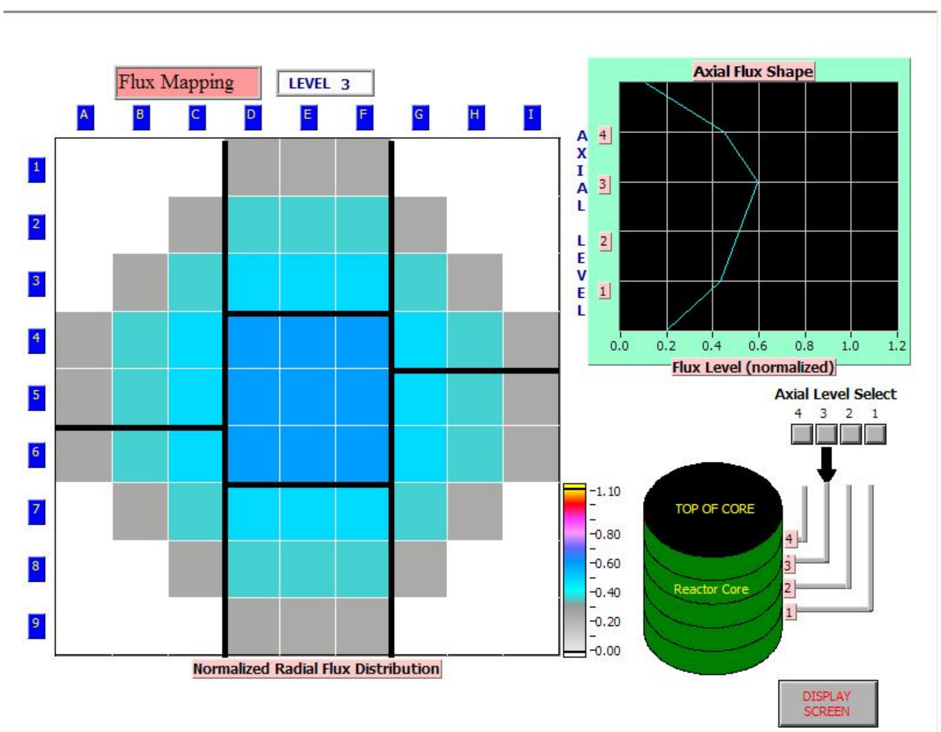
Follow the following steps to restart the Turbine Generator:

- 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.
- After turbine is in service, what happens to the steam bypass valve as the turbine power increases? Note the reactor pressure.
- 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 stabilizes, and then switch the governor control to "AUT"".
- 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 per sec.

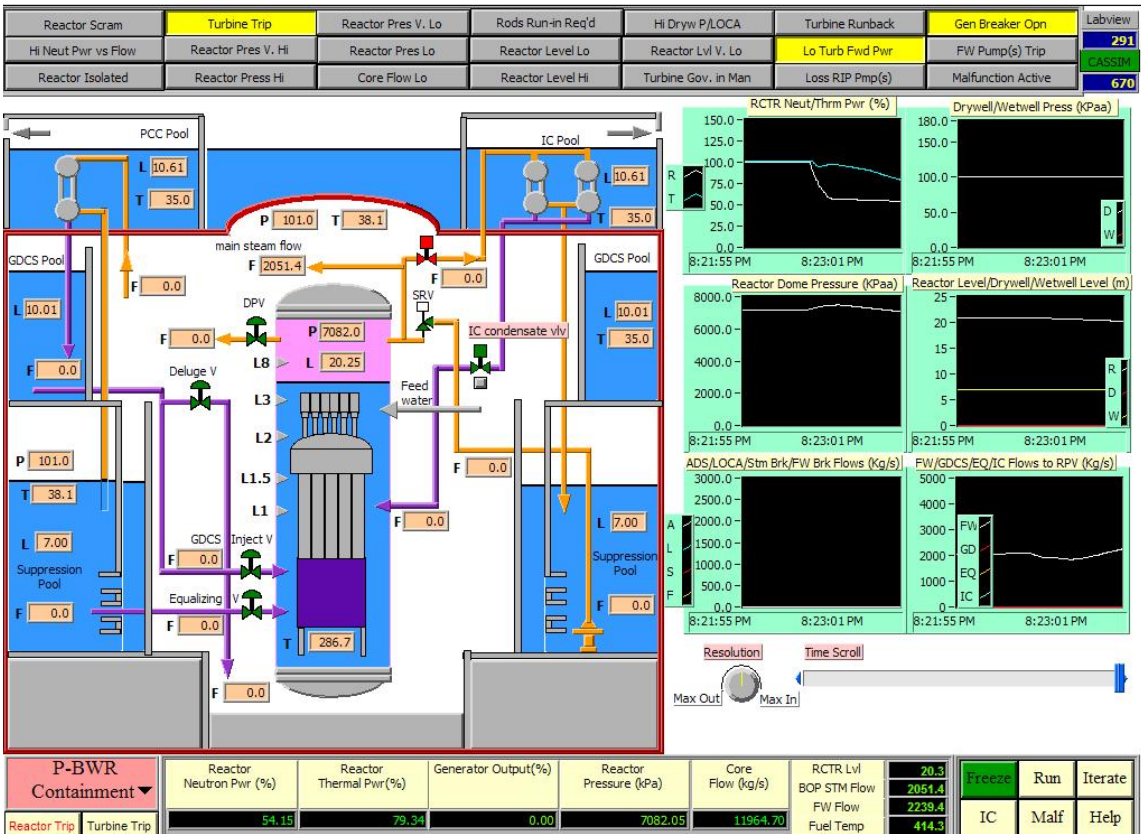
The Following snap shot shows the reactor power pulse due to void collapse from high reactor dome pressure, causing surge in reactivity increase. However, the closing of the Turbine Stop Valve initiates Turbine Bypass operation, and Selected Control Rods Run-in



The following snapshot shows core flux map after turbine trip. Provide explanation of the change in flux shape.



The following snap shot shows reactor level drops immediately after turbine trip, due to bubble collapse caused by high dome pressure. No Reactor Scram, nor SRV opening should be observed.



5.1.4 Reactor scram and quick 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.
- Wait until generator power is zero.
- Note the reactor neutron power. Reset reactor scram using control devices provided on "Power Flow Map & Controls" screen. Press the "YES" button next to SCRAM ST.
- The "YES" Button now turns to "NO", meaning the Reactor Scram Status is "NO SCRAM".
- Monitor the yellow cursor on the Power/Flow Map.
- Now pull the Control Rods out of the core by pressing the "PULL RODS" button to begin rods withdrawal.
- Observe any changes in control rods reactivity. **The rods should be withdrawing. Otherwise, go to the power flow map & controls screen, and press the "PULL RODS" Button again, you should see the "down" arrows showing the rods movement**.
- When the rods are withdrawing, go to the Passive BWR Reactivity & Setpoint Screen. Enter a new power setpoint say 1 %, at a rate of 100% present power per sec., 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 sec will yield an *effective* power rate of 1 % FP/sec. So the value of present power selected by user should depend on the present neutron power at that time.
- Record the time (using the display under the chart recorders) needed to withdraw all rods to Reset line.
- Go to "Passive BWR Reactivity & Setpoint" Screen; record the net total reactor reactivity. Is the reactor subcritical, critical or supercritical?

  Go to Passive BWR 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.
- 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 change. If this happens, press the button: "BRING REACTOR TO CRITICAL RANGE". Note: <u>This procedure is only applicable for this Educational Simulator, and</u>

does not apply for the real plant. Observe that the rods are withdrawn to bring the reactor close to criticality.

- Continue to raise power to 10% FP, 20 % in 10 % steps at a rate not more than maximum effective power rate of 1 % FP per sec. (by entering the appropriate value of present power per sec). Note after 20% FP is reached, you can enter % FP per sec. as the rate, instead of "present" power per sec. Note that the maximum rate is 1 % FP/sec., the suggested rate is 0.5 % FP/sec.
- If the turbine is tripped as a result of low power, reset turbine trip, synchronize and reload as follows:

(a) Go to Passive 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 stabilizes, and then switch the governor control to "AUTO".

(d) After the turbine power is equal to reactor power, go to "Passive BWR Reactivity & Setpoints" to increase reactor power to 100% in 10% steps at 0.5 % FP per sec.

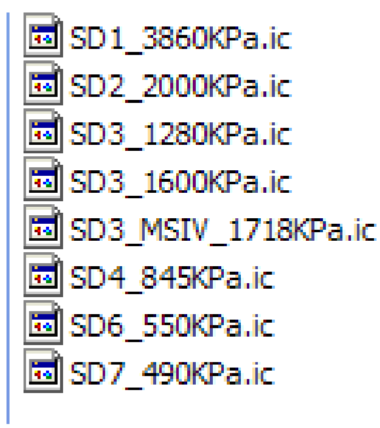
5.1.5 Reactor Shutdown Cooling

To shutdown the reactor after scram, follow the steps below:

- Confirm the reactor is scrammed, and turbine generator is tripped and on turning gear.
- Check reactor water level. It should recover after reactor scram.
- Go to Passive BWR Turbine Generator Screen, switch the BYPASS Valve mode to MANUAL, and use the pop-up to open the BYPASS VALVE to 2 % to bypass steam to condenser.
- Observe that the reactor pressure starts to decrease.
- Go to Passive BWR RWCU/SDC Screen, use the pop-up to switch from "CLEAN-UP" to cool-down.
- Provide explanation regarding the valving arrangement and the flow and temperature as shown on the Passive BWR RWCU/SDC Screen.

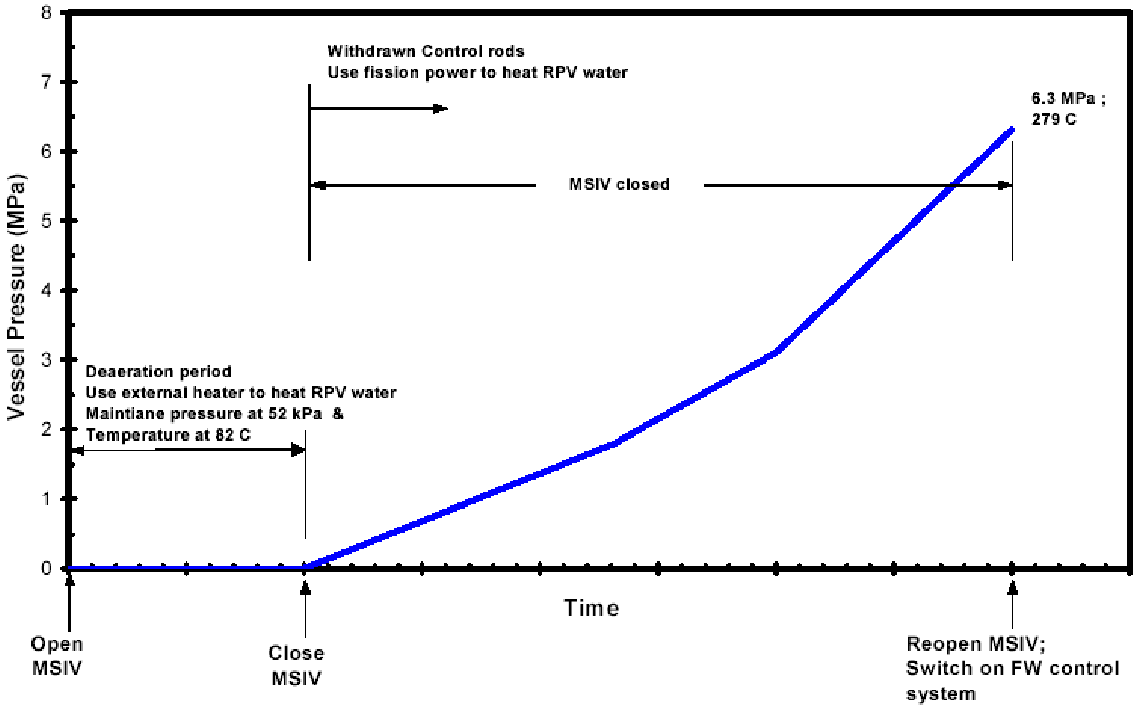
  When RWCU/SDC system is in the COOLDOWN Mode, V1 and V2 will be opened. Both the low and the high capacity speed-adjustable pumps will be turned on. CV1 will be opened to allow more flow in parallel to the demineralizer. V6 will be closed, and V7 will be opened to bypass the RHX. In this configuration, the hot coolant is cooled down via the tube side of NRHX. The shell side of NRHX has RCCW flow for cooling, whose flow rate and discharge temperature from NRHX is controlled by a controller.

- The following IC points are snapshots of shutdown cooling of the reactor pressure from rated pressure to 490 KPa.



- Load each of the IC points and observe if there are any changes in system status – ICS, MSIV etc. Provide explanations if there are any changes.
- After understanding the SDC procedure, load the Reactor Zero Power Hot - with reactor scrammed and turbine tripped, and practice Shutdown Cooling.

5.1.6 Reactor startup and warmup



Startup.xls

The above figure shows the stages of the startup process. In the De-aeration Period, the reactor coolant is de-aerated by drawing a vacuum on the main condenser and reactor vessel using mechanical vacuum pumps with the steam drain lines open. The reactor coolant is heated up to between 80 and 90°C with the Reactor Water Cleanup/Shutdown Cooling System RWCU/SDC) auxiliary heater and decay heat. The reactor pressure is reduced to about 50 to 60 kPa.

Following de-aeration, the Main Steam Isolation Valves (MSIVs) are closed to initiate the Startup Period. Control rods are withdrawn to criticality. Fission power is used to heat the reactor water, while maintaining the water level close to the top of the separators but well below the steam lines. Steaming at the free surface starts to pressurize the reactor vessel. The core region remains subcooled due to the large static head in the chimney and separators.

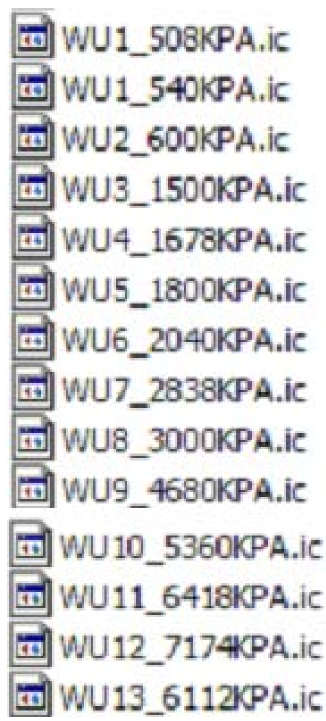
As the reactor heats up and pressurizes, the RWCU/SDC system heat exchangers are used to control the downcomer temperature, enhance coolant flow and reduce lower plenum stratification.

The MSIVs are reopened at the end of the Startup Period, when the pressure reaches 6.3 MPa. Subsequently, the turbine bypass valves are used to control pressure. The RPV power is increased and preparations made to roll the turbine.

In the actual plant startup, the reactor vessel will likely be pressurized during the Startup Period with the MSIVs open, and the turbine bypass and control valves closed, similar to the operating BWR plants. The character of the startup transient is not expected to change from what is presented here; however, it is possible that the heatup rate could be lower and the startup time could be longer.

Use the simulator to practice reactor warmup and startup. Follow the steps below:

- The following ICs record snapshots of the plant during various stages of warmup. Load up each of the IC and note the status of ICS and MSIV.



- Go to Passive BWR RWCU/SDC Screen. During plant startup from cold shutdown state with RWCU/SDC mode switch at "WARMUP", the feedwater and/or CRD water will be introduced to the reactor RPV. With this configuration, V1 and V2 will be opened, and both low and high capacity speed-adjustable pumps will be turned on. CV1 will be opened to allow more flow in parallel to the demineralizer. V6 is opened, V7 is closed, so that the coolant is heated by RHX, on its return to RPV.

  During heatup, reactor water level will increase due to thermal expansion. If the water level is higher than L4, it will be dumped, or overboarded, to the main condenser to maintain reactor water level at L4. Overboarding of reactor water is accomplished by opening V3, V4, and V5, and closing V6 and V7.

- After warmup, the RPV pressure is brought to saturation pressure by opening the vessel to the main condenser through the turbine bypass lines to promote deaeration of the reactor water. The RWCU/SDC system continues to remove excess water by dumping, or overboarding, to the condenser hotwell, if the water level is higher than L4.

- Load the cold IC SD7_490KPA.ic, and start the warmup process.

5.2 MALFUNCTION EXERCISES

5.2.1 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.

(a) Sequence of Events

- Loss of feedwater flow results in a reduction of vessel inventory, causing the vessel water level to drop.
- Water level continues to drop and when the level reaches the vessel level (L3), scram trip setpoint is reached, whereupon the reactor is shut down.
- 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 HPCRD injection is started and MSIV closure is initiated after 30 sec. delay.
- Water level will recover above L4. HPCRD injection will stop when level has reached L8.

(b) Practice Transient at Simulator

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.

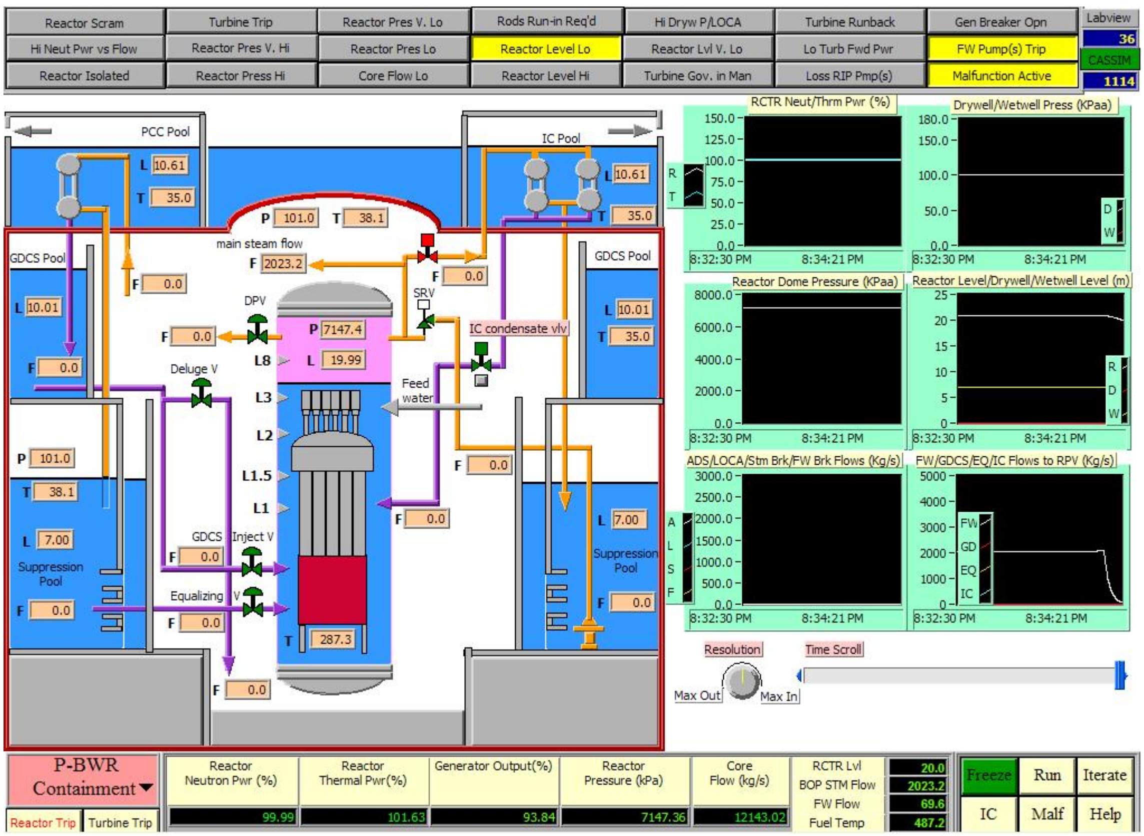
When this malfunction transient occurs:

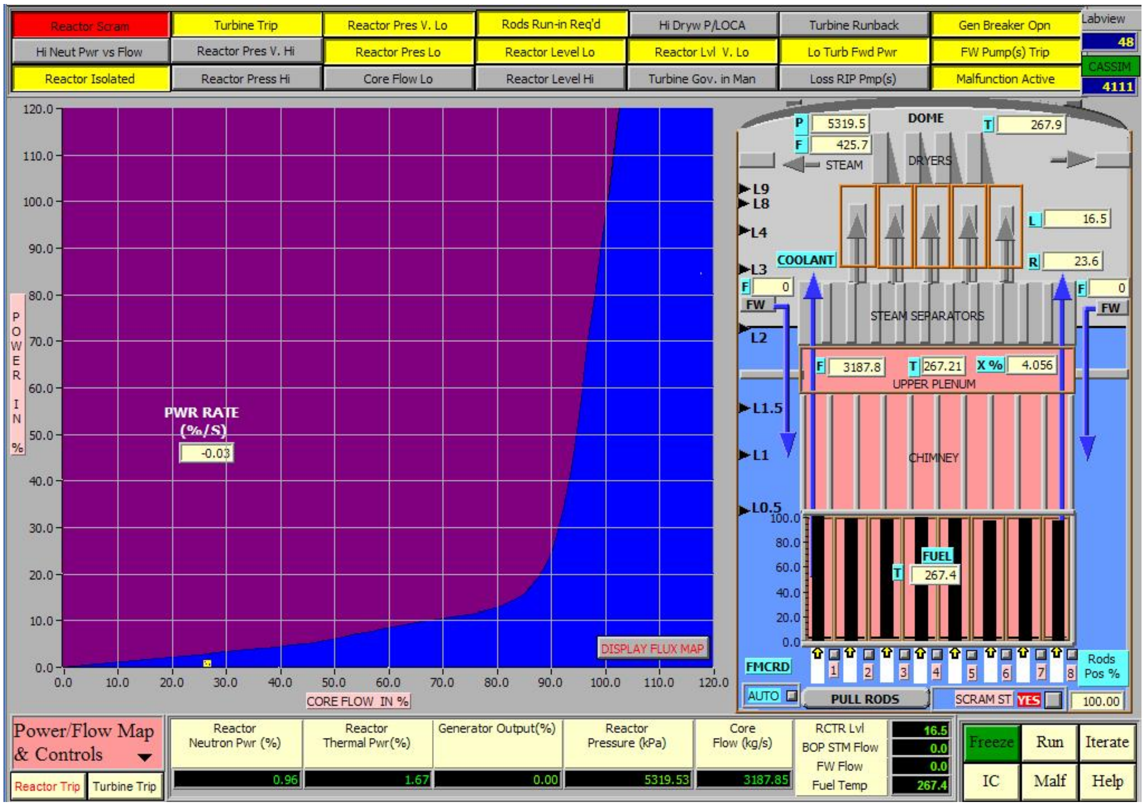
- On Passive BWR Feedwater & Extraction Steam Screen, observe that both the feedwater pumps stop.
- Go to Passive BWR Power/Flow Map Screen. The reactor level drops quickly due to loss of feedwater flow.
- Dome pressure is decreasing gradually as reactor water level drops. Provide explanation why this 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 MW drops.
- 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 reached L2, the HP-CRD injection is started and MSIV closure is initiated after 30 sec. Observe the HP-CRD flow in Passive BWR RWCU/SDC Screen. Observe MSIV closure in Screen Passive BWR Turbine Generator.
- Water level will recover above L4. HPCRD injection will stop when level has reached L8.

(c) Operator Action

In the real plant, the operator should ensure 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 at L3.
(2) Verify HPCRD initiation when water level reaches L2.
(3) Monitor MSIV closure.
(4) Monitor turbine coastdown, break vacuum as necessary.
(5) Complete scram report and investigate the root cause for "loss of all feedwater flow" event.




Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
48
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
4111
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
POWER IN %
PWR RATE (%/S)
-0.03
CORE FLOW IN %
DISPLAY FLUX MAP
DOME
DRYERS
STEAM
COOLANT
STEAM SEPARATORS
UPPER PLENUM
CHIMNEY
FUEL
FW
FMCRD
AUTO
PULL RODS
SCRAM ST
YES
Rods Pos %
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
0.96
Reactor Thermal Pwr(%)
1.67
Generator Output(%)
0.00
Reactor Pressure (kPa)
5319.53
Core Flow (kg/s)
3187.85
RCTR Lvl
16.5
BOP STM Flow
0.0
FW Flow
0.0
Fuel Temp
267.4
Freeze
Run
Iterate
IC
Malf
Help

5.2.2 Inadvertent Isolation Condenser Initiation

Manual startup of the four individual ICS is postulated for this transient (i.e. operator error). It begins with the opening of ICS condensate discharge valve, and full isolation condenser loop flow is established.

(a) Sequence of Events:

- Inadvertent startup of the isolation condenser causes a reactor pressure decrease from the initial conditions.
- The decrease in reactor pressure will cause SB&PC System to close turbine control valve, leading to decrease in turbine power.
- Closing of turbine control valve will, at some point, reverse the reactor pressure decrease, and the pressure starts increasing.
- Addition of cooler water to the downcomer causes a reduction in inlet enthalpy, which results in a slight power increase. However, the reactor power regulation system will insert control rods to maintain power to setpoint.
- The plant state is stabilized.

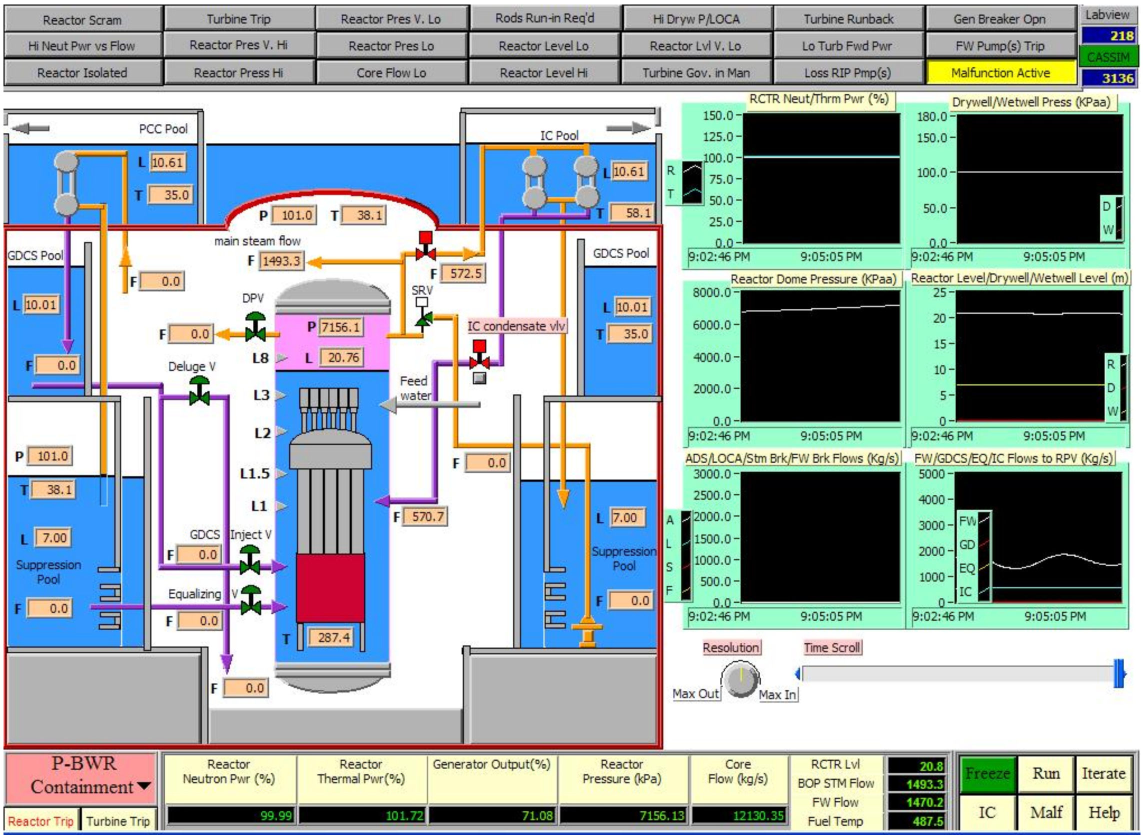
(b) Practice Transient at Simulator

Go to Passive BWR Containment Screen. Load the full power IC and run the simulator. Insert the malfunction “Inadvertent Isolation Condenser Initiation”.

- Monitor reactor pressure, and steam flow.
- Go to Passive BWR Turbine Generator Screen, monitor turbine control valve opening.
- Go to Passive BWR Power/Flow Screen, monitor water level, reactor power level.
- When the plant reaches a steady state, record the reactor power, turbine power, steam flow from dome, steam flow to ICS, steam flow to turbine control valve and bypass valve.

(c) Operator Action

- There is a control pop-up for ICS Condensate Valve. Discuss operator actions to recover from this transient, and to return the turbine power to its original value.



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
Labview
218
CASSIM
3136
PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
IC condensate vlv
Deluge V
Feed water
GDCS
Inject V
Equalizing V
Suppression Pool
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
9:02:46 PM
9:05:05 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR
Containment
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
99.99
101.72
71.08
7156.13
12130.35
RCTR Lvl
BOP STM Flow
FW Flow
Fuel Temp
20.8
1493.3
1470.2
487.5
Freeze
Run
Iterate
IC
Malf
Help

5.2.3 Inadvertent Opening of Bypass Valve

(a) Description of Event

- Inadvertent opening of Bypass Valves could due to operating error.
- When this error occurs, reactor pressure will decrease due to more than rated steam flow from reactor dome.
- The SB&PC system senses the pressure change and commands the turbine control valves to close, and thereby automatically reducing turbine load.
- The increased reactor pressure will activate Isolation Condenser System.
- Due to increased steam flow, the main steam line pressure will be lower than 5500 KPa, at which point MSIV closure is initiated.
- The closure of MSIV will scram the reactor.

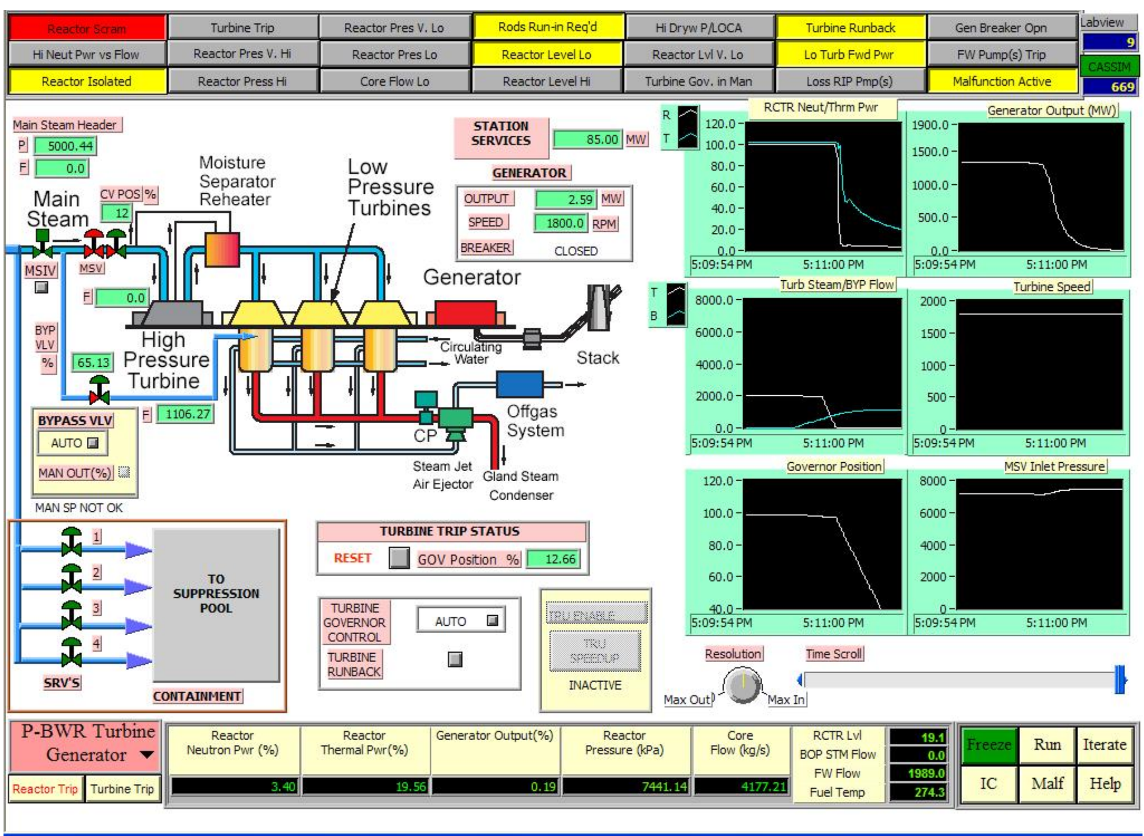
(b) Practice Transient at Simulator

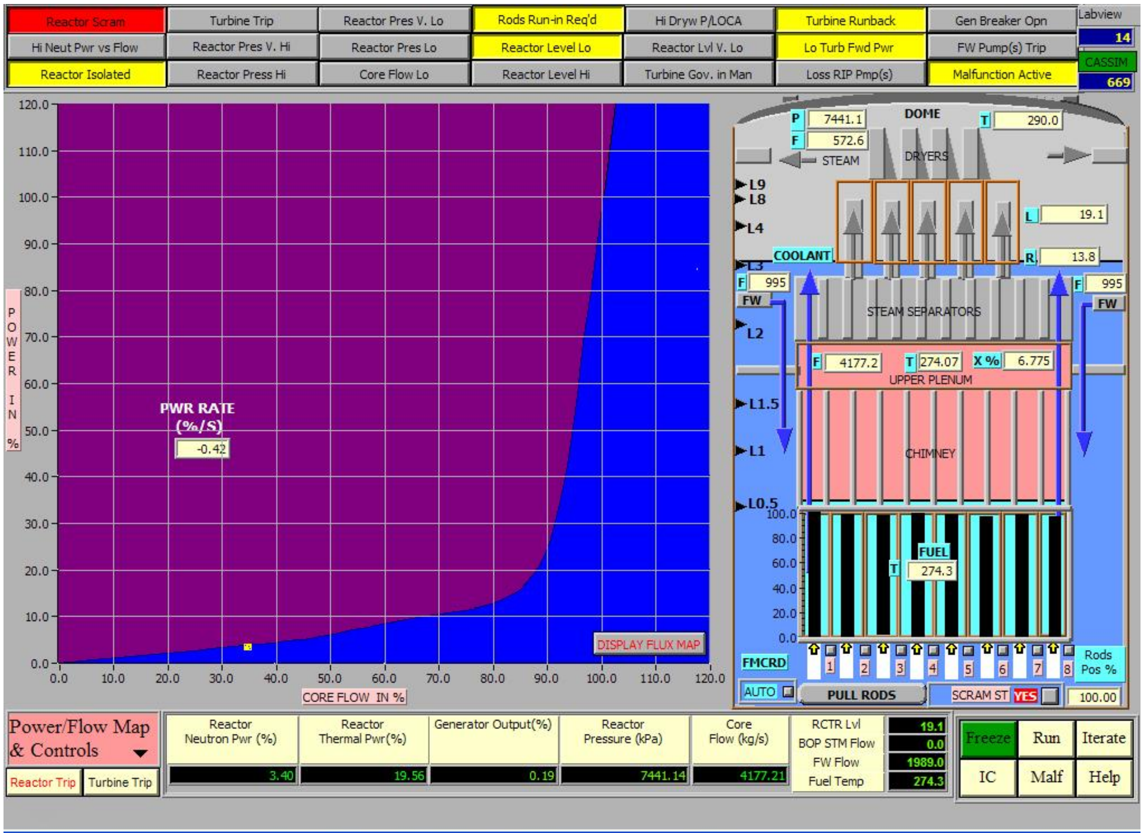
Go to Passive BWR Turbine Generator Screen. Load the 100 % IC and run the simulator. Insert the malfunction "Inadvertent Opening of Bypass Valve" and observe the transient:

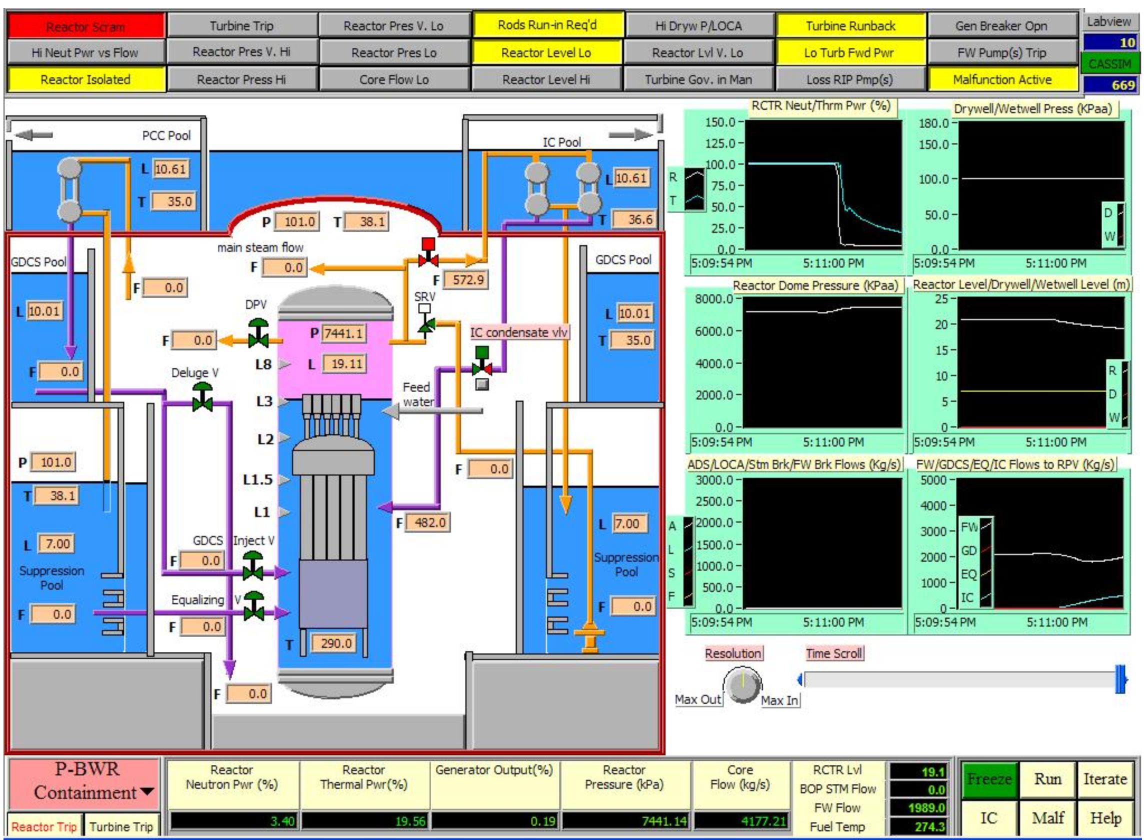
- Monitor the steam flow to Bypass Valve, steam flow via turbine control valve.
- Monitor reactor pressure. Is the ICS activated, at which pressure ?
- Monitor main steam line pressure at Passive BWR Turbine Generator Screen.
- At what main steam pressure will MSIV closure is initiated ?
- Is reactor scram ? By what parameter ?

(b) Operator Actions

Discuss possible operator actions to mitigate the malfunction event, without a reactor scram ?




Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
14
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
669
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
POWER IN %
PWR RATE (%/S)
-0.42
CORE FLOW IN %
DISPLAY FLUX MAP
DOME
DRYERS
STEAM
COOLANT
STEAM SEPARATORS
UPPER PLENUM
CHIMNEY
FUEL
FMCRD
AUTO
PULL RODS
SCRAM ST YES
Rods Pos %
100.00
Power/Flow Map & Controls
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
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
10
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
669
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
IC condensate vlv
Deluge V
Feed water
Suppression Pool
GDCS
Inject V
Equalizing V
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
Resolution
Time Scroll
Max Out
Max In
P-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
Freeze
Run
Iterate
IC
Malf
Help

5.2.4 Decreasing steam flow from dome due to pressure control failure

The ESBWR Pressure Control System consists of the Steam Bypass and Pressure Control (SB&PC) system which controls turbine control valves and turbine bypass valves to maintain reactor pressure.

If the Reactor Pressure Control System fails in such a way that the process variable input (pressure) for the controller fails to "low" value slowly, but the pressure transmitter reading for display is normal. As a consequence of this failure, the Reactor Pressure Control System is "fooled" into thinking that the reactor pressure is lower than the pressure setpoint of 7170 KPa. In response, the SB&PC system erroneously issues a decreasing steam demand control signal, causing slow closure of turbine control valves as well as inhibition of steam bypass flow, thus increasing reactor power and pressure.

(a) Sequence of Events:

- The malfunction simulates zero steam flow demand to SB & PC System
- Turbine control valves start to close. Reactor pressure starts increasing.
- As neutron flux increases, the reactor power regulation system will insert control rods momentarily to maintain reactor power setpoint.
- When the reactor pressure exceeds 7,447 KPa, the Isolated Condenser System (ICS) is activated.
- With ICS activated, reactor pressure starts decreasing.
- The ICS capacity will not handle all the turbine steam flow, as the turbine control valve closing. At some point ~ 7200 KPa, the reactor pressure will rise again.
- When pressure exceeds 7720 KPa, Selected Control Rods Run In is activated to decrease reactor power.
- Despite Rods Run In, there is still continued reactor power increase.
- Upon reaching reactor pressure of 7870 KPa, reactor scram is initiated.

(a) Practice Transient at Simulator:

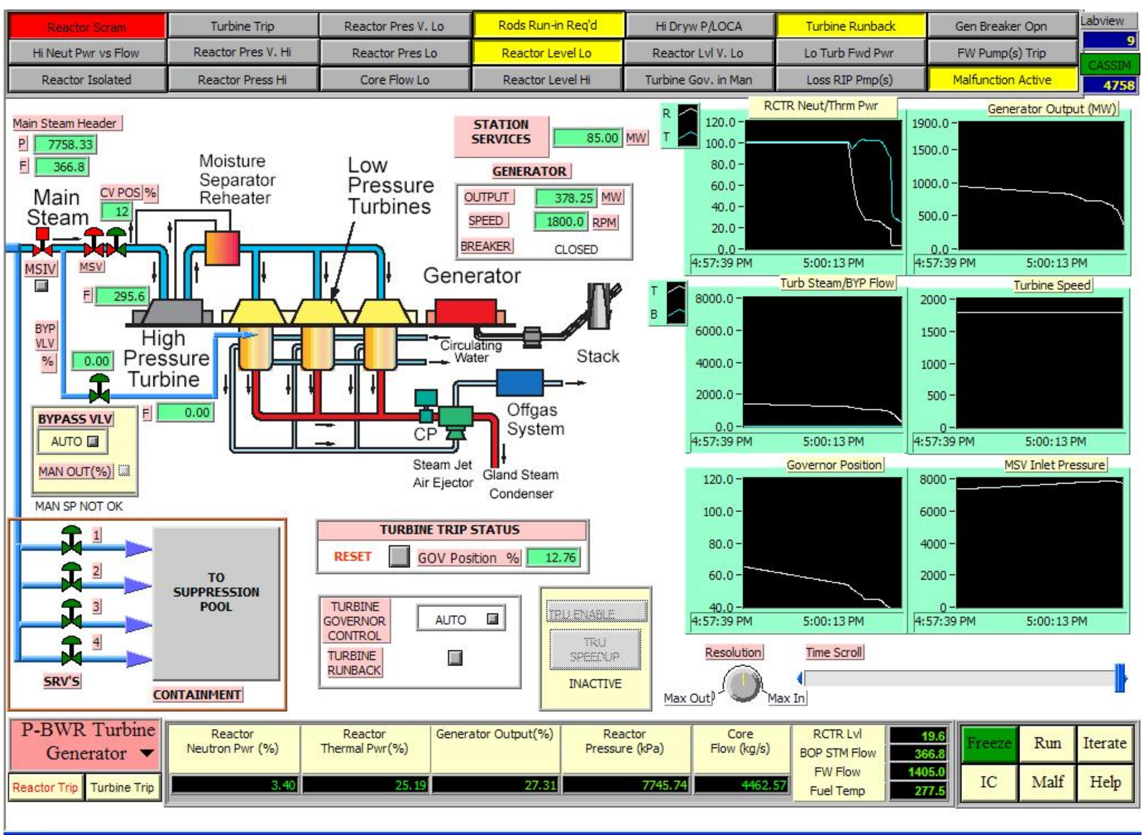
Go to the Power/Flow Map. Load the 100% FP IC. Run the simulator. Insert the malfunction.

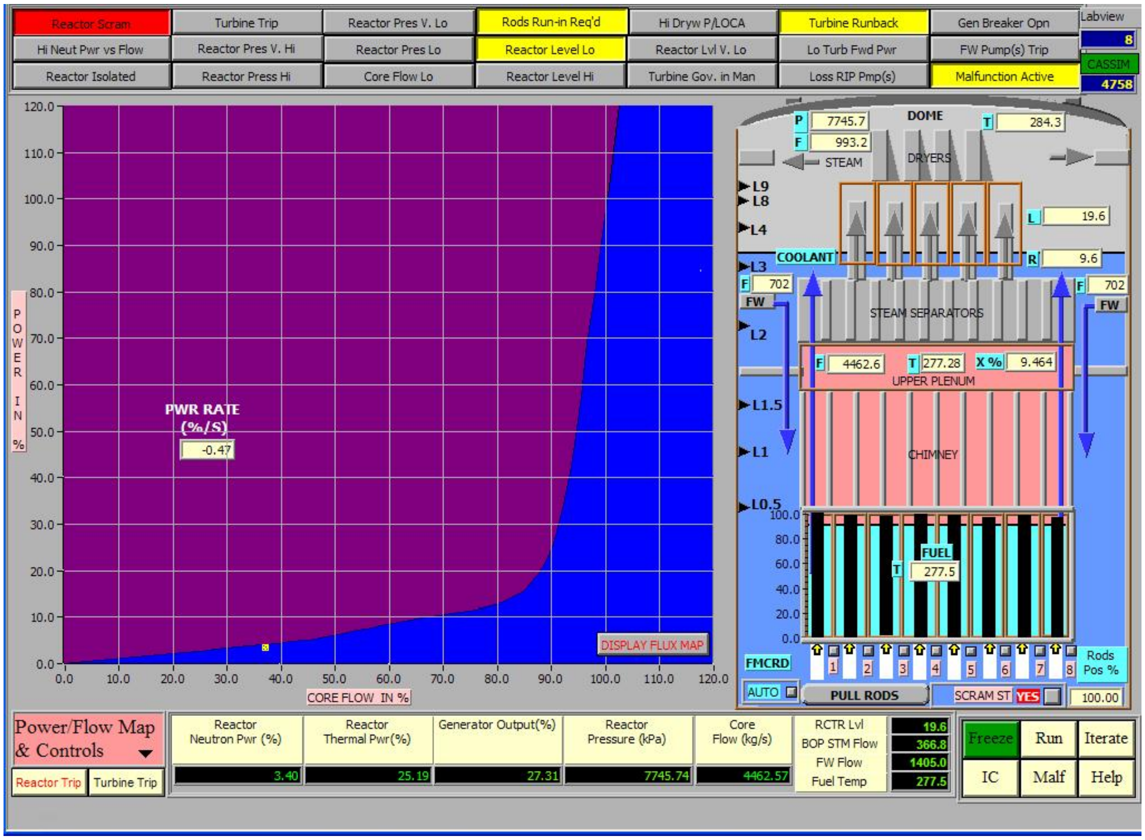
- On the Power/Flow Map Screen, observe the movement of the yellow cursor and record the reactor pressure reading - increasing or decreasing, as the malfunction event evolves.
- Observe changes in reactor power level, and the coolant flow. Provide explanation for the change, if any.
- As pressure increases, void fraction decreases. 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.

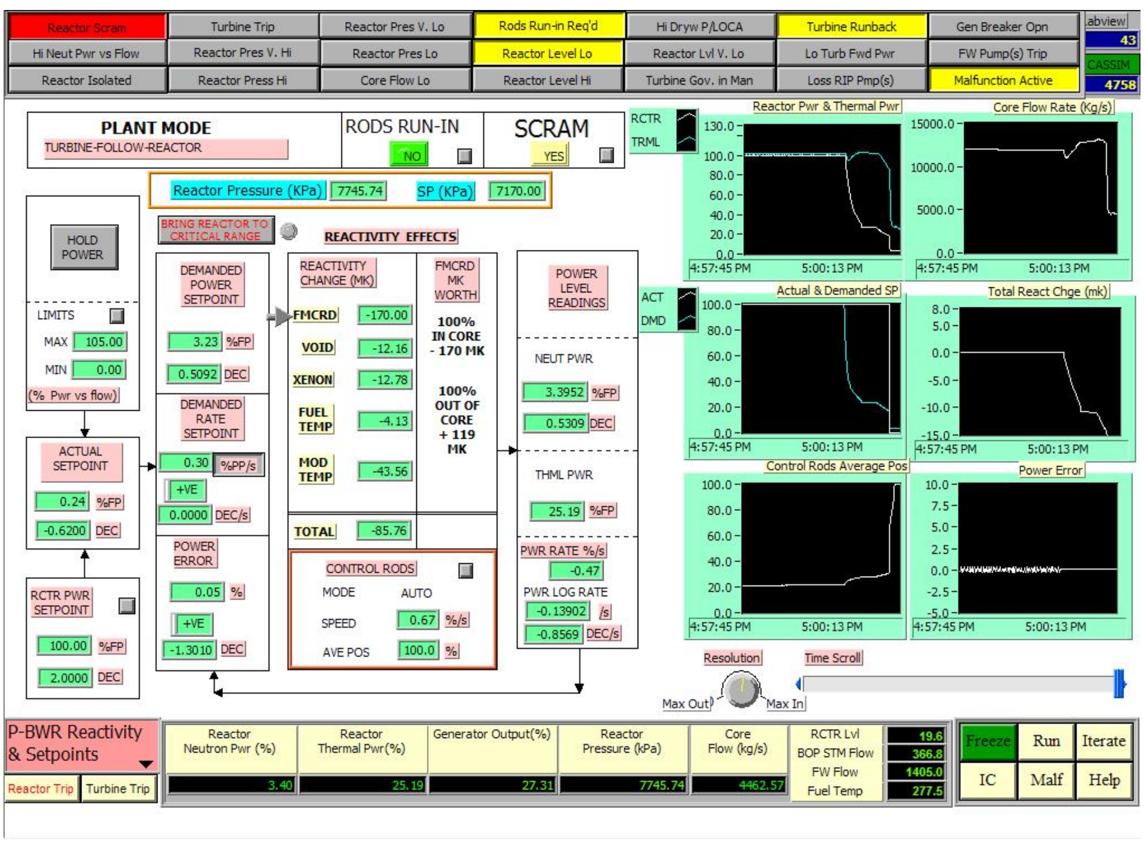
- Note the Generator Output (%) reading at the bottom of the screen - increasing or decreasing? Provide explanation. Compare this with the Reactor Thermal power (%). What is the difference? Explain the difference.
- As the malfunction event evolves, the reactor pressure increases to a point where ICS is activated. At what pressure does ICS come in ?
- Go to Passive BWR Containment Screen to observe ICS response. Note the reactor pressure begins to decrease. At what point, does the pressure increase again ?
- As this malfunction further evolves, alarm "Rods Run-in Required" will come on. Why ? Provide explanation.
- There will be an alarm "Reactor Press Hi". But after a short while, the reactor is scrammed. Explain why ?

(c) Operator Actions

- Discuss any possible operator intervention with this malfunction, taking into account the time allowed for operator to do the analysis of the root cause, plus the provisions of manual controls for turbine control valve and bypass valve




Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
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
4758
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
POWER IN %
PWR RATE (%/S)
-0.47
CORE FLOW IN %
DISPLAY FLUX MAP
DOME
STEAM
DRYERS
COOLANT
STEAM SEPARATORS
UPPER PLENUM
CHIMNEY
FUEL
277.5
FMCRD
AUTO
PULL RODS
SCRAM ST YES
Rods Pos %
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 3.40
Reactor Thermal Pwr(%) 25.19
Generator Output(%) 27.31
Reactor Pressure (kPa) 7745.74
Core Flow (kg/s) 4462.57
RCTR Lvl 19.6
BOP STM Flow 366.8
FW Flow 1405.0
Fuel Temp 277.5
Freeze
Run
Iterate
IC
Malf
Help


PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN
NO
SCRAM
YES
Reactor Pressure (KPa) 7745.74
SP (KPa) 7170.00
HOLD POWER
BRING REACTOR TO CRITICAL RANGE
REACTIVITY EFFECTS
DEMANDED POWER SETPOINT
3.23 %FP
0.5092 DEC
REACTIVITY CHANGE (MK)
FMCRD -170.00
VOID -12.16
XENON -12.78
FUEL TEMP -4.13
MOD TEMP -43.56
TOTAL -85.76
FMCRD MK WORTH
100% IN CORE - 170 MK
100% OUT OF CORE + 119 MK
POWER LEVEL READINGS
NEUT PWR
3.3952 %FP
0.5309 DEC
THML PWR
25.19 %FP
PWR RATE %/s
-0.47
PWR LOG RATE
-0.13902 /s
-0.8569 DEC/s
LIMITS
MAX 105.00
MIN 0.00
(% Pwr vs flow)
ACTUAL SETPOINT
0.24 %FP
-0.6200 DEC
DEMANDED RATE SETPOINT
0.30 %PP/s
+VE
0.0000 DEC/s
POWER ERROR
0.05 %
+VE
-1.3010 DEC
RCTR PWR SETPOINT
100.00 %FP
2.0000 DEC
CONTROL RODS
MODE AUTO
SPEED 0.67 %/s
AVE POS 100.0 %
Reactor Pwr & Thermal Pwr
Core Flow Rate (Kg/s)
Actual & Demanded SP
Total React Chge (mk)
Control Rods Average Pos
Power Error
4:57:45 PM
5:00:13 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 3.40
Reactor Thermal Pwr(%) 25.19
Generator Output(%) 27.31
Reactor Pressure (kPa) 7745.74
Core Flow (kg/s) 4462.57
RCTR Lvl 19.6
BOP STM Flow 366.8
FW Flow 1405.0
Fuel Temp 277.5
Freeze
Run
Iterate
IC
Malf
Help

5.2.5 Increasing steam flow from dome due to pressure control failure

The ESBWR Pressure Control System consists of the Steam Bypass and Pressure Control (SB&PC) system which controls turbine control valves and turbine bypass valves to maintain reactor pressure.

If the Reactor Pressure Control System fails in such a way that the process variable input (pressure) for the controller fails to "high" value slowly, but the pressure transmitter reading for display is normal. As a consequence of this failure, the Reactor Pressure Control System is "fooled" into thinking that the reactor pressure is higher than the pressure setpoint of 7170 KPa. In response, the SB&PC system erroneously issues an increasing steam demand control signal, causing slow opening of turbine control valves as well as the steam bypass valve, thus decreasing reactor power and pressure.

(a) Sequence of Events:

- The malfunction simulates increasing steam flow demand to SB & PC System
- Turbine control valves start to open. When turbine control valves have reached 100 % full opening, turbine bypass valve begins to open.
- Steam flow from steam dome is increasing and reactor pressure starts decreasing, same as reactor power.
- As neutron flux decreases due to increased void caused by decreasing reactor pressure, the reactor power regulation system will withdraw control rods momentarily to maintain reactor power setpoint.
- When the steam flow exceeds 120 % rated full power steam flow, MSIV closure will be activated, which triggers the reactor scram, as well as the ICS activation.
- With reactor scrammed, and ICS activated, reactor pressure start decreasing.
- Turbine will be runback due to decreasing reactor pressure, and will be tripped.

(b) Practice Transient at Simulator

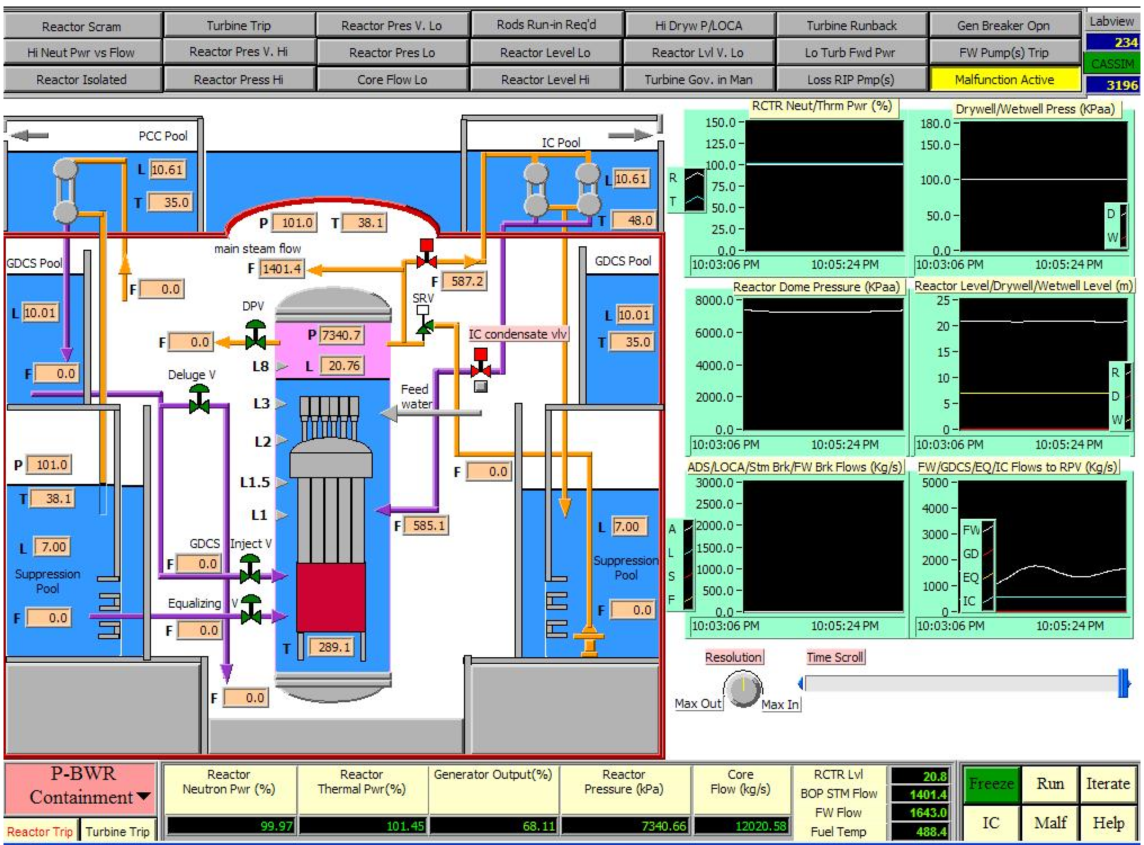
Go to Passive BWR Power/Flow Map Screen. Load the 100% FP IC. Run the simulator. Insert this malfunction. When this malfunction transient occurs:

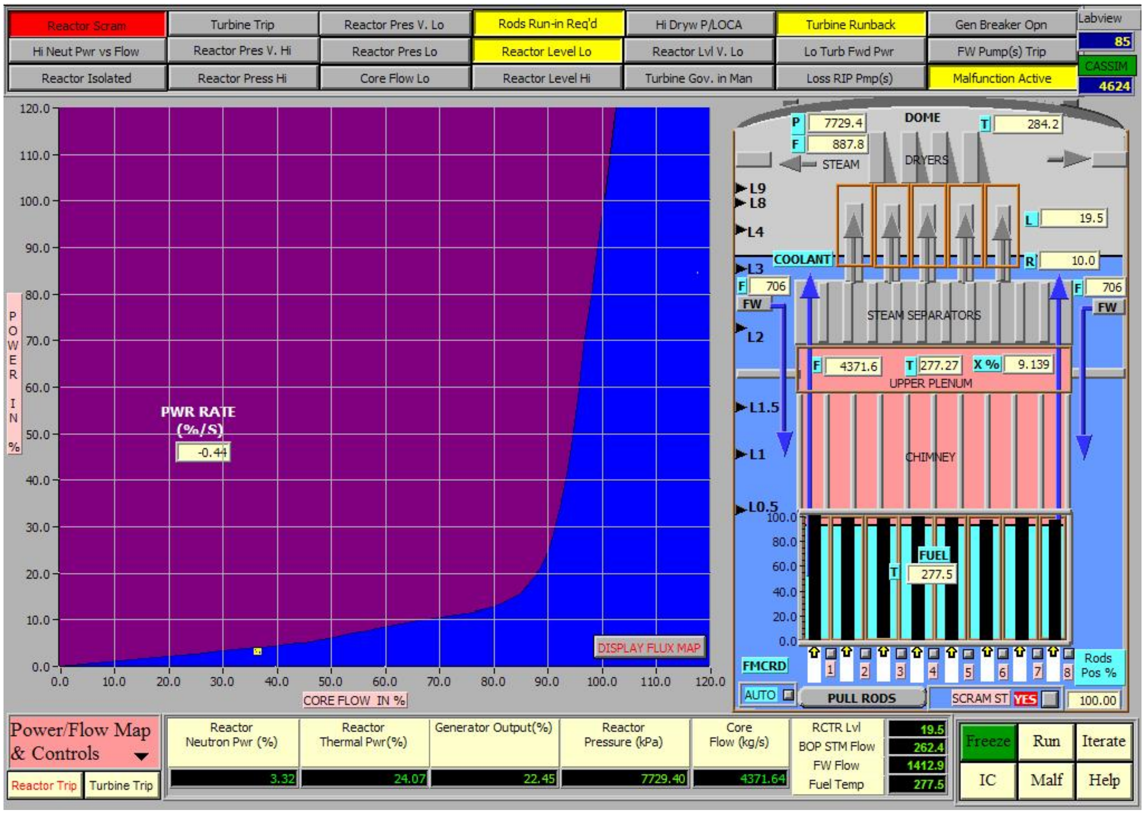
- On Passive BWR Power/Flow Map Screen, observe the movement of the yellow cursor and record the reactor pressure reading - increasing or decreasing, as the malfunction event evolves.
- Go to Passive BWR Turbine Generator Screen, monitor Turbine Control valve and Bypass Valve openings. Monitor steam flow from Dome.
- Observe that the reactor power is decreasing. Provide explanation why this is so.
- Reactor pressure will be decreasing further, and steam flow increases further.
- Is the reactor scrammed ? Analyze the event, how is the reactor scrammed ? By what parameter ? What Screen do you use to confirm the scram parameter ?

- Is the reactor isolated ? If it is, provide explanation. How to confirm isolation ?
- What other Engineered Safety Function (ESF) is activated ? Provide explanation. What Screen do you use to monitor Engineered Safety Function ?

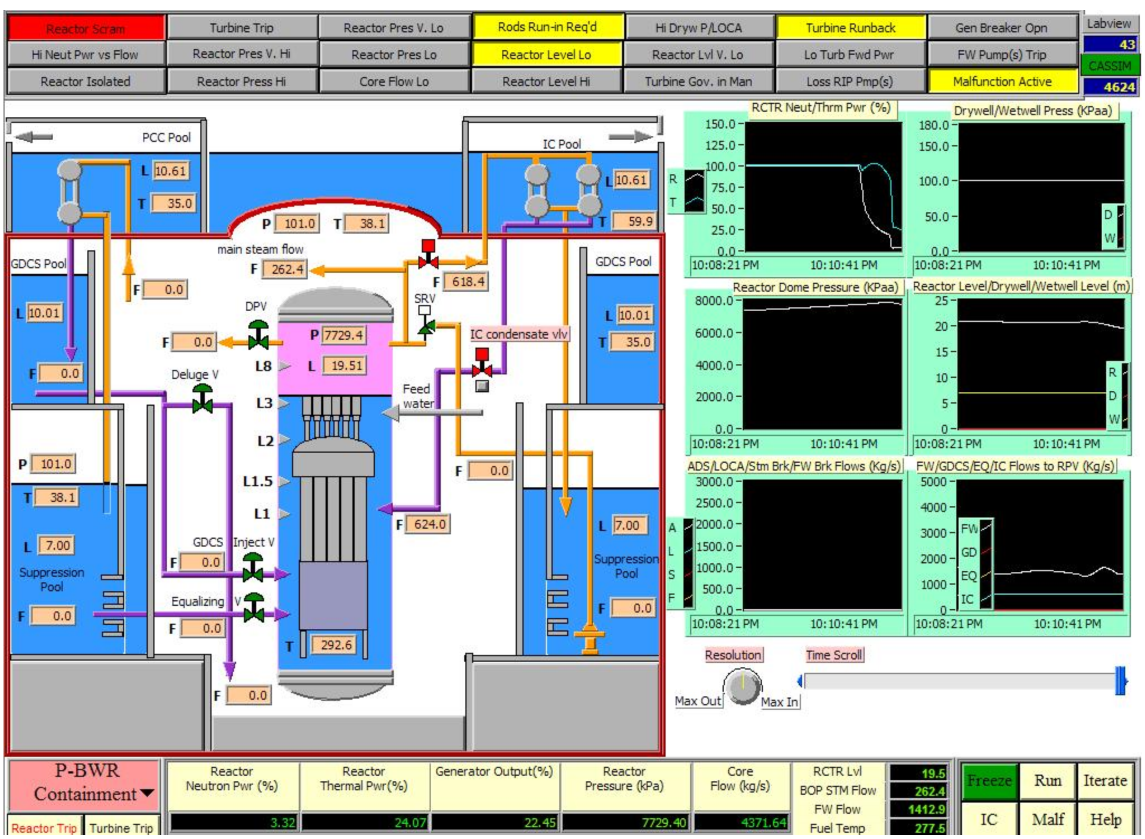
(c) Operator Actions

- Discuss any possible operator intervention with this malfunction, taking into account the time allowed for operator to do the analysis of the root cause, plus the provisions of manual controls for turbine control valve and bypass valve





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
PWR RATE (%/S)
-0.44
POWER IN %
CORE FLOW IN %
DISPLAY FLUX MAP
DOME
STEAM
DRYERS
COOLANT
STEAM SEPARATORS
UPPER PLENUM
CHIMNEY
FUEL
FMCRD
AUTO
PULL RODS
SCRAM ST YES
Rods Pos %
Power/Flow Map & Controls
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
Freeze
Run
Iterate
IC
Malf
Help



PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
IC condensate vlv
Deluge V
Feed water
GDCS Inject V
Equalizing V
Suppression Pool
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
Resolution
Time Scroll
Max Out
Max In
P-BWR Containment
Reactor Trip
Turbine Trip
Freeze
Run
Iterate
IC
Malf
Help

5.2.6 Turbine throttle PT fails low

This malfunction is turbine throttle pressure transmitter failing 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. But in actual fact, the turbine throttle pressure is not "low". The consequence is that the reactor pressure immediately shoots up rapidly. The increasing reactor pressure over 7170 KPa will open the turbine bypass valve.

(b) Sequence of Events:

- The malfunction simulates zero steam flow demand to SB & PC System
- Turbine control valves start to close. Reactor pressure starts increasing. The SB & PC starts opening turbine bypass valves. Eventually turbine will be tripped.
- As neutron flux increases, the reactor power regulation system will insert control rods momentarily to maintain reactor power setpoint.
- When the reactor pressure exceeds 7,447 KPa, the Isolated Condenser System (ICS) is activated.
- With ICS activated, reactor pressure starts decreasing.
- The ICS capacity will not handle all the turbine steam flow, as the turbine control valve closing. At some point (~ 7200 KPa), the reactor pressure will rise again.
- When pressure exceeds 7720 KPa, Selected Control Rods Run In is activated to decrease reactor power.
- The transient will settle at some intermediate reactor power and reactor pressure, with ICS active.

(b) Practice Transient at Simulator:

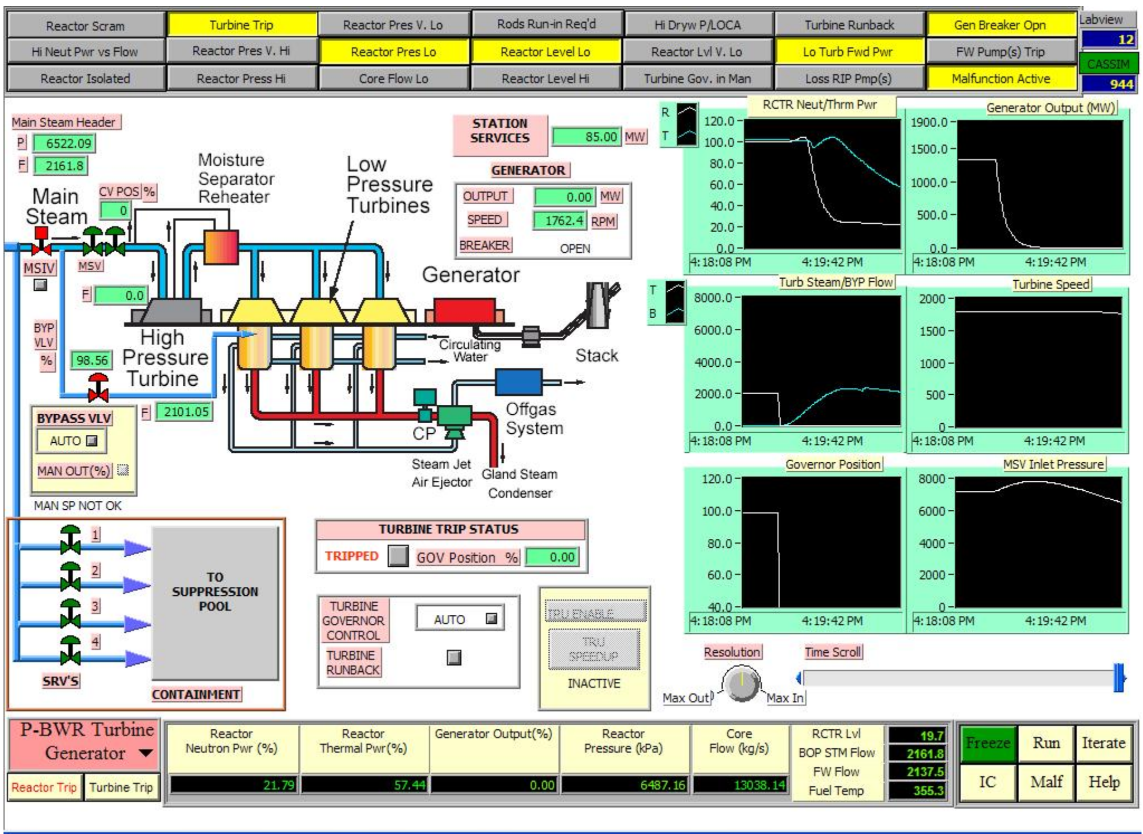
Go to the Passive BWR Power/Flow Map. Load the 100% FP IC. Run the simulator. Insert the malfunction.

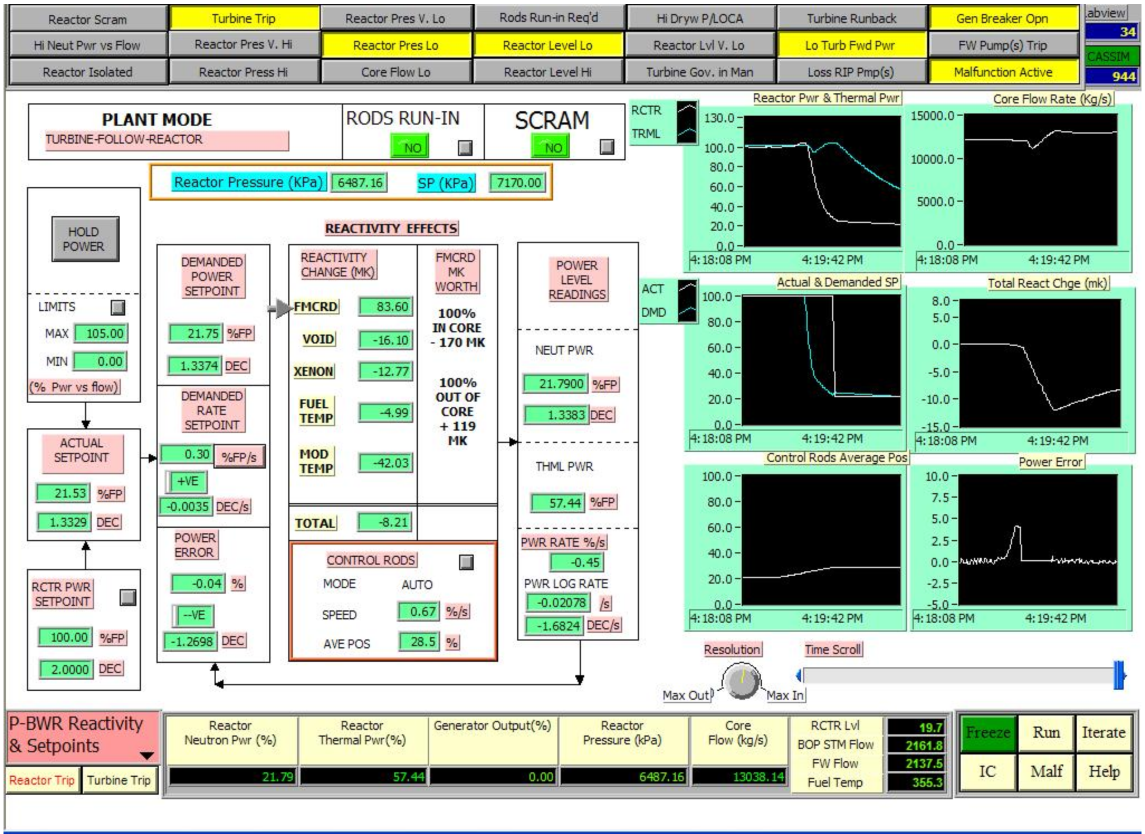
- On the Passive BWR Power/Flow Map Screen, observe the movement of the yellow cursor and record the reactor pressure reading - increasing or decreasing, as the malfunction event evolves.
- Observe changes in reactor power level, and the coolant flow. Provide explanation for the change, if any.
- As pressure increases, void fraction decreases. As a result, the reactor neutron flux increases. Provide explanation.

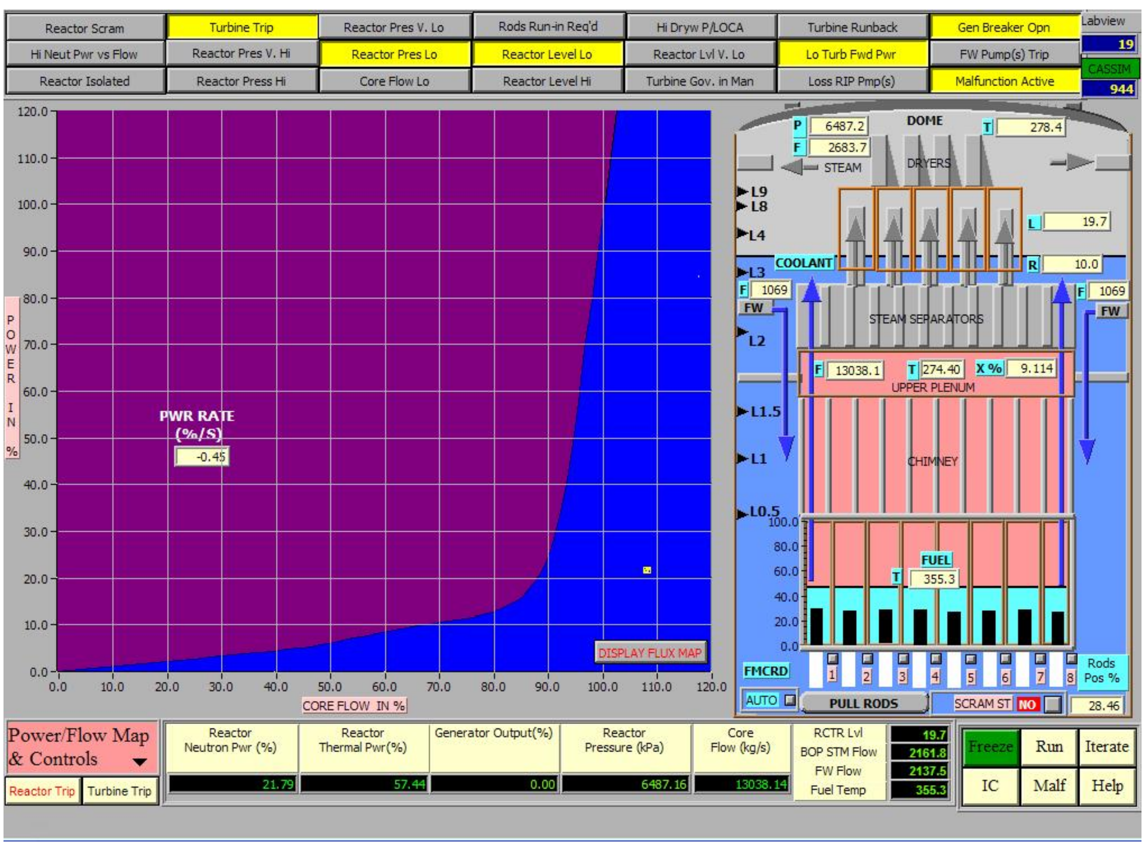
- As soon as the reactor power is higher than the target setpoint, the Reactor Power Control system will attempt to decrease reactor power by inserting control rods.
- Note the Generator Output (%) reading at the bottom of the screen - increasing or decreasing? Provide explanation. Compare this with the Reactor Thermal power (%). What is the difference? Explain the difference.
- As the malfunction event evolves, the reactor pressure increases to a point where Isolated Condenser System (ICS) is activated. At what pressure does ICS come in ?
- Go to Passive BWR Containment Screen to observe ICS response. Note the reactor pressure begins to decrease. At what point, does the pressure increase again ?
- As this malfunction further evolves, alarm "Rods Run-in Required" will come on. Why ? Provide explanation.

(c) <u>Operator Actions</u>

- Discuss any possible operator intervention with this malfunction, taking into account the time allowed for operator to do the analysis of the root cause, plus the provisions of manual controls for turbine control valve and bypass valve




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
Labview 34
CASSIM 944
PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN NO
SCRAM NO
Reactor Pressure (KPa) 6487.16
SP (KPa) 7170.00
HOLD POWER
LIMITS
MAX 105.00
MIN 0.00
(% Pwr vs flow)
ACTUAL SETPOINT
21.53 %FP
1.3329 DEC
RCTR PWR SETPOINT
100.00 %FP
2.0000 DEC
DEMANDED POWER SETPOINT
21.75 %FP
1.3374 DEC
DEMANDED RATE SETPOINT
0.30 %FP/s
+VE
-0.0035 DEC/s
POWER ERROR
-0.04 %
--VE
-1.2698 DEC
REACTIVITY EFFECTS
REACTIVITY CHANGE (MK)
FMCRD 83.60
VOID -16.10
XENON -12.77
FUEL TEMP -4.99
MOD TEMP -42.03
TOTAL -8.21
FMCRD MK WORTH
100% IN CORE - 170 MK
100% OUT OF CORE + 119 MK
CONTROL RODS
MODE AUTO
SPEED 0.67 %/s
AVE POS 28.5 %
POWER LEVEL READINGS
NEUT PWR
21.7900 %FP
1.3383 DEC
THML PWR
57.44 %FP
PWR RATE %/s
-0.45
PWR LOG RATE
-0.02078 /s
-1.6824 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
4:18:08 PM
4:19:42 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 21.79
Reactor Thermal Pwr(%) 57.44
Generator Output(%) 0.00
Reactor Pressure (kPa) 6487.16
Core Flow (kg/s) 13038.14
RCTR Lvl 19.7
BOP STM Flow 2161.8
FW Flow 2137.5
Fuel Temp 355.3
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
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
Labview 19
CASSIM 944
POWER IN %
PWR RATE (%/S)
-0.45
DISPLAY FLUX MAP
CORE FLOW IN %
DOME
P 6487.2
F 2683.7
T 278.4
STEAM
DRYERS
L9
L8
L4
L 19.7
R 10.0
COOLANT
L3
F 1069
FW
STEAM SEPARATORS
L2
F 13038.1
T 274.40
X % 9.114
UPPER PLENUM
L1.5
L1
CHIMNEY
L0.5
FUEL
T 355.3
FMCRD
AUTO
PULL RODS
SCRAM ST NO
Rods Pos %
28.46
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 21.79
Reactor Thermal Pwr(%) 57.44
Generator Output(%) 0.00
Reactor Pressure (kPa) 6487.16
Core Flow (kg/s) 13038.14
RCTR Lvl 19.7
BOP STM Flow 2161.8
FW Flow 2137.5
Fuel Temp 355.3
Freeze
Run
Iterate
IC
Malf
Help

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

During normal operation, a spurious signal can cause one SRV to open inadvertently. The cause of inadvertent safety relief valve (SRV) opening can be attributed to malfunction of the valve or an operator-initiated opening of the SRV. The steam of this SRV is discharged in the suppression pool. If the subsequent manual closure of the SRV is not obtained, then the suppression pool temperature increases reaching the scram setpoint, finally scramming the reactor.

(a) Sequence of Events:

- Inadvertent opening of one SRV allows steam to be discharged into the suppression pool.
- The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.
- The SB&PC system senses the nuclear system pressure decrease and within a few seconds closes the Turbine Control Valves enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level.
- The turbine power will settle at a lower value than the initial value. Eventually**,** the plant automatically scrams on high suppression pool temperature, if this situation is not corrected.

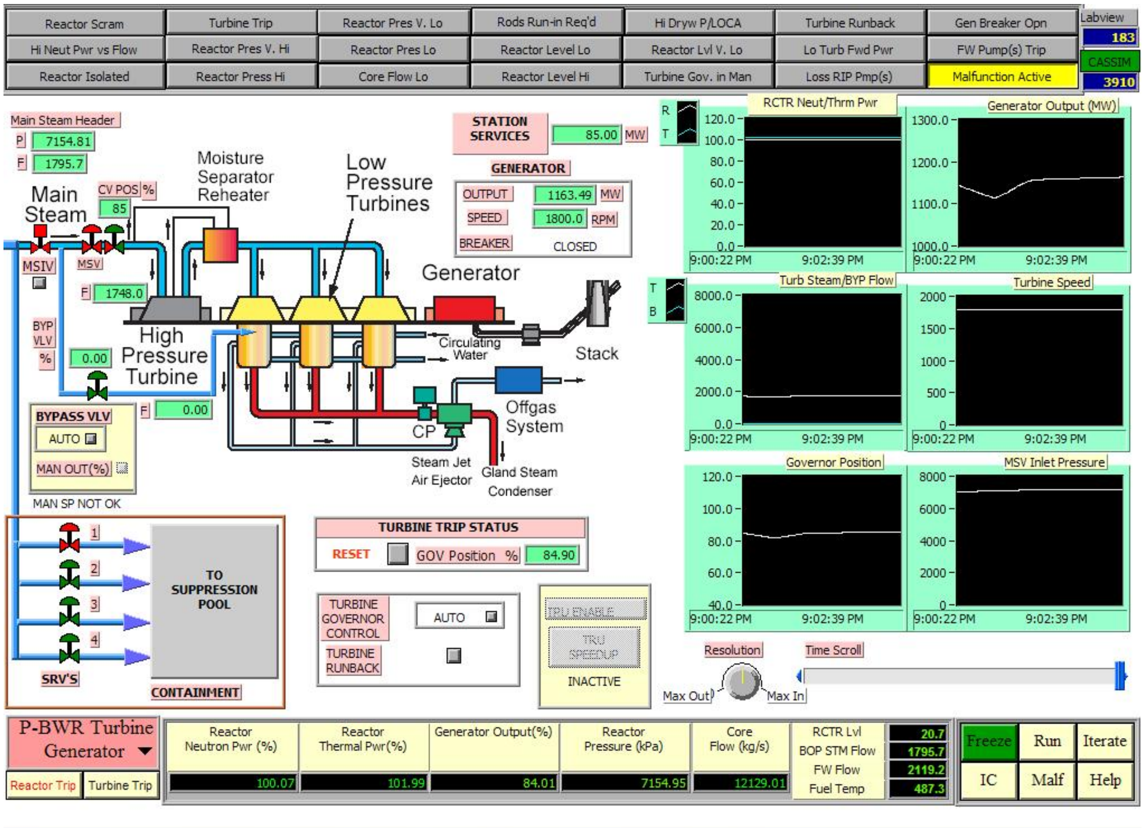
(b) Practice Transient at Simulator

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

- On the Passive BWR Turbine Generator Screen, observe that SRV #1 opens to the suppression pool, and also note that turbine runback is initiated.
- Reactor power decreases initially in responses 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.
- When the transient settles down a bit, note the following parameters: reactor pressure, generator output (%), reactor neutron power (%).
- Go to Passive BWR Containment Screen, monitor Wetwell temperature.

(c) <u>Operator Actions</u>

- Discuss any possible operator intervention with this malfunction.



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
Labview
CASSIM
Main Steam Header
Main Steam
MSIV
MSV
CV POS
Moisture Separator Reheater
Low Pressure Turbines
High Pressure Turbine
Generator
STATION SERVICES
GENERATOR
OUTPUT 1163.49 MW
SPEED 1800.0 RPM
BREAKER CLOSED
Circulating Water
Stack
Offgas System
Steam Jet Air Ejector
Gland Steam Condenser
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
TO SUPPRESSION POOL
SRV'S
CONTAINMENT
TURBINE TRIP STATUS
RESET
GOV Position % 84.90
TURBINE GOVERNOR CONTROL
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
RCTR Neut/Thrm Pwr
Generator Output (MW)
Turb Steam/BYP Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
Resolution
Time Scroll
Max Out
Max In
P-BWR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 100.07
Reactor Thermal Pwr(%) 101.99
Generator Output(%) 84.01
Reactor Pressure (kPa) 7154.95
Core Flow (kg/s) 12129.01
RCTR Lvl 20.7
BOP STM Flow 1795.7
FW Flow 2119.2
Fuel Temp 487.3
Freeze
Run
Iterate
IC
Malf
Help

5.2.8 Feedwater level control valve fails open

(a) Sequence of Events:

- With malfunction “feedwater level control valve failing open”, there will be excess FW flow.
- With excess FW flow, the water level rises to the high water level reference point (Level 8), at which time the FW pumps are tripped, the main turbine is tripped.
- Selected Control Rods Run In is activated to reduce reactor power quickly. Note: Reactor scram will occur at L9, if water level continues to rise.
- With FW flow running out, water level will drop can reach L3, scramming the reactor.
- The water level gradually drops, and can reach the Low Level reference point (Level 2), activating the ICS for long-term level control and the HP CRD system to permit a slow recovery to the desired level

(b) Practice Transient a Simulator

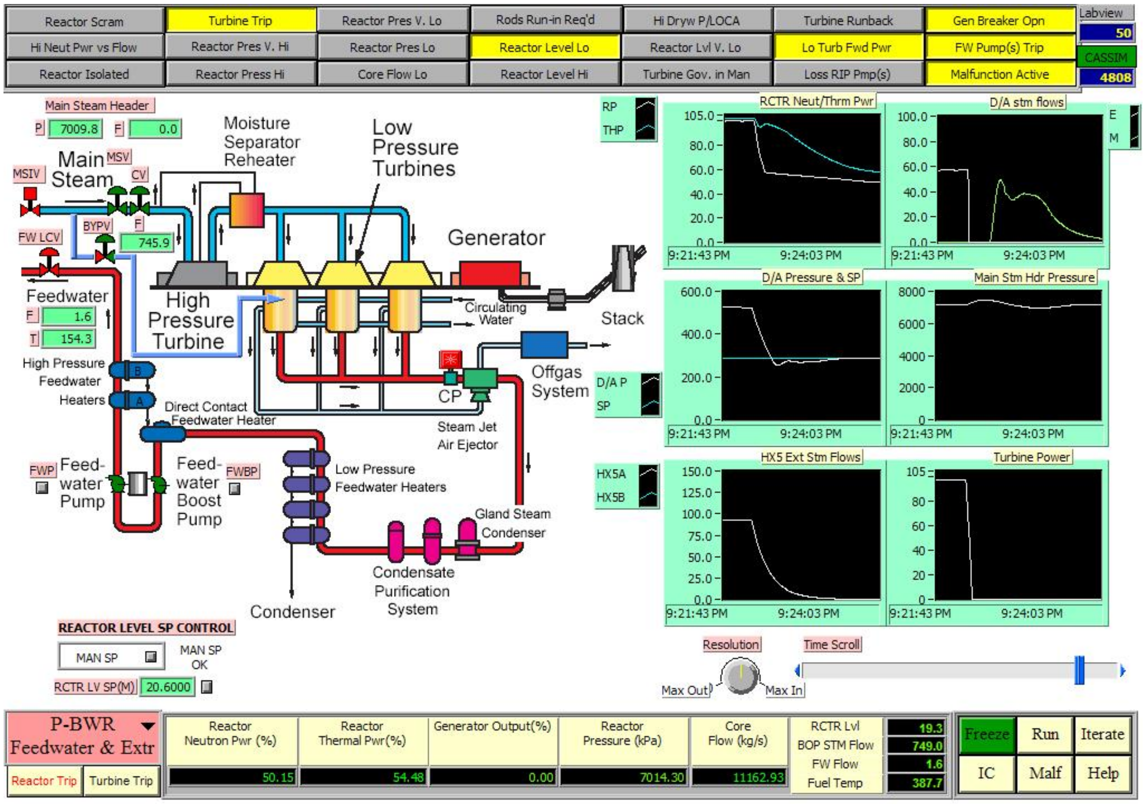
Go to Passive BWR Feedwater & Extr Steam Screen. Load the 100 % FP IC. Run the simulator. Insert the malfunction. Note that this malfunction causes the Feedwater Level Control Valve to fail open 100%. The consequence is that reactor water level will increase immediately.

.

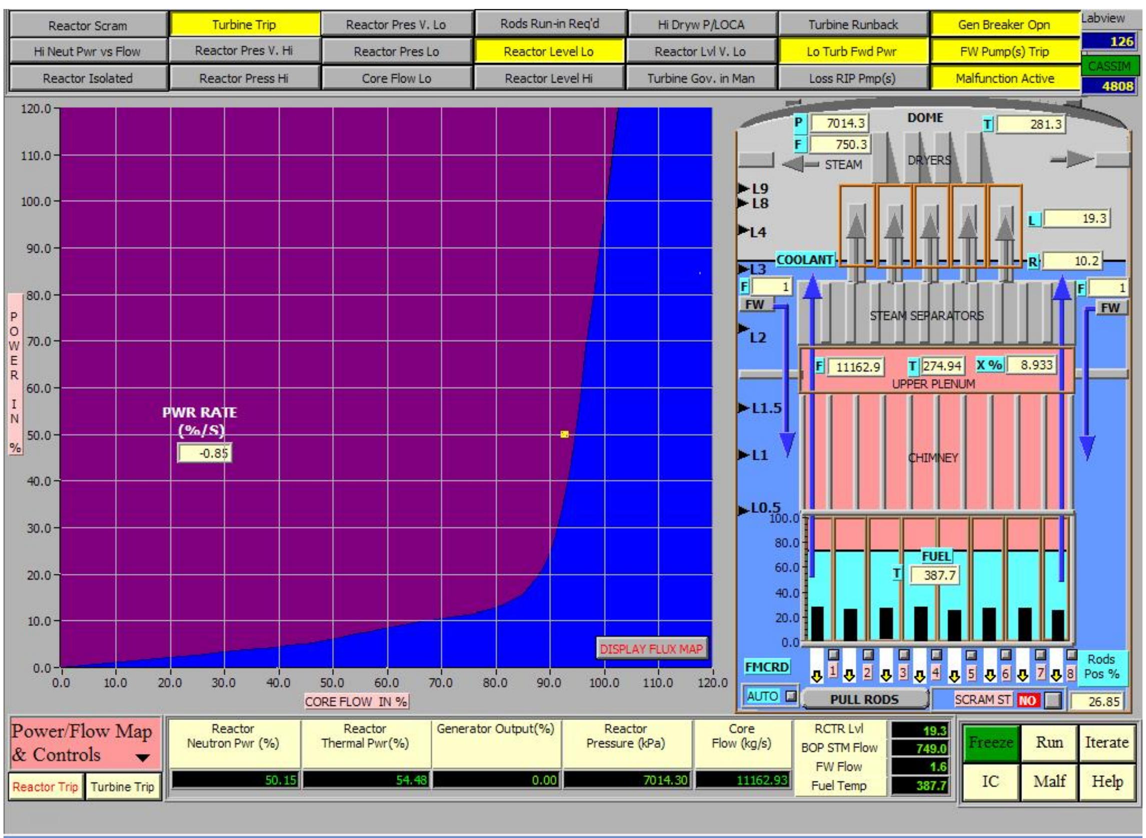
- On the Passive BWR Feedwater & Extr Steam Screen, observe that FW LCV flow is increasing.
- Go to Passive BWR Power/Flow Map Screen, note the reactor water level, reactor neutron power, reactor pressure, FW flow.
- Let the malfunction event runs for about 5 minutes. Note the reactor water level, reactor neutron power, reactor pressure, FW flow.
- Reactor water level has increased, due to the imbalance between the steam flow and the feedwater flow. Has 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, FW will also be tripped on L8, followed by Selected Control Rods Run In.
- Observe water level further decrease, and monitor protective actions.

(b) Operator Actions

Discuss any operator actions possible.



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
Labview
50
CASSIM
4808
Main Steam Header
P 7009.8 F 0.0
Moisture Separator Reheater
Low Pressure Turbines
Main Steam
MSIV
MSV
CV
BYPV
F 745.9
FW LCV
Generator
Feedwater
F 1.6
T 154.3
High Pressure Turbine
Circulating Water
Stack
High Pressure Feedwater Heaters
Direct Contact Feedwater Heater
CP
Offgas System
Steam Jet Air Ejector
FWP
Feed-water Pump
Feed-water Boost Pump
FWBP
Low Pressure Feedwater Heaters
Gland Steam Condenser
Condensate Purification System
Condenser
REACTOR LEVEL SP CONTROL
MAN SP
MAN SP OK
RCTR LV SP(M) 20.6000
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
9:21:43 PM
9:24:03 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Feedwater & Extr
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 50.15
Reactor Thermal Pwr(%) 54.48
Generator Output(%) 0.00
Reactor Pressure (kPa) 7014.30
Core Flow (kg/s) 11162.93
RCTR Lvl 19.3
BOP STM Flow 749.0
FW Flow 1.6
Fuel Temp 387.7
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
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
Labview
126
CASSIM
4808
POWER IN %
PWR RATE (%/S)
-0.85
DISPLAY FLUX MAP
CORE FLOW IN %
DOME
P 7014.3
F 750.3
T 281.3
STEAM
DRYERS
L9
L8
L4
L 19.3
COOLANT
L3
R 10.2
FW
STEAM SEPARATORS
L2
F 11162.9
T 274.94
X % 8.933
UPPER PLENUM
L1.5
L1
CHIMNEY
L0.5
FUEL
T 387.7
FMCRD
Rods Pos %
AUTO
PULL RODS
SCRAM ST NO
26.85
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 50.15
Reactor Thermal Pwr(%) 54.48
Generator Output(%) 0.00
Reactor Pressure (kPa) 7014.30
Core Flow (kg/s) 11162.93
RCTR Lvl 19.3
BOP STM Flow 749.0
FW Flow 1.6
Fuel Temp 387.7
Freeze
Run
Iterate
IC
Malf
Help

5.2.9 Turbine trip with bypass valve failed closed

A variety of turbine or nuclear system malfunctions initiate turbine trips. Some examples are high velocity separator drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum, and reactor high water level.

After the main turbine is tripped, turbine bypass valves are supposed to open in their fast opening mode by the SB&PC system. In the event that multiple failures cause all bypass valves fail to open on demand, turbine trip at high power with failure of all bypass valves produces the sequence of events as follows:

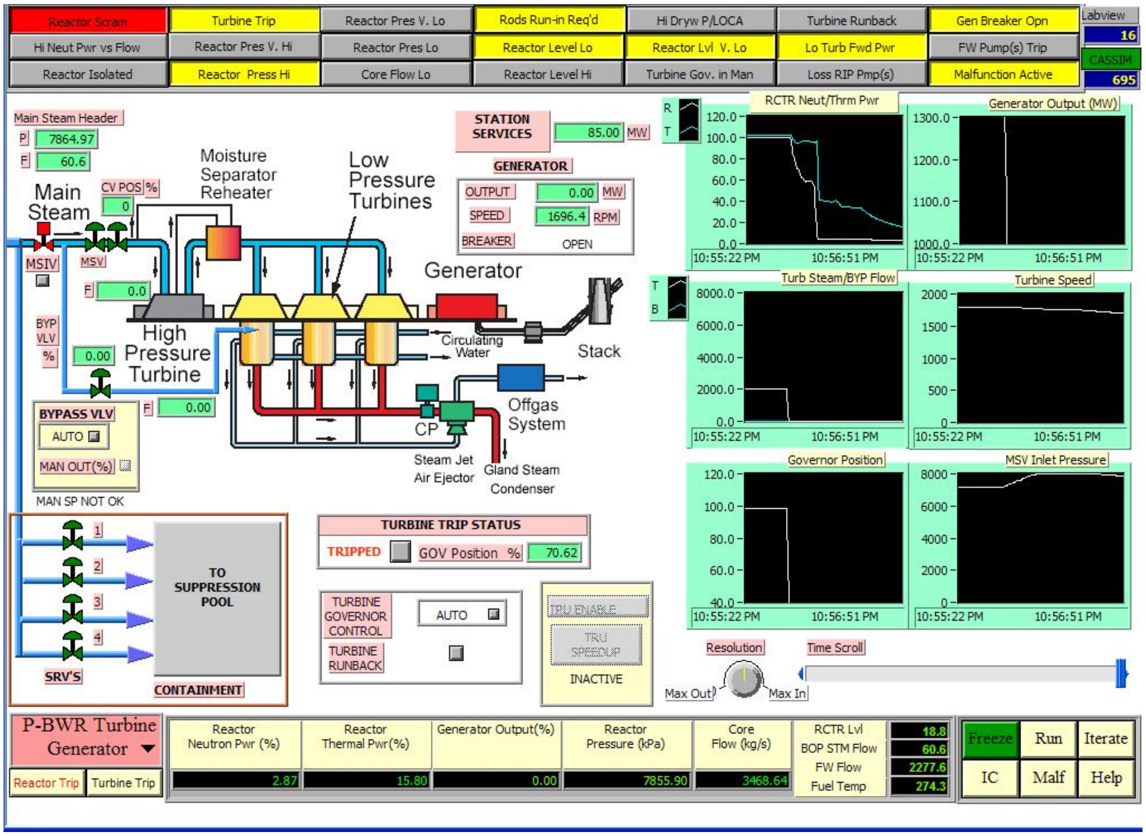
(a) Sequence of Events:

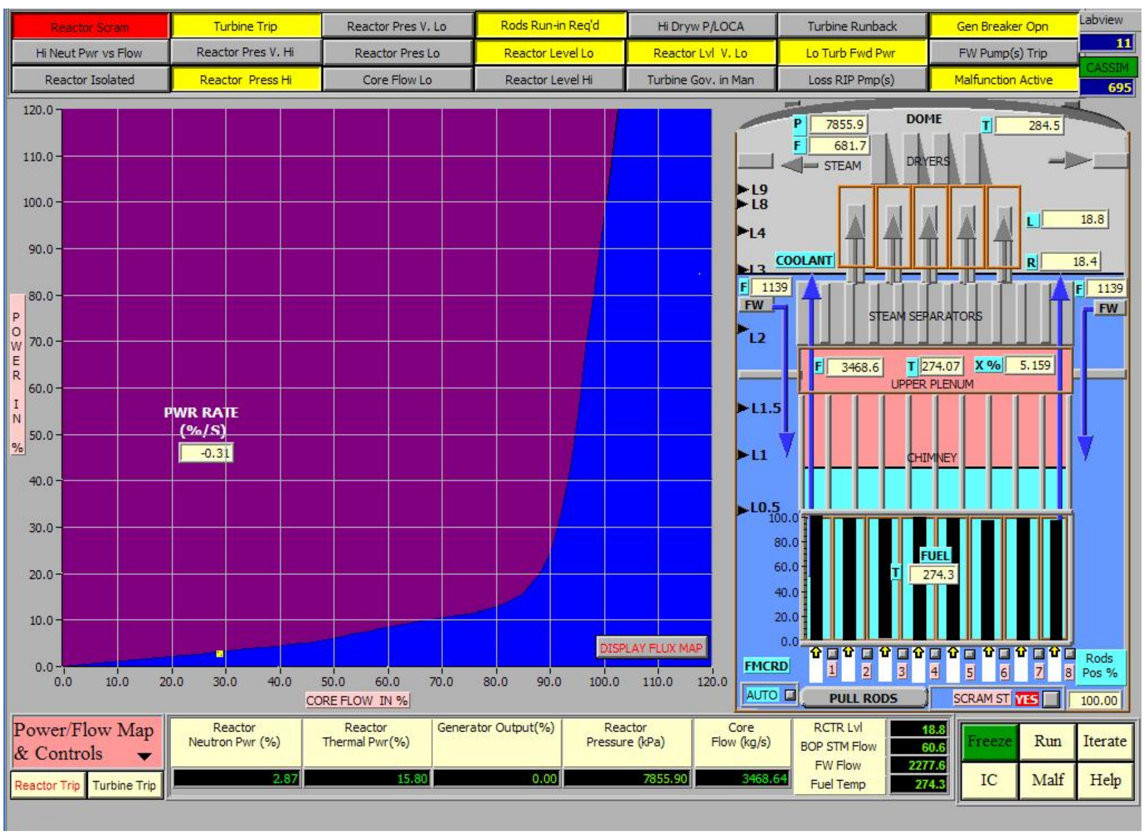
- Turbine trip initiates closure of main stop valves.
- Turbine bypass valves fail to operate.
- Reactor high pressure triggers Selected Control Rods Run In.
- Reactor power reduced.
- But reactor pressure continues to be high, activating ICS, the high pressure will scram reactor.
- Reactor pressure drops after scram and water level drops quickly to L3, due to void collapse. L3 will initiate reactor scram, but the reactor scram is already scrammed.

(b) Practice Transient at Simulator

Go to BWR Turbine Generator Screen. Load the 100% FP IC. Run the simulator. Note that this malfunction causes two failures to occur at the same time: (1) turbine trip (2) the turbine steam bypass valve failed closed. Insert this malfunction.

- 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. Is SRV opened ?
- Reactor pressure will rise rapidly, leading to “Reactor Pres. Hi” alarm, followed by Selected Control Rods Run In. At what pressure does Rods Run-in come in ?
- What is the reactor power ? What is the reactor pressure ? What safety functions are activated with high reactor pressure ? Provide explanations for how these safety functions are activated.
- Confirm the reactor scram parameter by going to the reactor 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
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
Labview
16
CASSIM
695
Main Steam Header
P 7864.97
F 60.6
Main Steam
CV POS % 0
MSIV
MSV
Moisture Separator Reheater
Low Pressure Turbines
STATION SERVICES 85.00 MW
GENERATOR
OUTPUT 0.00 MW
SPEED 1696.4 RPM
BREAKER OPEN
Generator
High Pressure Turbine
BYP VLV % 0.00
Circulating Water
Stack
Offgas System
CP
Steam Jet Air Ejector
Gland Steam Condenser
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
TO SUPPRESSION POOL
SRV'S
CONTAINMENT
TURBINE TRIP STATUS
TRIPPED
GOV Position % 70.62
TURBINE GOVERNOR CONTROL
AUTO
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
RCTR Neut/Thrm Pwr
Generator Output (MW)
Turb Steam/BYP Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
10:55:22 PM
10:56:51 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.87
Reactor Thermal Pwr(%) 15.80
Generator Output(%) 0.00
Reactor Pressure (kPa) 7855.90
Core Flow (kg/s) 3468.64
RCTR Lvl 18.8
BOP STM Flow 60.6
FW Flow 2277.6
Fuel Temp 274.3
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
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
Labview
11
CASSIM
695
POWER IN %
PWR RATE (%/S)
-0.31
DISPLAY FLUX MAP
CORE FLOW IN %
DOME
P 7855.9
F 681.7
T 284.5
STEAM
DRYERS
L 18.8
R 18.4
COOLANT
F 1139
FW
STEAM SEPARATORS
F 3468.6
T 274.07
X % 5.159
UPPER PLENUM
CHIMNEY
FUEL
T 274.3
FMCRD
AUTO
PULL RODS
SCRAM ST YES
Rods Pos %
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.87
Reactor Thermal Pwr(%) 15.80
Generator Output(%) 0.00
Reactor Pressure (kPa) 7855.90
Core Flow (kg/s) 3468.64
RCTR Lvl 18.8
BOP STM Flow 60.6
FW Flow 2277.6
Fuel Temp 274.3
Freeze
Run
Iterate
IC
Malf
Help

5.2.10 Inadvertent withdrawal of one bank of rods

The causes of an inadvertent withdrawal of one bank of rods are either:

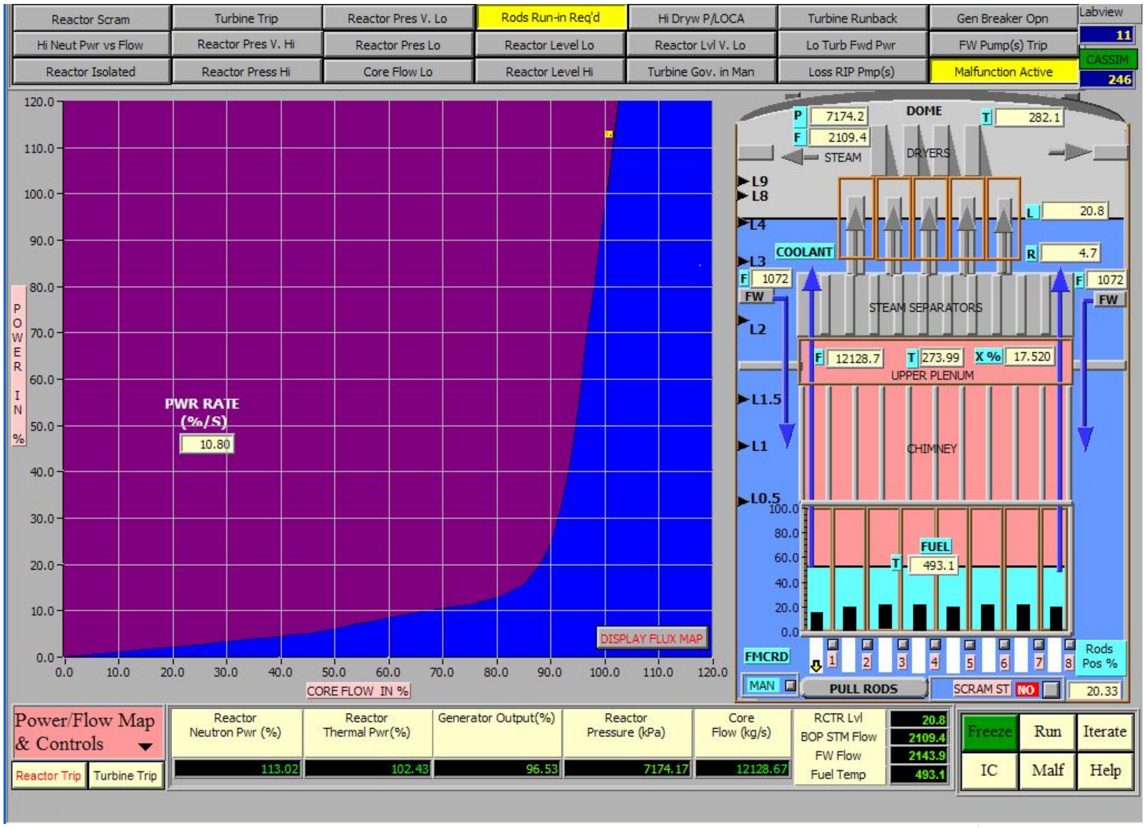
(1) A procedural error by the operator in which a bank of control rods is withdrawn continuously, or
(2) A malfunction of the automated rod withdrawal sequence control logic during automated operation in which a bank of control rods is withdrawn continuously.

(a) Practice Transient at Simulator

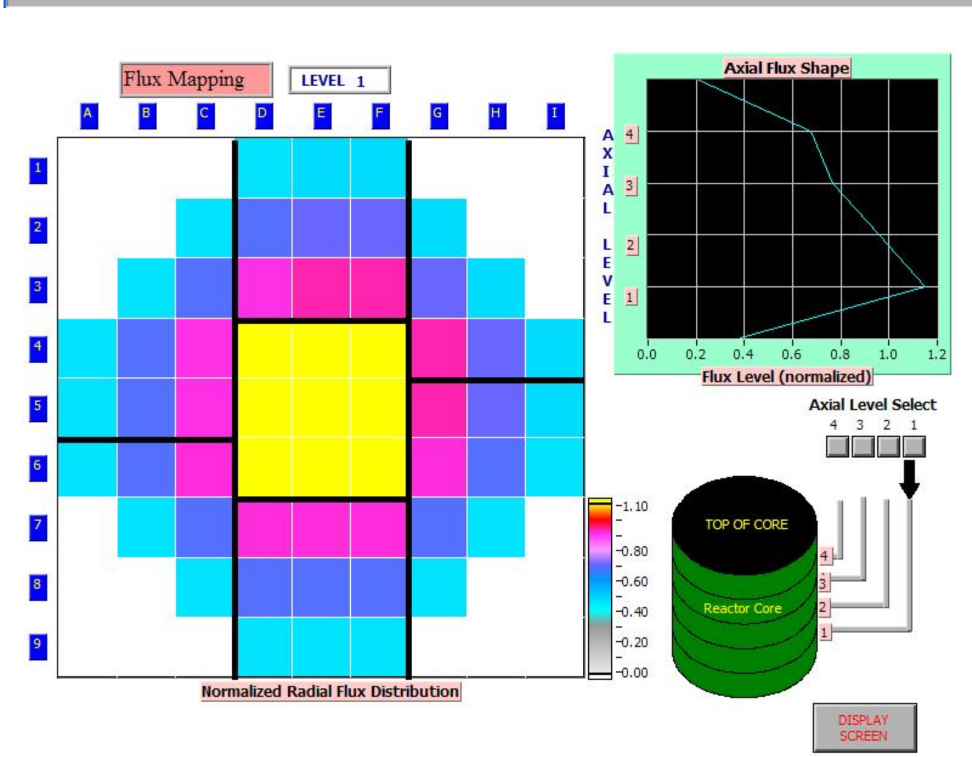
Go to Passive BWR Power/Flow Map Screen. Load the 100% FP IC. Run the simulator. Note that this malfunction causes inadvertent withdrawal of control rods Bank #1. The consequence is that the reactor will suddenly have positive reactivity change due to the withdrawal of Bank #1 control rods.

Insert this malfunction. When this malfunction transient occurs:

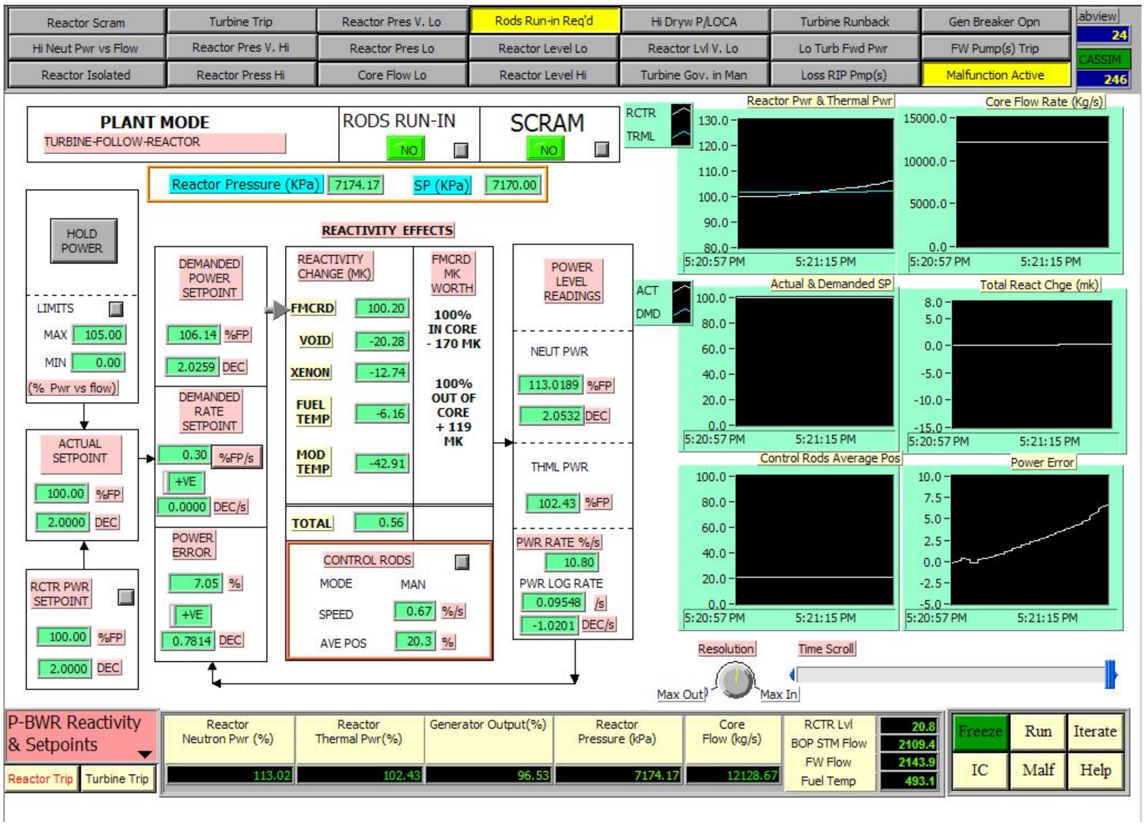
- On the Passive BWR Power/Flow Map Screen, observe the movement of the Control Rods Bank #1.
- Note the arrow pointer for control rods Bank #1 and note its position in core. Observe that reactor power is increasing.
- In response to the reactor power increase, what would the reactor power control system 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 sudden increase in reactor power – Selected Control Rods Run In activated.
- 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.



Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
11
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
246
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
POWER IN %
PWR RATE (%/S)
10.80
DISPLAY FLUX MAP
CORE FLOW IN %
DOME
P 7174.2
F 2109.4
T 282.1
STEAM
DRYERS
L9
L8
L4
L 20.8
COOLANT
R 4.7
L3
F 1072
FW
STEAM SEPARATORS
L2
F 12128.7
T 273.99
X % 17.520
UPPER PLENUM
L1.5
L1
CHIMNEY
L0.5
FUEL
T 493.1
FMCRD
Rods Pos %
MAN
PULL RODS
SCRAM ST NO
20.33
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
113.02
Reactor Thermal Pwr(%)
102.43
Generator Output(%)
96.53
Reactor Pressure (kPa)
7174.17
Core Flow (kg/s)
12128.67
RCTR Lvl 20.8
BOP STM Flow 2109.4
FW Flow 2143.9
Fuel Temp 493.1
Freeze
Run
Iterate
IC
Malf
Help



Flux Mapping
LEVEL 1
A B C D E F G H I
1 2 3 4 5 6 7 8 9
Normalized Radial Flux Distribution
Axial Flux Shape
AXIAL LEVEL
Flux Level (normalized)
Axial Level Select
4 3 2 1
TOP OF CORE
Reactor Core
DISPLAY SCREEN



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
PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN
SCRAM
Reactor Pressure (KPa) 7174.17
SP (KPa) 7170.00
REACTIVITY EFFECTS
HOLD POWER
LIMITS
MAX 105.00
MIN 0.00
(% Pwr vs flow)
DEMANDED POWER SETPOINT
106.14 %FP
2.0259 DEC
DEMANDED RATE SETPOINT
0.30 %FP/s
+VE
0.0000 DEC/s
POWER ERROR
7.05 %
+VE
0.7814 DEC
ACTUAL SETPOINT
100.00 %FP
2.0000 DEC
RCTR PWR SETPOINT
100.00 %FP
2.0000 DEC
REACTIVITY CHANGE (MK)
FMCRD 100.20
VOID -20.28
XENON -12.74
FUEL TEMP -6.16
MOD TEMP -42.91
TOTAL 0.56
FMCRD MK WORTH
100% IN CORE - 170 MK
100% OUT OF CORE + 119 MK
CONTROL RODS
MODE MAN
SPEED 0.67 %/s
AVE POS 20.3 %
POWER LEVEL READINGS
NEUT PWR
113.0189 %FP
2.0532 DEC
THML PWR
102.43 %FP
PWR RATE %/s
10.80
PWR LOG RATE
0.09548 /s
-1.0201 DEC/s
Reactor Pwr & Thermal Pwr
Core Flow Rate (Kg/s)
Actual & Demanded SP
Total React Chge (mk)
Control Rods Average Pos
Power Error
Resolution
Time Scroll
Max Out
Max In
P-BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 113.02
Reactor Thermal Pwr(%) 102.43
Generator Output(%) 96.53
Reactor Pressure (kPa) 7174.17
Core Flow (kg/s) 12128.67
RCTR Lvl 20.8
BOP STM Flow 2109.4
FW Flow 2143.9
Fuel Temp 493.1
Freeze
Run
Iterate
IC
Malf
Help

5.2.11 Inadvertent insertion of one bank of rods

The causes of an inadvertent insertion of one bank of rods are either:

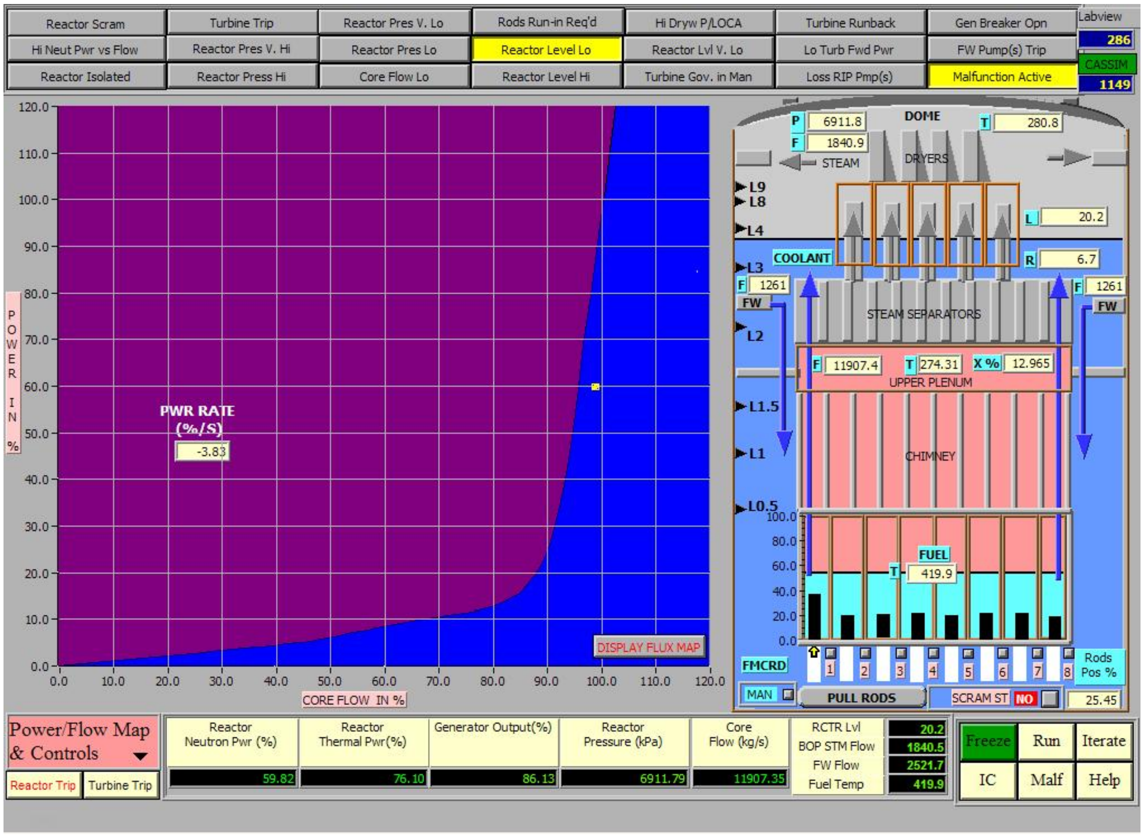
- A procedural error by the operator in which a bank of control rods is inserted continuously, or
- A malfunction of the automated rod insertion sequence control logic during automated operation in which a bank of control rods is inserted continuously.

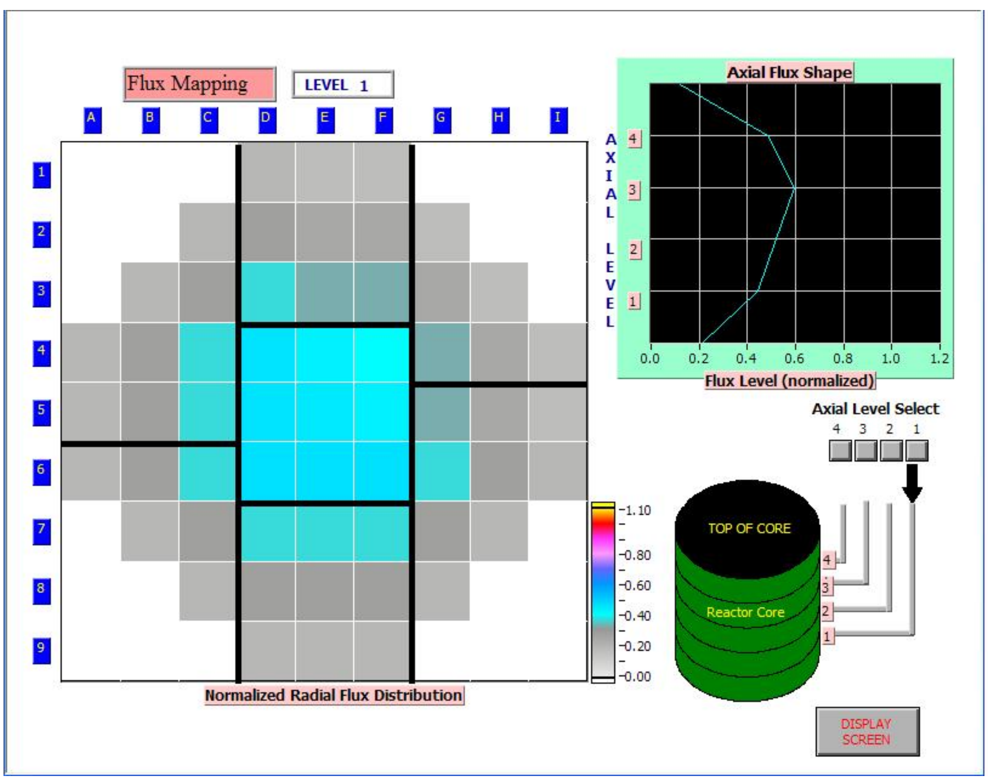
(a) Practice Transient at Simulator

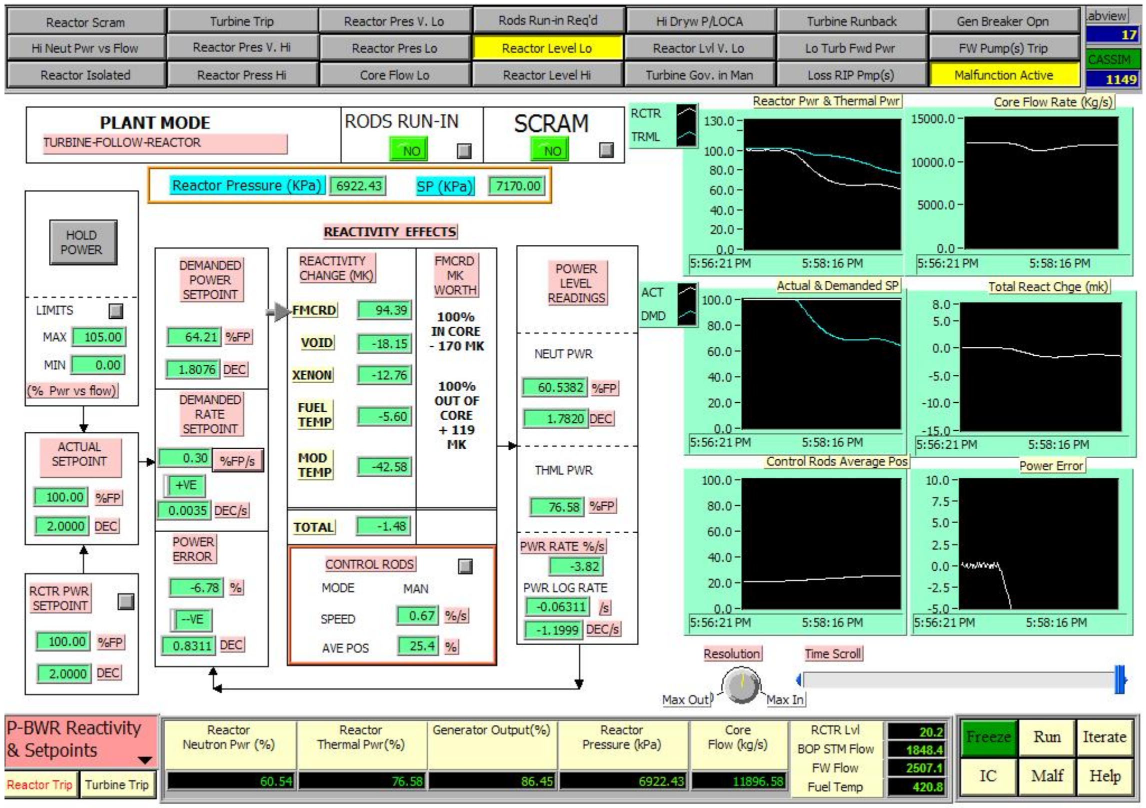
Go to Power/Flow Map Screen. Load the 100% FP IC. Run the simulator. Note that this malfunction causes inadvertent insertion of control rods Bank #1. The consequence is that the reactor will suddenly have negative reactivity change due to the insertion of Bank #1 control rods.

Insert this malfunction. When this malfunction transient occurs:

- On the Power/Flow Map Screen, observe the movement of the control rods.
- Note the arrow pointer for control rods Bank #1 and note its position in 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 when compared with previous measurement.
- The reactor power control system recognizes that increasing the coolant flow rate is unable to maintain 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 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, turbine control system will runback turbine in order to restore main steam pressure.


Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Drywl P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
286
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
1149
POWER IN %
PWR RATE (%/S)
-3.83
CORE FLOW IN %
DISPLAY FLUX MAP
DOME
P 6911.8
F 1840.9
T 280.8
STEAM
DRYERS
L9
L8
L4
L 20.2
L3
COOLANT
R 6.7
F 1261
FW
STEAM SEPARATORS
L2
F 11907.4
T 274.31
X % 12.965
UPPER PLENUM
L1.5
L1
CHIMNEY
L0.5
FUEL
T 419.9
FMCRD
Rods Pos %
MAN
PULL RODS
SCRAM ST NO
25.45
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
59.82
Reactor Thermal Pwr(%)
76.10
Generator Output(%)
86.13
Reactor Pressure (kPa)
6911.79
Core Flow (kg/s)
11907.35
RCTR Lvl 20.2
BOP STM Flow 1840.5
FW Flow 2521.7
Fuel Temp 419.9
Freeze
Run
Iterate
IC
Malf
Help


Flux Mapping
LEVEL 1
A B C D E F G H I
1 2 3 4 5 6 7 8 9
Normalized Radial Flux Distribution
Axial Flux Shape
AXIAL LEVEL
Flux Level (normalized)
Axial Level Select
4 3 2 1
TOP OF CORE
Reactor Core
DISPLAY SCREEN


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
Labview 17
CASSIM 1149
PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN NO
SCRAM NO
Reactor Pressure (KPa) 6922.43
SP (KPa) 7170.00
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
64.21 %FP
1.8076 DEC
DEMANDED RATE SETPOINT
0.30 %FP/s
+VE
0.0035 DEC/s
POWER ERROR
-6.78 %
--VE
0.8311 DEC
REACTIVITY EFFECTS
REACTIVITY CHANGE (MK)
FMCRD 94.39
VOID -18.15
XENON -12.76
FUEL TEMP -5.60
MOD TEMP -42.58
TOTAL -1.48
FMCRD MK WORTH
100% IN CORE - 170 MK
100% OUT OF CORE + 119 MK
CONTROL RODS
MODE MAN
SPEED 0.67 %/s
AVE POS 25.4 %
POWER LEVEL READINGS
NEUT PWR
60.5382 %FP
1.7820 DEC
THML PWR
76.58 %FP
PWR RATE %/s
-3.82
PWR LOG RATE
-0.06311 /s
-1.1999 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
5:56:21 PM
5:58:16 PM
Resolution
Max Out
Max In
Time Scroll
P-BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 60.54
Reactor Thermal Pwr(%) 76.58
Generator Output(%) 86.45
Reactor Pressure (kPa) 6922.43
Core Flow (kg/s) 11896.58
RCTR Lvl 20.2
BOP STM Flow 1848.4
FW Flow 2507.1
Fuel Temp 420.8
Freeze
Run
Iterate
IC
Malf
Help

5.2.12 Inadvertent reactor isolation

Various steamline and nuclear system malfunctions, or operator actions, can initiate MSIV closure. Examples are:

- Manual action (purposely or inadvertent);
- Spurious signals such as low pressure, low reactor water level, low condenser vacuum;
- Equipment malfunctions, such as faulty valves or operating mechanisms.

A closure of one MSIV may cause an immediate closure of all other MSIVs, depending on reactor conditions.

(a) <u>Sequence of Events</u>:

- MSIV closure initiates a reactor scram trip via position signals to the RPS.
- The same signal also initiates the operation of isolation condensers, which prevent the lifting of SRVs.
- Closure of MSIV causes the dome pressure to increase.
- The isolation condenser operation terminates the pressure increase.
- The anticipatory scram prevents any change in the thermal margins.

(b) <u>Practice Transient at Simulator</u>

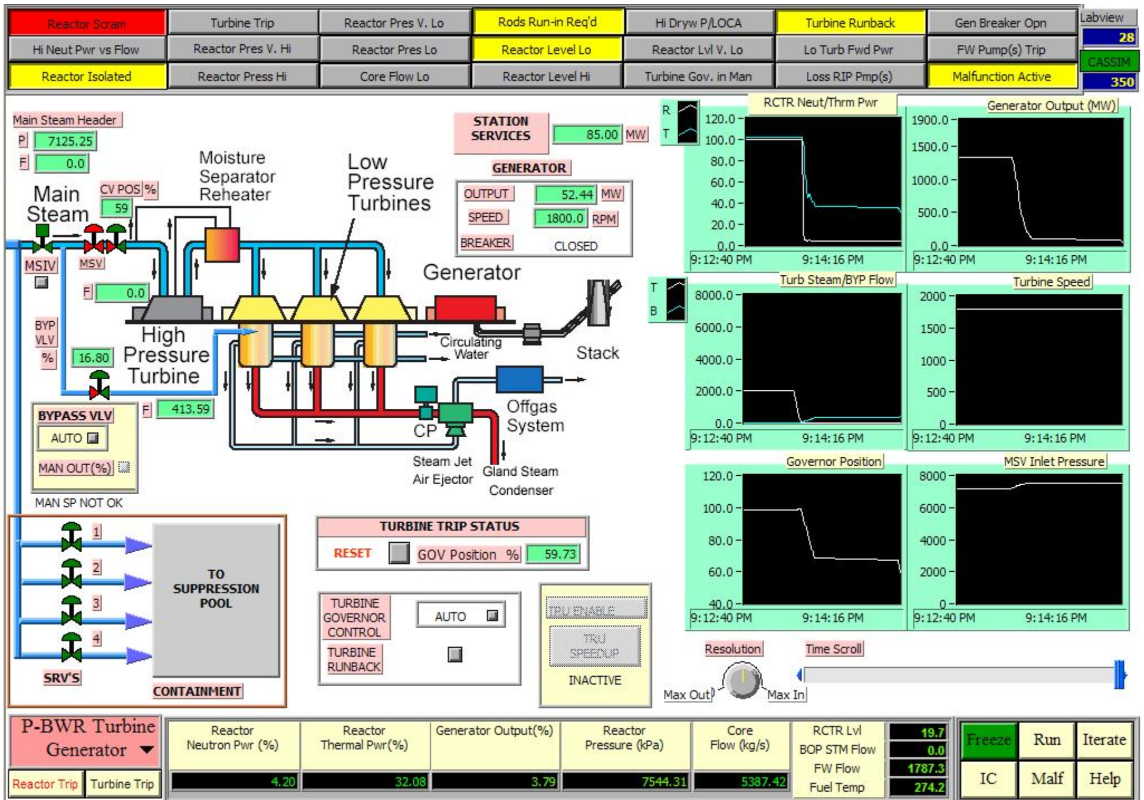
Go to Passive BWR Plant Overview Screen. Load the 100% FP IC and run the simulator. Note that this malfunction causes inadvertent closing of the reactor vessel isolation valve. The consequence is that the reactor vessel steam supply is isolated from the turbine generator.

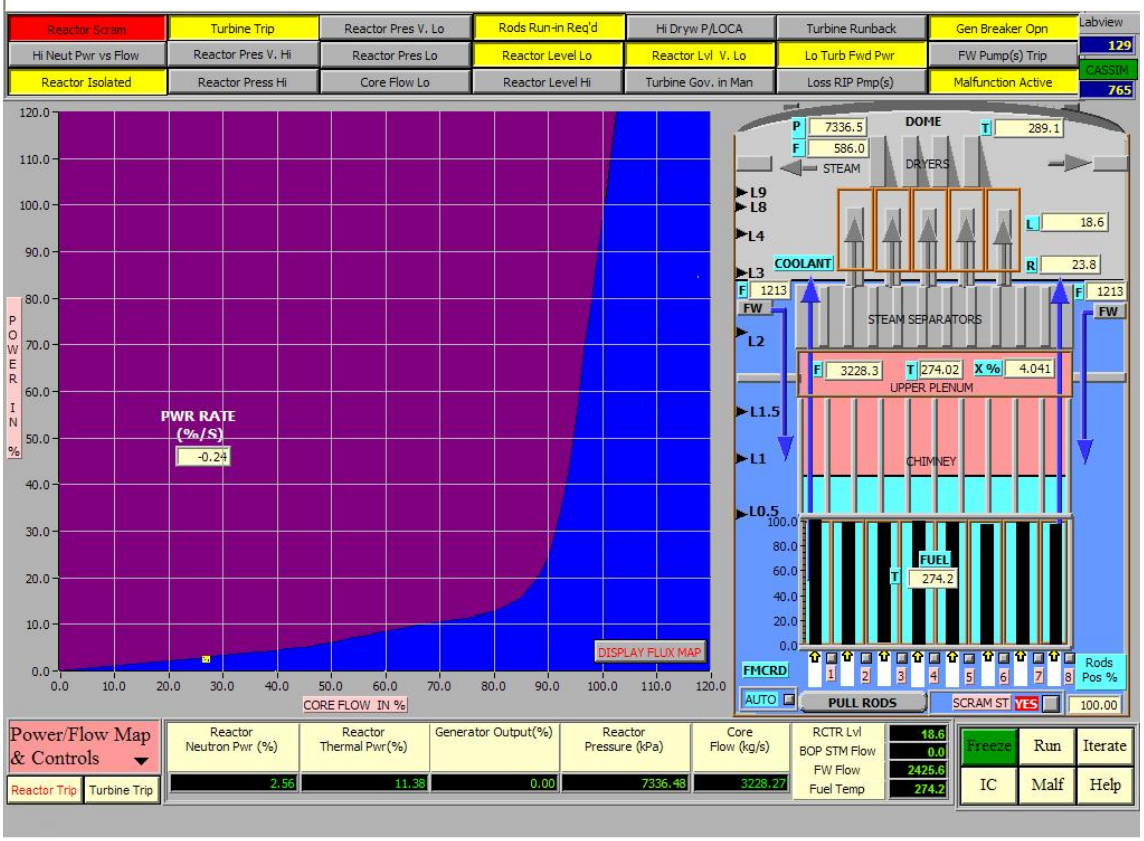
Insert the malfunction. When this malfunction transient occurs:

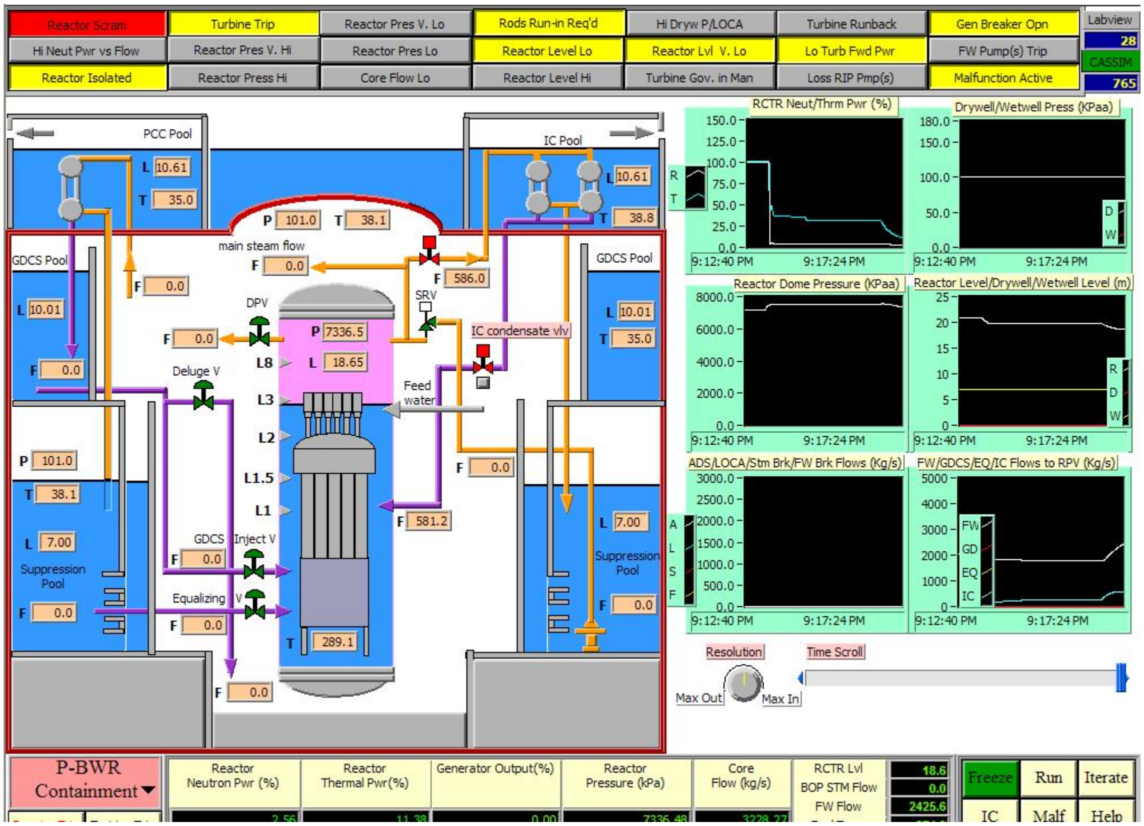
- On the Passive BWR Plant Overview Screen, observe the status of the reactor isolation valve.
- Note the steam flow from the reactor dome. Observe the status of SRV to suppression pool.
- At the bottom of the screen, note the following parameters, as the event evolves:

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

- As the reactor isolation valve closes, the reactor pressure increases rapidly. Monitor pressure rise.
- When the reactor isolation valve is fully closed, the reactor is scrammed due to reactor isolation. Confirm this by reviewing the Passive BWR scram parameter screen.
- 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.
- What safety function is activated ? Provide explanation. Monitor reactor pressure. When is pressure increase terminated ?


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
Labview
28
CASSIM
350
Main Steam Header
P 7125.25
F 0.0
Main Steam
CV POS % 59
MSIV
MSV
Moisture Separator Reheater
Low Pressure Turbines
High Pressure Turbine
Generator
STATION SERVICES 85.00 MW
GENERATOR
OUTPUT 52.44 MW
SPEED 1800.0 RPM
BREAKER CLOSED
BYP VLV % 16.80
F 413.59
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
Circulating Water
Stack
CP
Offgas System
Steam Jet Air Ejector
Gland Steam Condenser
TO SUPPRESSION POOL
SRV'S
CONTAINMENT
TURBINE TRIP STATUS
RESET
GOV Position % 59.73
TURBINE GOVERNOR CONTROL
AUTO
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
RCTR Neut/Thrm Pwr
Generator Output (MW)
Turb Steam/BYP Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
9:12:40 PM
9:14:16 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 4.20
Reactor Thermal Pwr(%) 32.08
Generator Output(%) 3.79
Reactor Pressure (kPa) 7544.31
Core Flow (kg/s) 5387.42
RCTR Lvl 19.7
BOP STM Flow 0.0
FW Flow 1787.3
Fuel Temp 274.2
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
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
Labview
129
CASSIM
765
POWER IN %
PWR RATE (%/S)
-0.24
DISPLAY FLUX MAP
CORE FLOW IN %
P 7336.5
F 586.0
DOME
T 289.1
STEAM
DRYERS
L9
L8
L4
L3
L2
L1.5
L1
L0.5
L 18.6
R 23.8
COOLANT
F 1213
FW
STEAM SEPARATORS
F 3228.3
T 274.02
X % 4.041
UPPER PLENUM
CHIMNEY
FUEL
T 274.2
FMCRD
Rods Pos %
AUTO
PULL RODS
SCRAM ST YES
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.56
Reactor Thermal Pwr(%) 11.38
Generator Output(%) 0.00
Reactor Pressure (kPa) 7336.48
Core Flow (kg/s) 3228.27
RCTR Lvl 18.6
BOP STM Flow 0.0
FW Flow 2425.6
Fuel Temp 274.2
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
28
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
765
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
IC condensate vlv
Deluge V
Feed water
GDCS
Inject V
Equalizing V
Suppression Pool
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
9:12:40 PM
9:17:24 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Containment
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
RCTR Lvl
BOP STM Flow
FW Flow
Freeze
Run
Iterate
IC
Malf
Help

5.2.13 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) Description of Events

The passive BWR plant is designed such that no single operator error or equipment failure shall cause a loss of up to 55.6 ° C (100 ° F) of the feedwater heating capability of the plant. The loss of FW heating up to 55.6 ° C will cause an increase in core inlet subcooling, and core power increase 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 Δ T 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.

(b) Practice Transient at Simulator

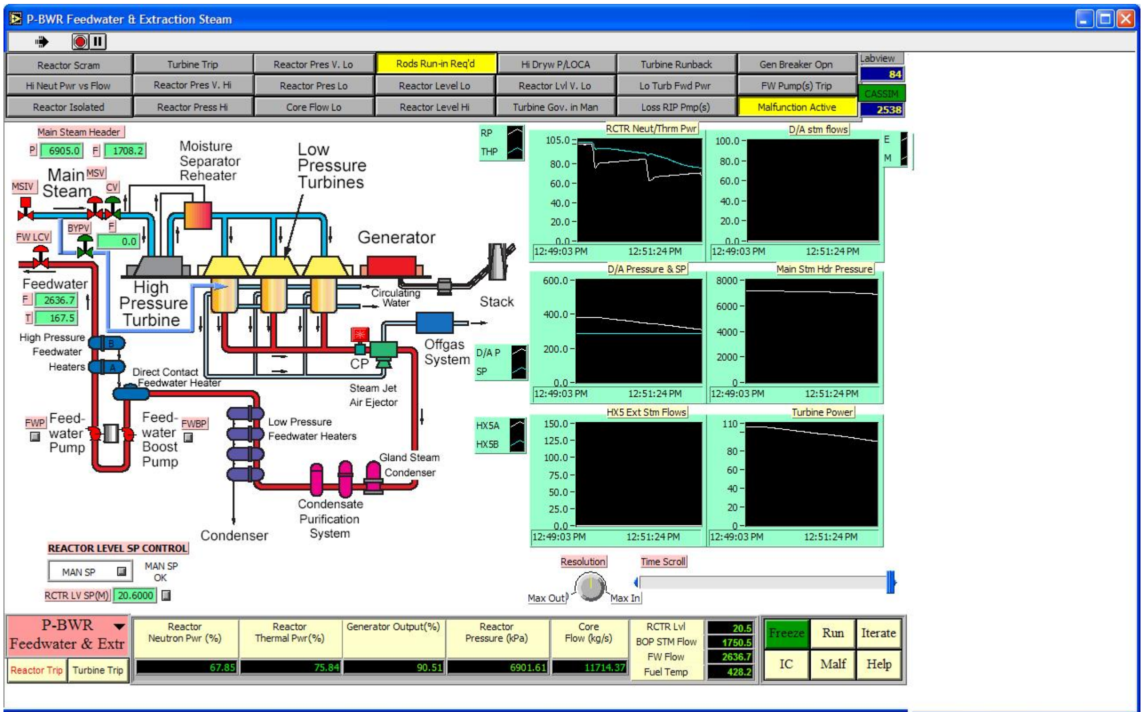
Load the 100% FP IC. Run the simulator.

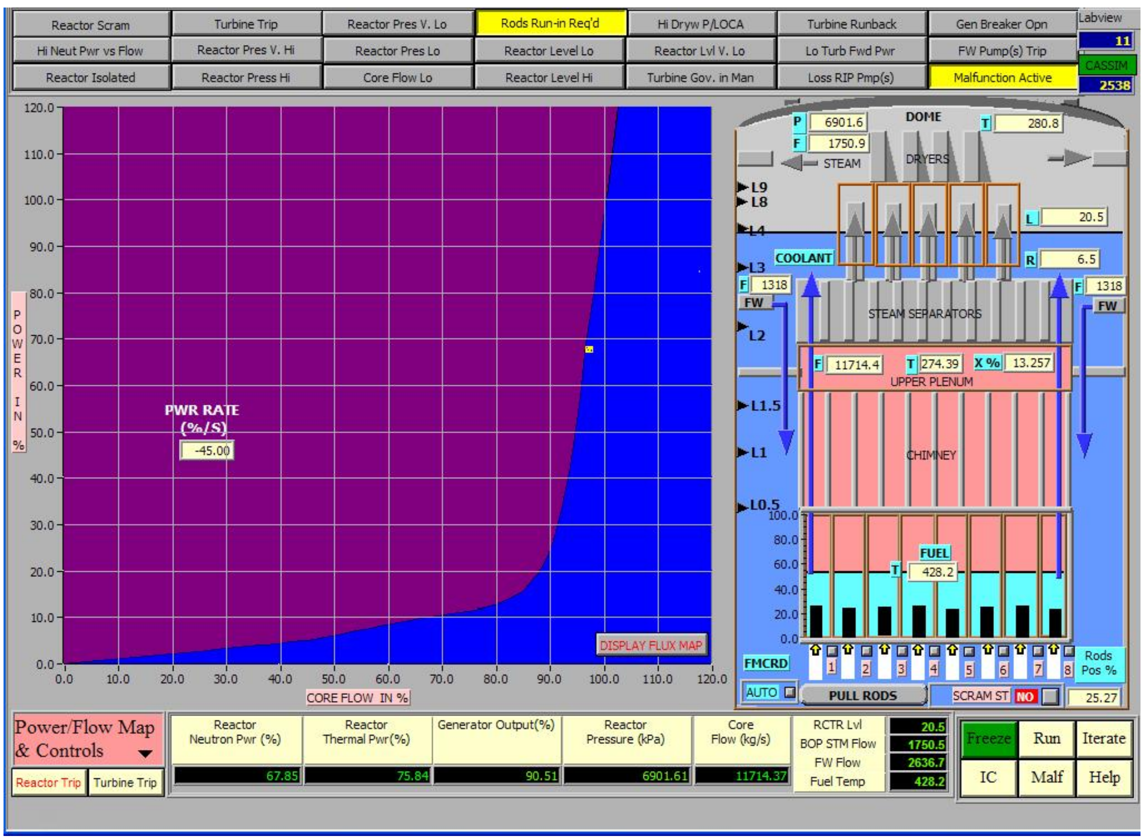
- Go to Passive BWR Feedwater & Extr Steam Screen. Record the extraction steam flows to Deaerator and HP heaters. Record the feedwater temperature going to the reactor.
- Insert the above malfunction. This malfunction causes all the extraction steam isolation valves for FW Heaters to close. The consequence is total loss of feedwater heating.

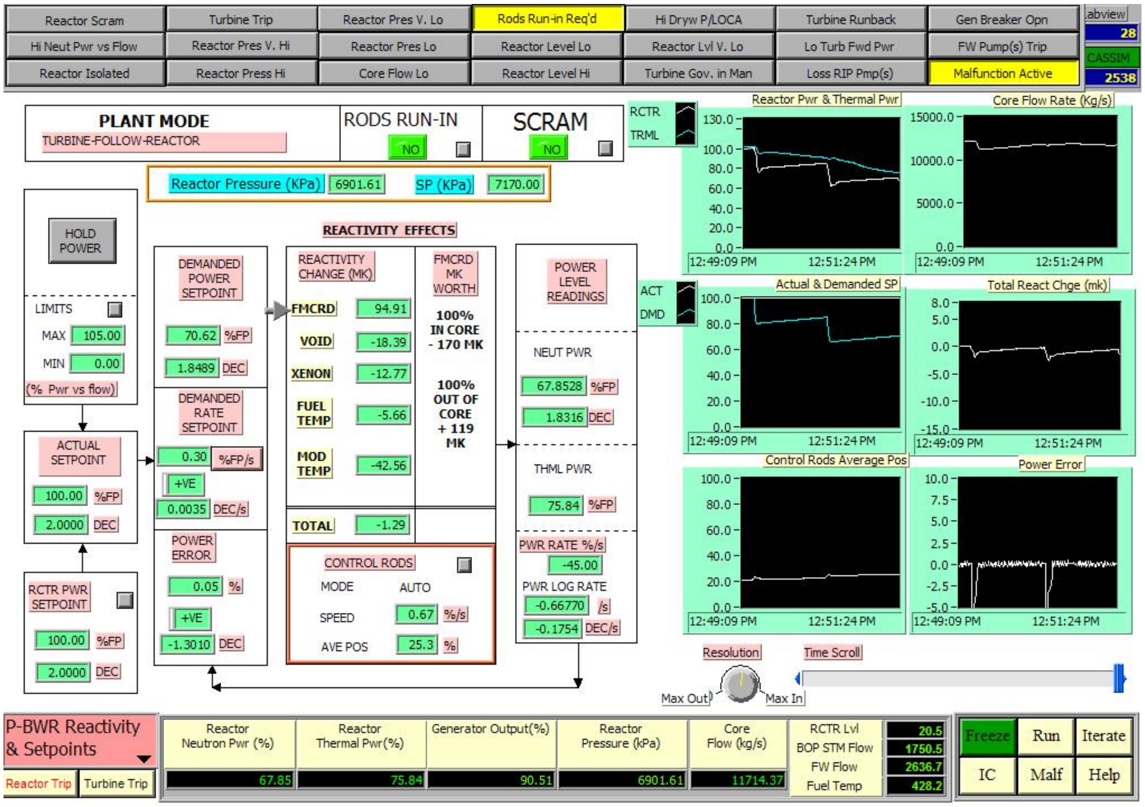
- Record the following parameters as the malfunction event evolves:

	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 on turbine. Hence the generator output (%) increases.
- Does the reactor power change at this time?
- How much does the feedwater temperature drop since the malfunction event begins?
- 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 the malfunction event runs for 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 and observe 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. Record the equilibrium reactor power and generator power.


P-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
84
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
2538
Main Steam Header
P 6905.0
F 1708.2
Moisture Separator Reheater
Low Pressure Turbines
Main Steam
MSIV
MSV
CV
Generator
FW LCV
BYPV
0.0
Feedwater
F 2636.7
T 167.5
High Pressure Turbine
Circulating Water
Stack
High Pressure Feedwater Heaters
Offgas System
CP
Direct Contact Feedwater Heater
Steam Jet Air Ejector
FWP
Feed-water Pump
Feed-water Boost Pump
FWBP
Low Pressure Feedwater Heaters
Gland Steam Condenser
Condensate Purification System
Condenser
REACTOR LEVEL SP CONTROL
MAN SP
MAN SP OK
RCTR LV SP(M) 20.6000
RCTR Neut/Thrm Pwr
D/A stm flows
D/A Pressure & SP
Main Stm Hdr Pressure
HX5 Ext Stm Flows
Turbine Power
12:49:03 PM
12:51:24 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Feedwater & Extr
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
67.85
Reactor Thermal Pwr(%)
75.84
Generator Output(%)
90.51
Reactor Pressure (kPa)
6901.61
Core Flow (kg/s)
11714.37
RCTR Lvl 20.5
BOP STM Flow 1750.5
FW Flow 2636.7
Fuel Temp 428.2
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
11
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
2538
POWER IN %
PWR RATE (%/S)
-45.00
DISPLAY FLUX MAP
CORE FLOW IN %
P 6901.6
F 1750.9
DOME
T 280.8
STEAM
DRYERS
L9
L8
L4
L 20.5
COOLANT
R 6.5
L3
F 1318
FW
STEAM SEPARATORS
L2
F 11714.4
T 274.39
X % 13.257
UPPER PLENUM
L1.5
L1
CHIMNEY
L0.5
FUEL
T 428.2
FMCRD
Rods Pos %
AUTO
PULL RODS
SCRAM ST NO
25.27
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%)
67.85
Reactor Thermal Pwr(%)
75.84
Generator Output(%)
90.51
Reactor Pressure (kPa)
6901.61
Core Flow (kg/s)
11714.37
RCTR Lvl 20.5
BOP STM Flow 1750.5
FW Flow 2636.7
Fuel Temp 428.2
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
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
PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN
NO
SCRAM
NO
Reactor Pressure (KPa) 6901.61
SP (KPa) 7170.00
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
70.62 %FP
1.8489 DEC
DEMANDED RATE SETPOINT
0.30 %FP/s
+VE
0.0035 DEC/s
POWER ERROR
0.05 %
+VE
-1.3010 DEC
REACTIVITY EFFECTS
REACTIVITY CHANGE (MK)
FMCRD 94.91
VOID -18.39
XENON -12.77
FUEL TEMP -5.66
MOD TEMP -42.56
TOTAL -1.29
FMCRD MK WORTH
100% IN CORE - 170 MK
100% OUT OF CORE + 119 MK
CONTROL RODS
MODE AUTO
SPEED 0.67 %/s
AVE POS 25.3 %
POWER LEVEL READINGS
NEUT PWR
67.8528 %FP
1.8316 DEC
THML PWR
75.84 %FP
PWR RATE %/s
-45.00
PWR LOG RATE
-0.66770 /s
-0.1754 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
12:49:09 PM
12:51:24 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 67.85
Reactor Thermal Pwr(%) 75.84
Generator Output(%) 90.51
Reactor Pressure (kPa) 6901.61
Core Flow (kg/s) 11714.37
RCTR Lvl 20.5
BOP STM Flow 1750.5
FW Flow 2636.7
Fuel Temp 428.2
Freeze
Run
Iterate
IC
Malf
Help

5.2.14 Loss of Condenser Vacuum

(a) Description of Event

Various system malfunctions can cause a loss of condenser vacuum due to some single equipment failure (e.g. condenser vacuum pump trip).

The loss of condenser vacuum initially does not affect the vessel. When the turbine trip setpoint for condenser vacuum is reached, turbine trip will be active to close main turbine control valve, simultaneously with a bypass valve opening. Six seconds later, the low vacuum setpoint produces closure of the bypass valve. With a small delay, the MSIV also closes, with reactor scram initiated.

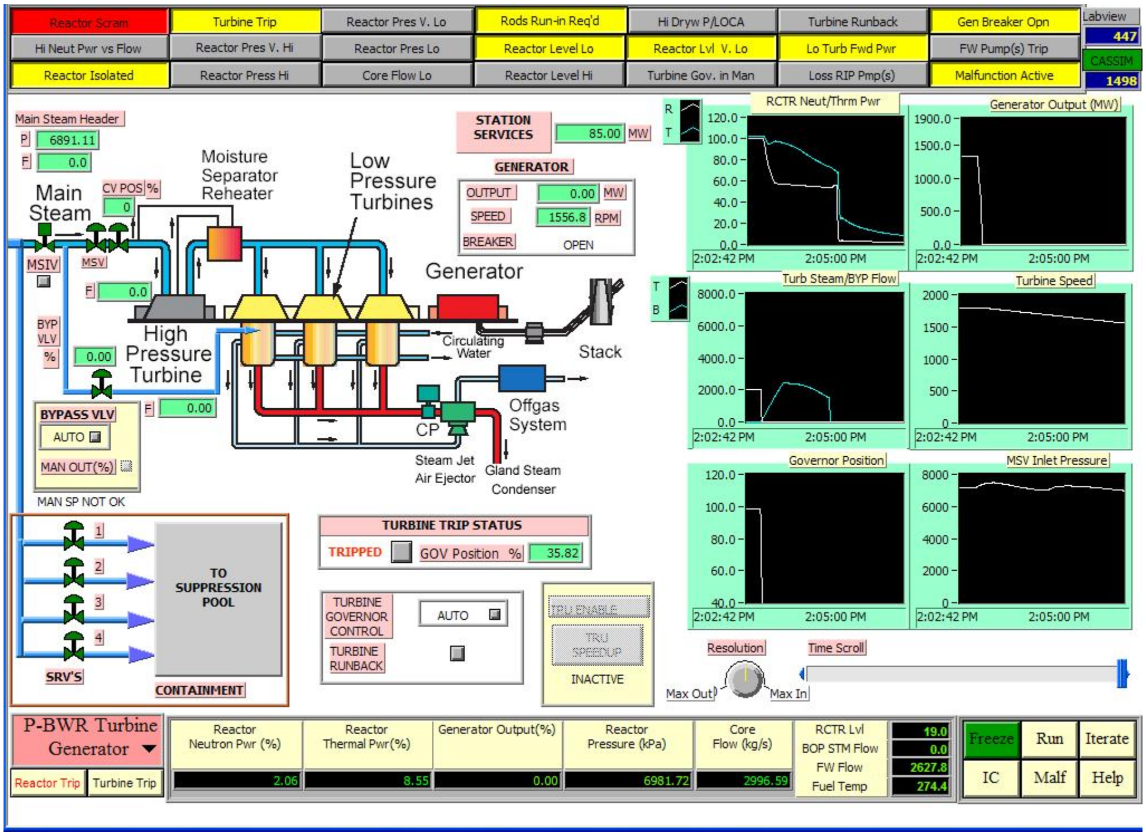
(b) Practice Transient at Simulator

Go to Passive BWR Turbine Generator Screen and load the 100 % FP IC. Run the simulator and insert the malfunction: Loss of Condenser Vacuum, and observe the sequence of events as follows:

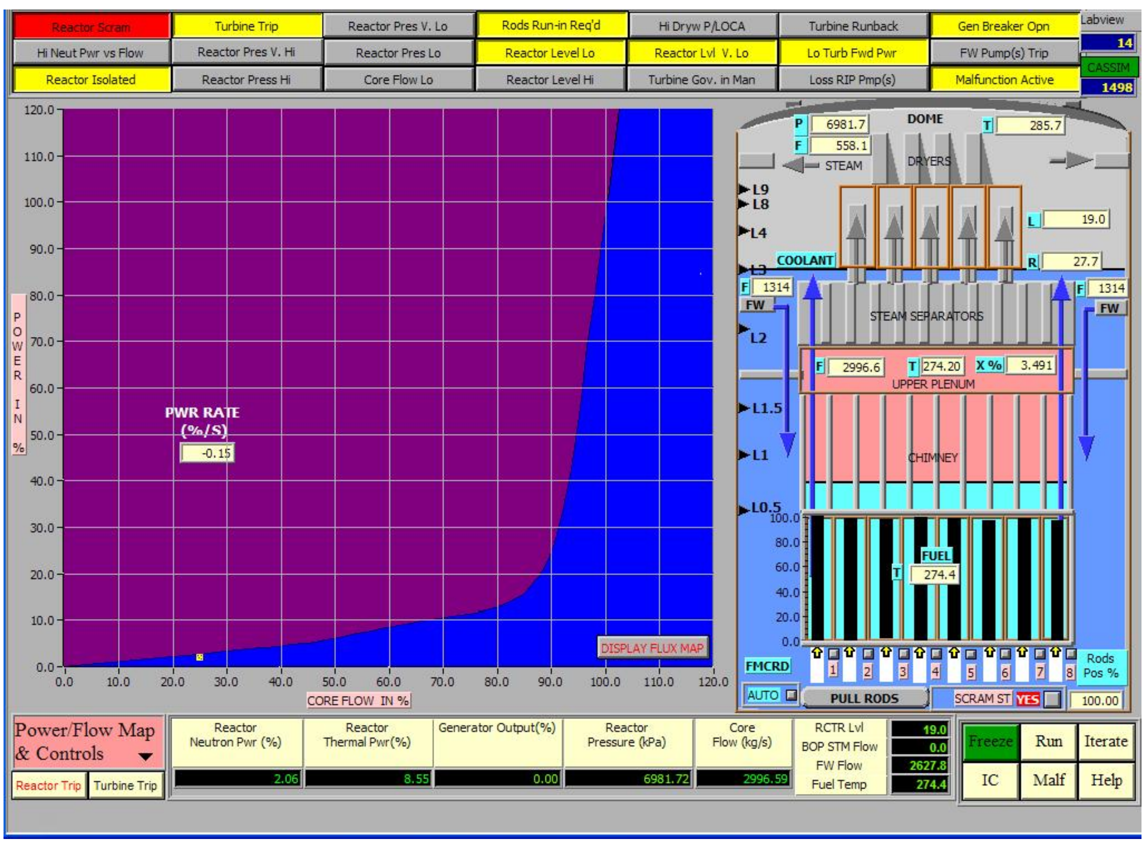
- Low condenser vacuum malfunction initiates turbine trip.
- Turbine trip initiates Selected Control Rods Run-in, reducing reactor power immediately. Record at which reactor pressure when Selected Control Rods Run-In occurs.
- Turbine Bypass Valves start to open to regulate reactor pressure. Monitor steam flow via Bypass Valves.
- Monitor reactor power, reactor pressure, vessel water level at Passive BWR Power/Flow Screen.
- After a 6 sec. delay, low condenser vacuum forces turbine bypass valve closure. Confirm this at Passive BWR Turbine Generator Screen.
- At the same time, reactor scram is initiated by reactor high pressure. Confirm scram parameters at Passive BWR Scram Parameter Screen.
- Low condenser vacuum initiates MSIV closure. Confirm this at Passive BWR Turbine Generator Screen.
- MSIV closure activates Isolation Condenser System. Reactor pressure increase is stopped and starts decreasing. Confirm this at Passive BWR Containment Screen.
- Water level drops to L2, and HP CRD is activated. Monitor HP-CRD flow at Passive BWR RWCU/SDC Screen. Water level starts recovering.

(c) Operator Actions

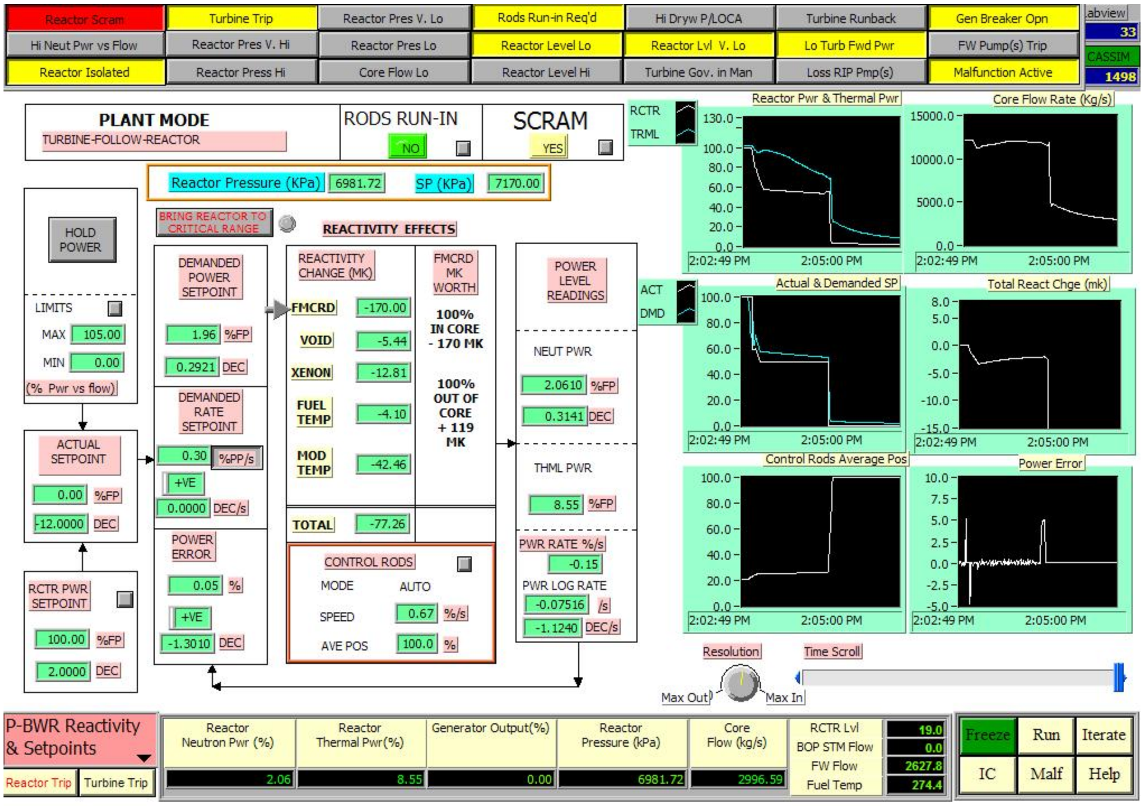
Discuss what operator actions are necessary with this malfunction event.

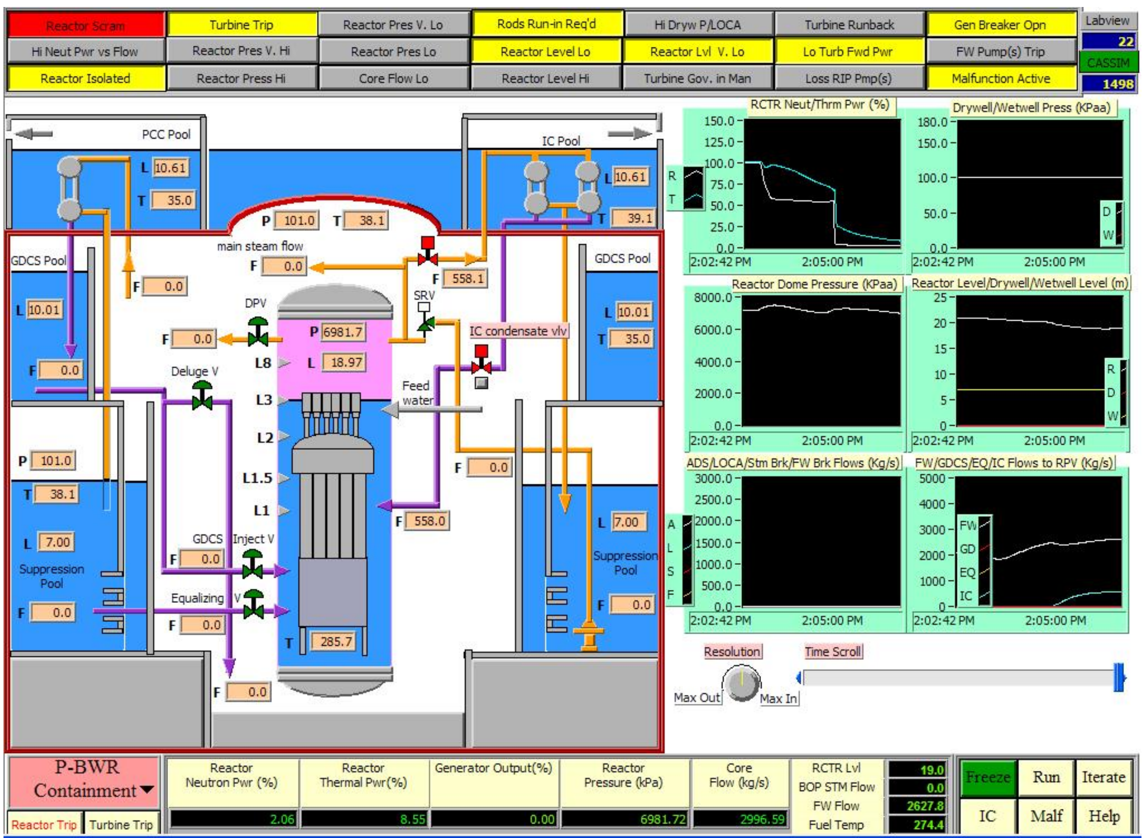


Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
447
CASSIM
1498
Main Steam Header
6891.11
Moisture Separator Reheater
Low Pressure Turbines
STATION SERVICES
85.00 MW
GENERATOR
OUTPUT 0.00 MW
SPEED 1556.8 RPM
BREAKER OPEN
Main Steam
MSIV
MSV
High Pressure Turbine
Generator
Circulating Water
Stack
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
CP
Offgas System
Steam Jet Air Ejector
Gland Steam Condenser
TO SUPPRESSION POOL
SRV'S
CONTAINMENT
TURBINE TRIP STATUS
TRIPPED
GOV Position % 35.82
TURBINE GOVERNOR CONTROL
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
RCTR Neut/Thrm Pwr
Generator Output (MW)
Turb Steam/BYP Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
2:02:42 PM
2:05:00 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.06
Reactor Thermal Pwr(%) 8.55
Generator Output(%) 0.00
Reactor Pressure (kPa) 6981.72
Core Flow (kg/s) 2996.59
RCTR Lvl 19.0
BOP STM Flow 0.0
FW Flow 2627.8
Fuel Temp 274.4
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
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
Labview
14
CASSIM
1498
POWER IN %
PWR RATE (%/S)
-0.15
DISPLAY FLUX MAP
CORE FLOW IN %
DOME
6981.7
558.1
285.7
STEAM
DRYERS
L9
L8
L4
L3
L2
L1.5
L1
L0.5
COOLANT
19.0
27.7
1314
FW
STEAM SEPARATORS
2996.6
274.20
X %
3.491
UPPER PLENUM
CHIMNEY
FUEL
274.4
FMCRD
AUTO
PULL RODS
SCRAM ST YES
Rods Pos %
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.06
Reactor Thermal Pwr(%) 8.55
Generator Output(%) 0.00
Reactor Pressure (kPa) 6981.72
Core Flow (kg/s) 2996.59
RCTR Lvl 19.0
BOP STM Flow 0.0
FW Flow 2627.8
Fuel Temp 274.4
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
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
PLANT MODE
TURBINE-FOLLOW-REACTOR
RODS RUN-IN
NO
SCRAM
YES
Reactor Pressure (KPa) 6981.72
SP (KPa) 7170.00
HOLD POWER
BRING REACTOR TO CRITICAL RANGE
REACTIVITY EFFECTS
LIMITS
MAX 105.00
MIN 0.00
(% Pwr vs flow)
ACTUAL SETPOINT
0.00 %FP
-12.0000 DEC
RCTR PWR SETPOINT
100.00 %FP
2.0000 DEC
DEMANDED POWER SETPOINT
1.96 %FP
0.2921 DEC
DEMANDED RATE SETPOINT
0.30 %PP/s
+VE
0.0000 DEC/s
POWER ERROR
0.05 %
+VE
-1.3010 DEC
REACTIVITY CHANGE (MK)
FMCRD -170.00
VOID -5.44
XENON -12.81
FUEL TEMP -4.10
MOD TEMP -42.46
TOTAL -77.26
FMCRD MK WORTH
100% IN CORE - 170 MK
100% OUT OF CORE + 119 MK
CONTROL RODS
MODE AUTO
SPEED 0.67 %/s
AVE POS 100.0 %
POWER LEVEL READINGS
NEUT PWR
2.0610 %FP
0.3141 DEC
THML PWR
8.55 %FP
PWR RATE %/s
-0.15
PWR LOG RATE
-0.07516 /s
-1.1240 DEC/s
Reactor Pwr & Thermal Pwr
Core Flow Rate (Kg/s)
Actual & Demanded SP
Total React Chge (mk)
Control Rods Average Pos
Power Error
2:02:49 PM
2:05:00 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Reactivity & Setpoints
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.06
Reactor Thermal Pwr(%) 8.55
Generator Output(%) 0.00
Reactor Pressure (kPa) 6981.72
Core Flow (kg/s) 2996.59
RCTR Lvl 19.0
BOP STM Flow 0.0
FW Flow 2627.8
Fuel Temp 274.4
Freeze
Run
Iterate
IC
Malf
Help


PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
IC condensate vlv
Deluge V
Feed water
GDCS
Inject V
Equalizing V
Suppression Pool
P 6981.7
L 18.97
T 285.7
F 558.1
F 558.0
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
2:02:42 PM
2:05:00 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Containment
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.06
Reactor Thermal Pwr(%) 8.55
Generator Output(%) 0.00
Reactor Pressure (kPa) 6981.72
Core Flow (kg/s) 2996.59
RCTR Lvl 19.0
BOP STM Flow 0.0
FW Flow 2627.8
Fuel Temp 274.4
Freeze
Run
Iterate
IC
Malf
Help

Brief Review of Passive BWR ECC Features

Before running the transients for the following malfunctions: # 15 (Steam line break inside drywell), #16 (Feedwater line break inside drywell), #17 (Reactor vessel bottom break - 3000 kg/sec LOCA), it is appropriate to review the passive ECC functions that are discussed in Chapter 2. The followings only provide a brief review and summary. It is recommended for simulator users to go back to Chapter 2 for specific details, if necessary.

(1) Passive ECC Features

The passive BWR ECC features include the following systems:

(a) Gravity-Driven Cooling System (GDCS)

The GDCS provides flow to the annulus region of the reactor through dedicated nozzles. It provides gravity-driven flow from three separate water pools located within the drywell at an elevation above the active core region. It also provides water flow from the suppression pool to meet long-term post-LOCA core cooling requirements. The system provides these flows by gravity forces alone (without reliance on active pumps) once the reactor pressure is reduced to near containment pressure.

(b) Automatic Depressurization System (ADS)

The ADS provides reactor depressurization capability in the event of a pipe break. The ADS is a function of the Nuclear Boiler System (NBS). The depressurization function is accomplished through the use of Safety Relief Valves (SRVs) and Depressurization Valves (DPVs).

(c) Isolation Condenser System (ICS)

The ICS provides additional liquid inventory upon opening of the condensate return valves to initiate the system. The IC system also provides initial depressurization of the reactor before ADS in event of loss of feed water, such that the ADS can take place from a lower water level.

(d) Standby Liquid Control System (SLCS)

The SLC system provides reactor additional liquid inventory in the event of DPV actuation. This function is accomplished by firing squib type injection valves to initiate the SLC system.

Key Summary of Passive ECC Systems:

- The ECCS systems ADS and GDCS are designed to accomplish only one function, to cool the reactor core following a LOCA.
- The ECCS system SLCS is designed to be used during an ATWS (Anticipated Transient Without Scram),
- The ECCS system ICS is designed to avoid unnecessary use of other Engineered Safety Functions (ESFs) for residual heat removal.
- Both SLCS and ICS provide additional liquid inventory upon actuation.

(2) Acceptance Criteria

The design objectives of the passive safety system are to meet the applicable acceptance criteria for nuclear safety, such as the US NRC 10 CFR 50.46 for Design Basis Accidents (DBAs) :

- **Criterion 1: Peak Cladding Temperature (PCT)**
  The calculated maximum fuel element cladding temperature shall not exceed 2200°F, which is equivalent to 1204°C.
- **Criterion 2: Maximum Cladding Oxidation**
  The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- **Criterion 3: Maximum Hydrogen Generation**
  The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated, if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- **Criterion 4: Coolable Geometry**
  Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- **Criterion 5: Long-Term Cooling**
  After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be remove for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is assured for any LOCA where the water level can be restored and maintained at a level above the top of the core. For SBWR, the core never uncovers during a design basis LOCA event due to flow from the GDCS pools. The Passive BWR ECCS maintains the water level in the vessel above the core for a period of greater than 30 days following a LOCA.

(3) **ECC System Performance Expectation during Accidents**

In general, the system response to an accident can be described as:

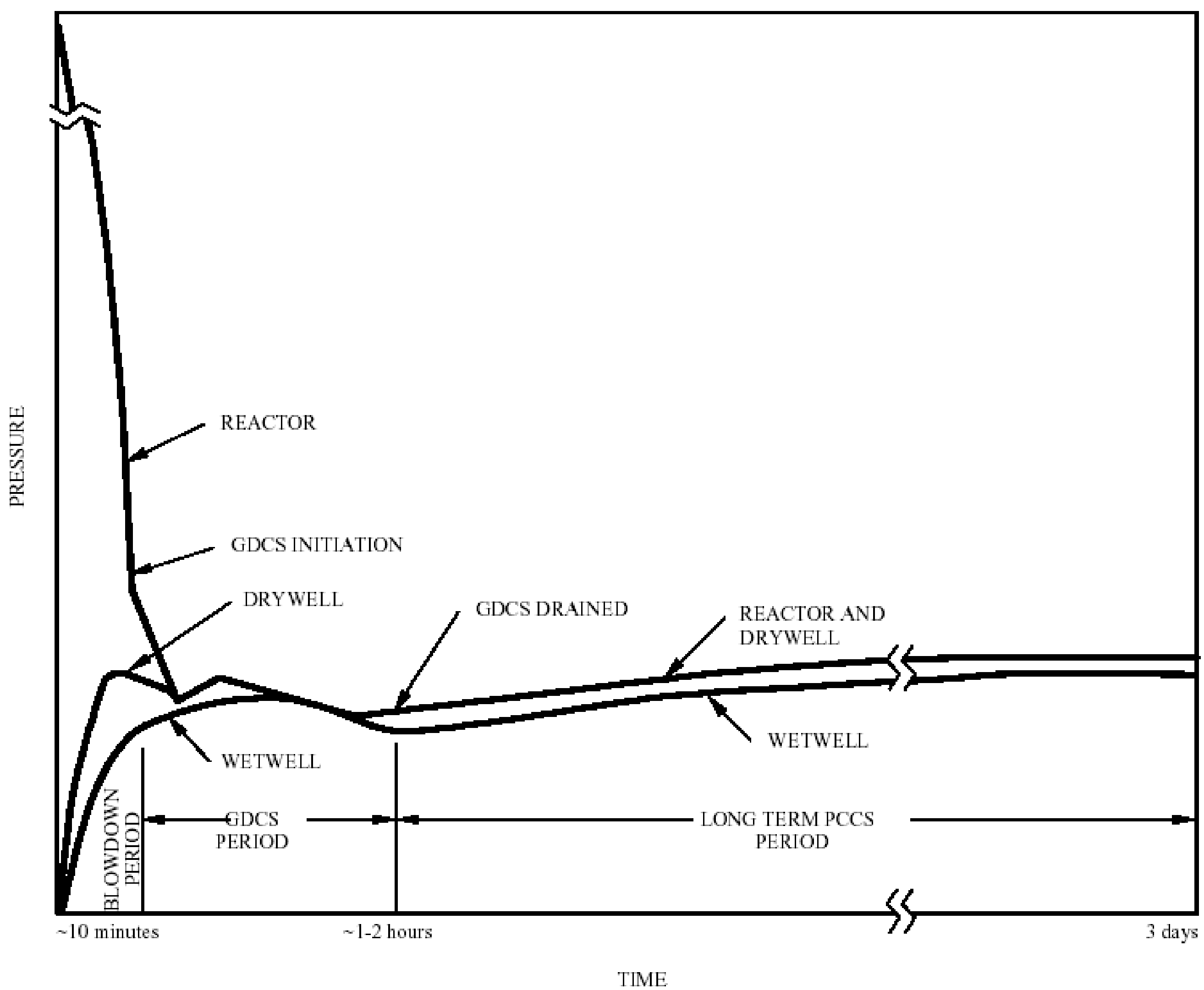
- Receiving an initiation signal;
- A small lag time (to open all valves and depressurize the vessel); and
- The GDCS flow entering the vessel.
- Operator action is not required for 72 hours, except as a monitoring function, following any LOCA.

ECCS actuation setpoints and time delays should cover the various “break” accidents. The key ones are:

(1) The ADS actuation logic includes a delay time to confirm the presence of a low water level (Level 1) initiation signal.

(2) The GDCS flow delivery rates for the various breaks should be analyzed.

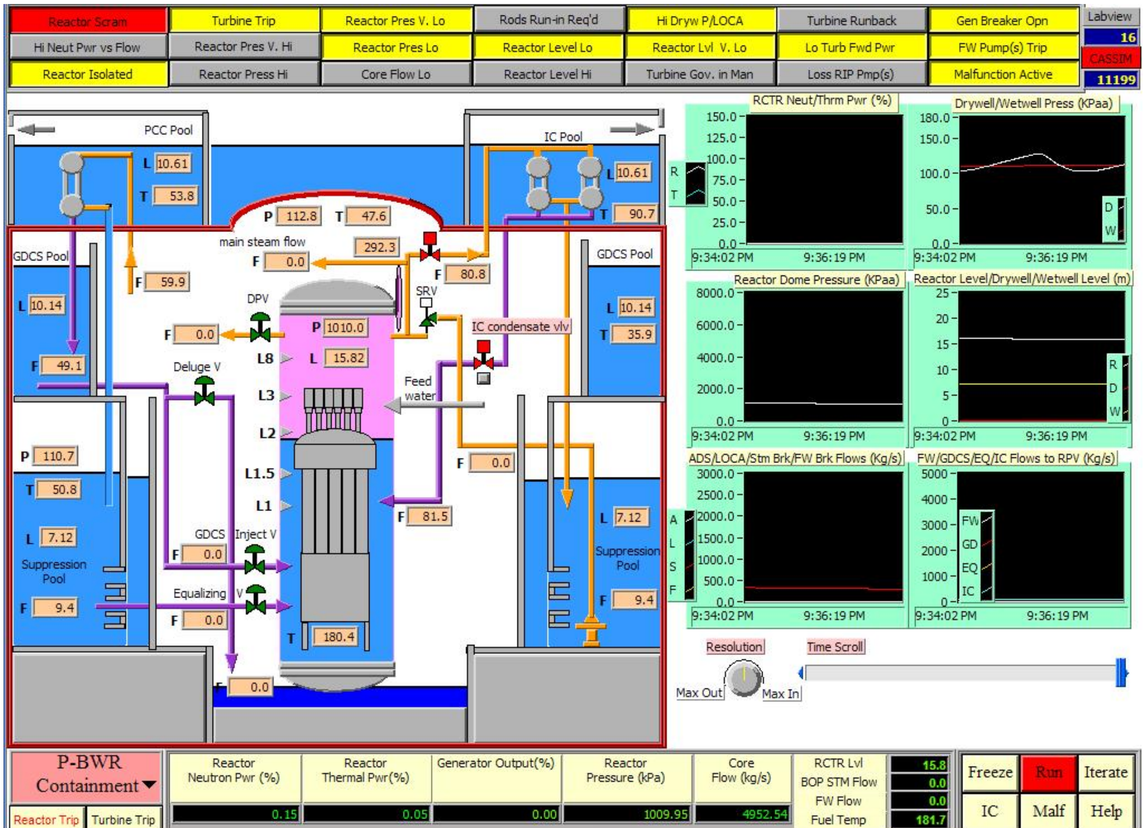
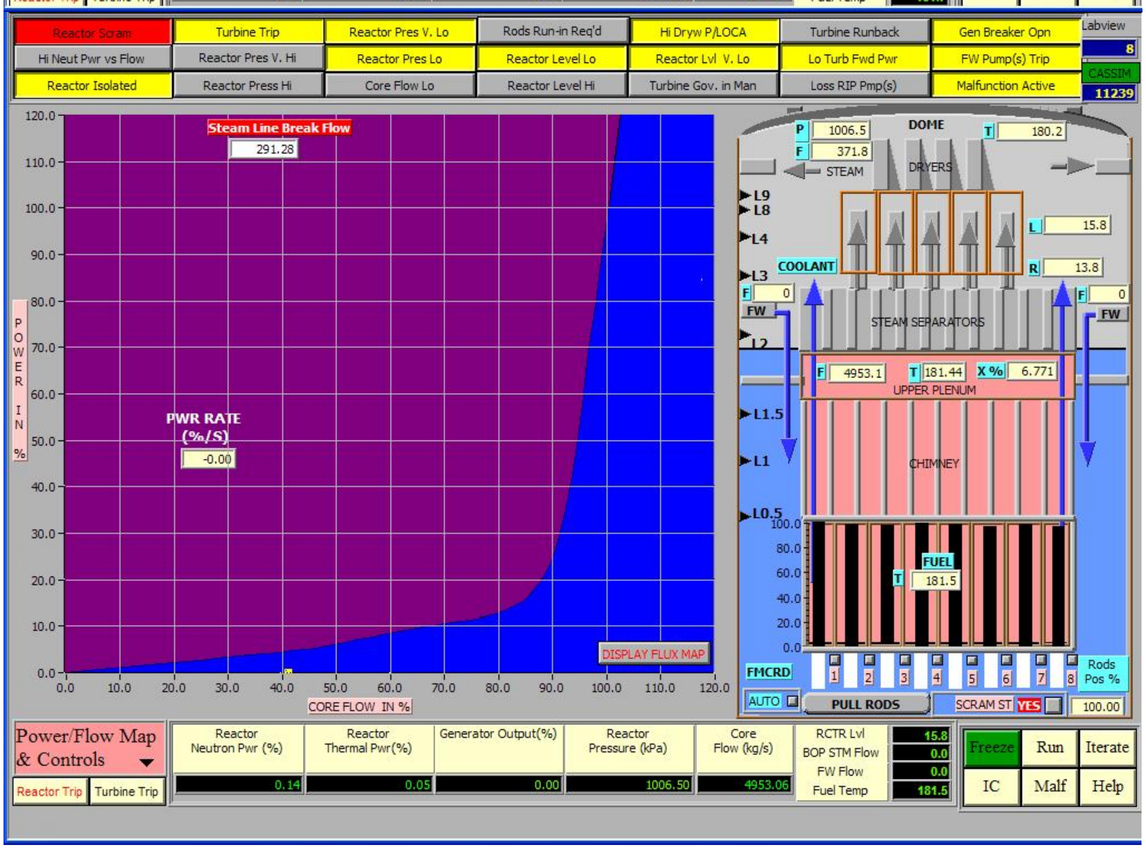
(3) Piping and instrumentation for the GDCS and ADS should be addressed appropriately.



5.2.15 Steam line break inside drywell

Practice Transient at Simulator

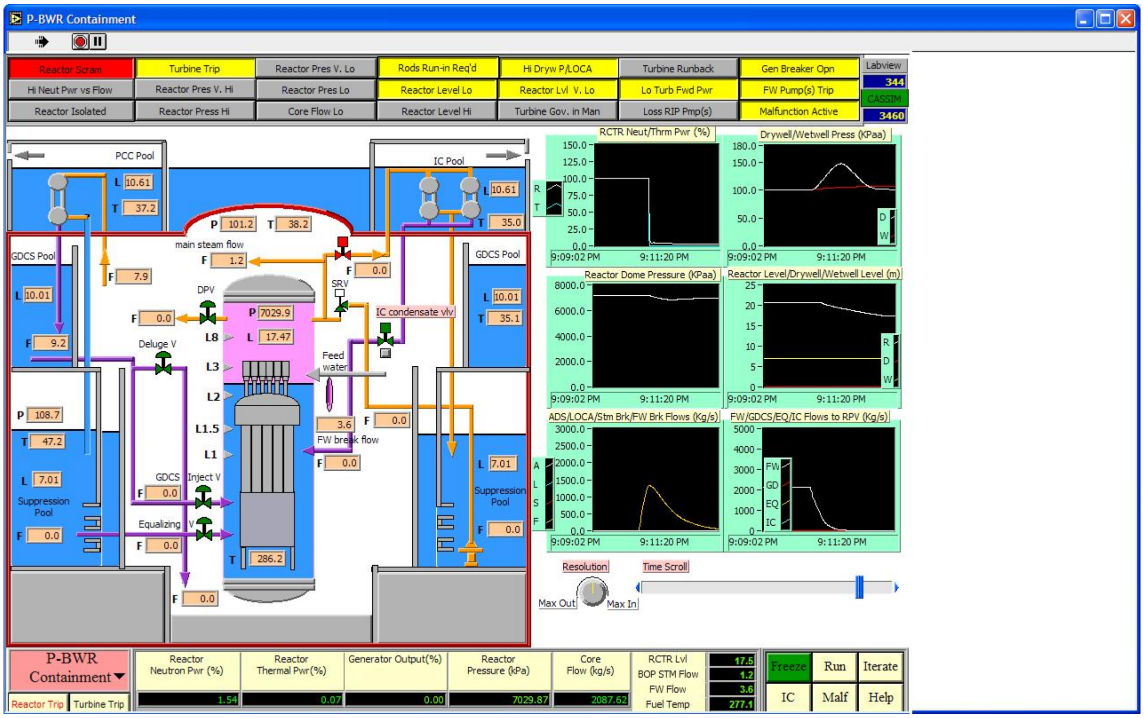
- Go to Passive BWR Power/Flow Screen and load the 100% FP IC. Run the simulator.
- This malfunction causes a main steam line break in the main steam line (before the main steam isolation valve) inside containment drywall.
- The break flow into the drywell will increase pressure rapidly, resulting in depressurization of the reactor dome pressure, as well as pressurization of the drywell.
- The consequence is the detection of LOCA condition as a result of drywell pressurization.
- LOCA signal scrams the reactor and trips the FW pumps. FW flow decreases.
- As the reactor dome depressurizes, there is more boiling in core, hence more void. Observe there is "Reactor Lo Pres" alarm.
- Drywell vents are clearing. Steam in drywell enters wetwell through vents. Drywell pressure starts to decrease.
- Low main steam line pressure setpoint reached. MSIV closure initiated to isolate reactor. Isolation Condenser initiated by MSIV closure, leading to more decrease in reactor pressure, with steam flowing to ICS.
- The turbine throttle pressure decreases rapidly, leading to turbine. The result is turbine trip by low turbine forward power (alarm "Lo Turb Fwd Pwr")
- Water level L3 reached. Reactor scram initiated again on L3. But reactor is already scrammed.
- At this point, FW flow stops. The water level continues to drop but at a lower pace, due to water boiling off, and escaping through the steam line break into containment drywell.
- Eventually water level reaches L1. ADS/GDCS/SLC timer initiated; SRV actuated to depressurize.
- DPV actuation begins at 50 sec. after confirmed L1.
- GDCS timed out – 150 sec. after confirmed L1. GDCS injection valves open.
- Vessel pressure decreases below the maximum injection pressure of GDCS. GDCS flow into the vessel begins. Water level starts to rise.
- Vessel level remains higher than L0.5. Therefore equalizing line valves do not open for this malfunction.

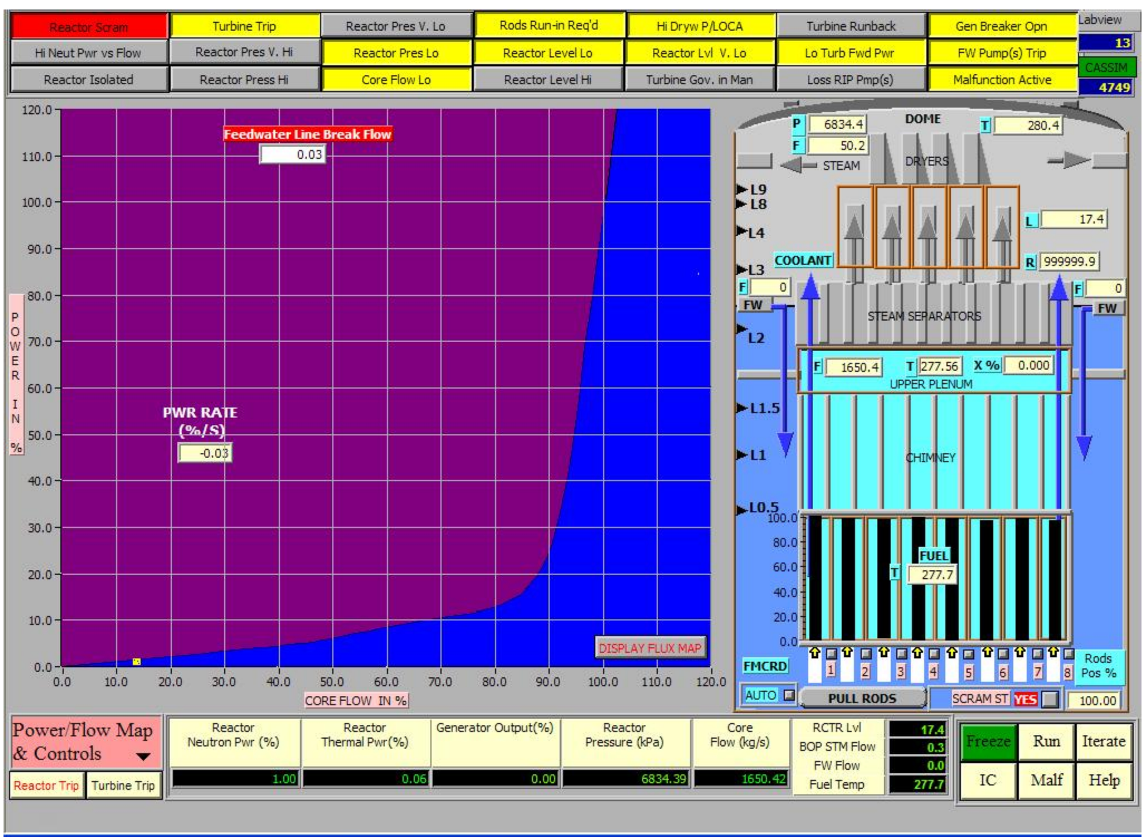

Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
CASSIM
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
Deluge V
IC condensate vlv
Feed water
GDCS Inject V
Equalizing V
Suppression Pool
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
Resolution
Time Scroll
Max Out
Max In
P-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
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
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
Steam Line Break Flow
PWR RATE (%/S)
POWER IN %
CORE FLOW IN %
DISPLAY FLUX MAP
DOME
DRYERS
STEAM
COOLANT
STEAM SEPARATORS
UPPER PLENUM
CHIMNEY
FUEL
FMCRD
AUTO
PULL RODS
SCRAM ST
Rods Pos %
Power/Flow Map & Controls
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
Freeze
Run
Iterate
IC
Malf
Help

5.2.16 Feedwater line break inside drywell

Practice Transient at Simulator

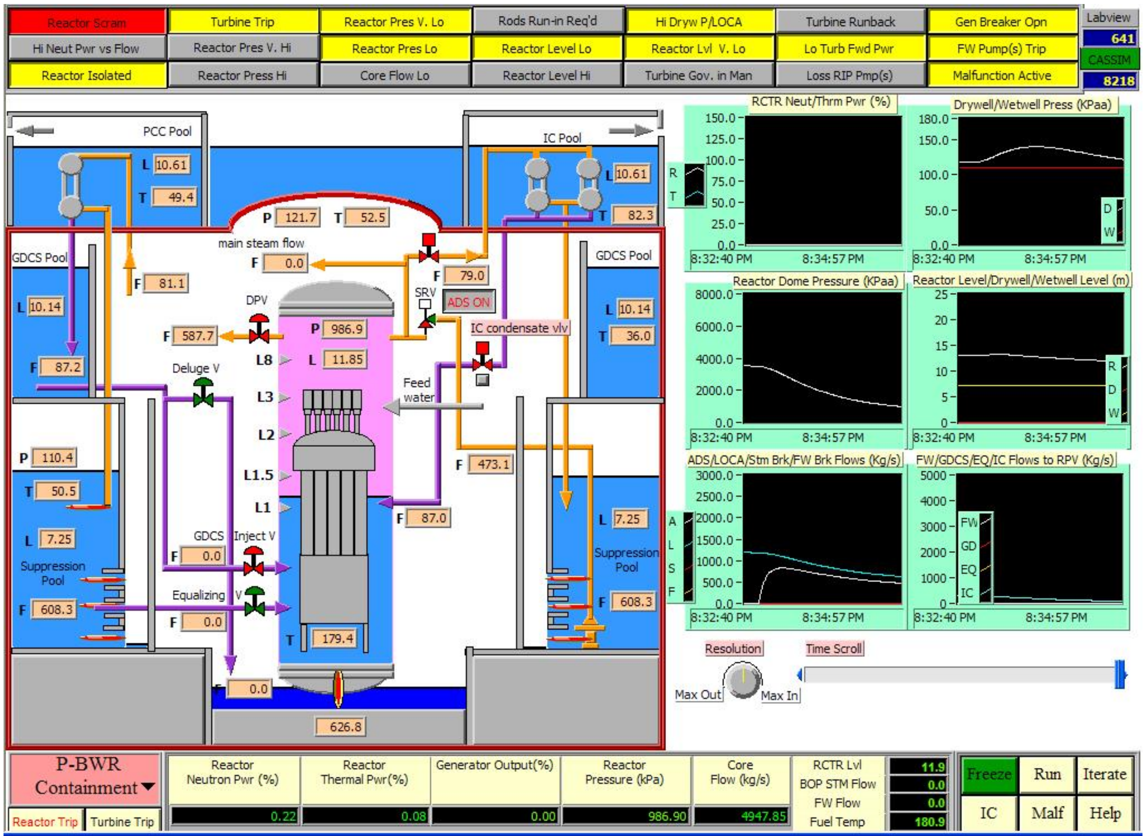
- Go to Passive BWR Power/Flow Screen and load the 100% FP IC. Run the simulator.
- This malfunction causes a feedwater line break in the supply line inside containment drywall. There is a non-return check valve upstream of break, so that steam from the dome will not flow back to the feedwater supply line, except for some steam leakage.
- The feedwater break flow into the drywell will increase drywell pressure rapidly.
- The consequence is the detection of LOCA condition as a result of drywell pressurization.
- FW pumps trip on LOCA signal, decreasing FW flow through the break inside containment.
- Drywell vents are clearing. Steam in drywell enters wetwell through vents. Drywell pressure starts to decrease.
- The turbine throttle pressure decreases rapidly, leading to turbine runback. The result is turbine trip by low turbine forward power (alarm "Lo Turb Fwd Pwr").
- Water level L3 reached. Reactor scram initiated again on L3. But reactor is already scrammed.
- Water level drops below L3, and the transient stabilizes with slow pressure decreases due to steam leakage.

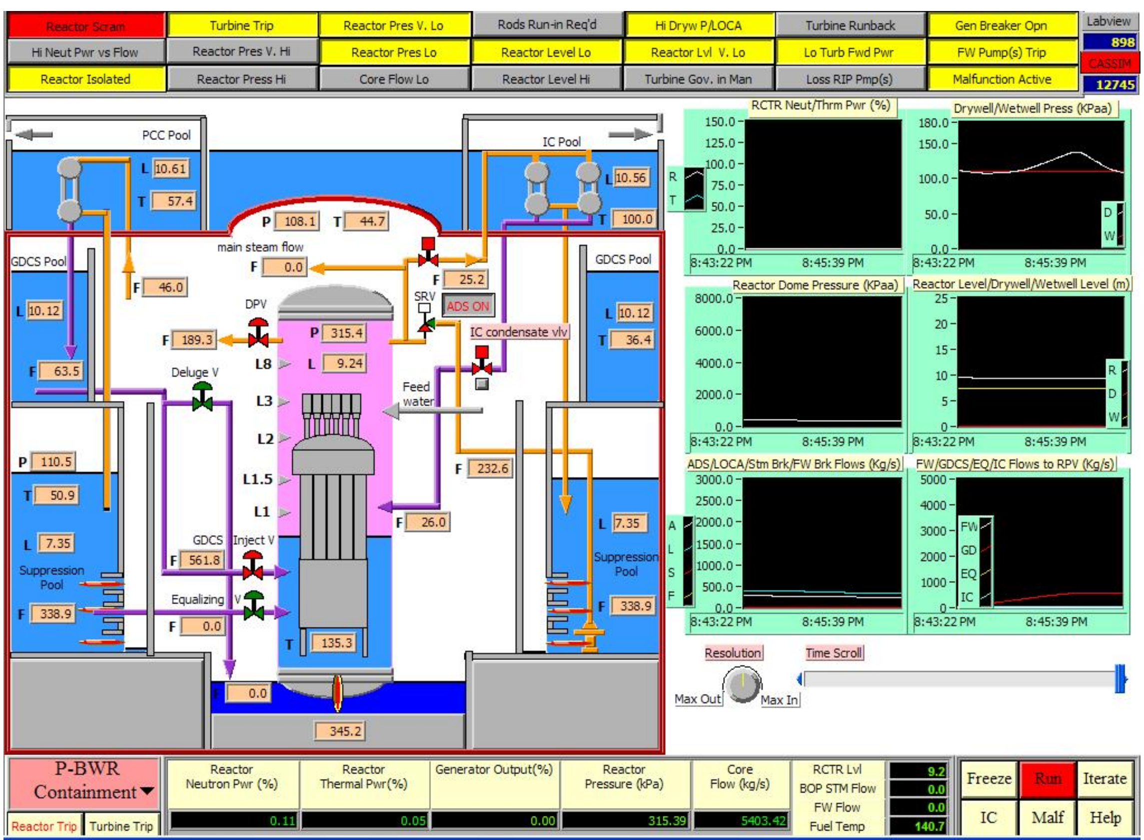

P-BWR Containment
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
CASSIM
PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
IC condensate vlv
Deluge V
Feed water
FW break flow
GDCS Inject V
Equalizing V
Suppression Pool
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
Resolution
Time Scroll
Max Out
Max In
P-BWR Containment
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
RCTR Lvl
BOP STM Flow
FW Flow
Fuel Temp
Freeze
Run
Iterate
IC
Malf
Help
Reactor Trip
Turbine Trip

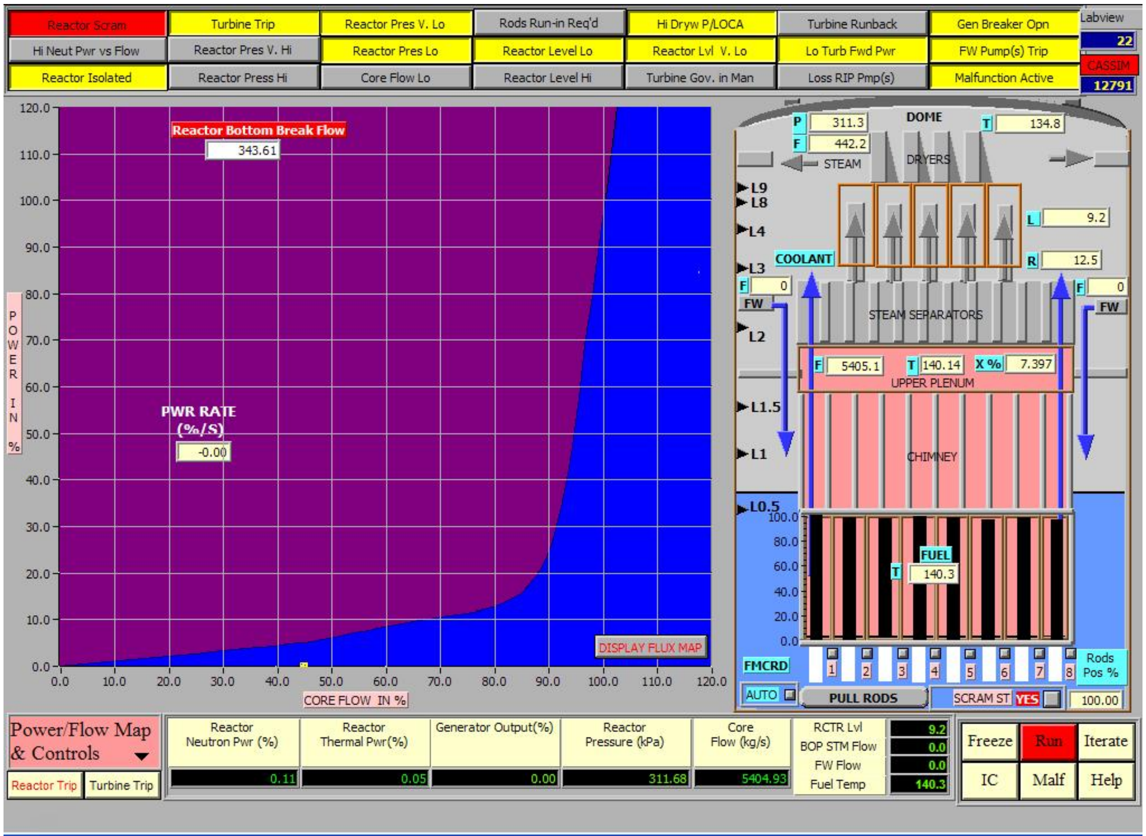

Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
CASSIM
Feedwater Line Break Flow
PWR RATE (%/S)
POWER IN %
CORE FLOW IN %
DISPLAY FLUX MAP
DOME
STEAM
DRYERS
COOLANT
STEAM SEPARATORS
UPPER PLENUM
CHIMNEY
FUEL
FMCRD
AUTO
PULL RODS
SCRAM ST
Rods Pos %
Power/Flow Map & Controls
Reactor Neutron Pwr (%)
Reactor Thermal Pwr(%)
Generator Output(%)
Reactor Pressure (kPa)
Core Flow (kg/s)
RCTR Lvl
BOP STM Flow
FW Flow
Fuel Temp
Freeze
Run
Iterate
IC
Malf
Help
Reactor Trip
Turbine Trip

5.2.17 Reactor vessel bottom break - 1660 kg/sec LOCA

- First go to Power/Flow Screen and load the 100% FP IC. Run the simulator.
- This malfunction causes a "crack" opening at the Reactor Vessel bottom, resulting in a LOCA event with increasing break flow reaching 1660 Kg/sec into the containment drywell.
- The bottom break flow into the drywell will increase pressure rapidly. The consequence is the detection of LOCA condition as a result of drywell pressurization.
- LOCA signal scrams the reactor and trips the FW pumps. FW flow is decreasing.
- Low steam pressure runs back turbine, leading to a turbine trip by "low forward power".
- Drywell vents are clearing. Steam in drywell enters wetwell through vents. Drywell pressure starts to decrease.
- Water level L3 reached. Reactor scram initiated again on L3. But reactor is already scrammed.
- Water level L2 reached – L2 initiates MSIV closure and IC initiation.
- Water level has reached L1. ADS/GDCS/SLC timer initiated; SRV actuated to depressurize RPV pressure to suppression pool.
- DPV actuation begins at 50 sec. after confirmed L1.
- GDCS timed out – 150 sec. after confirmed L1. GDCS injection valves open.
- Vessel pressure decreases below the maximum injection pressure of GDCS. GDCS flow into the vessel begins. Water level starts to rise.
- Vessel level remains higher than L0.5. Therefore equalizing line valves do not open for this malfunction.


Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Labview
641
CASSIM
8218
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
PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
ADS ON
IC condensate vlv
Deluge V
Feed water
GDCS Inject V
Equalizing V
Suppression Pool
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
8:32:40 PM
8:34:57 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Containment
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.22
Reactor Thermal Pwr(%) 0.08
Generator Output(%) 0.00
Reactor Pressure (kPa) 986.90
Core Flow (kg/s) 4947.85
RCTR Lvl 11.9
BOP STM Flow 0.0
FW Flow 0.0
Fuel Temp 180.9
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
898
CASSIM
12745
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
PCC Pool
IC Pool
GDCS Pool
main steam flow
DPV
SRV
ADS ON
IC condensate vlv
Deluge V
Feed water
GDCS Inject V
Equalizing V
Suppression Pool
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
8:43:22 PM
8:45:39 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Containment
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.11
Reactor Thermal Pwr(%) 0.05
Generator Output(%) 0.00
Reactor Pressure (kPa) 315.39
Core Flow (kg/s) 5403.42
RCTR Lvl 9.2
BOP STM Flow 0.0
FW Flow 0.0
Fuel Temp 140.7
Freeze
Run
Iterate
IC
Malf
Help

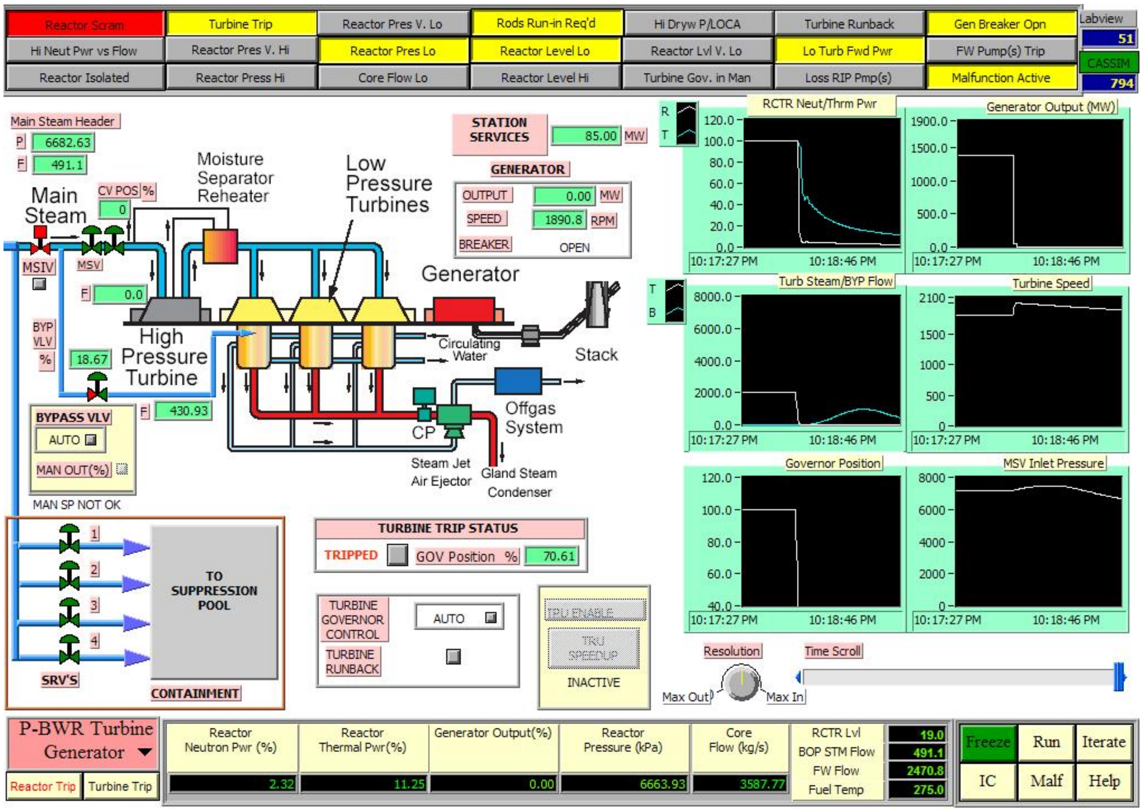


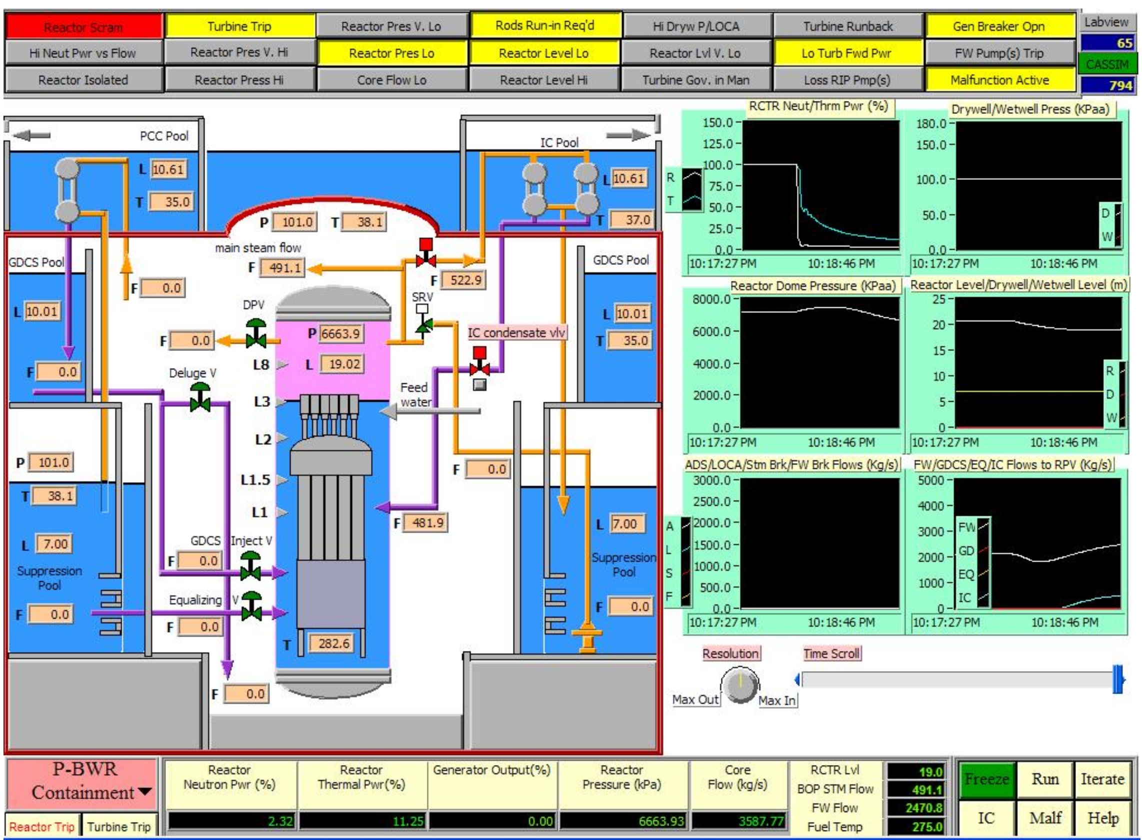
Reactor Scram
Turbine Trip
Reactor Pres V. Lo
Rods Run-in Req'd
Hi Dryw P/LOCA
Turbine Runback
Gen Breaker Opn
Hi Neut Pwr vs Flow
Reactor Pres V. Hi
Reactor Pres Lo
Reactor Level Lo
Reactor Lvl V. Lo
Lo Turb Fwd Pwr
FW Pump(s) Trip
Reactor Isolated
Reactor Press Hi
Core Flow Lo
Reactor Level Hi
Turbine Gov. in Man
Loss RIP Pmp(s)
Malfunction Active
Labview
22
CASSIM
12791
Reactor Bottom Break Flow
343.61
PWR RATE (%/S)
-0.00
POWER IN %
CORE FLOW IN %
DISPLAY FLUX MAP
DOME
DRYERS
STEAM
COOLANT
STEAM SEPARATORS
UPPER PLENUM
CHIMNEY
FUEL
140.3
FMCRD
AUTO
PULL RODS
SCRAM ST YES
Rods Pos %
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.11
Reactor Thermal Pwr(%) 0.05
Generator Output(%) 0.00
Reactor Pressure (kPa) 311.68
Core Flow (kg/s) 5404.93
RCTR Lvl 9.2
BOP STM Flow 0.0
FW Flow 0.0
Fuel Temp 140.3
Freeze
Run
Iterate
IC
Malf
Help

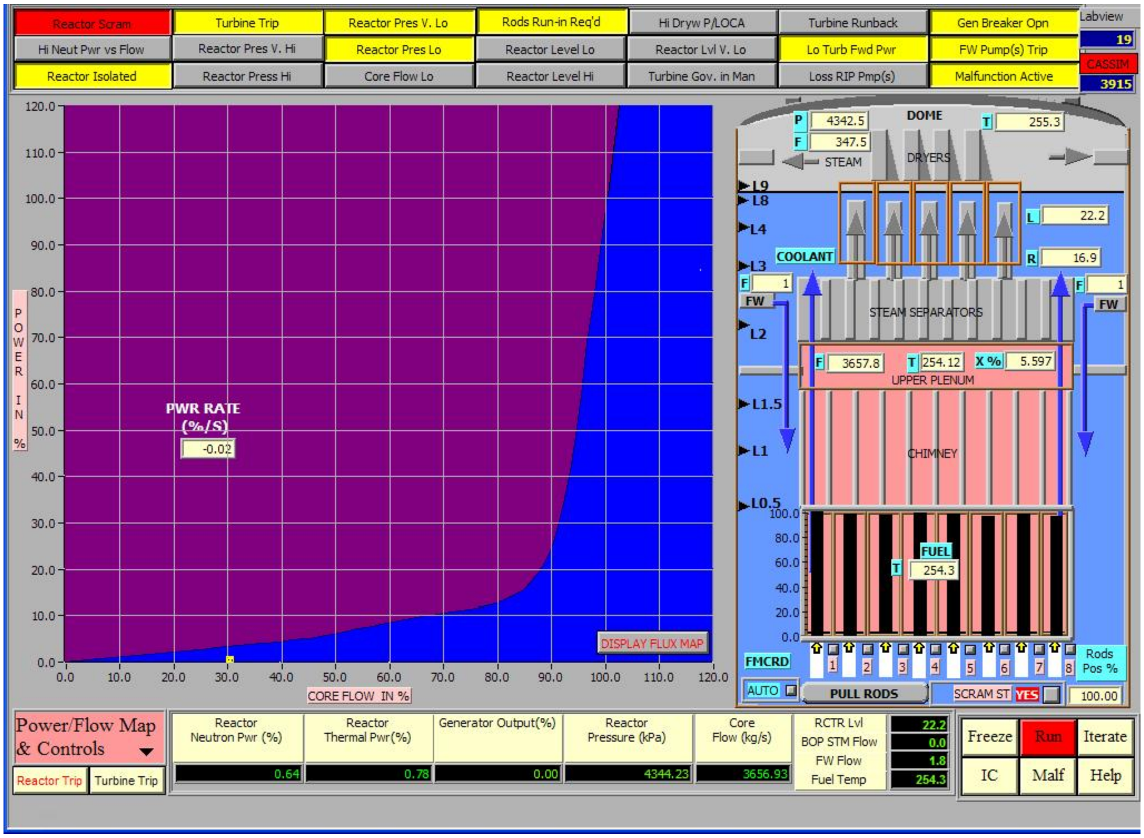
5.2.18 Load rejection

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 with this malfunction, causing trip on the generator, and subsequent trip on the turbine. In addition, the reactor is scrammed by power/load unbalance.

- Before initiating the malfunction, first go to the Power/Flow Map Screen.
- Load the 100% FP IC. Run the simulator.
- Insert the malfunction, and observe the turbine speed increases due to loss of load.
- Turbine trip occurs very quickly, followed by reactor scram by "Turbine Power/Load Unbalance — Loss of Line" trip logic.
- Reactor pressure increases rapidly, BP & PC system opens the bypass valve to relieve steam pressure.
- The high steam pressure activates Isolation Condenser. Reactor pressure decreases.
- Eventually low reactor pressure activates MSIV closure.
- Water level recovers to L4. FW flow decreases.
- Due to continued vessel depressurization via ICS, the water has more void, thus raising the level above L4.
- Pressure continues to decrease via steam condensation at ICS.
- Transient stabilizes


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
Labview
51
CASSIM
794
Main Steam Header
P 6682.63
F 491.1
Main Steam
CV POS %
MSIV
MSV
BYP VLV %
18.67
Moisture Separator Reheater
Low Pressure Turbines
High Pressure Turbine
STATION SERVICES
85.00 MW
GENERATOR
OUTPUT 0.00 MW
SPEED 1890.8 RPM
BREAKER OPEN
Generator
Circulating Water
Stack
Offgas System
CP
Steam Jet Air Ejector
Gland Steam Condenser
F 430.93
BYPASS VLV
AUTO
MAN OUT(%)
MAN SP NOT OK
TO SUPPRESSION POOL
SRV'S
CONTAINMENT
TURBINE TRIP STATUS
TRIPPED
GOV Position % 70.61
TURBINE GOVERNOR CONTROL
AUTO
TURBINE RUNBACK
TRU ENABLE
TRU SPEEDUP
INACTIVE
RCTR Neut/Thrm Pwr
Generator Output (MW)
Turb Steam/BYP Flow
Turbine Speed
Governor Position
MSV Inlet Pressure
10:17:27 PM
10:18:46 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Turbine Generator
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.32
Reactor Thermal Pwr(%) 11.25
Generator Output(%) 0.00
Reactor Pressure (kPa) 6663.93
Core Flow (kg/s) 3587.77
RCTR Lvl 19.0
BOP STM Flow 491.1
FW Flow 2470.8
Fuel Temp 275.0
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
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
Labview
65
CASSIM
794
PCC Pool
IC Pool
L 10.61
T 35.0
T 37.0
P 101.0
T 38.1
main steam flow
F 491.1
F 522.9
SRV
DPV
IC condensate vlv
GDCS Pool
L 10.01
T 35.0
P 6663.9
L 19.02
L8
L3
L2
L1.5
L1
Deluge V
Feed water
F 481.9
GDCS Inject V
Equalizing V
T 282.6
L 7.00
Suppression Pool
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)
FW/GDCS/EQ/IC Flows to RPV (Kg/s)
10:17:27 PM
10:18:46 PM
Resolution
Time Scroll
Max Out
Max In
P-BWR Containment
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 2.32
Reactor Thermal Pwr(%) 11.25
Generator Output(%) 0.00
Reactor Pressure (kPa) 6663.93
Core Flow (kg/s) 3587.77
RCTR Lvl 19.0
BOP STM Flow 491.1
FW Flow 2470.8
Fuel Temp 275.0
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
19
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
3915
POWER IN %
CORE FLOW IN %
PWR RATE (%/S)
-0.02
DISPLAY FLUX MAP
DOME
P 4342.5
F 347.5
T 255.3
STEAM
DRYERS
COOLANT
L 22.2
R 16.9
STEAM SEPARATORS
F 3657.8
T 254.12
X % 5.597
UPPER PLENUM
CHIMNEY
FUEL
T 254.3
FMCRD
Rods Pos %
AUTO
PULL RODS
SCRAM ST YES
100.00
Power/Flow Map & Controls
Reactor Trip
Turbine Trip
Reactor Neutron Pwr (%) 0.64
Reactor Thermal Pwr(%) 0.78
Generator Output(%) 0.00
Reactor Pressure (kPa) 4344.23
Core Flow (kg/s) 3656.93
RCTR Lvl 22.2
BOP STM Flow 0.0
FW Flow 1.8
Fuel Temp 254.3
Freeze
Run
Iterate
IC
Malf
Help

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9. Shiralkar, B., et al., "Natural Circulation in ESBWR," Paper No. ICONE15-10439, Proceedings of 15th International Conference on Nuclear Engineering, Nagoya, Japan, April 2007.
10. ESBWR Design Control Document Tier 2 – Chapter 4 REACTOR, submission by GE Energy to US NRC for ESBWR Design Certification. 26A6642AP, Revision 3 February 2007

Overall Design	
Site Envelope	
Safe shutdown earthquake, g	0.3 envelope
Wind design, km/h	225
Maximum tornado, km/h	531
Max dry bulb/wet bulb ambient temperature, °C	46/27
Thermal and Hydraulic	
Rated Power, MWt	4500
Generator Output, MWe	1600
Steam flow rate, Mkg/h	8.76
Core coolant flow rate, Mkg/h	36.0
System operating pressure, MPa	7.17
Average core power density, kW/l	54.3
Maximum linear heat generation rate, kW/m	44.0
Average linear heat generation rate, kW/m	15.1
Minimum critical power ratio (MCPR)	1.4 - 1.5*
Core average exit quality, %	17.0
Feedwater temperature, °C	215.6

* Depending on the cycle length

Core Design	
Fuel Assembly	
Number of fuel assemblies	1132
Fuel rod array	10 x 10
Overall length, cm	379
Weight of UO2 per assembly, kg	144
Number of fuel rods per assembly	92
Rod diameter, cm	1.026
Cladding material	Zircaloy-2
Fuel Channel	
Thickness corner/wall, mm	3.05/1.91
Dimensions, cm	14 X 14
Material	Zircaloy-2
Reactor Control System	
Method of variation of reactor power	Moveable control rods
Number of control rods	269
Shape of control rods	Cruciform
Pitch of control rods, cm	31
Type of control rod drive	Bottom entry electric hydraulic fine motion
Rod step size, mm	36.5
Number of hydraulic accumulators	135
Hydraulic scram speed, sec to 60% insert	1.15
Electric drive speed, mm/sec	30
Type of temporary reactivity control	Burnable poison; gadolinia urania fuel rods
High pressure coolant injection 1/2 pumps, m^3/h	118/235
Incore Neutron Instrumentation	
Total number of LPRM detectors	256
Number of incore LPRM penetrations	76
Number of LPRM detectors per penetration	4
Number of SRNM penetrations	12

Emergency Core Cooling	
Gravity Driven Core Cooling	
Number of loops	4
Number of pumps	0
Flow rate, m^3/h	500*
Automatic Depressurization	
Number of relief valves	10
Number of depressurization valves	8
Passive Containment Cooling System	
Number of loops	6
Heat removal duty per loop, MWt	11
Standby Liquid Control	
Number of accumulators	2
B10 enrichment, %	94
Capacity per accumulator, m^3	7.8
Initial flow rate per accumulator, m^3/h	66

* At runout

Reactor Vessel and Internals	
Reactor Vessel	
Material	Low-alloy steel/ stainless and Ni-Cr-Fe alloy clad
Design pressure, MPag	8.62
Inside diameter, m	7.1
Inside height, m	27.6
Steam Separators and Dryers	
Separator type	AS-2B
Number of separators	379
Dryer type	Chevron
Main Steam	
Number of steam lines	4
Diameter of steam lines, cm	70
Number of safety/relief valves	18
Number of depressurization valves	8
Isolation Condenser	
Number of loops	4
Capacity of each loop, MWt	34
Number of safety/relief valves	18

Containment	
Primary	
Type	Pressure suppression
Construction	Reinforced concrete with steel liner
Drywell	Concrete cylinder
Wetwell	Concrete cylinder
Design pressure, MPa	0.31
Design leak rate, % free volume/day	0.5*
Drywell free volume, m^3	7206
Wetwell free volume, m^3	5467
Suppression pool water volume, m^3	4383
Number of vertical vents	10
Vertical vent diameter, m	1.2
Number of horizontal vents/vertical vent	3
Horizontal vent diameter, m	0.7
Reactor Building	
Type	Low leakage
Construction	Reinforced concrete/steel
Design in leakage rate at 6.4 mm water, %/day	100

* Excluding MSIV leakage

Auxiliary Systems	
Reactor Water Cleanup/Shutdown Cooling	
Number of trains	2
Number of pumps per train high/low capacity	1/1
Type	Canned rotor
Flow rate per train (cleanup mode), m^3/h/ % of feedwater	116/1
No. of regenerative heat exchangers per train	2
No. of non-regenerative heat exchangers per train	3
Return water temperature (cleanup mode), °C	227
Flow rate (shutdown mode), m^3/h	1365
Heat removal duty (shutdown), MWt	55
Fuel and Auxiliary Pools Cooling	
Number of trains	2
Number of pumps/train	1
Flow rate per pump, m^3/h	250
Number of heat exchangers/train	1
Total heat removal capability, MWt	4.0
Backup LPCI flow/train, m^3/h	454
Reactor Component Cooling Water	
Number of trains	2
Capacity of each train, %	100
Number of pumps per train	3
Number of heat exchangers per train	3
Flow rate per loop (normal), m^3/h	1250
Heat removal duty (normal), MWt	31
Flow rate per loop (shutdown), m^3/h	2500
Heat removal duty (shutdown), MWt	86
Plant Service Water	
Number of trains	2
Capacity of each train, %	100
Number of pumps per train	2
Flow rate per loop (normal), m^3/h	9085
Drywell Cooling	
Number of trains	2
Flow rate per train, m^3/h	72800
Number of fans per train	4
Heat removal duty per train, MWt	1.78

6. PASSIVE BWR SIMULATOR MODEL DESCRIPTIONS

This Section provides a model description of the Passive Boiling Water Reactor (BWR) Simulator presented in this Manual.

The Passive BWR model is comprised of :

(1) Process modeling based on approximations of lumped and distributed parameters to consider:

- 3 D spatial reactor neutronics;
- Natural circulation thermal-hydraulic processes in the core;
- Passive ECC system model;
- Balance of Plants models;

(2) Simplified logic modeling for Control Systems for NSSS and BOP systems.

The main features of the model are:

- The main components of the model are: vessel dome, downcomer, lower plenum, core (channel and fuel), upper plenum, pressure and level controls.
- The vessel dome, downcomers, recirculation loops and neutron process models are based on lumped parameter approximations.
- Further consideration of the model is the natural-circulation path in the internal circuit of the reactor, which governs the dynamic simulation performance of the Passive BWR plant. The multiple parallel channels in core are lumped into eight average channels. The drift-flux model for two-phase is used to consider velocity difference between liquid- and vapour phases.
- Fuel temperature model is based on distributed parameter approximations. A multi-node fuel pin model is developed to describe the heat transfer process. Three regions are considered in the heat transfer analysis: the first region corresponds to heat transfer in the fuel; the second region corresponds to heat transfer in the gap and the third region corresponds to heat transfer in the clad, whose temperatures are determined by the rate of heat convection due to core flow.
- Reactor power is calculated from a three-dimensional spatial reactor kinetics model with 12 reactor nodes. Each reactor node is a point kinetic model with six groups of delayed neutrons. The reactivity of each reactor node is a function of localized Doppler effect, void fraction, moderator temperature and control rod reactivity, at that node. Each reactor node has a provision for coupling coefficient which is calculated according to the Avery spatial kinetic method to approximate spatial reactor behavior.
- Passive ECC system models and containment model are simplified models, with intent to demonstrate the passive safety features of the ECC systems.
- Balance of Plant models are also simplified models, with intent to demonstrate BOP coupling effects to nuclear system models.

To demonstrate the applicability of the model, thermal-hydraulic expert’s comments are solicited and incorporated into the model during model development, to the extent possible within the scope of the current development.

6.1 MODELING SYSTEM DESCRIPTION

The model configuration and the flow paths are illustrated in Figure 17. The reactor natural-circulation loop circulates the required coolant flow through the reactor core. The flow within the reactor vessel provides a continuous internal natural-circulation path for a major portion of the core coolant flow. The core flow is taken from the vessel and discharged into the lower core plenum. The coolant water passes along the individual fuel rods inside the fuel channel where it boils and becomes a two-phase steam/water mixture. In the core, the two-phase fluid generates upward flow through the axial steam separators while the steam continues through the dryers and flows directly out through the steam lines into the turbine-generator. The condensate flow is then returned through the feedwater heaters by the condensate-feedwater pumps into the vessel. Finally, the water, which is separated from the steam in the steam separators, flows downward in the periphery of the reactor vessel and mixes with the incoming main feed flow from the turbine.

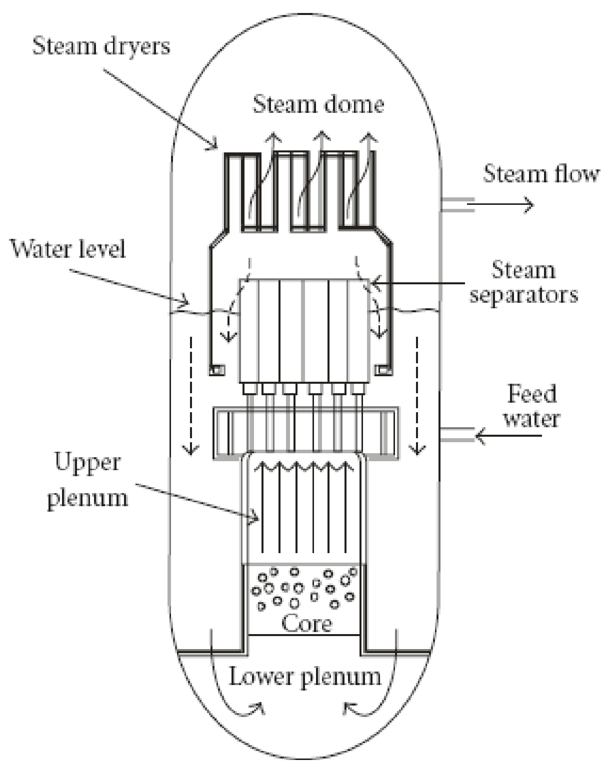

Figure 17 – components and flow paths for the reactor core

6.2 MATHEMATICAL MODEL – NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

Figure 18 is a schematic diagram of the core model where the arrangement of the computational cells of the core model is shown. The reactor vessel is divided into five nodes. Two of these nodes, the vessel dome and the downcomer, have variable volume according to the vessel water level. The three fixed volume nodes are the lower plenum, the upper plenum and steam separators, and the reactor core and chimney. The details for the reactor core model are as follows:

- The reactor core is subdivided into four quadrants radially.
- Each quadrant has three axial levels: lower core, middle core, upper core divisions, to form twelve reactor neutronic nodes.
- The T-H model has eight lumped flow channels, with two channels in each core quadrant. Each flow channel has its own lumped fuel volume, lumped coolant volume, and interfacing variables from the axial reactor nodes (reactor flux), the downcomer and vessel dome models.
- The T-H parameters calculated along each channel's axial length: fuel temperature; coolant temperature; void fraction, provide inputs to the axial reactor node models as noted above.
- The distribution of rod reactivity from the banks of control rods among the 12 reactor nodes is dependent on the control rods position.

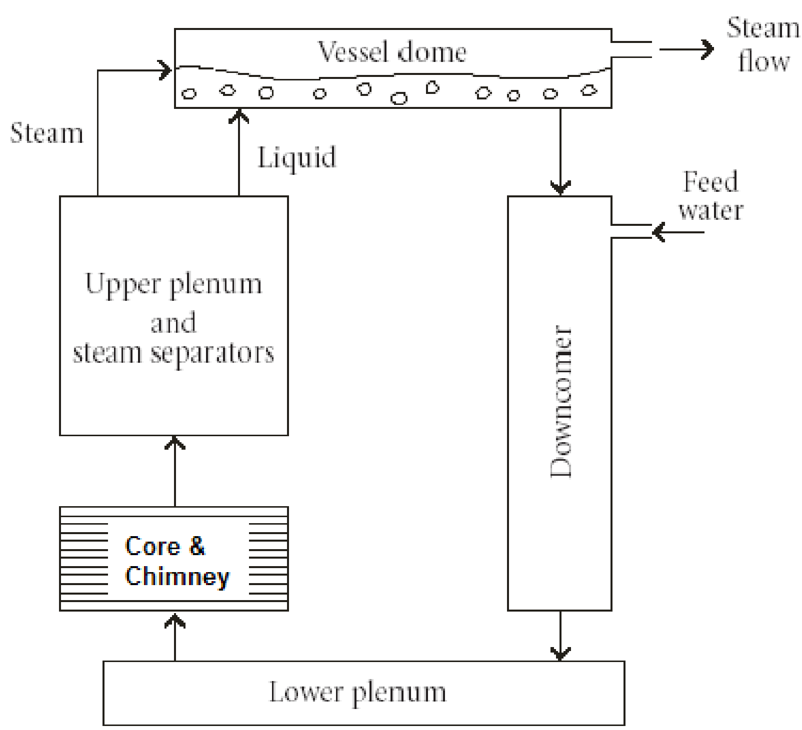

Fig 18 – Model control volumes

6.3 MODEL ASSUMPTIONS

The following assumptions were adopted in the development of the passive BWR NSSS models:

- The point-kinetic equations are used to describe the evolution of neutron population and precursor concentration.
- Delayed neutrons are modeled using six groups.
- Spatial reactor kinetic simulation follows Avery spatial reactor kinetics approximation.
- Fuel rod is described by a distributed parameters model, which includes fuel, gap and cladding.
- The multiple parallel channels in the core are lumped into eight average channels. The drift-flux model for two-phase is used to consider velocity difference between liquid- and vapour phases.
- The axial power profile is modeled as a superposition of the axial profiles from the average flux of the reactor nodes at various axial levels. The radial reactor power profile at designated axial level is modeled as a superposition of the radial (quadrant) flux intensity for the reactor nodes at that axial level, following an assumed cosine flux shape.
- The downcomer section is assumed to be filled with single-phase fluid. The thermalhydraulics system itself consists of five different sections: lower plenum, core, upper plenum and steam separators, and downcomer. The heated section is the core and the others are unheated sections.
- The gas-phase is saturated at the vessel pressure.

6.4 REACTOR POWER MODEL

6.4.1 Three-Dimensional Reactor Spatial Kinetic Model

The reactor neutronic model for the Passive PWR simulator is a three dimensional spatial kinetic reactor model using nodal approach based on Avery's coupled region kinetics theory (Modeling reference #1).

The reactor core is divided into a number of nodes axially and radially. The usual considerations for the choice of the nodes are the core symmetry and the accuracy required in the description of neutron flux distributions, and the execution time of the nodal kinetic model.

For this simulator, the Passive BWR reactor core is divided into 12 nodes: 4 nodes in the upper core; 4 nodes in the middle core; 4 nodes in the lower core. Each node represents a quadrant of the cross section of the core at lower, middle and upper regions.

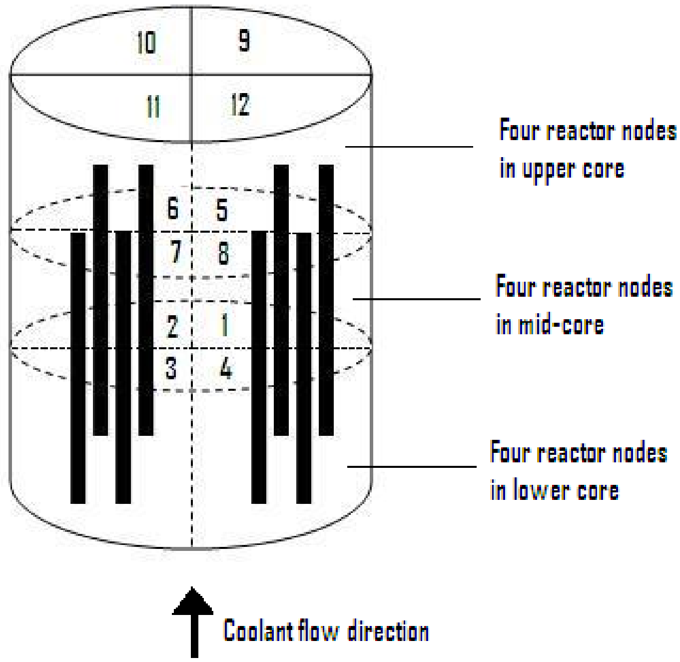

Figure 19 : 3-D Spatial Reactor Kinetic Model Diagram

The essential modeling details for the 3-D spatial kinetic model for the passive BWR core are discussed as follows:

(a) 12-point kinetic models are used respectively to simulate the 12 reactor nodes inside the core as per Figure 19 above.

(b) Each reactor node is point kinetic model, which calculates the neutron power based on 6 different neutron delay groups. The net reactivity for the node is based on the balance of reactivity feedback effects versus the reactivity of the control devices.

- The reactivity feedback effects include: nodal reactivity of fuel; nodal reactivity of void fraction at the node, nodal concentration of xenon, nodal fuel temperature (Doppler effect), nodal moderator temperature, nodal reactivity coupling effects (explained below).
- The reactivity of control devices includes: control rods reactivity worth at the reactor node location in question, and nodal reactivity effects due to safety shutdown devices, if applicable.

(c) The 12 reactor nodes are coupled together by the nodal reactivity coupling effects described in Section 3.2.2 - Coupled Reactor Model. Hence, if localized control rod reactivity changes due to selected control rod movement near the reactor node in question, the neutronic power at that node will change. This nodal neutronic power change or called nodal flux gradient, will impact a coupling effect on the neighboring reactor nodes' reactivity. The effect can be quantified by a factor known as "coupling coefficient", defined as the probability of a neutron born in node j (the reactor node in question), producing a fission neutron in node i (neighboring nodes) in the next generation. Henceforth, there are axial coupling effects for reactor nodes arranged axially, as well as radial coupling effects for reactor nodes arranged radially.

6.4.2 Coupled Reactor Model

For each reactor node in Figure 19 nodal configuration, the temporal nodal fluxes are computed by the following nodal kinetic equations using the Avery formulation.

For reactor node i,

$$l_i \frac{dN_i}{dt} = (1-\beta)\sum_{j=1}^{12} K_{ij} N_j - N_i + \sum_{j=1}^{12} K_{ij} \sum_{m=1}^{6} \lambda_m C_{mj} \qquad \text{.....(1)}$$

$$\frac{dC_{mj}}{dt} = \beta_m N_j - \lambda_m C_{mj} \qquad \text{............(2)}$$

Where:

i, j = 1, 2, …….12 (reactor node number)

m = 1, 2, ….6 (delayed neutron group number)

N_i = Neutronic fluxes in Node i, respectively (nodal fluxes)

l_m = Decay constants of the m^{th} delayed neutron group

b = Total delayed neutron fraction

β_m = Delayed neutron fraction of the m^{th} group

Kij = "Coupling coefficient" determining the probability of a neutron born in node j producing a fission neutron in node i in the next generation.

$\lambda mCmj$ = Partial power of node j contributed from the m^{th} delayed neutron group.

Cmj = Concentration of delayed neutron group m in node j

l_i = Mean neutron life time

Equation (1) can be rewritten by regrouping the coupling coefficients for node i,

$$\frac{dN_i}{dt} = \left\{(1-\beta)K_{ii} - 1\right\}\frac{N_i}{l_i} + \frac{K_{ii}}{l_i}\sum_{m=1}^{6}\lambda_m C_{mi} +$$

$$\frac{(1-\beta)}{l_i}\sum_{j=1, j\neq i}^{12} K_{ij} N_j + \frac{1}{l_i}\sum_{j=1, j\neq i}^{12} K_{ij}\sum_{m=1}^{6}\lambda_m C_{mj}$$

$$i = 1,2,3.....12 \qquad \text{...(3)}$$

The above respective terms represent the various contributions of neutronic flux changes in node i from the following sources:

(a)

$$\left\{(1-\beta)K_{ii}-1\right\}\frac{N_i}{l_i}$$

is the rate of neutronic flux changes in node i due to the node reactivity.

(b)

$$\frac{K_{ii}}{l_i}\sum_{m=1}^{6}\lambda_m C_{mi}$$

is the rate of neutronic flux changes in node i due to its concentration of delayed neutron groups.

(c)

$$\frac{(1-\beta)}{l_i}\sum_{j=1,j\neq i}^{12}K_{ij}N_j$$

is the rate of neutronic flux changes in node i due to the coupling effects of the neutronic fluxes in the other 11 nodes.

(d)

$$\frac{1}{l_i}\sum_{j=1,j\neq i}^{12}K_{ij}\sum_{m=1}^{6}\lambda_m C_{mj}$$

is the rate of neutronic flux changes in node i due to coupling effects from the concentration of delayed neutron groups in the other 11 nodes.

By introducing the definition of reactivity ΔKi = (Kii -1)/Kii, equation (3) can be written as:

$$\frac{dN_i}{dt}=\frac{(\Delta K_i-\beta)}{\Lambda_i}N_i+\sum_{m=1}^{6}\lambda^{*}_m C_{mi}+\alpha_i\sum_{j=1,j\neq i}^{12}K_{ij}N_j$$
$$\frac{1}{l_i}\sum_{j=1,j\neq i}^{12}K_{ij}\sum_{m=1}^{6}\lambda_m C_{mj} \qquad \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots...(4)$$

Where

$$\Lambda_i = \frac{l_i}{K_{ii}}$$

$$\alpha_i = \frac{(1-\beta)}{l_i}$$

$$\lambda_m^* = \frac{\lambda_m}{l_i}$$

Equation (4) is almost identical to the point kinetic model for reactor node i, with the exception of an extra node coupling source terms:

$$\alpha_i \sum_{j=1, j\neq i}^{12} K_{ij} N_j \quad \text{and} \quad \frac{1}{l_i} \sum_{j=1, j\neq i}^{12} K_{ij} \sum_{m=1}^{6} \lambda_m C_{mj}$$

The "node coupling effects" can be integrated into "the node i reactivity term Δρii" and the point kinetic equations for node i can be written:

$$\frac{dN_i}{dt} = \frac{(\Delta\rho_{ii} + \sum_{j=1, j\neq i}^{12} \Delta\rho_{ij} - \beta)}{\Lambda_i} N_i + \sum_{m=1}^{6} \lambda_m^* C_{mi} \quad \text{.....…….......(5)}$$

Where

Δρii is the node i reactivity change.

Δρij is the reactivity change for node i due to coupling effects in node j

Equation (4) and (5) will be identical if

For node 1 & 2, $$\frac{\Delta\rho_{12}\cdot N_1}{\Lambda_1} = \alpha_1 K_{12} N_2 + \frac{1}{l_1}(K_{12}\sum_{\substack{NODE2\\ m=1}}^{6}\lambda_m C_m)$$

For node 1 & 3, $$\frac{\Delta\rho_{13}\cdot N_1}{\Lambda_1} = \alpha_1 K_{13} N_3 + \frac{1}{l_1}(K_{13}\sum_{\substack{NODE3\\ m=1}}^{6}\lambda_m C_m)$$

For node 1 & 4, $$\frac{\Delta\rho_{14}\cdot N_1}{\Lambda_1} = \alpha_1 K_{14} N_4 + \frac{1}{l_1}(K_{14}\sum_{\substack{NODE4\\ m=1}}^{6}\lambda_m C_m)$$

For node 1 & 12, $$\frac{\Delta\rho_{1,12}.N_1}{\Lambda_1} = \alpha_1 K_{1,12} N_{12} + \frac{1}{l_1}(K_{1,12}\sum_{\substack{NODE\,12\\ m=1}}^{6}\lambda_m C_m)$$

Repeat these similar equations for all other 11 nodes.

Therefore the equation for Δρij, the reactivity change for node i due to coupling effects in node j is:

$$\Delta\rho_{ij} = \Lambda_i K_{ij}\left(\alpha_i \frac{N_j}{N_i} + \frac{1}{l_i}\frac{\sum_{\substack{NODEj\\ m=1}}^{6}\lambda_m C_m}{N_i}\right) \text{...........(6)}$$

It can be seen that equation (6) involves the calculation of coupling coefficient Kij. These coefficients Kij define the probability of a neutron born in node j producing a fission neutron in node i in the next generation. The approximation methods for the coupling coefficients for coupled reactor follow the Avery Approximation Methods as detailed in Modeling Reference #1.

The overall 3-D spatial kinetic model can be diagrammatically represented by the following model diagram (figure 20) for the 12 spatially distributed reactor nodes.

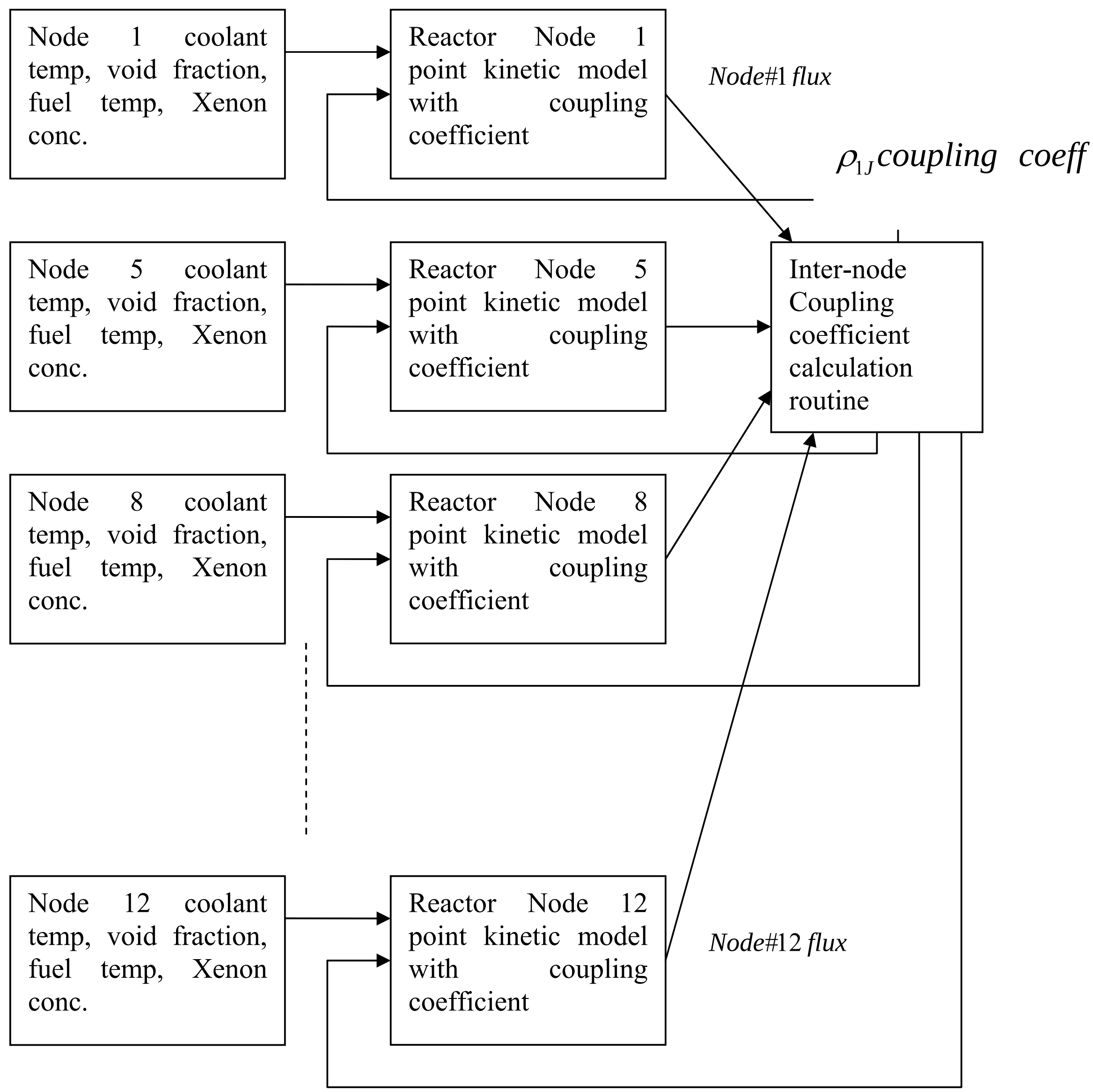

Figure 20 – Spatial Reactor Model Diagram.

The Reactor data used for simulation basis only is as follows:

Reactor Kinetics	
Prompt Neutron Lifetime	2.4 x 10-4 seconds
Effective Delayed Neutron Fraction	0.0072
Effective Delayed Neutron Precursor Decay Constant	0.1 sec-1 (Keff >1), 0.05 sec-1 (Keff <1)
Reactivity Coefficients Note: Core age and fission product poisons are not simulated.	
Moderator Temperature Coefficient (dK/K/deg F)	- 0.8 x 10-4 to -1.5 x 10-4 (70 deg F to 545 deg F)
Moderator Void Coefficient (dK/K/% voids)	-1.2 x 10-3 to -2.4 x 10-3 (10 percent to 80 percent voids)
Fuel Temperature Coefficient (dK/K/deg F)	-1.8 x 10-5 to -0.5 x 10-5 (500 deg F to 3000 deg F and 0% to 80% voids)

6.4.3 Control Rods Model

As discussed above, the reactivity change for each reactor node includes reactivity effects from the banks of "Control" Rods (total 269 control rods). Specific ONLY for this simulator, these control rods are arranged in eight banks (or called gangs), as shown below (figure 21):

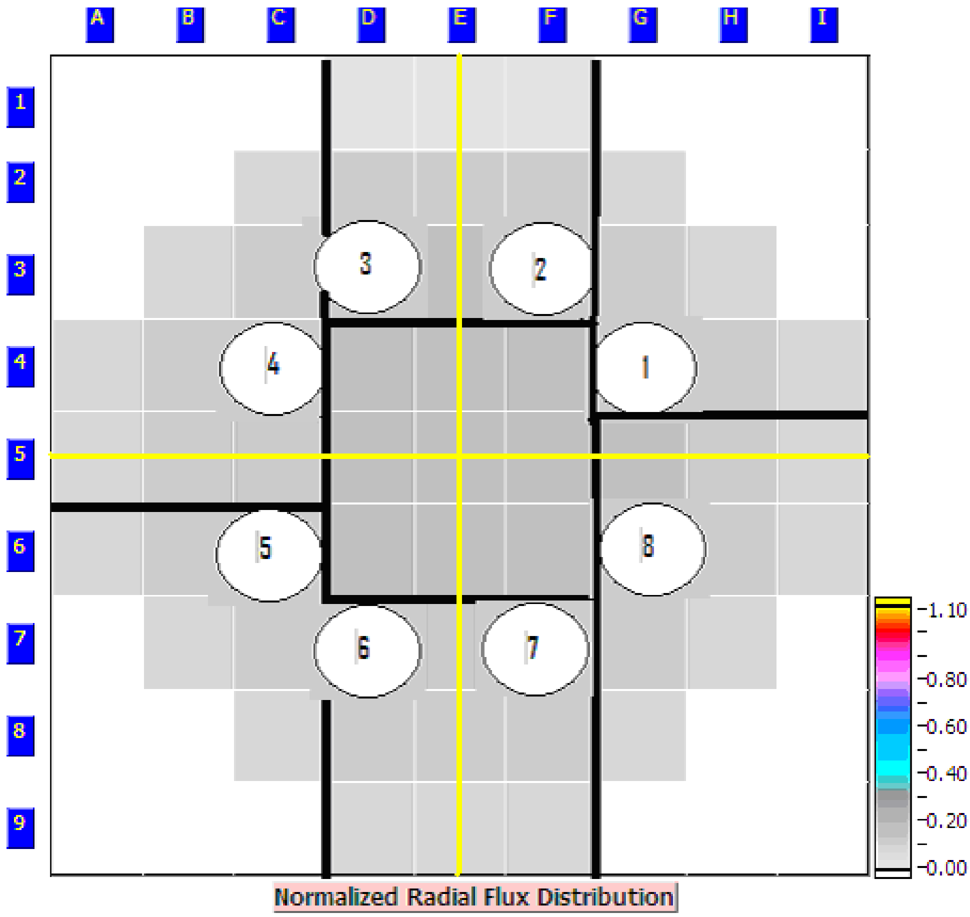

Figure 21 – Control Rods Bank arrangement.

The control rods bank arrangement is:

- Bank #1 and #2 situated in axial node 1, 5 , 9 (1st quadrant)
- Bank #3 and #4 situated in axial node 2, 6, 10 (2nd quadrant)
- Bank #5 and #6 situated in axial node 3, 7 , 11 (3rd quadrant)
- Bank #7 and #8 situated in axial node 4, 8 , 12 (4th quadrant)

The distribution of rod reactivity from the eight banks among the reactor nodes is dependent on the rods position.

- Referring to the figure 22 below, when the control rods are fully inserted, the rods normalized reactivity worth to all the nodes at lower, middle and upper level is the same, 1.

- As the control rods are withdrawn from their fully inserted positions, say 20 % withdrawal from core, the rods normalized reactivity worth to the lower and middle nodes remains at 1, while the normalized reactivity worth to upper nodes drops to 0.9.

- As the control rods are 50 % withdrawn from the core, the rods normalized reactivity worth to the lower drops to 0.9, and for middle nodes, it drops to 0.4. For the upper nodes, it drops to 0.3.

- As the control rods are withdrawn further to 100 % withdrawal from the core, the rods normalized reactivity worth to the lower, middle and upper nodes drops to 0.

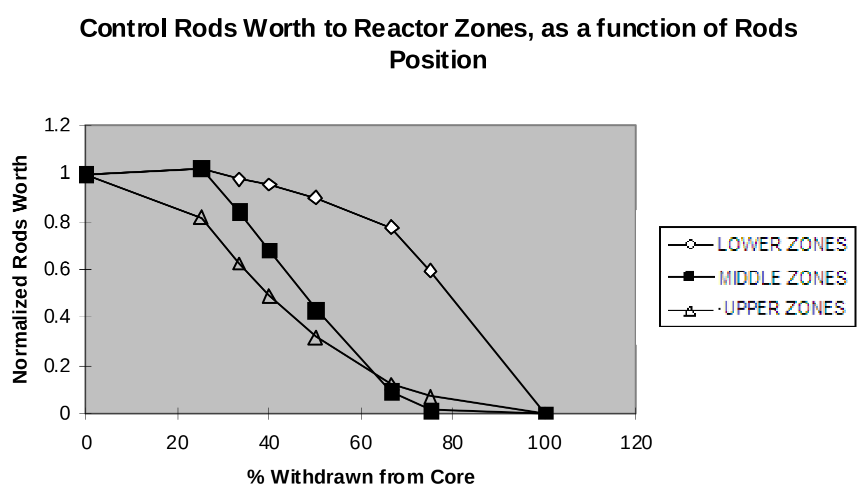

Figure 22 - Control rods worth with respect to reactor nodes at lower, middle and upper level.

- In addition to the above, the control rods reactivity worth has more contribution to the reactor nodes in the same quadrant where the rods banks are located, when compared to reactor nodes in other quadrants. For example, Bank #1 and #2 rods have more influence to axial node 1, 5 , 9 (1st quadrant), than axial nodes 2, 6, 10 (2nd quadrant), and so forth.

6.4.4 Radial and Axial Flux Shapes

- At a given axial level, the radial flux for each the cell below is determined by the reactor fluxes from the 4 quadrant reactor nodes at that particular axial level.
- Within each quadrant, e.g. quadrant #1, the flux intensity for each of the cell in that quadrant (i.e. E1 to E5, F1 to F5, G1 to G5, H1 to H5, I1 to I5) is determined by the flux level of the reactor node at that quadrant (e.g. Node #1). An assumed "cosine-like" flux shape determines the weights for each cell's intensity, with higher intensity for cells at the central region, and with increasingly less intensity for cells near the reactor edge.

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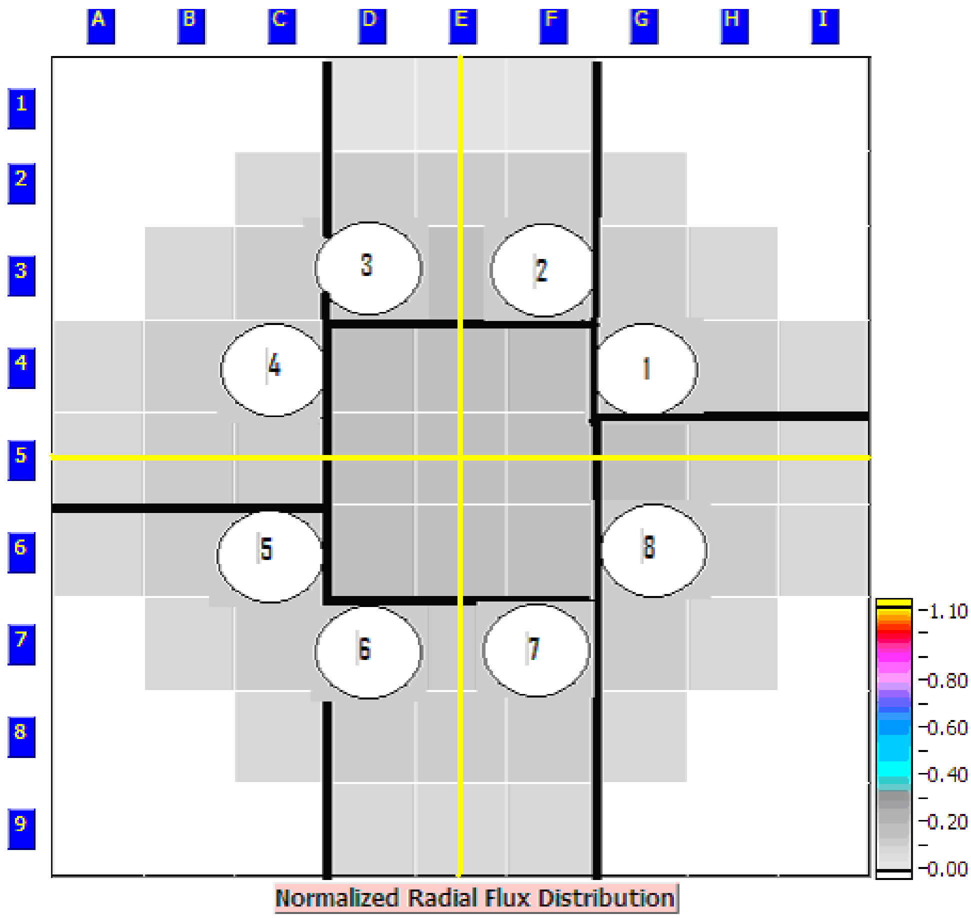
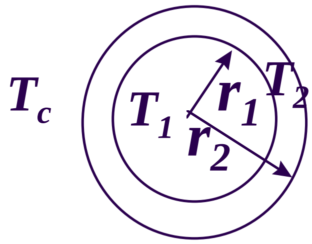


- The axial flux shape along the core's axis is determined respectively by the average fluxes of the reactor nodes at lower, middle and upper core level.

6.5 FUEL MODEL

A lumped parameter technique is used for calculating the heat transfer from UO_2 fuel rods:



Cross-section of a fuel pellet, enclosed by metal fuel clad.
Reactor coolant gets heat transfer from fuel clad.

For fuel elements in a reactor, the transient fuel meat temperature and fuel clad temperature are given by:

$$C_1 \frac{dT_1}{dt} = \dot{Q}_n - \frac{T_1 - T_2}{R_1} \text{ ……………(7)}$$

$$C_2 \frac{dT_2}{dt} = \frac{T_1 - T_2}{R_1} - \frac{T_2 - T_c}{R_2} \text{ …………(8)}$$

Where

$\dot{Q}_n$ = nuclear heating of 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 is UO_2 thermal conductivity;

h_g is 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 fuel pellet including the clad.

Note: the above fuel pin model is incorporated into each of the eight flow channels as noted above.

6.6 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 3 separate decay product groups exist, each with a different decay time constant.

$$P = N_{FLUX} - \sum_{i=1}^{3} (\gamma_i * N_{FLUX} - D_i) \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots(9)$$

$$\frac{dD_i}{dt} = \lambda_i \; * (\gamma_i * N_{FLUX} - D_i) \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots(10)$$

Where

P = thermal power released from fuel (normalized)

N_{FLUX} = neutron flux (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 decay heat calculation is used, as input - reactor power P, by the “Fuel Heat Transfer to Coolant” module to calculate respective coolant temperature and fuel temperature in the core.

Note: the above decay heat model is incorporated into each of the 12 reactor nodes as noted above.

6.7 THERMALHYDRAULIC MODEL

The heat generated in the fuel rod is conducted and convected to the coolant in the flow channel. The single phase coolant enters the bottom of the channel with a velocity $v_{inlet}(t)$ and temperature T_{inlet}, and then starts boiling at a certain level—called the boiling boundary $\mu(t)$— at which the coolant reaches the saturation temperature T_{sat}.

Above the boiling boundary, the coolant is a mixture of two phases, i.e. water and steam. The flow channel is accordingly divided into two regions, the single-phase and two-phase regions.

The three-dimensional mass, energy, and momentum equations in the single-phase and two-phase regions, that describe the fluid mechanics in the channel, are averaged over the cross-section of the flow channel to arrive at equations that depend only on a single spatial variable (axial position z) and time.

After the full three-dimensional mass, momentum and energy equations have been sectionally averaged, there result one-dimensional axial and time-dependent partial differential equations (PDEs). The system of equations using drift flux model can be summarized below:

- 3 equations for mass, momentum and energy continuity in the single-phase region, and ;
- 3 equations for mass, momentum and energy continuity in each of the two phases of the two-phase regions, i.e. a total of 6 equations in the two phases.
- 3 constitutive relationships for the mass, momentum and energy transfer between the two phases are needed to solve the 6 equations in the two-phase region.
- Additional approximations are made to reduce the 6 equations to 3 equations in the two-phase region by assuming a drift velocity V_{gj} between the two phases, assuming a thermal equilibrium (i.e. both phases are at the saturation temperature at the prevailing pressure), and allowing for the concentration parameter C_o, which represents the global effect due to the radially nonuniform void distribution and the velocity profiles, to be different from 1.

Weighted Residual Approximation:

In order to reduce the space- and time-dependent PDEs to time-dependent ODEs, a weighted residuals procedure is used with *reasonable* quadratic spatial approximations for the space- and time dependent single-phase enthalpy and the two-phase quality (Reference #2, #3).

With this approximation, the equation sets used in the reduced T-H ODEs model are derived from Reference #2, #3 and summarized below.

6.7.1 Summary of T-H System Equations

Since the thermal-hydraulic model has been described in detail in Reference #2 (HEM model) and #3 (Drift Flux Model extension), only the final ODEs are presented here.

(a) Single Phase Region

For the single phase region, the time-dependent, spatially quadratic distribution for the enthalpy is approximated as proposed in Reference #2 :

$$h(z,t) \cong h_{inlet} + a_1(t)z + a_2(t)z^2 \quad \text{……….(11)}$$

Where z, t are dimensionless distance and time as defined in Appendix of Reference #2, and a_1 and a_2 are weighted residual approximation coefficients.

(b) Two Phase Region

In the two phase region, the quadratic dependence of the quality in the axial direction is approximated as proposed in Reference #2 :

$$x(z,t) \cong N_\rho N_r [s_1(t)(z-\mu(t)) + s_2(t)(z-\mu(t))^2] \quad \text{…(12)}$$

where

$$N_\rho = \frac{\rho_g^*}{\rho_f^*} \qquad N_r = \frac{\rho_f^*}{\rho_f^* - \rho_g^*}$$

s_1 and s_2 are weighted residual approximation coefficients.

The drift flux relationship between the void fraction and the equilibrium quality x(z,t) can be written as a sum of the void fraction due to HEM model and a correction term (Reference #3):

$$\alpha(z,t) = \frac{1}{C_0}[\alpha_{HEM}(z,t) - V_{gj}\alpha_{CORR}(z,t)] \quad \text{….(13)}$$

Where C_0 is the void distribution parameter; V_{gj} is drift velocity. The void distribution parameter C_o and the average drift velocity V_{gj} are represented by a correlation for each of the two-phase flow regimes, bubbly and slug in vertical pipes.

Where

$$\alpha_{HEM}(z,t) = \frac{x(z,t)N_r}{x(z,t) + N_\rho N_r} \quad \text{……….(14)}$$

$$\alpha_{CORR}(z,t) = \frac{x(z,t)N_r}{(x(z,t)+N_\rho N_r)(C_0 j(z,t)+V_{gj})}$$

where j is the superficial velocity …………………….(15)

(c) Convective heat transfer coefficients

- The convective heat transfer coefficient for the single-phase liquid and vapour is obtained from the classical Dittus-Boelter correlation.
- The heat transfer coefficient in both subcooled boiling and nucleate boiling regimes is calculated with the Chen correlation.

(d) Reduced Order ODEs

In summary, for each flow channel, five coupled time-dependent ODEs result from the integration of the one-dimensional time-dependent continuity, energy and momentum equations in the single and two-phase regions, using the weighted residuals procedure as detailed above (Reference #2, 3):

$$\frac{da_1(t)}{dt} = \frac{6}{\mu(t)}[N_\rho N_r N_{pch,1\phi}(t) - v_{inlet}(t)a_1(t)] - 2v_{inlet}(t)a_2(t)$$

$$\frac{da_2(t)}{dt} = \frac{6}{\mu^2(t)}[N_\rho N_r N_{pch,1\phi}(t) - v_{inlet}(t)a_1(t)]$$

$$\frac{ds_1(t)}{dt} = \frac{1}{ff_5(t)}[ff_1(t)\frac{d\mu(t)}{dt} + ff_2(t)\frac{dv_{inlet}(t)}{dt} + ff_3(t)\frac{dN_{pch,2\phi}(t)}{dt} + ff_4(t)]$$

$$\frac{ds_2(t)}{dt} = \frac{1}{ff_{10}(t)}[ff_6(t)\frac{d\mu(t)}{dt} + ff_7(t)\frac{dv_{inlet}(t)}{dt} + ff_8(t)\frac{dN_{pch,2\phi}(t)}{dt} + ff_9(t)]$$

$$\frac{dv_{inlet}(t)}{dt} = \frac{1}{ff_{14}(t)}[ff_{11}(t)\frac{d\mu(t)}{dt} + ff_{12}(t)\frac{dN_{pch,2\phi}(t)}{dt} + ff_{13}(t)]$$

………..(16)

where a_1(t) and a_2(t) are the coefficients of the linear and quadratic terms for liquid enthalpy profile in the axial direction along the channel up to the boiling boundary.

s_1(t) and s_2(t) are the coefficients of the linear and quadratic terms for the quality profile in the axial direction along the channel in the two phase region.

The expression ff_n (t) , n = 1, …14 are complicated intermediate quantities, which depend on the phase variables, the operating parameters and the design parameters (Reference #3).

The boiling boundary is determined by $$\mu(t) = \frac{2N_\rho N_r N_{sub}}{a_1(t) + \sqrt{a_1^2(t) + 4a_2(t)N_\rho N_r N_{sub}}}$$

……………………………(17)

$N_{pch,1\phi}$ is phase change number for single phase, which is proportional to the fuel bundle wall heat flux, and is directed related to the surface temperature of the fuel rod.

N_{sub} is the sub-cooling number $$N_{sub} = \frac{(h^*_{sat} - h^*_{inlet})(\rho^*_f - \rho^*_g)}{(h^*_g - h^*_f)\rho^*_g} \qquad \text{………………………}(18)$$

Summary:

- A simulation module is created for the above 5 coupled ODEs, representing the dynamic thermal-hydraulic of a flow channel: enthalpy (z,t), void fractions (z,t).
- This module has inputs from axial reactor nodes: flux, and from the Control Rod Model: reactivity worth.
- The output for this module calculates coolant enthalpy (z,t), void fractions (z,t).
- This module is repeated 7 times to simulate the 7 flow channels.

6.8 VESSEL DOME MODEL

The vessel dome is modelled as a two-region volume, one region being liquid and the other vapour. The two regions are assumed to be at the same pressure but not necessarily at the same temperature. The dynamic model used to obtain the pressure in the vessel dome is based on balances of mass and energy:

$$\frac{dP}{dt} = -\frac{\upsilon_f \sum_j W_{jf} + (\frac{\partial \upsilon_f}{\partial h_f})_p \sum_j W_{jf}(h_{jf} - h_f)}{Q + Q^{\wedge}} + \frac{\upsilon_g \sum_j W_{jg} + (\frac{\partial \upsilon_g}{\partial h_g})_p \sum_j W_{jg}(h_{jg} - h_g)}{Q + Q^{\wedge}}$$

…………(19)

where

$$Q = m_f [\upsilon_f (\frac{\partial \upsilon_f}{\partial h_f})_p + (\frac{\partial \upsilon_f}{\partial p})_{h_f}] \qquad Q^{\wedge} = m_g [\upsilon_g (\frac{\partial \upsilon_g}{\partial h_g})_p + (\frac{\partial \upsilon_g}{\partial p})_{hg}]$$

……….(20)

- W_{jf} and W_{jg} are the mass flow rates into or out of the liquid and vapour regions respectively,
- h_{jf} and h_{jg} are the enthalpy of the fluid entering or leaving the liquid and vapour regions, respectively.
- The mass of liquid , mass of vapour, enthalpy of liquid and enthalpy of vapour are given by the following balance equations:

$$\frac{dm_f}{dt} = \sum_j W_{jf}$$

$$\frac{dm_g}{dt} = \sum_j W_{jg}$$

…………………….………….(21)

$$\frac{dh_f}{dt} = \frac{1}{m_f} \sum_{jf} W_{jf}(h_{jf} - h_f) + \upsilon_f \frac{dp}{dt}$$

$$\frac{dh_g}{dt} = \frac{1}{m_g} \sum_{jg} W_{jg}(h_{jg} - h_g) + \upsilon_g \frac{dp}{dt}$$

…………………...……………(22)

- The flashing and condensation flows are given by:

$$W_{flash} = \frac{dm_f}{dt}[\frac{h_f - h_{fsat}}{h_{gsat} - h_{fsat}}] \quad \text{if } h_f > h_{fsat}$$

$$W_{cond} = \frac{dm_g}{dt}[\frac{h_{gsat} - h_g}{h_{gsat} - h_{fsat}}] \quad \text{if } h_{gsat} > h_g$$

……….……..(23)

- The density and temperature in the downcomer is calculated by the functional relationship with pressure and liquid enthalpy, as determined by the steam table functions.

- The level in the vessel is calculated as a function of liquid volume, i.e., where the liquid volume is given by

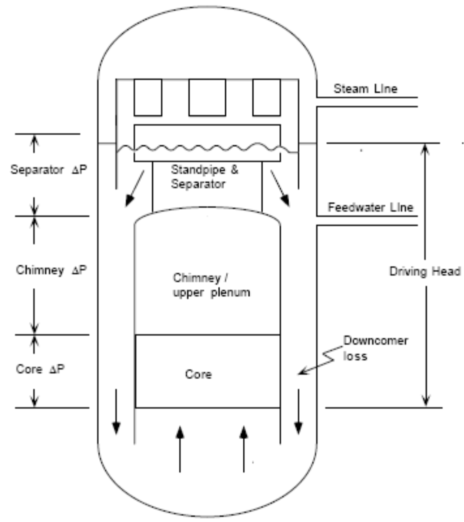
$$V_f = \frac{m_f}{\rho_f}$$

………………….(24)

6.9 NATURAL CIRCULATION LOOP

Natural Circulation Flow

- Core flow depends on
  > driving head
  > losses through the loop
- Driving head
  > proportional to core + chimney height
    - Void Fraction
- Loop losses
  > downcomer
    - Single-phase Δp, handbook loss coefficient
  > core (fuel bundle)
    - Two-phase Δp, test data/correlation
  > chimney ~ small
  > separator
    - Two-phase Δp, test data/correlation



Schematic of Flow and Pressure Drops in a Reactor

Balancing the gravity head (driving head) available and total loop pressure drop obtained by integration of the momentum balance leads to the model for the natural circulation.

The natural circulation model includes the pressure drops and flows from the downcomer, lower and upper plenum, reactor core and steam separators, in order to obtain the following momentum balance:

$$\frac{dW_{circ}}{dt} = \left(\frac{1}{\sum_{i=1}^{n} \frac{l_i}{A_i}}\right)\left(-K_{psn}\frac{W_{circ}^2}{\rho_{dw}} - K_{sep}\frac{W_{sep}^2}{\rho_{sep}} - \Delta p_{core} + \Delta p_{gravity}\right)$$

………………(25)

where

- $\sum_{i=1}^{n} \frac{l_i}{A_i}$ is the inertial term.
- ρ_{dw} is the downcomer density; ρ_{sep} is the steam-separator density; *Wsep* is the flow mass through the steam separator.
- *Kpsn* is the support core plate loss coefficient; *Ksep* is the separator loss coefficient, Δ Pcore is the core pressure drop, and Δ Pgravity is the pressure drop due to gravity.

- The total core pressure drop is the sum of the frictional, acceleration and gravitational components.

6.10 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:

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

……….(26)

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

$$\frac{dh_{fw}}{dt} = \frac{h_{fwh} - h_{fw}}{\tau} \qquad \text{…………(27)}$$

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.

6.11 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,

$$\frac{dM_h}{dt} = W_{DOME} - (W_T + W_D + W_B) \quad \text{……..…………….(28)}$$

$$\frac{dU_h}{dt} = W_{DOME}.h_{DOME} - (W_T + W_D + W_B).h_h \quad \text{…..………(29)}$$

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:

$$v_h = \frac{V_h}{M_h} \qquad \text{…………..(30)}$$

$$u_h = \frac{U_h}{M_h} \qquad \text{………….(31)}$$

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

$$P = f(v_h,\ u_h) \qquad \text{…………….(32)}$$

The flow between the reactor dome and the main steam system has the following form:

$$P_{DOME} - P_h = K_V \frac{1}{2}\frac{W|W|}{\rho_h A_V^2} + K_{NZ} \frac{1}{2}\frac{W|W|}{\rho_h A_{NZ}^2} \quad \text{……(33)}$$

Where

P_{DOME} = reactor steam dome pressure

P_h = main steam 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 (33) 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.

6.12 PASSIVE ECC SYSTEMS

In contrast to the classical modern BWR designs, circulation inside the passive BWR pressure vessel is gravity driven (natural circulation); there are no pumps driving the recirculation flow. To achieve good natural circulation, the passive BWR has a tall adiabatic chimney, above the core.

In case of a break in the primary coolant system, a mixture of steam and water is blown down, from the pressure vessel into the containment. Emergency core cooling water must first be added to the RPV to keep the core covered. The steam pressurizes the containment, and transports the decay heat from the reactor core to the containment atmosphere.

Consequently, this decay heat must be removed from the containment. The passive BWR uses also gravity or natural circulation-driven, passive safety systems to provide emergency coolant to the core, to keep the core cooled and to remove decay heat from both the primary system and/or the containment.

The main systems performing these tasks, are the Gravity-Driven Cooling System (GDCS), the Isolation Condenser System (ICS), and the Passive Containment Cooling System (PCCS).

Emergency core cooling water is provided to the core by the GDCS. This system consists of three water pools situated above the top of the core, from which makeup coolant can flow by gravity to replenish the coolant lost from the Reactor Pressure Vessel (RPV). However, the GDCS can operate only after depressurization of the RPV; therefore, the passive BWR is equipped with an Automatic Depressurization System that performs this function.

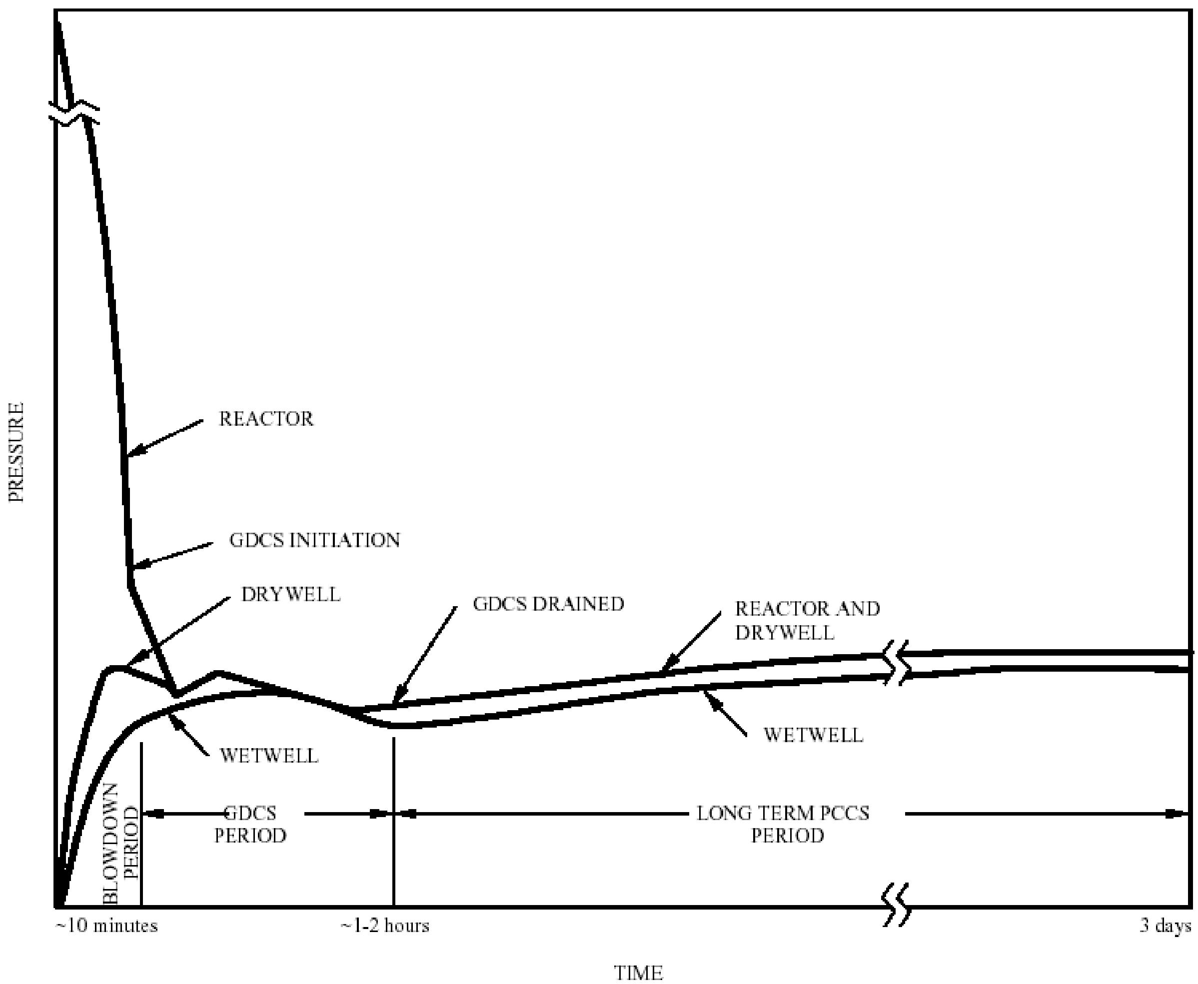
Decay heat removal from the primary system while it is intact or under high pressure is performed by the ICS. The ICS consists of three Isolation Condensers (IC) located in a pool on the top of the reactor building. When redundant condensate return valves are opened, steam from the primary system flows into the tubes of the ICs, condenses, and returns to the RPV, removing stored energy.

Decay heat is removed from the drywell (DW) by the PCCS, which employs three PCC condensers also located in interconnected IC pool compartments on top of the reactor building. The PCC condenser tubes are permanently connected to the DW. A mixture of steam and non-condensible gases (nitrogen present in the containment during normal operation) may enter the PCC condensers. The steam will condense, while the non-condensible gases must be vented to assure proper operation of the condensers. This is accomplished by conveying and venting the non-condensible gases into the suppression pool (SP) in the Suppression Chamber (SC) (or Wetwell)

Since the DW volume is connected directly to the SP either via the main pressure suppression vents or through the PCC condensers and their vent lines, the path that the steam will follow depends on the pressure differences between the DW volume and the two possible venting points.

During the long-term containment cooling period, direct opening of the main vents and condensation of the steam in the SP must be avoided, since the SP is not provided with a safety grade cooling system; the steam must be condensed in the PCC (or IC) condensers and any non-condensibles vented to the SC.

During the low-pressure, long-term decay heat removal period, the decay heat is removed from the core by natural circulation inside the RPV. The steam that is produced is blown into the containment, condenses in the PCCS System, and the condensate returns to the RPV as makeup flow. Low-pressure natural circulation takes place during the long-term decay heat removal period from the containment.



Only a summary of the modeling approach for passive ECC is provided here, because there is intensive use of generic simulation library modules in the CASSIM Simulation System (proprietary software).

For example, in modeling of Isolation Condenser System (ICS), Passive Containment Cooling Systems (PCCS), we make use of the Generic Simulation Algorithm in CASSIM Simulation Software, the STMHTR Algorithm (#117), Steam heater (see below), with modifications:

In modeling the GDCS tank, we make use of the Generic Simulation Algorithm, WTRTANK (#137), Water Tank with Heating, with modifications.

In modeling gravity flow from the GDCS tank, Generic Algorithm #140, CONDUCT_FI, Incompressible Flow Conductance is used.

As noted above, the Isolation Condenser System (ICS) and Passive Containment Cooling System (PCCS) are simplified models. They are not high fidelity models taking into detailed account of the complex feedback between noncondensible holdup, its impact on heat removal, and hence on condenser pressure. However, it should be noted these areas are current research topics for passive safety research at leading universities and institutions.

6.13 CONTAINMENT MODEL

The containment model for the passive BWR plant is identical to the containment model developed for the BWR simulator with Active Safety Systems (see Modeling Reference # 4). Therefore it is not repeated here.

6.14 LOGIC MODEL FOR CONTROL AND PROTECTION SYSTEMS

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

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

6.14.1 Control rods control system

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

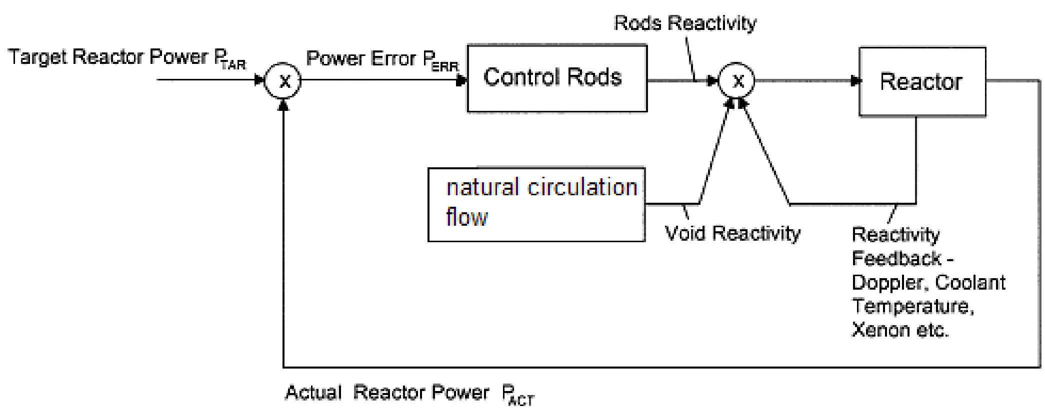


FIG. 8. Simplified block diagram for reactor power control.

- 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 control rods controller.

- 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. The simplified explanation of the rods control logic is:

  Control rods will be moved "in" or "out" automatically in controlling reactor power, ONLY IF the absolute value of power error, ABS(P_{ERR}), is greater than a certain deadband.

6.14.2 Reactor pressure control system

The reactor pressure is automatically controlled to be constant. See detailed description in section " Passive 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.

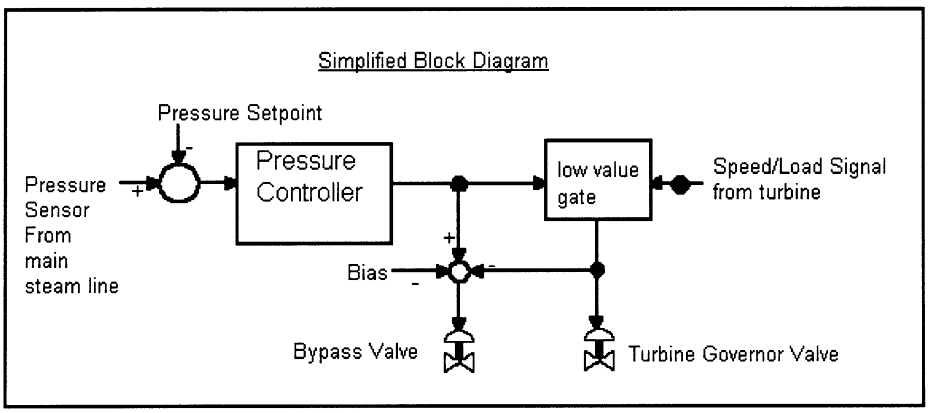


FIG. 9. Simplified reactor pressure controller block diagram.

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

6.14.3 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:

$$M_L = K_{CL} * (e_L + (1/\tau) \int e_L \, dt) \quad \ldots\ldots\ldots\ldots.(34)$$

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.

$$M_{FS} = K_{CF} * (e_{FS} + (1/\tau) \int e_{FS}\, dt) \quad \ldots\ldots\ldots\ldots(35)$$

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 provides the adjusted feedwater flow rate to the steam generators.

6.14.4 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.

6.14.5 Turbine steam bypass control system

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.

6.14.6 Protection system

(1) **Reactor scrams:**

- High neutron flux/low core flow —
  If at any time, the current reactor power exceeds 115 % 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 —L3
  Low level scram SP < 19 m ; (normal level = 20.6 m)
- Reactor pressure high —
  Scram SP > 7870 KPa
- Reactor water level high — L9
  Scram SP > 22.4 m; (normal level = 20.6 m)
- Main steam isolation valve (MSIV) closed - reactor isolated
- Turbine power/load unbalance — loss of line
- Manual scram

(2) **Control rods "Blocked"**

- 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"**

- 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.
- If the feedwater temperature is less than the reference temperature (calibrated at the load setting) by more than 16.7 degrees, the control rods will be inserted into the core to reduce power quickly.
- The "Control Rods Run-in" will be stopped until the power has been reduced to 10% less than the current value.

(4) **Main steam isolation valve (MSIV) closed - Reactor isolated**

- Low steam line pressure < 5,500 KPa
- Low water level L2 = 16.5 m sensed for more than 29 sec.
- High steam flow > 2551 Kg/sec.
- Loss of condenser vacuum for more than 6 sec.

(5) **Isolation Condenser Activation**

- MSIV Closure
- Water level at L2 = 16.5 m
- Water level at L1.5 = 13 m
- Reactor pressure > 7447 KPa

(6) **Passive Emergency core cooling injection**

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.

The sequence of events upon detection of LOCA signal:

1. Reactor will be scrammed, and isolated.
2. MSIV closure initiates ICS activation.
3. LOCA steam vapor and gas mixture in Drywell will enter the Suppression Pool (Wetwell) through vents to relieve Drywell pressure. Drywell pressure will decrease.
4. Reactor pressure is first depressurized through steam condensation at ICS
5. Water inventory at RPV drops due to LOCA. When level drops to L1.5 (13 m), Automatic Depressurization System is activated to open SRVs and DPV to depressurize reactor pressure.
6. The Standby Liquid Control system provides reactor additional liquid inventory in the event of DPV actuation. This function is accomplished by firing squib type injection valves to initiate the SLC system. This system is not modeled.
7. Once the reactor pressure is reduced to near containment pressure, the Gravity Driven Cooling System provides cooling flows to RPV by gravity forces alone (without reliance on active pumps)

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

(1) Reactor protection — control rods blocked; control rods run-in, and reactor scram.
(2) SLC will be activated in the event of ATWS.
(3) Containment isolation — in the event of steam line break, feedwater line break, LOCA.
(4) ECCS actuation — ICS for decay heat removal; PCC to remove steam from drywell; Drywell Vents open to Suppression Pool to allow quick pressure relief for containment, and to discharge non-condensable (e.g. hydrogen) to Wetwell.
(5) ADS actuation to depressuruze reactor pressure to a level to bring in GDCS flows by gravity.
(6) Suppression pool cooling - to provide long term cooling of the isolated containment in case of LOCA.

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ABBREVIATIONS AND ACRONYMS

ADS	Automatic Depressurization System
ALWR	Advanced Light Water Reactor
AOO	Anticipated Operational Occurrences
ARI	Alternate Roc Insertion
ATWS	Anticipated Transients Without Scram
BiMAC	Basemat Internal Melt Arrest and Coolability (Device)
BMP	Basemat Melt Penetration
BWR	Boiling Water Reactor
CBP	Containment Bypass and Leakage
CCFP	Conditional Containment Failure Probability
CCI	Corium-Concrete Interactions
CDF	Core Damage Frequency
CET	Containment Event Tree
C&FS	Condensate and Feedwater System
CLCH	Convection-Limited Containment Heating (model for DCH)
CMS	Containment Monitoring System
COL	Combined Construction and Operating License
COP	Containment Over-Pressurization
COPS	Containment Over-pressure Protection System
CPET	Containment Phenomenological Event Tree
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDH	Control Rod Drive Hydraulic Subsystem
CRSS	Center for Risk Studies and Safety
CSET	Containment Systems Event Tree
CV	Containment Vessel
DBA	Design Basis Accident
DBD	Design Basis Accident
DCD	Design Control Document
DCH	Direct Containment Heating
DPV	Depressurization Valve
DW	Drywell
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineering Safety Features
EVE	Ex-Vessel Steam Explosion
FAPCS	Fuel and Auxiliary Pool Cooling System
FCI	Fuel-Coolant Interaction
FMCRD	Fine Motion Control Drive
FWCS	Feedwater Control System
FWL	Feedwater Line
GDCS	Gravity-Driven Cooling System
GE	General Electric (Company)
GE-NE	GE Nuclear Energy
H2C	Hydrogen Combustion
HCU	Hydraulic Control Unit
HP	High Pressure (core-melt Scenario)
HPME	High Pressure Melt Ejection

I&C	Instrumentation and Control
IC	Isolation Condenser
IDGR	Industry Degraded Core Rulemaking Program
ICS	Isolation Condenser System
IGT	Instrumentation Guide Tube
IVR	In-Vessel Retention (severe accident management scheme)
LDW	Lower Drywell
LOCA	Loss-of-Coolant-Accident
LBB	Leak Before Break
LP	Low Pressure (core-melt Scenario)
LWR	Light Water Reactors
MAAP	Modular Accident Analysis Program
MCCI	Molten Corium-Concrete Interactions
MCOPS	Manual Containment Over-Pressurization System
MCR	Main Control Room
MSIV	Main Steam Isolation Valve
MSL	Main Steam-Line
NBS	Nuclear Boiler System
NRC	Nuclear Regulatory Commission
O&M	Operation and Maintenance
PCC	Passive Containment Cooling
PCCS	Passive Containment Cooling System
PIRT	Phenomena Identification and Ranking Table
PRA	Probabilistic Risk Assessment
PRMS	Process Radiation Monitoring System
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RCCR	Reinforced Concrete Containment Vessel
RCCV	Reinforced Concrete Containment Vessel
RCIS	Rod Control Information System
RCPB	Reactor Coolant Pressure Boundary
ROAAM	Risk-Oriented Accident Analysis Methodology
RPV	Reactor Pressure Vessel
RWCU/SDC	Reactor Water Cleanup/Shutdown Cooling System
SA	Severe Accident
SAM	Severe Accident Management
SAMS	Severe Accident Management Strategy
SAT	Severe Accident Treatment
SBWR	Simplified Boiling Water Reactor
SLC	Standby Liquid Control System
SP	Suppression Pool
SPC	Suppression Pool Cooling
SRV	Safety Relief Valve
SSC	Structure, Systems and Components
SSE	Safe Shutdown Earthquake
TDH	Torispherical Drywell Head
TMSS	Turbine Main Steam System
UCCS	Upper Cylindrical Containment Section
UCSB	University of California, Santa Barbara (CRSS)
UDW	Upper Drywell
URD	Utility Requirement Documents
VB	Vacuum Breaker
WW	Wetwell