PRACTICAL COURSE ON REACTOR INSTRUMENTATION

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1. The TRIGA Mark II Instrumentation

The TRIGA Mark-II reactor was installed by General Atomic (San Diego, California, U.S.A.) in the years 1959 through 1962, and went critical for the first time on March 7, 1962. Operation of the reactor since that time has averaged 220 days per year, without any long outages. The TRIGA-reactor is purely a research reactor of the swimming-pool type that is used for training, research and isotope production (Training, Research, Isotope Production, General Atomic = TRIGA). Throughout the world there are more than 50 TRIGA-reactors in operation, Europe alone accounting for 10 of them.

The TRIGA-reactor Vienna has a maximum continuous power output of 250 kW (thermal). The heat produced is released into a channel of the river Danube via a primary coolant circuit (deionised, distilled water at temperatures between 20 and 40 °C) and a secondary coolant circuit (ground water at temperatures between 12 and 18 °C), the two circuits being separated by a heat exchanger.

The reactor core consists of some 80-fuel elements (3.75 cm in diameter and 72.24 cm in length), which are arranged in an annular lattice. Two fuel elements have thermocouples implemented in the fuel meat, which allow to measure the fuel temperature during reactor operation. At nominal power (250 kW), the centre fuel temperature is about 200 °C. Because of the low reactor power level, the burn-up of the fuel is very small and most of the fuel elements loaded into the core in 1962 are still there. Should these fuel elements ever become unserviceable, they will be sent back to the United States.

Inside the fuel element cladding (aluminium or steel), the fuel is in the form of a uniform mixture of 8 wt% uranium, 1 wt% hydrogen and 91 wt% zirconium, the zirconium-hydride, being the main moderator. Since the moderator has the special property of moderating less efficiently at high temperatures, the TRIGA-reactor Vienna can also be operated in a pulsed mode (with a rapid power rise to 250 MW for roughly 40 milliseconds). The power rise is accompanied by an increase in the maximum neutron flux density from 1×10^{13} $cm^{-2}s^{-1}$ (at 250 kW) to 1×10^{16} $cm^{-2}s^{-1}$ (at 250 MW). This negative temperature coefficient of reactivity, as it is called, brings the power level back to a lower power level. In the so-called pulse mode it is first necessary to operate the reactor on a low power level of about 10 W, than rapidly insert a positive reactivity into the reactor. After insertion of the positive reactivity (maximum 2$ in the case of the TRIGA Vienna) into the reactor operating at 10 W, the power will increase rapidly to about 250 MW and than the power rapidly drops down (duration of pulse is ~40ms) to reach a stabilised low power level depending on the feedback value. The maximal pulse rate is 12 per hour, since the temperature of the fuel elements rises to about 360 °C during the pulse and, therefore, the fuel is subjected to strong thermal stress.
The reactor is controlled by three control rods, which contain boron carbide as absorber material. When these rods are fully inserted into the reactor core, the neutrons continuously emitted from a start-up source (Sb-Be photoneutron source) are absorbed by the rods and the reactor remains sub-critical. If the absorber rods are withdrawn from the core (two of them by an electric motor and one pneumatically, the number of fissions in the core and the power level increases. The start-up process takes roughly one minute for the reactor to reach a power level of 250 kW from the sub-critical state. The reactor can be shut-down either manually or

automatically by the safety system. It takes about 1/10 of a second for the control rods to fall into the core.

The reactor is controlled by four nuclear channels, their signals are displayed both at a colour graphic-monitor and at bar graph indicators.

a) The auto-ranging wide-range channel NM-1000 controls the reactor power from the source level (around 5 mW) up to nominal power of 250 kW. It uses a fission chamber operated in so-called Campbell mode; the signal is controlled by a microprocessor.

b) Two independent linear channels, NMP-Ch and NMP-Ph control the reactor power from the source level up to nominal power. The signals pass over a range switch, which selects the power range. If the signal of one of these two channels exceeds the selected power range for more than 5%, the reactor is shut down automatically. Both channels use compensated ionisation chambers as sensors.

c) For the control of reactor pulse operation an uncompensated ionisation chamber is used. This chamber measures the shape of the reactor pulse, which is displayed on the graphic monitor. Further pulse data like integrated power are calculated from this signal.

In accordance with its purpose as a research reactor, the TRIGA Mark-II is equipped with a number of irradiation devices:

5 reflector irradiation tubes
1 central irradiation tube
1 pneumatic transfer system (transfer time 3 s)
1 fast pneumatic transfer system (transfer time 20 ms)
4 neutron beam holes
1 thermal column
1 neutron radiography facility

In the reflector irradiation tubes 10 containers can be irradiated simultaneously. In the central irradiation tube samples up to 38.4 mm in diameter can be exposed to neutrons at a neutron flux density of 10^{13} $cm^{-2}s^{-1}$, while the pneumatic transfer system allows to transfer the materials to be activated into the reactor from a chemistry laboratory and back again after the required period of irradiation, without the experimentalist having to leave his working place. The four neutron beam tubes permit extraction of neutron beams of all energies into the reactor hall for the purpose of neutron and solid-state physics experiments. The thermal column is used to extract with a thermal spectrum into the reactor hall, unlike the beam holes, the space between the reactor core and the hall is in this case filled with graphite to slow down the neutrons.
The neutron radiography facility is used to investigate components by neutron irradiation similar to X-ray radiography. However, neutrons show especially hydrogen or neutron absorber material in solid matter.

1.1 TECHNICAL DATA

1. REACTOR CORE

fuel-moderator material	8 wt% uranium 91 wt% zirconium 1 wt% hydrogen
uranium enrichment	20% uranium-235
fuel element dimensions	3.75 cm in diameter 72.24 cm in length
cladding	0.76 mm aluminum or 0.51 mm steel
active core volume	max. 49.5 cm diameter, 35.56 cm high
core loading	2.3 kg of uranium-235

2. REFLECTOR

material	graphite with aluminum cladding
radial thickness	30.5 cm
top and bottom thickness	10.2 cm

3. CONSTRUCTION

reactor mounting	heavy and standard concrete 6.55 m high 6.19 m wide 8.76 m long
reactor tank	1.98 m in diameter 6.40 m in depth

4. SHIELDING

radial:	30.5 cm of graphite; 45.7 cm of water and at least 206 cm of heavy concrete
vertical:	above the core 4.90 m of water and 10.2 cm graphite; underneath the core 61.0 cm water, 10.2 cm graphite and at least 91 cm standard concrete.

5. IRRADIATION DEVICES

(1) four beam holes 15.2 cm in diameter
(2) one central irradiation tube (middle of core)
(3) five reflector irradiation tubes
(4) one pneumatic transfer system (near core edge)
(5) a thermal column with cross section 1.22x1.22 m and length 1.68 m
(6) experimental tank with surface area 2.44x2.74 m and depth 3.66 m; connected to the reactor by means of a neutron radiography collimator 0.61x0.61 m in cross section and 1.22 m long.

6. CONTROL SYSTEM

- Two boron carbide control rods with electric motor and rack and pinion drive;
- One boron carbide pulse rod with compressed air drive (5 bars);
- Maximum reactivity insertion rate - time rate of change (excluding pulse operation): 0.04% δk/k per second
- Total rod worth about 4.2% **δ**k/k.

7. CHARACTERISTICS IN CONTINUOUS OPERATION

Thermal power output:	250 kW
Fuel element cooling:	natural convection of the tank water below 100 kW, pump circulation cooling above 100 kW
tank water cooling:	heat exchanger
thermal flux:	$1x10^{13}$ $cm^{-2}s^{-1}$ in the central irradiation tube $1.7x10^{12}$ $cm^{-2}s^{-1}$ in the irradiation tubes
prompt temperature coefficient:	$-1.2x10^{-4}$ **δ**k/k°C
mean prompt neutron lifetime:	$6.0x10^{-5}$ s.

8. CHARACTERISTICS IN TRANSIENT OPERATION

peak power	250 MW
prompt pulse energy yield	10 MW s
prompt pulse lifetime	40 ms
total energy yield	16 MW s
minimal period	10 ms
maximum reactivity insertion	1.4% **δ**k/k = 2$
maximum repetition frequency	12/h
number of fissions during a pulse	$3x10^{17}$
maximum fuel temperature:	
during the pulse	240 °C
9 seconds after the pulse	360 °C

2. Calibration of the nuclear channels

2.1 Introduction:

For the maintenance program of a research reactor a maintenance schedule has to be established which lists all systems and components necessary for a safe reactor operation. These are, however, not only the direct safety related system and components but also auxiliary systems and components. The frequency of maintenance depends on the importance of the components and also on operation experience but it will usually be at least once a year.

The calibration of the nuclear channels of a research reactor should be done once a year. For this calibration, the current of the according ionisation chamber will be simulated, and put into the Data Acquisition Computer (DAC). The strength of the signal will be in the same range as during reactor operation at 10 W, 100 W, 1 kW, till up to 250 kW. At 250 kW the maximum current should be 1 mA. By choosing various current strengths, the electronic components of the DAC can be tested in different power ranges and the whole electronic systems can be tested for ageing, drifting or failures.

The TRIGA Mark II reactor Vienna has four nuclear channels:

wide range, period and logarithmic channel NM 1000:	fission chamber (Campell channel)
Linear channel 1: NMP-CH	compensated ionisation chamber
Linear channel 1: NMP-PH	compensated ionisation chamber
Pulse channel: NPP	uncompensated ionisation chamber

The power level of these channels is shown in Fig. 2.1

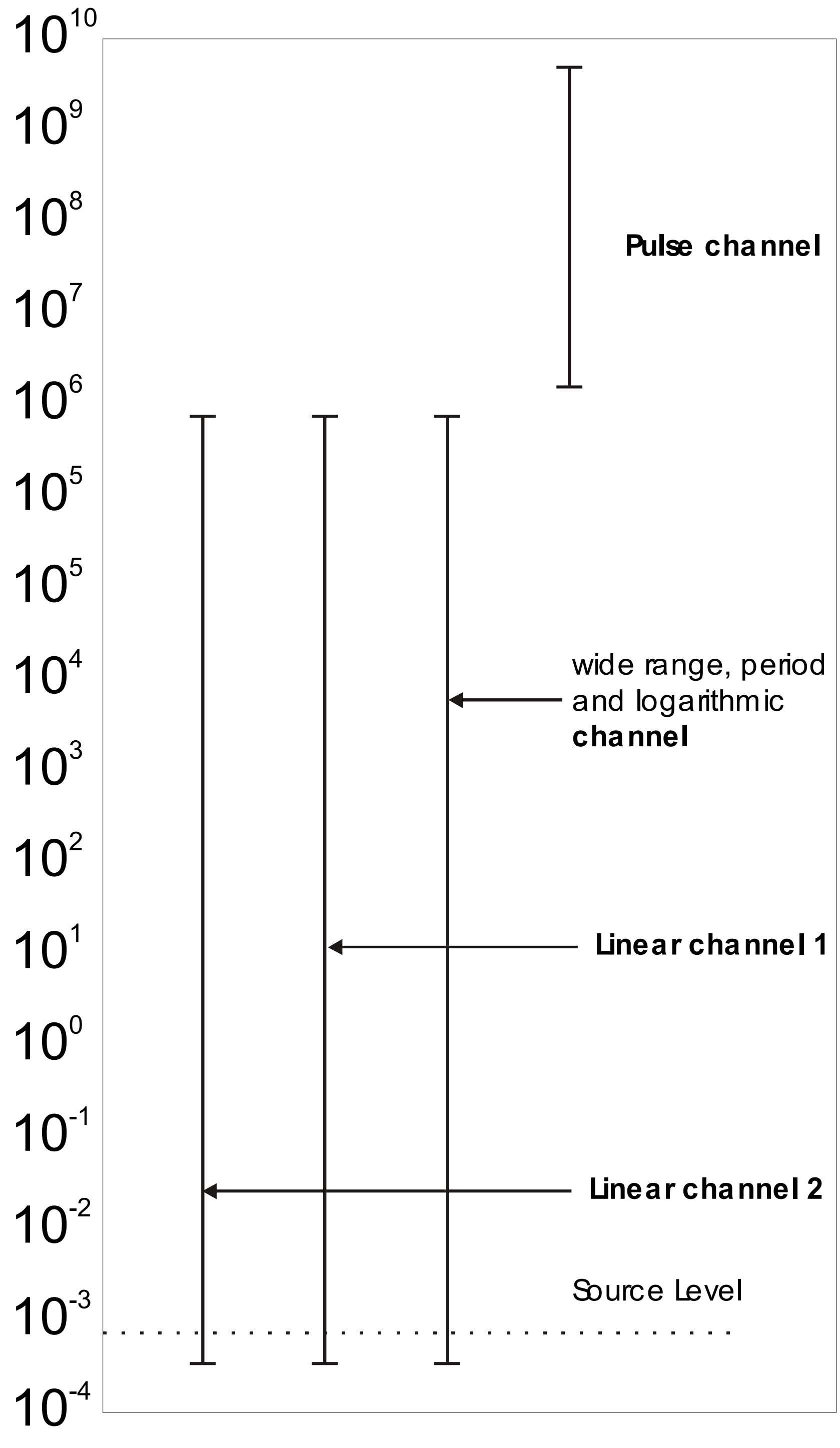

Fig. 2.1 Range of the nuclear channels

As source a Keithley programmable current source is used. With this current source, electric currents from ± 0.5 pA up to ± 101 mA can be chosen. This signal will be applied to the channel which to be tested.

2.1 POWER CALIBRATION AND TEMPERATURE COEFFICIENT OF REACTIVITY

Date: ....
DD MM YY

Reactor power	Transient rod pos.	Shim rod position	Reg. Rod position	Water temp.	Fuel element temp.	$\Delta\rho$ [¢]	ΔT [°C]
10 W							
100 kW							

t [min]	T [°C]	t [min]	T [°C]	t [min]	T [°C]
0		35		70	
5		40		75	
10		45		80	
15		50		85	
20		55		90	
25		60		95	
30		65		100	

Temperature coefficient: $\frac{\Delta\rho}{\Delta T}$ = ...[¢.°C^{-1}]

Reactor power: $P[kW] = \frac{\Delta T[°C] \text{ during one hour}}{5.19°C} \cdot 100 =$

2.2 NUCLEAR CHANNELS LINEARITY CHECK

Date: ....
DD MM YY

Reactor Power 100 kW

NM 1000	Graphic monitor		Status Window	Bar Graph
	Linear:	kW	W	no entry
	% Log P	%	%	%
	% Power	%	%	%
NMP CH %1		%	%	%
NMP CH %2		%	%	%
		NMP CH - PH Power check	%	-----

2.3 NUCLEAR CHANNELS LINEARITY CHECK NMP-Ph

Date: ....
DD MM YY

Current in [A]	Position of Range Switch in [kW/W]	Instrument	Display Graphic-monitor	Display Status Window	Display Bar Graph
1.10^{-3}	250				
4.10^{-4}	250				
1.10^{-4}	250				
1.10^{-4}	25				
4.10^{-5}	25				
1.10^{-5}	25				
1.10^{-5}	2,5				
4.10^{-6}	2,5				
1.10^{-6}	2,5				
1.10^{-6}	0,25				
4.10^{-7}	0,25				
1.10^{-7}	0,25				
1.10^{-7}	25 W				
4.10^{-8}	25 W				
1.10^{-8}	25 W				
1.10^{-8}	2,5 W				
4.10^{-9}	2,5 W				
1.10^{-9}	2,5 W				
1.10^{-9}	250 mW				
4.10^{-10}	250 mW				
1.10^{-10}	250 mW				
1.10^{-10}	25 mW				
4.10^{-11}	25 mW				
1.10^{-11}	25 mW				
1.10^{-11}	2,5 mW				

Attention: Check immediately the reactor scram at 250 kW+10%

2.4 NUCLEAR CHANNELS CHECK OF THE NM-1000

Date: ....
DD MM YY

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1. **NM-1000 in Calibration state # 4 :**
   Push F5, F8, 4, ENTER in Microprocessorbox ☐
2. **The following values should be displayed:**
   a) Push F1 → 10% (Microprocessorbox) ☐
   b) LOG Bar Graph → 10% ☐
   c) % Power Bar Graph → 10% ☐
   d) Recorder → 10% ..... kW
   e) at Graphic monitor
      Log Bar → 10% ☐
      Lin Bar → 25 kW ☐
      % PWR → 10% ☐
   f) Rod Withdrawal Prohibit
      Status Window ☐
      Warning Window ☐
      Graphic monitor ☐
3. **NM-1000 in Calibration state #5:**
   Push F5, F8, 5, ENTER in Microprocessorbox ☐
4. **The following values should be displayed:**
   a) Push F1 → 110% (Microprocessorbox) ☐
   b) LOG Bar Graph ~ 100% ☐
   c) % Power Bar Graph → 110% ☐
   d) Recorder → 110% ☐
   e) at Graphic monitor
      Log Bar ~ 100% ☐
      Lin Bar → 275 kW ..... kW
      % PWR → 110% ☐
   f) Rod Withdrawal Prohibit
      Status Window ☐
      Warning Window ☐
   Graphic monitor ☐
   g) NM-1000 Power-HI Scram in Scram Window ☐
   h) NM-1000 Period Scram in Scram Window ☐
5. **NM-1000 in operation state:**
   Push F5, F8, 0, ENTER ☐
6. **Push ACK-button**
7. **Remove source from core** ☐
8. **In Warning Window Rod Withdrawal Prohibit**
   Yes should be announced
   Rod removal not possible (only T and R) ☐
9. **Source into Core** ☐
10. **Press ACK-button, Rod Withdrawal possible**
    Prohibit erased

Signature

2.5 Reactor Scrams

Date: ....
DD MM YY

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1. NM-1000:

Rods up	☐
F5, F8, Enter	☐
NM-1000 110%	☐
Graphic monitor display	☐
Scram window display	☐
F5, F8, 0, Enter	☐
Reset Scram	☐

2. NMP-PH

Rods up	☐
1 mA no Scram Rods up	☐
1,1 mA Scram	☐
NM-PH: Graphic monitor	.... %
Bar Graph	.... %
Graphic monitor display	☐
Scram window display	☐
Reset Scram	☐

3. NPP

Rods up	☐
$3{,}3.10^{-6}$ A no Scram	..... A
$3{,}43.10^{-6}$ A Scram	..... A
Graphic monitor display	☐
Scram window display	☐
Reset Scram	☐

4. Additional Tests

Rods up	☐
Manual Scram	☐
Reset	☐
Rods up	☐
Key switch of Scram	☐
Reset	☐
Rods up	☐
220 V Scram (DAC TB 5-23 open)	☐

2.6 FUEL TEMPERATURE CHANNELS

Date: ....
DD MM YY

Thermocouple Identification	Fuel element Number	TC Position	Core position
1	5284 TC	o	C6
2	5284 TC	m	C6
3	8257 TC	m	E13
4	5284 TC	u	C6
5	8257 TC	o	E13
6	8257 TC	u	E13

Used instrument GA CL-300-1000 C in DAC-drawer backside

Calibration:	Display at graphic monitor [°C]			Display at Status Window [°C]					
	1	2	3	1	2	3	4	5	6
0 °C									
50 °C									
100 °C									
150 °C									
200 °C									
250 °C									
300 °C									
350 °C									
400 °C									

Scram: should be at 360 °C, Transient rod GA CL 305 Voltage Analyzer (0-100 mV, Source Position)

			1	2	3	4	5	6
real scram temperature:	°C	....						
Voltage:	mV	....						
Transient rod Scram at			☐	☐	☐	☐	☐	☐

Remarks
Ambient temperature: .... °C

Signature:

2.7 WATER TEMPERATURE CHANNELS

Date: ....
DD MM YY

Used Instrument: GA CL 301-250 C in DAC-drawer backside

Position [°C]	Display at Graphic monitor [°C]	Display at Status Window [°C] Pool Temp Scram	Tank In Temp Alarm	Tank Out Temp Scram
0				
25				
50				
75				
100				

Used instrument: Heli-Pot, Resistance increased slowly until Scram is triggered

		Graphic-Window	Status Window
Announcement:	Rod up		
	Pool Temperature Scram triggered at	.....°C = ... Ω	.....°C = ... Ω
	Display at Graphic monitor	☐	
	Display at Scram Window	☐	
Announcement:			
	Tank In Temp Alarm triggered at		.....°C = ... Ω
	Display at Graphic monitor		☐
	Display at Scram Window		☐
Announcement:	Rod up		
	Tank Out Temp Scram triggered at		.....°C = ... Ω
	Display at Graphic monitor		☐
	Display at Alarm Window		☐

Remarks:

Signature:

3. Rod drop time of the control rods

If for any reason the reactor has scrammed, it is of utmost importance, that the control rods drop within a very short time into the reactor core. The TRIGA Mark II reactor has three control rods, shim rod, regulating rod and transient rod.

To start up the reactor, the absorber rods are withdrawn from the core. The shim and the regulating rod are moved by an electric motor with a magnetic coupling. The transient rod is removed pneumatically. The total vertical movement of the three control rods is around 50 cm. If the reactor is shut down automatically with a SCRAM, the current to the two magnets is disconnected, and the air of the transient rod pressure system is released. In such a case the absorber rods will drop into the reactor core. The Scram signal of the instrumentation defines the start of a time interval. If the rods reach the lower end, a micro switch sends a signal to the DAC. This signal stops the measurement of the rod drop time.

3.1 Rod drop time of the control rods

Date: ....
DD MM YY

Rod drop time [ms]

Rod Position	Transient Rod		Shim Rod		Regulating Rod	
	Previous Year	This year	Previous Year	This year	Previous Year	This year
500						
400						
300						
200						
100						

4. Neutron flux density measurement using compensated ionisation chambers (CIC)

4.1 Introduction

The neutron flux density is the most important parameter to be controlled during reactor operation. Its range varies from source level at shut down reactor to 10^{13} n $cm^{-2}.s^{-1}$ or even more at full power operation. In addition four radiation types are present in a reactor core, which is prompt and delayed gamma radiation and prompt and delayed neutrons all with a large variety of energies.

In the shut down situation mainly delayed gamma radiation and some source neutrons are present, during the start up procedure the ratio between gammas and neutrons changes constantly and at full power the neutron signal dominates the gamma induced signal for several decades. Therefore very special detectors have to be used at different power levels in order to measure the neutron induced signal.

In the shut down and low power mode the optimal detectors are fission chambers (FC), at medium power level compensated ionisation chambers (CIC) are mainly used and at full power level uncompensated ionisation chambers (UIC) can be applied. An UIC is similar to a CIC except it does not include the compensating electrode, it just a chamber with a boron coated electrode. UIC are used at high reactor power level where the gamma induced signal is negligible compared to the neutron induced signal.

4.2 Compensated ionisation chambers:

In order to improve neutron detection at low neutron fluence rates, these detectors are able to reduce the gamma influence on the detector to about 1% of the total signal. A CIC is a differential chamber comprising three concentric electrodes, which define two chambers in one housing. The first chamber is located between the external positive polarized electrode and the central electrode, which collects the signal. The surfaces of the electrodes are coated with a boron deposit and are therefore sensitive to neutron and gamma rays. (Fig.5.1)

Probably the most frequently used reaction for the conversion of slow neutron into directly detectable particles is the $^{10}B(n,\alpha)$ reaction. The reaction may be written

$$^{10}_{5}B+^{1}_{0}n\rightarrow^{7}_{3}Li^{*}+^{4}_{2}He+5^{0}_{-1}e \qquad Q=2.31\ \text{MeV}$$

and

$$^{10}_{5}B+^{1}_{0}n\rightarrow^{7}_{3}Li+^{4}_{2}He+5^{0}_{-1}e \qquad Q=2.792\ \text{MeV}$$

where the branching indicates that the reaction product $^{7}_{3}Li$ may be left in its ground state ore in its first excited state.[1] When thermal neutrons are used to induce the reaction, about 94% of all reactions lead to the excited state and rest 6% reactions products leads directly to the ground state. In the nuclear reaction created energy Q

[1] The excited lithium nucleus quickly returns (T~10^{-13} s) to its ground state with emission of a 0.482 MeV gamma ray. This photon always escapes and does not contribute the response of the detector.

(2.31 MeV respectively 2.792 MeV) imparted to the reaction products (^7Li and α) are essentially higher in compared with the incoming of the slow neutron.
Alpha particles and ^{7}Li ionise gas and produce charge pulse or an ionisation current due to the neutron reaction in boron. Also gamma-radiation from fission products produces electrons in the chamber through the following reactions:

- Photo-
- Compton- and
- Pair-production effect

However this gamma induced current is very small compared to neutron induced current.

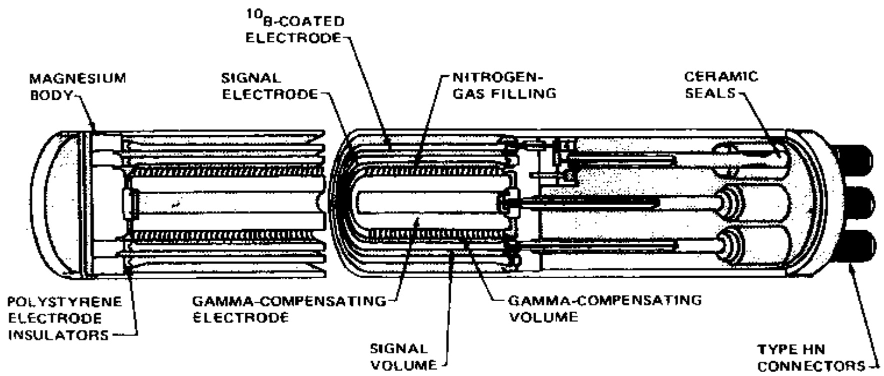


Fig. 4.1: Cross section through a CIC

The second chamber is located between the internal negative polarized electrode and the signal electrode. The surfaces are NOT coated with a boron deposit and are therefore only sensitive to gammy rays. The current of these two chambers are subtracted from each other due to their opposite polarization and the resulting current is therefore mainly proportional to the neutron induced signal only as shown below:

Chamber one with boron coating:	n+γ sensitive (+HV)
Chamber two without boron:	γ sensitive (-HV)

By subtracting the two signals, the gamma signal is compensated and the chamber is only sensitive to neutrons. (Fig. 4.2)

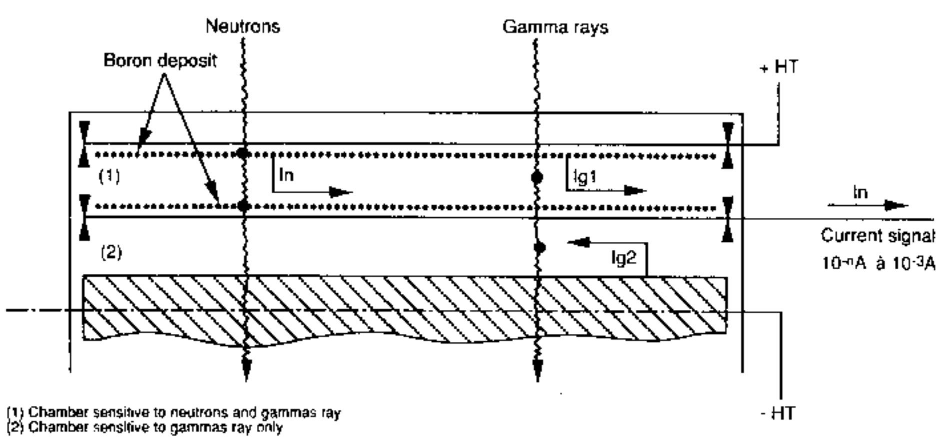


Fig. 4.2: Compensated ionisation chamber mimic diagram

$$n+\gamma - \gamma \rightarrow n \text{ (signal)}$$

The main characteristics of a CIC are the following.

- It is mainly sensitive to neutrons
- A typical neutron sensitivity is 10^{-14} A/nv to 10^{-13}A/nv
- The chamber has to be compensated after installation in the reactor in the presence of a high gamma field but in the absence of neutrons (i.e. soon after reactor shut down with removed neutron source)

Fig. 4.3 shows a typical compensation plot of a CIC.

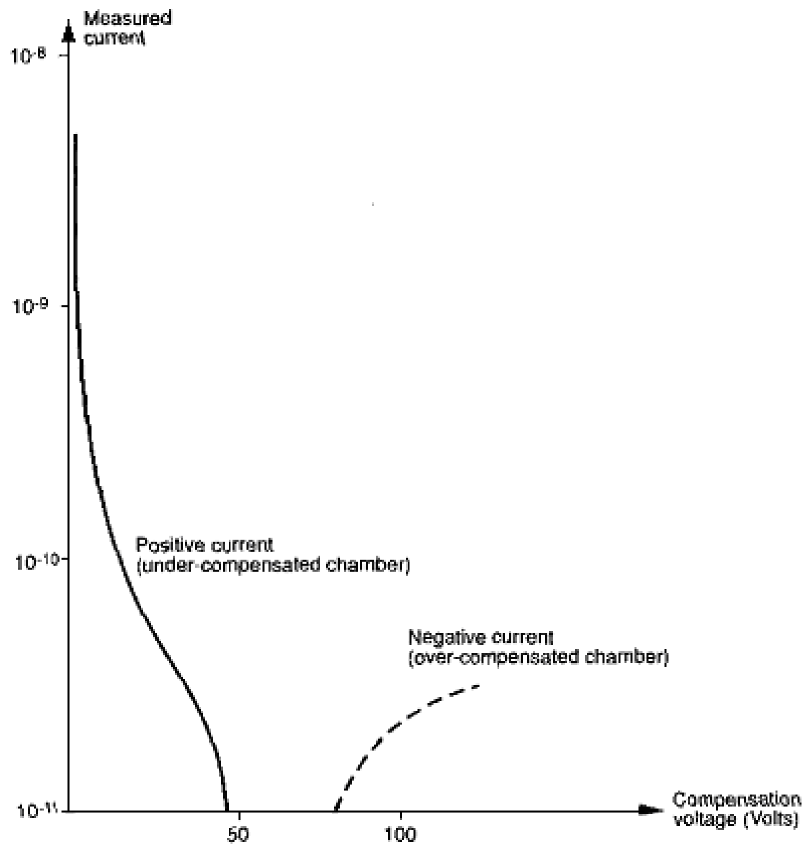


Fig. 4.3: Typical plot of a compensation curve

4.3 Practical exercises

1. While the reactor is shut down and the neutron source is removed from the core the negative compensation voltage is varied between 0 V up to –150V. The positive HV is kept constant at + 600V. (Use table 4.1.)

2. At 4 different reactor power levels (10 W, 1 kW, 10 kW, 25 kW) the positive high voltage is increased stepwise, the resulting chamber current is read until the plateau is reached. The compensating high voltage is kept constant at the value determined under 1. (Use table 4.2.), see Fig. 4.4

3. At fixed positive and negative high voltage the reactor power is increased in steps the chamber current is read. (Use table 4.3.)

Table 4.1: Setting of the compensation voltage

+ HV set at +600 V	
compensating voltage [V]	current [A]
0	
-1	
-2	
-3	
-4	
-5	
-10	
-20	
-30	
-40	
-50	
-100	

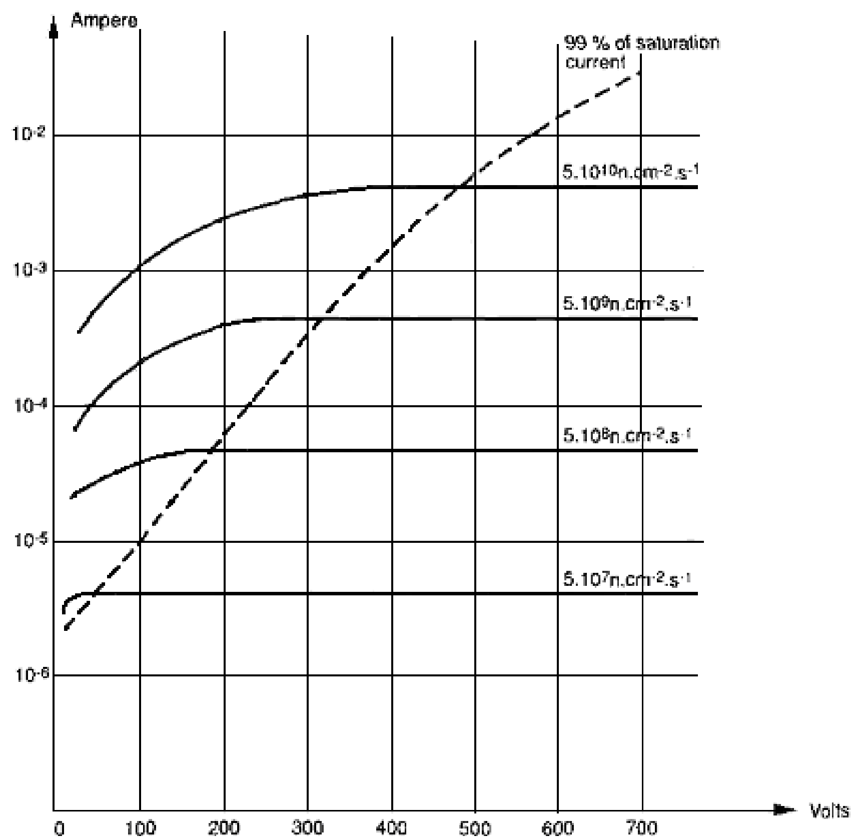


Fig. 4.4: Saturation curve array

Table 4.2: Chamber current as a function of + HV and reactor power

+ HV	Reactor power 10 W (current A)	Reactor power 1 kW (current A)	Reactor power 25 kW (current A)
+4			
+5			
+6			
+7			
+8			
+9			
+10			
+20			
+30			
+40			
+50			
+100			
+150			
+200			
+300			
+400			
+500			

Table 4.3: Chamber current as a function of reactor power

+HV set at: V   -HV set at: V

Reactor power	current [A]

5. Neutron flux density measurement with fission chambers (FC)

5.1 Introduction:

The structure of fission chamber is similar to that of a ionisation chamber, the main differences are as follows:

- The electrode deposit is made of a small amount (appr. 1 mg) of 93% enriched uranium
- The neutron interaction with uranium results in a fission process with a much higher reaction energy than with the neutron-boron interaction
- The interaction of the chamber with gamma radiation is smaller therefore FC can be readily applied to detect a small amount of neutrons in a high gamma background field (i.e. in the reactor shut-down or start-up mode, reactor power level below 10 W)
- Using a pulse high discriminator the small pulses from gamma interaction can be filter away from the large neutron induced pulses
- The typical neutron sensitivity is 10^{-14} A/nv to 10^{-12} A/nv
- Fission chambers can be operated either in the pulse mode or in the current mode, in special cases they could also use current fluctuations (Campbell mode)

5.2 Fission chambers:

In research reactors and small nuclear power plants (NPP) fission chambers are positioned outside around the core to monitor the neutrons. In large NPP the information collected from outside the core on the neutron flux density distribution inside the core is not sufficient therefore miniature FC have been developed to be positioned inside the core. These miniature FC have the size of a pencil and are usually positioned in tubes entering the core from top or bottom. In large NPP there may be typically around 100 such FC distributed inside the core. The signals are collected and evaluated in a computer system to optimize the power distribution, but the signals may also be used for fast reactor shut-down in case of any power deviation from demand power. (Fig. 5.1, Fig. 5.2)

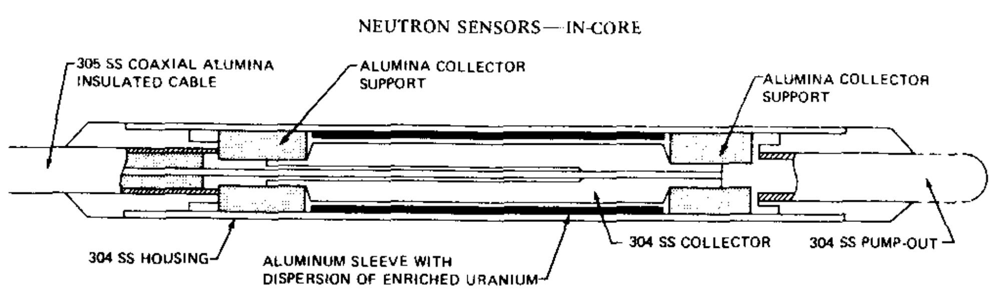


Fig. 5.1 Cross section through miniature in-core fission chamber

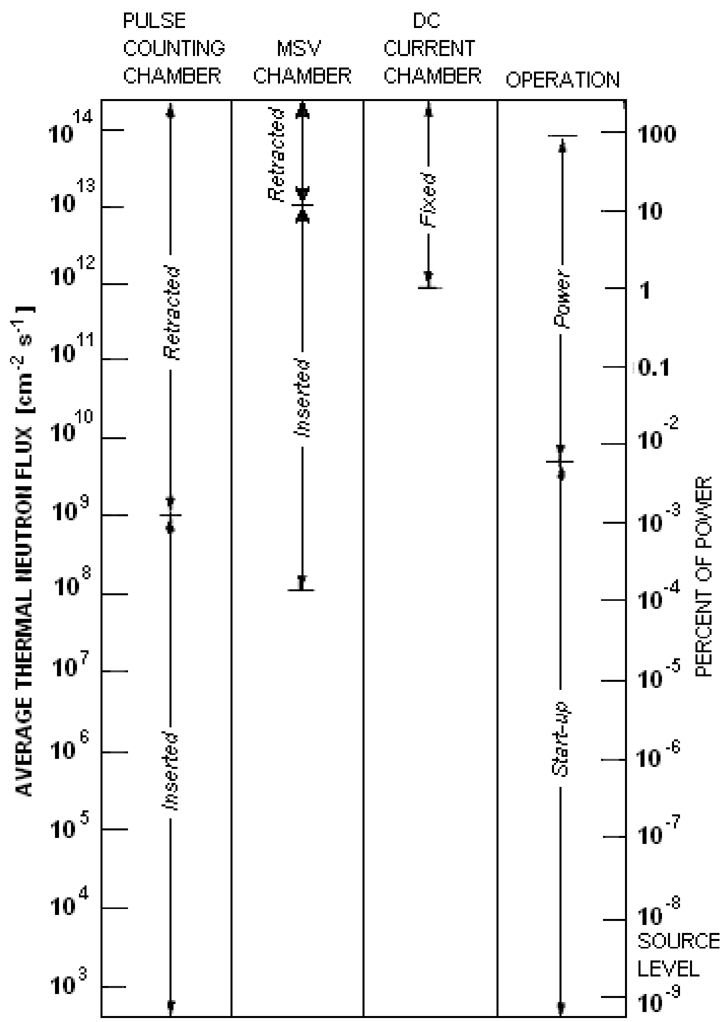


Fig. 5.2: Ranges of in-core fission chambers

5.3 Practical exercises

1. While the reactor is shut down and the neutron source is removed from the core (no neutrons available for detection) the pulses of the fission chamber (gamma background and noise) are counted as a function of the discriminator setting. When no more pulses are counted the discriminator is set at that position. (Use table 5.1)

2. The neutron source is inserted into the core, the reactor is operated at a low power level (1 W) and the FC is lowered to bottom of the guide tube. Now the counts are registered for about 30 s (three times calculating the average count rate), then the FC is removed in 10 cm steps from the core and the measurement is repeated until the FC is so far away from the core that no more counts are registered. (Use table 5.2)

3. When the reactor is shut down, the count rate of the FC as a function of time is measured to follow the neutron flux density decrease. (Use table 5.3)

Table 5.1: Range of neutrons in the core region using a fission chamber FC 165

Position from lowest end [cm]	Counts per 30s
0	
+10	
+20	
+30	
+40	
+50	
+60	
+70	
+80	
+90	
+100	
+110	
+120	
+130	
+140	
+150	
+160	
+170	
+180	
+190	
+200	

Table 5.2: Fission chamber count rate as a function of the discriminator setting Type FC 165

Discriminator setting	Counts per 30s
0	
0.2	
0.4	
0.6	
0.8	
1	
1.2	
1.4	
1.6	
1.8	
2	
2.5	
3	
3.5	
4	

Table 5.3: Fission chamber count rate after reactor shut down

Time after shut down [min]	Fission chamber count rate [30s]

6. Neutron flux density measurement with self-powered neutron detectors (SPND)

6.1 Introduction

The principle of measuring electrons directly after neutron capture was known and patented as early as 1938. Neutron detectors operating without any external voltage were reported in 1961, but the practical application in neutron flux measurement techniques was first reported in 1964. During the following years a large number of papers on self-powered neutron detectors were published. As these detectors operate without any external voltage they are usually called self-powered neutron detectors (SPND).

The years between 1969 and 1974 were used especially for the development of new emitter materials. Besides the most important emitter materials Co, Rh and V also the other emitter materials such as Ag, Al, Au, ^{11}B, Ce, Er, Gd, Hf, Pd, Pt, Os, Ta, Tl, W and Yb were investigated. Another part of the developmental work concentrated on new geometric detector forms where detector bundles with different emitter lengths were used to measure the axial power distribution in the core.

Table 6.1: Some characteristic data of emitter materials

Isotope	Natural abundance (%)	Neutron cross section (b)	Half-life	Maximum β-decay MeV	Burn up in % per month at 10^{14} $cm^{-2}s^{-1}$
^{27}Al	100	0.23	2.3 min	2.9	0.006
^{107}Ag	48.65	35	2.3 min	1.8	0.9
^{109}Ag	51.35	89	24 s	2.8	2.3
^{11}B	80.2	0.005	0.02 min	13.4	
^{51}V	99.76	4.8	3.76 min	2.5	0.12
^{103}Rh	100	150	4.4 min/42 s	2.5	3.9
^{59}Co	100	37	10^{-14} s	-	1.0
^{55}Mn	100	13.3	2.58 h	2.8	0.34
^{7}Li	92.6	0.037	0.85 s	13.0	
Natural Er	100	162			4.0
Natural Hf	100	102			3.2
Natural Pt	100	10			0.25

The most common detector form is a coaxial cylinder with an emitter in the center surrounded completely by an insulator. The outer sheath acts as case and collector (Fig.6.1 and 6.2**).** The detector geometry can be adapted for special applications and also small plates, hooks or spirals are offered by the industry. Table 6.1 gives a summary of the commonly used emitter materials for self-powered neutron detectors.

The emitter material must have an appropriate neutron absorption cross-section, which usually is a compromise between detector sensitivity and detector burn-up. The detector signal decreases with burn-up, the speed of burn-up being governed by the cross-section value (Fig. 6.1). Other emitter material selection criteria are the energy of the radiation emitted after a neutron capture, the half-life of the produced nuclides, their daughter products and the melting point. According to these criteria the whole nuclide chart has been searched for possible emitter materials. The resulting nuclides have been mentioned above.

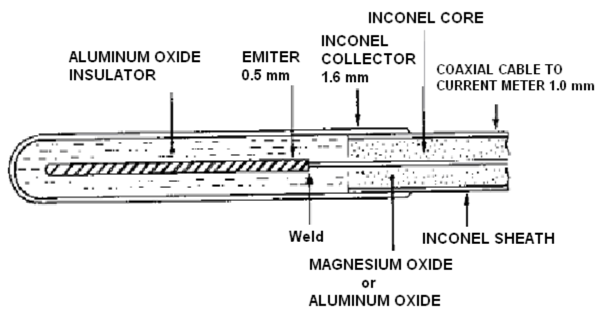


Fig. 6.1: Self powered neutron detector

The insulator usually consists of a high-temperature resistant material (like Al_2O_3, MgO or BeO) and must have a thickness that allows the produced electrons - (n,γ)(γ,e) - to reach the collector. An optimization of the detector geometry especially the emitter diameter and the insulator thickness is necessary.

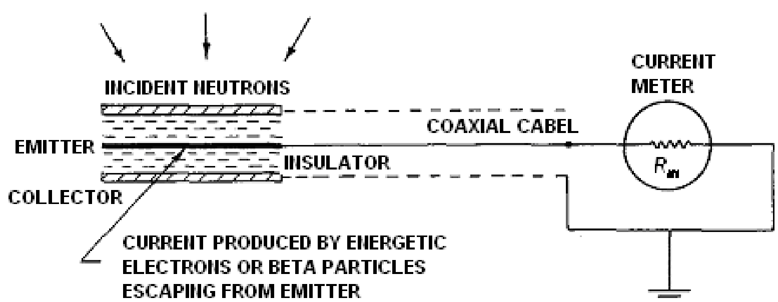


Fig. 6.2: Self powered neutron detector

The collector has to collect the emitted electrons and gives the detector its mechanical stability. It must be corrosion resistant and should not undergo dimensional changes at high temperatures. After a neutron capture in the emitter a (n,γ)(γ,e) reaction produces electrons. The constant loss of electrons from the emitter produces a current between emitter and collector, which can be measured directly by an ammeter. The current is proportional to the rate of absorption of neutrons in the emitter and thus proportional to the local neutron flux.

6.2 Slow or delayed SPND

In some emitter materials the electrons are produced with a time delay according to the half-life of the nuclide. The detector, therefore, gives a delayed response to a neutron flux variation and a constant signal is measured only after the saturation activity of the emitter material has been reached. Typical delayed self-powered neutron detectors are Rh and V detectors, which have a response time of about 5 min. For this reason they cannot be used in the reactor protection system. (Fig. 6.3)

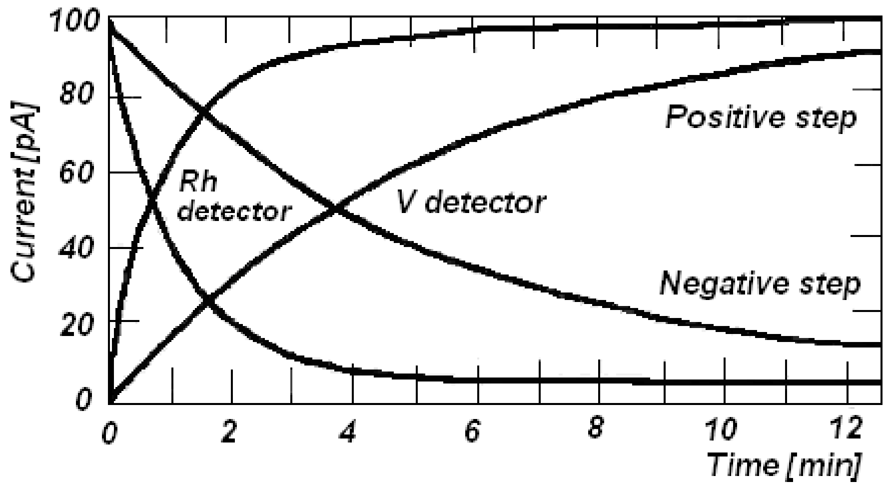


Fig. 6.3: SPND response to step change in neutron flux

6.3 Prompt SPND

If the (n,γ)(γ,e) reactions take place immediately (i.e. 10^{-14} s for a Co detector), the detector responds promptly to any neutron flux change and can be used in the reactor protection system. As the prompt self-powered neutron detectors are very important for power reactor instrumentation, the signal composition will be analyzed more closely.

The main part of the signal is prompt and results, e.g., with a Co detector from the reaction $^{59}Co(n,\gamma)^{60*}Co$. In addition, heavier nuclides like ^{60}Co and ^{61}Co are produced in the emitter material. The gamma radiation emitted from this nuclear reaction produces electrons in the emitter by either photo, Compton, pair-production processes or by internal conversion, which penetrate the insulator and are collected by the collector. Now the prompt signal $I_{pr,n}$ resulting from the direct nuclear reaction has to be distinguished from the delayed signal I_β from the produced heavier isotopes. These isotopes usually emit β-radiation and delayed gamma radiation, both producing an unwanted delayed detector signal together with the prompt signal.

6.4 Other influences on the SPND signal

Other disturbances are neutron capture gamma rays from the reactor core and the prompt ($I_{pr,\gamma}$) and delayed ($I_{del,\gamma}$) gamma radiation from nuclear fission. These radiation components penetrate the detector and cable from outside and produce secondary electrons registered as detector noise.

The polarity of this noise signal depends on the atomic numbers of collector, insulator and emitter. Therefore, the total detector signal is composed according to:

$$I_{tot} = I_{pr,n} + I_{pr,\gamma} + I_{del,\gamma} + I_\beta$$

From the above-stated components contained in $I_{pr,\gamma}$ one component (prompt gamma radiation from fission) depends on the fission rate and the other component (neutron capture gamma rays) depends on the location where the detector is installed in the reactor core. Therefore, $I_{pr,\gamma}$ depends on the reactor type and on the detector core position. For one special location of a new Co detector $I_{pr,n}$ is about 83% and $I_{pr,\gamma}$ about 17% of the total signal. For other emitter materials these values depend on the atomic number of the emitter. The composition of the two different noise signals ($I_{del,\gamma}$ and I_{β}) can produce accidentally a total signal I_{tot}, which may be approximately constant over several years even if the true detector signal $I_{pr,n}$ decreases according to the detector burn-up. Thus the signal to noise ratio is decreased while the total signal is constant. The behaviour of I_{tot} depends very much on the local neutron flux and the above mentioned effect of burn-up compensation by the creation of noise signals is predominant at a flux level of about $1x10^{14}$ $cm^{-2}s^{-1}$. From a safety point of view this effect is not acceptable, therefore, a periodic calibration of the detector signal as well as an investigation of its composition is necessary. This can be done either during reactor operation with movable detector systems or during reactor shutdown periods where the prompt detector signal is zero.

6.5 Practical exercise

A SPND is installed in the reactor core and the reactor is started to a low power level. According to the selected emitter material the current of the SPND will increase due to the activation of the emitter material. It takes about 10 half-lives until a constant signal is obtained. Then the reactor is shut down and the decrease of the signal is observed which again depends on the half-life of the emitter material. From these values the emitter material can be determined.

7. Pressurized water reactor simulator

7.1 Introduction

This text assumes that the participant already has some basic knowledge of the function of a Pressurized Water Reactor (PWR). Therefore no attempt has been made to provide detailed descriptions of the PWR system. Such information is found in nuclear engineering textbooks such as in /www.world-nuclear.org/ >issues and information briefs/. However, details are provided where necessary to describe the functionality and the interactive features of the individual simulator screen, which relates to the specific PWR subsystems.

The text covers basic NPP plant operations, like plant load following, reactor trips or scrams (=immediate reactor shut down) and recovery (e.g. turbine- or reactor scram). In addition, it covers plant responses to malfunction events. Some malfunction events lead to reactor scram or turbine trip. Other serious malfunctions (e.g. LOCA) lead to accident situations, causing actuation of the Passive Core Cooling Safety System.

Pressurized water reactors were initially designed for use in submarines. Knolls Atomic Power Laboratory and Westinghouse Bettis Laboratories performed the research and development work. As a result of this initial R & D work, a commercial PWR was designed and developed for nuclear power plant applications. Eventually, several commercial PWR suppliers emerge: Westinghouse, Babcock and Wilcox; and Combustion Engineering in the U.S.; Siemens (Kraftwerk Union) in West Germany; and Framatome in France. Subsequently, Mitsubishi in Japan and Agip Nucleari in Italy became PWR Licensees.

Over the past three decades, many PWRs were in service, accumulating thousands of reactor years of operating experience. In recent years, new generations of advanced PWR nuclear power plants have been developed, building upon the past success, as well as applying lessons learned from past operating experience. The advanced PWR design incorporates efforts by utilities, and the regulators to establish standardized solutions to meet their requirements. This is because the advanced PWR design has to be suitable for deployment in many countries and the design must also be economical. In this context, important programmes in the development of advanced PWRs were initiated in the mid 1980s in the United States. In 1984, the Electric Power Research Institute (EPRI), in cooperation with US Department of Energy (DOE), and with the participation of US nuclear plant designers, and several foreign utilities, initiated a programme to develop utility requirements to guide the advanced PWR design. As a result of this effort, utilities requirements were established for large PWRs having ratings of 1200 MWe to 1300 MWe, and for mid-size PWRs in the 600 MW range.

The effort for advanced PWR design was led by Westinghouse for the AP-600 and AP-1000 design, which received NRC certification in 1999. The AP-600 and AP-1000 design include the following key developments:

(1) larger core, resulting in lower (25 % less) power density;
(2) lower fuel enrichment, and the use of radial reflector for better neutron economy;
(3) longer fuel cycle;
(4) 15 % more safety margin for DNB and LOCA;

(5) reduced worth control rods to achieve load following capability without substantial use of boron;
(6) passive core cooling system which includes core depressurization, safety injection, and residual heat removal;
(7) passive containment cooling system;
(8) in-vessel retention.

This PWR simulator is largely based on a 600 MWe advanced PWR design, similar to AP 600.

7.1.1 Prominent Characteristics of PWR

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

(1) The core normal operating conditions are liquid phase water;
(2) Steam generation occurs only in the secondary phase of the power cycle, namely, the steam generators.
(3) The primary system pressure is maintained by a pressurizer that utilizes electric heaters for heating and pressurization, and sprays for cooling and depressurization.
(4) The reactor power control is achieved by the combination of a heavy-worth bank of control rods dedicated to axial flux shape control, and reduced worth control rods position adjustments to maintain average coolant temperature during power changes. Liquid boron is only used under the limiting cases of the rods control system. It is dissolved in the primary system to keep the power distribution and level under control in the core. With such implementation of the reactor power control system, it permits PWR to have load following operations, including frequency control, to respond to grid requirement, without substantial use of liquid boron.
(5) The PWR fuel rods are smaller and packed in larger bundles.
(6) The PWR control rods are inserted in the bundles, rather than between bundles.
(7) The entire core flow is normally pumped through the recirculation pumps.

Because there is no boiling in the PWR core during normal operations and most abnormal and normal plant transients, there is not a large density change in the core, as compared with the BWR core, during transients. This means that pressurization transients contribute little density reactivity feedback in the PWR core and consequently little power increase. On the other hand, flow coast-down transients get little density change negative feedback, making this type of transient, which is limiting in PWR, the most severe in terms of thermal challenge to the system.

As well, because there is no void reactivity feedback in the PWR to damp the Xenon and Iodine fission product build-up, the PWR is subject to Xenon oscillations.

A typical 600 MWe PWR design is shown in Figure 7.1. This figure shows a PWR system with two steam generators, four recirculation loops and a pressurizer in the system. The primary coolant is circulated through a recirculation pump into the core through the bottom and out the top into the discharge plenum. The heated water then flows down through the steam generator where the heat is transferred to the secondary system. The primary coolant is then taken from the bottom of the steam generator into the recirculation pump to repeat the cycle.

The secondary coolant leaves the steam generator as superheated steam. It passes through the turbine where the energy is delivered to drive the turbine-generator unit. The remaining heat is removed in the condenser where the secondary coolant is returned to the liquid phase. From the condenser, the secondary coolant is pumped as feed water through various heating and pumping stages until it reaches the steam generators where it picks up energy again from the primary coolant. From here on the power cycle repeats again.

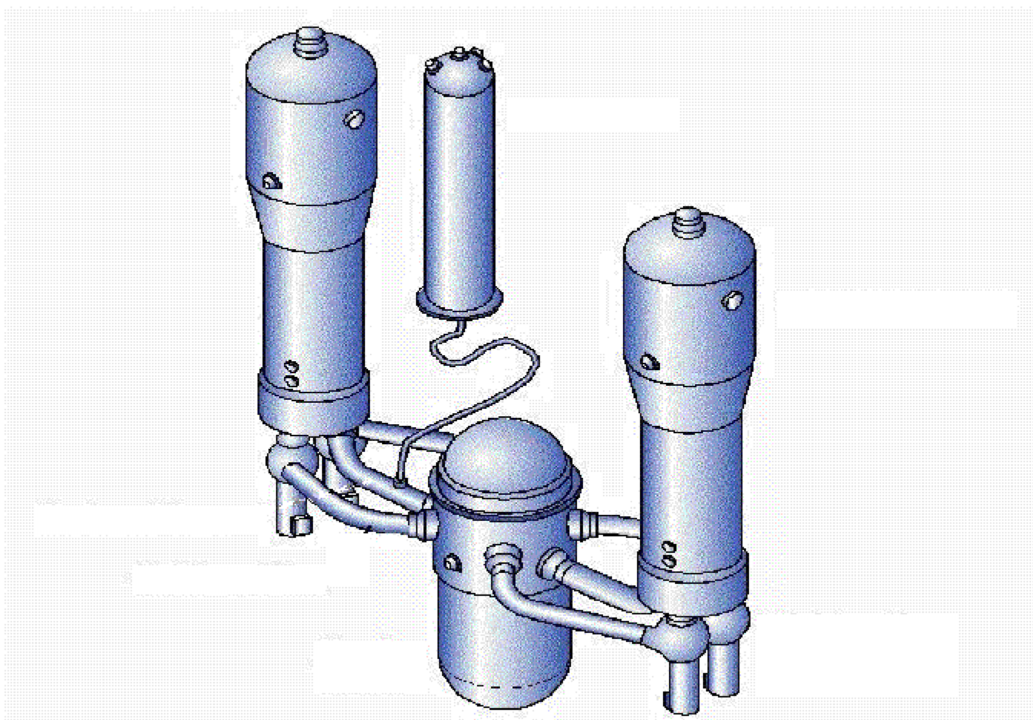


Figure 7.1 - A typical 600 MWe Pressurized Water Reactor NPP

7.2 600 MWe Pressurized Water Reactor Simulator

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

The interaction between the user and the Simulator is via a combination of monitor displays, mouse and keyboard. Parameter monitoring and operator controls implemented via the plant display system at the generating station are represented in a virtually identical manner on the Simulator. Control panel instruments and control devices, such as push-buttons and hand-switches, are shown as stylised 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 Pressurized Water Reactor (PWR).

Table 7.1: Summary of Simulator Features.

SYSTEM	SIMULATION SCOPE	DISPLAY PAGES	OPERATOR CONTROLS	MALFUNCTIONS
REACTOR	• neutron flux levels over a range of 0.001 to 110% full power, 6 delayed neutron groups • decay heat (3 groups) • all reactivity control devices - “dark” rods; “grey” rods; boron control. • Xenon/Iodine poison • reactor power control system • reactor shutdown system	• PWR Power Control • PWR Control Rods & SD Rods • PWR Trip Parameters	• reactor power and rate of change (input to control computer) • manual control of reactivity devices - control rods and boron addition/removal • reactor trip • reactor setback • reactor step back	• reactor setback and step back fail • one bank of Dark control rods drop into the reactor core
REACTOR COOLANT	• main circuit coolant loop with four pumps, two steam generators, four equivalent “lumped” reactor coolant channels • pressure and inventory control which includes pressurizer, coolant letdown condenser, charge & letdown control, and pressure relief • operating range is from zero power hot to full power	• PWR Reactor Coolant System • PWR Coolant Inventory & Pressurizer • PWR Inventory Control • PWR Pressure Control	• reactor coolant pumps • coolant makeup pumps • pressurizer pressure control: heaters; spray; pressure relief valve • pressurizer level control by regulating coolant feed & bleed flow • isolation valves for: coolant feed and bleed	• Pressurizer Pressure Relief Valve fails open • charging (feed) valve fails open • letdown (bleed) valve fails open • pressurizer heaters #2 to # 6 turned "ON" by malfunction • reactor header break
STEAM & FEEDWATER	• boiler dynamics, including shrink and swell effects • steam supply to turbine and reheater • turbine by-pass to condenser • extraction steam to feed heating • steam generator pressure control • steam generator level control • boiler feed system	• PWR Feedwater & Extraction Steam	• feed pump on/off operation • boiler level controller mode: Auto or manual • level control setpoint during Auto operation • level control valve opening during manual operation • extraction steam valves opening	• all level control isolation valves fail closed • one level control valve fails open • one level control valve fails closed • all feed pumps trip • all steam safety valves open • steam header break • steam flow transmitter fails
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	• PWR Turbine Generator	• turbine trip • turbine run-back • turbine run-up and synchronization • condenser steam discharge valves	• turbine spurious trip • turbine spurious run-back
OVERALL UNIT	• fully dynamic interaction between all simulated systems	• PWR Plant Overview • PWR Control		

	• overall unit power control with reactor leading mode; or turbine leading mode • unit annunciation & time trends • computer control of all major system functions	Loops • PWR MW Demand SP & SGPC		
SAFETY SYSTEM	•	• PWR Passive Core Cooling		

7.2.1 SIMULATOR STARTUP

- select program 'PWR' for execution
- click anywhere on 'PWR Simulator" screen
- click 'OK' to 'Load Full Power IC?'
- the Simulator will display the 'PWR Plant Overview' screen with all parameters initialized to 100% Full Power
- at the bottom right hand corner click on 'Run' to start the simulator

7.2.2 SIMULATOR INITIALIZATION

If at any time it is necessary to return the Simulator to one of the stored Initialization Points, do the following:

- 'Freeze' the Simulator
- click on 'IC'
- click on 'Load IC'
- click on 'FP_100.IC' for 100% full power initial state
- click 'OK' to 'Load C:\PWR\FP_100.IC'
- click 'YES' to 'Load C:\PWR\FP_100.IC'
- click 'Return'
- Start the Simulator operating by selecting 'Run'.

7.2.3 LIST OF PWR SIMULATOR DISPLAY SCREENS

1. Plant Overview
2. Control Loops
3. Control/Shutdown Rods & Reactivity
4. Reactor Power Control
5. Trip Parameters
6. Reactor Coolant System
7. Coolant Inventory & Pressurizer
8. Coolant Inventory Control

9. Coolant Pressure Control
10. Turbine Generator
11. Feedwater & Extraction Steam
12. MW Demand SP & SGPC
13. Passive Core Cooling
14. Trends

7.2.4 SIMULATOR DISPLAY COMMON FEATURES

The PWR Simulator is made up of 14 interactive display screens or pages. All of these screens have the same information at the top and bottom of the displays, as follows:

- top of the screen contains 21 plant alarms and annunciations; these indicate important status changes in plant parameters that require operator actions;
- top right hand corner shows the simulator status:
  - ⇒ the window under ‘Labview’ (this is the proprietary software that generates the screen displays) has a counter that is incrementing when Labview is running; if Labview is frozen (i.e. the displays cannot be changed) the counter will not be incrementing;
  - ⇒ the window displaying ‘CASSIM’ (this is the proprietary software that computes the simulation responses) will be green and the counter under it will not be incrementing when the simulator is frozen (i.e. the model programs are not executing), and will turn red and the counter will increment when the simulator is running;
- to stop (freeze) Labview click once on the ‘STOP’ sign at the top left hand corner; to restart ‘Labview’ click on the ⇒ symbol at the top left hand corner;
- to start the simulation click on ‘Run’ at the bottom right hand corner; to ‘Stop’ the simulation click on ‘Freeze’ at the bottom right hand corner;
- the bottom of the screen shows the values of the following major plant parameters:
  - ⇒ Reactor Neutron Power (%)
  - ⇒ Reactor Thermal Power (%)
  - ⇒ Generator Output (%)
  - ⇒ Primary Coolant Pressure (kPa)
  - ⇒ Core Flow (kg/sec)
  - ⇒ Main Steam Pressure (kPa)
  - ⇒ BOP Steam Flow (kg/sec)
- the bottom left hand corner allows the initiation of two major plant events:
  - ⇒ ‘Reactor Trip’
  - ⇒ ‘Turbine Trip’

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

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

7.2.5 PWR Plant Overview

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

- REACTOR is a 3-D spatial kinetic model with six groups of delayed neutrons; the decay heat model uses a three group approximation; reactivity calculations include reactivity control and safety devices, Xenon, fuel temperature, moderator temperature, Boron. The parameters displayed are:
  - ⇒ Neutron Power (% full power)
  - ⇒ Reactor Thermal Power (% full power)
- Reactor coolant main loop, with four cold legs (CL1, CL2, CL3, CL4); two hot legs (HL1, HL2); pressure and inventory control systems are shown on the Plant Overview display, additional details will be shown on subsequent displays. The parameters displayed are:
  - ⇒ Reactor core pressure (kPa)
  - ⇒ Reactor core flow (kg/sec)
  - ⇒ Average reactor coolant temperature (°C)
  - ⇒ Average fuel temperature (°C)
  - ⇒ Pressurizer Level (m) and Pressure (kPa)
  - ⇒ Flow to/from Pressurizer (kg/sec)
  - ⇒ Status of the four Reactor Coolant Pumps (RCP#1, 2, 3, 4)
- The two Steam Generators are individually modeled, along with Balance of Plant systems. The parameters displayed are:
  - ⇒ Boiler 1, 2 Level (m)
  - ⇒ Boiler 1, 2 steam flow (kg/sec)
  - ⇒ Boiler 1, 2 steam pressure (kPa)
  - ⇒ Boiler 1, 2 steam temperature (°C)
  - ⇒ Total flow (kg/sec) and opening status of the four Steam Relief Valves (SRV's). The four SRV's are represented by one valve symbol - that is, in the event that any SRV opens, the valve symbol colour will be red; green when all SRV's are closed.
  - ⇒ Moisture Separator and Reheater (MSR) Drains Flow (kg/sec)
  - ⇒ Status of control valves is indicated by their colour: green is closed, red is open
  - ⇒ Main Steam Stop Valves (MSV) status
  - ⇒ Condenser Steam Bypass (Dump) Valves status and % open

- Generator output (MW) is calculated from the steam flow to the turbine
- Condenser and Condensate Extraction Pump (CEP) are not simulated
- Simulation of the feedwater system is simplified; the parameters displayed on the Plant Overview screen are:
  - ⇒ Total Feedwater flow to the Steam Generators (kg/sec)
  - ⇒ Average Feedwater temperature after the High Pressure Heaters (HPHX)
  - ⇒ Status of Boiler Feed Pumps (BFP) is indicated as red if any pumps are 'ON' or green if all the pumps are 'OFF'

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

7.2.6 PWR Control Loops

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

In the "Turbine Leading" Control Mode, generator power is controlled according to the power demanded by means of a remote reference value (e.g. operator input), and/or by a value derived from the actual generator frequency deviation from the grid. Using this deviation from setpoint, the reactor power is adjusted using average coolant temperature control. This Mode of control is typically used for baseload operation with constant or scheduled load; as well as load following operation with a frequency control function. It is important to note that steam generator pressure is maintained constant during this Control Mode operation.

In the "Reactor Leading" Control Mode, the reactor power control is determined by operator input, and/or plant upset conditions (e.g. Turbine Trip), which in turns will set new average coolant temperature setpoint, hence adjusting the reactor power to match the power setpoint. The water-steam system, consisting of the turbine with its bypass system, and the steam generators, will follow such reactor power change and adjust in power by maintaining the steam generator pressure constant.

In support of these two Control Modes and plant safety functions, the PWR has the following Control Loops as illustrated by the "PWR Control Loops Screen" in the simulator:

(1) Reactor Power Demand SP

Reactor Power Demand Setpoint (SP) is determined by operator input and/or by the automatic limitation functions such as the Reactor Stepback, which requires a step change in power reduction, or Reactor Setback, which requires power reduction at a fixed rate. The automatic limitation functions are triggered by specific reactor/coolant process conditions, which exceed alarm setpoints.

(2) Reactor Power Control

Reactor Power Control in the PWR can be accomplished by *Core Reactivity Regulation* and *Power Distribution Control*. *Core Reactivity Regulation* accounts for reactivity changes due to power level changes, and transient xenon level resulting from the power level changes. It is achieved by a combination of control rod position adjustment, and boron concentration adjustment. The control rods that perform the core reactivity regulation are reduced strength rods, known as “Grey” rods. They are moved up or down, when the deviation between primary power (P_{av}) and the reference power (P_{ref}) obtained from the turbine load (secondary power; turbine first stage pressure), exceeds the predetermined setpoint.

Power Distribution Control is performed to maintain the core thermal margin within operating and safety limits. Power distributions, as determined by the core neutron power axial shapes, are monitored and controlled during power maneuvers. In advanced PWRs, a bank of high reactivity worth, known as “Dark” rods, is dedicated to axial power shape control. As the “Dark” rods are inserted into or withdrawn from the core, the axial power shape is bottom or top shifted respectively. Hence, with the utilization of the bank of “Dark” rods, axial power shape control can be accomplished. That means during power maneuvers if the axial power distribution is top skewed, insertion of “Dark” rods would be required. Conversely, withdrawal would be required, when the axial power distribution is bottomed skewed.

(3) Control Rods Actuation

The rod control system - “Grey” and “Dark” rods, receives rod speed and direction signals from the Reactor Power Control system. The rod speed demand signal varies over a range depending on the input signal level. Manual control is provided to move a bank in or out at a prescribed speed. In Automatic mode, the rod motion is controlled by the Reactor Power Control system. The rods are withdrawn (or inserted) in a predetermined, programmed sequence. The shutdown banks are always held in the fully withdrawn position during normal operation, and are moved to this position at a constant speed by manual control prior to criticality. A reactor trip signal causes them to fall by gravity into the core.

Only the control banks move under automatic control. Each control bank is divided into smaller groups of control rods to obtain smaller incremental reactivity changes per step. All the control rods in a group are electrically in parallel so that they move simultaneously. Individual position indication is provided for each rod. A variable speed drive provides the ability to insert small amounts of reactivity at low speeds to give fine control of reactor average coolant temperature, as well as to furnish control at high speeds to correct larger temperature transients.

(4) Boron Control

The boron concentration control system is used for relatively long term and slow core reactivity control. With the combined use of “Gray” and “Dark” rods for core reactivity regulation and core power distribution, boron concentration control is used only if necessary, so that the required rod worth is maintained for safe shutdown margin, as well the control rods are kept within the rod position limitations by the control bank rod insertion limit.

(5) Primary Coolant Pressure Control

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

(6) Primary Coolant Inventory & Makeup Control

The Primary Coolant Inventory & Makeup Control is performed by the pressurizer level control system. It provides the capability of establishing, maintaining and restoring the pressurizer water level to the target value which is a function of the average coolant temperature. It maintains the coolant level in the pressurizer within prescribed limits by adjusting the flow of the charging (feed) and let-down (bleed) system, thus controlling the reactor coolant water inventory.

(7) MW Demand Setpoint Demand

Mega Watts (MW) Demand Setpoint is determined by operator input. This input will used as reference target for raising or lowering the turbine load.

(8) Steam Generator Pressure Control

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

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

(9) Steam Generator Level Control

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

(10) Turbine Governor Control

The Turbine Governor Control system will regulate the steam flow through the turbine to meet turbine load target by controlling the opening of the turbine governor valve.

(11) Core Cooling Control

The Passive Core Cooling system uses three sources of water to maintain core cooling (a) Core Makeup Tanks (CMTs) (b) Accumulators (c) In-containment Refueling Water Storage Tank (IRWST). All of these injection sources are connected directly to two nozzles on the reactor vessel. Using gravity as a motivating force, these cooling sources are designed to provide rapid cooling of the reactor core from small leaks to large loss-of-coolant accidents (LOCAs).

7.2.7 PWR Control Rods and Shutdown Rods

The screen shows the status of the Shutdown System (SDS), as well as the reactivity contributions of each device and physical phenomenon that is relevant to reactor operations.

- The positions of each of the two SDS SHUTDOWN ROD banks are shown relative to their normal (fully withdrawn) position. In this PWR Simulator, the reactivity worth for each SDS SHUTDOWN ROD bank is - 35.365 mk, so the total reactivity worth for the two SDS SHUTDOWN ROD banks, when fully inserted in core is - 70.73 mk.
- REACTOR TRIP status is shown as NO (green) or YES (yellow), the trip can be reset here; note that SDS RESET must also be activated before Reactor Power Control (RPC) will begin withdrawing the Shutdown Rods.
- The REACTIVITY CHANGE (mk) of each device and parameter from the initial 100% full power steady state is shown. These include:

  1. SHUTDOWN RODS
  2. GRAY RODS
  3. DARK RODS
  4. XENON
  5. FUEL TEMPERATURE Reactivity Feedback
  6. MODERATOR TEMPERATURE & BORON Reactivity Feedback

  ⇒ Note that reactivity is a computed parameter, and not a measured parameter. It can be displayed on a simulator but is not directly available at an actual plant.

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

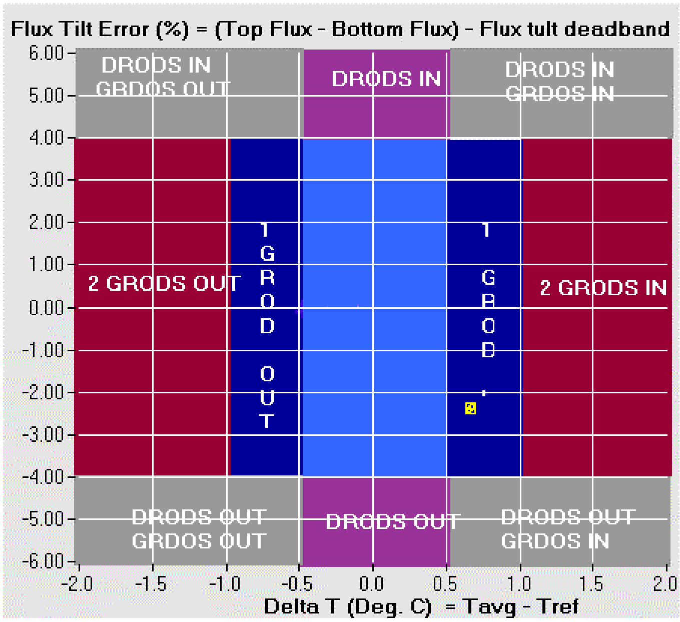
This screen also shows the Control Rods Movement Diagram, and the status of the three reactivity control devices that are under the control of the Reactor Power Control System (RPS) - “Gray” control rods; “Dark” control rods; boron concentration control.

- The Control Rods Movement Diagram displays the Operating Point in terms of Flux Tilt Error (ΔI) - Y axis of the Diagram, and Coolant Temperature difference (ΔT) - X axis of the Diagram, where

  FLUX TILT ERROR, ΔI = (TOP FLUX - BOTTOM FLUX) - FLUX TILT DEADBAND

  ΔT = COOLANT AVERAGE TEMPERATURE T_{avg} - REFERENCE COOLANT TEMPERATURE T_{ref} - TEMPERATURE DEADBAND T_{db}

- Regions A and C in the following figure show cases of skewed axial power distributions - region A is top-skewed; region C is bottom skewed. The ΔI has exceeded the target band of ΔI_{db} (4%) from its reference value ΔI_{ref} in both regions. Hence in region A, the "Dark" rods would be inserted to compensate for the top-skewed flux; whereas in region C, "Dark" rods would be withdrawn to compensate for the bottom-skewed flux.



In this PWR Simulator, there are four banks of "Dark" rods. They are positioned near the top of the reactor core and have a strong influence on axial power shape. They move together and a few centimeters of movement are needed for effective axial power distribution control. The reactivity worth of a bank of "Dark" rods is - 54.5 mk, so the total reactivity worth for the four banks of "Dark" rods is - 218 mk if they are fully inserted into the core.

NOTE: the four banks of "Dark" rods are normally controlled by the Reactor Power Control (RPC) in "Auto" mode. The control of "Dark" rods can be switched to "Manual" mode where each Bank can be controlled individually with the control button for "IN", "STOP", "OUT".

- In regions B and D, the T_{avg} exceeds the deadband T_{db} (0.5 degree C) from its reference value T_{ref} because of the change in reactivity. In region B, because T_{avg} is lower than T_{ref} by the width of the deadband, "Gray" rods would be withdrawn, one bank at a time, to increase reactivity. Conversely in region D, "Grey" rods would be inserted, one bank at a time, to reduce reactivity, because T_{avg} is higher than T_{ref} by the width of the deadband.

- In this PWR Simulator, there are four banks of "Grey" control rods, each bank's reactivity worth is slightly different to enable finer reactivity control at high power: Bank #1 - 6.25 mk; Bank #2 - 5 mk; Bank # 3 - 3 mk; Bank # 4 - 1.75 mk. So the total reactivity worth for all the "Grey" rods is -16 mk. For core power increase, Bank #1 "Grey" Rods will be withdrawn first, followed by Bank #2, Bank #3, and Bank #4. For core power decrease, the sequence for insertion of the banks of "gray" rods will be the reverse.

- In the event that the T_{avg} exceeds the *second* deadband T_{db} (1 degree C) from its reference value T_{ref} due to rapid changes in reactivity, two banks of "Grey" rods would be moved simultaneously to account for the rapid change in reactivity.

NOTE: The four banks of "Grey" rods are normally controlled by the Reactor Power Control (RPC) in "Auto" mode. The control of "Grey" rods can be switched to

"Manual" mode where each Bank can be controlled individually with the control button for "IN", "STOP", "OUT".

- In region E, both the "Dark" rods and "Grey" rods are used simultaneously until the core condition can be transformed into any of the A, B, C, or D regions. Then the reactivity regulation or power shape control can be obtained according to the previously described control logic of each region.
- It should be mentioned that in the event that the "Grey" rods are fully withdrawn, or fully inserted, and core reactivity regulation is still required for reactor power control, the "Dark" rods can be used in a limited way for temporary support to the "Grey" rods.

  As well, the boron concentration control system can be used for relatively long term and slow core reactivity control. However, boron concentration control is used only if necessary, so that the required rod worth is maintained for safe shutdown margin, as well the control rods are kept within the rod position limitations by the control bank rod insertion limit. AUTO/MANUAL control buttons are provided for boron control

  The Screen also shows the reactor core normalized flux intensity map in colour.

- The flux intensity scale is from 0 (grey colour) - 1.2 (red colour).
- The core is divided into 4 quadrants, representing 4 lumped reactor channels. Each lumped channel has 3 sections - lower core, mid-core, and upper core sections. Thus in a simplified way, the 3 dimensional reactor core can be made up of 12 core sections. Each core section's flux intensity is represented by a colour map.
- In conjunction with the flux map of the core, the flow path of the reactor coolant through the core is also shown. Reactor coolant from the U tubes steam generators enters the reactor pressure vessel (RPV) at the respective cold legs entry points- CL1, CL2, CL3, CL4. The reactor coolant then travels down the core downcomer and enters into the core lower plenum, mixes with other reactor coolant streams, before entering the reactor core fuel channels.
- The reactor coolant carries the heat energy from the fuel pellets as it travels up the core channels, exits the core at the upper plenum, and mixes with other coolant streams before leaving the Reactor Pressure Vessel at the two "hot" legs -HL1, HL2.

7.2.8 PWR Reactor Power Control

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

- The status of reactor control is indicated by the four blocks marked MODE, SETBACK, STEPBACK AND SCRAM. They are normally blue but will turn red when in the abnormal state.

  ⇒ MODE will indicate whether the reactor is under TURBINE LEADING to REACTOR LEADING control, this status can also be changed here.

⇒ SETBACK status is indicated by YES or NO; Setback is initiated automatically under the prescribed conditions by RPC, but at times the operator needs to initiate a manual Setback, which is done from this page on the Simulator: the Target value (%) and Rate (%/sec) need to be input.

⇒ STEPBACK status is indicated by YES or NO; Stepback is initiated automatically under the prescribed conditions by RPC, but at times the operator needs to initiate a manual Stepback, which is done from this page on the Simulator: the Target value (%) needs to be input.

⇒ SCRAM status is indicated by YES or NO; scram is initiated by the Shutdown System, if the condition clears, it can be reset from here. Note however, that the scrammed Shutdown System must also be reset before RPC will pull out the shutdown rods, this must be done on the Shutdown Rods Page

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

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

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

⇒ The ACTUAL SETPOINT is set equal to the accepted “REACTOR POWER SETPOINT’ *TARGET* under RPC control in “REACTOR LEADING” mode.

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

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

⇒ DEMANDED POWER SETPOINT is the incremental power target, which is set equal to Current Reactor Power (%) + Rate (% / s) * Program Cycle Time (sec). In this way, the DEMANDED POWER STEPOINT is “ramping” towards the REACTOR POWER SETPOINT Target, at the accepted rate of change.

From the DEMANDED POWER SETPOINT, a Reference Reactor Coolant Temperature (T_{REF}) is obtained from the “T_{ref} versus Power” characteristic curve. T_{REF} is then compared with $T_{AVG,}$ average coolant temperature to determine the temperature difference ΔT.

As well, the POWER ERROR is also determined from Current Reactor Power minus Demanded Power Setpoint. From this, the rate of change of the POWER ERROR between successive RPC Program Cycles will provide the “derivative” term to be used in the control algorithm.

The sum of coolant temperature difference ΔT and the Power Error Derivative, with appropriate gains, will be used as control signal to drive the "Gray" Control Rods, as described in previous Section 2.3. The auto/manual mode (changeable by user), rod speed, and the average position of the "Gray" rods are displayed on this Screen.

Flux detectors are distributed throughout the reactor core to measure the average TOP FLUX (average of the flux intensity of the top four quadrants), and the average BOTTOM FLUX (average of the flux intensity of the bottom four quadrants). The difference minus the deadband yields FLUX TILT ERROR, Δ I, which is used as control signal to drive the "Dark" Rods, as described in Section 2.3. The auto/manual mode (changeable by user), rod speed, and the average position of the "Dark" rods are displayed on this Screen.

⇒ The Rate of Change in Reactor Power is displayed, as result of the Control Rods movement.

⇒ The following time trends are displayed:

- Reactor Power, Thermal Power and Turbine Power (%)
- Coolant ΔT Error (Deg. C)
- Actual and Demanded SP (%)
- Flux Tilt Error (%)
- Dark & Grey Rods Average Position in Core (%)
- Core Reactivity Change (Δk) - mk

7.2.9 PWR Trip Parameters

This Screen displays the parameters that cause REACTOR SCRAM, REACTOR STEPBACK, and REACTOR SETBACK.

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

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

⇒ IMPORTANT NOTE: in this Simulator, certain Trip Parameters can be "disabled" by means of a "ENABLE/DISABLE" switch associated with that parameter. This is ONLY for educational purpose. Its purpose is to allow simulator user to study the various levels of defense actions built into the design in support of reactor safety - that is, in the unlikely event that certain trip parameters malfunction, other trip parameters will come into action, as a consequence. In a realistic NPP, "disabling" of trip parameters is NOT allowed or may be impossible by design.

The TRIP PARAMETERS for REACTOR SCRAM are:

⇒ Low Reactor Outlet Header (Hot legs) Pressure Trip - Trip Setpoint = 14,380 kPa.

⇒ Low Steam Generator Level Trip - Trip Setpoint = 11.94 M

⇒ High Reactor Outlet Pressure Trip - Trip Setpoint = 16,200 kPa
⇒ High Neutron Flux Trip - Trip Setpoint = 120 % of Neutron Flux at full power
⇒ High Log Rate Trip - Trip Setpoint = 8 % /s
⇒ Low Coolant Flow Trip - Trip Setpoint = 2,000 Kg/s
⇒ Low Pressurizer Level Trip - Trip Setpoint = 2.7M
⇒ Low Feedwater Discharge Header - Trip Setpoint = 5200 kPa
⇒ Manual Trip

The causes for REACTOR STEPBACK are:
⇒ High Reactor Coolant Pressure (initiated at P > 16051 kPa; target 2 % FP)
⇒ Loss of one Reactor Coolant Pump (target 2 % FP)
⇒ Loss of two Reactor Coolant Pumps (target 2 % FP)
⇒ High Log Rate (initiated when d(InP)/dt) > 7 %/s; target 2 % FP)
⇒ Manual Stepback (initiated by operator; target set by operator)
⇒ Hi Zone Flux (initiated if Zone Flux is > 115 % of nominal zone flux at full power)

The causes for REACTOR SETBACK are:
⇒ Main Steam Header Pressure Hi - setback if > 6150 kPa
⇒ Hi Pressurizer Level - setback if > 12 M
⇒ Manual Setback in Progress
⇒ Low Steam Generator Level - setback if < 12M
⇒ Low Deaerator Level - setback if < 2M
⇒ Hi Flux Tilt - setback if > 20 %
⇒ Hi Zonal Flux - setback if > 110 %

7.2.10 PWR Reactor Coolant System

This screen shows a layout of the Reactor Coolant System (RCS): two steam generators, four recirculation loops, a pressurizer, and a letdown condenser in the system.

The primary coolant is circulated through four recirculation pumps into the core through the bottom of the Reactor Pressure Vessel (RPV), through four entry points, commonly known as the “cold” legs. There is a pipe that connects one “cold” leg to the letdown condenser. Its purpose is to “bleed” off some reactor coolant from the main circuit in order to maintain inventory, if necessary.

After entering the RPV, the coolant then travels through the fuel channels in the core, and out at the top into the discharge plenum, and exits the Reactor Pressure Vessel at two exit points, commonly known as “hot” legs. The two “hot” legs are connected to two steam generators respectively. As well, there is a pipe connecting one “hot” leg to the pressurizer.

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

The system components and parameters shown are:

- average fuel temperature (°C); average coolant temperature (°C); average core flow (kg/s)); ΔT across the core = coolant outlet temperature - coolant inlet temperature.
- reactor coolant pump's discharge flow (kg); discharge pressure (kPa); discharge temperature (°C)
- reactor coolant pump pop-up control which allows 'START', 'STOP' and 'RESET' operations
- Pressure (kPa), flow (kg/s) and temperature (°C) at the "hot" legs outlet of the Reactor Pressure Vessel.
- Coolant flow (kg/s) to the pressurizer from one "hot" leg. The flow will be shown as +ve if the coolant flows from "hot" leg to the pressurizer; it will be shown as -ve if vice-versa.
- For each steam generator (SG) - feedwater flow (kg/s); feedwater level in drum (m); steam drum pressure (kPa); main steam flow from SG to main steam header (kg/s). For SG1, the feed flow (kg/s) from Chemical & Volume Control System (CVS) is shown. More explanation of this feed flow will be provided in the PWR Coolant Inventory & Pressurizer Screen.
- In the pressurizer, there are five electric on/off heaters, and one variable heater. They are controlled by the Coolant Pressure Control System. The color will be red when heater is 'on'; green when off. The following process parameters are shown: pressurizer vapor pressure (kPa); pressurizer liquid level (m); spray flow into the pressurizer (kg/s), to control pressure; pressure relief flow (kg/s) to letdown condenser to relief over-pressure in the pressurizer.
- The following time trends are displayed:
  - ⇒ The four cold legs temperatures (°C)
  - ⇒ The four cold legs inflow into Reactor Pressure Vessel (kg/s)
  - ⇒ The two hot legs temperatures
  - ⇒ The coolant feed (charging) flow (kg/s); the coolant bleed (letdown) flow (kg/s)
  - ⇒ The four cold legs pressures (kPa)
  - ⇒ Reactor Power (%)

7.2.11 PWR Coolant Inventory and Pressurizer

This screen shows the coolant pressure control system, including the pressurizer, letdown condenser, pressure relief, feed (charging) and bleed (letdown) circuits and coolant makeup storage tank.

- Starting with the coolant makeup storage tank at the bottom left hand corner, its level is displayed in meters. The tank supplies the flow and suction pressure for the Feed (or Charging) pumps P1 and P2: normally one pump is running, the pop-up menu allows START, STOP and RESET operations.
- The Flow (kg/sec) and Temperature (°C) of the feed (charging) flow are displayed. The feed flow then passes through the feed isolation valve MV18

before entering Steam Generator #1, at the suction point of the reactor coolant pumps.

- Flow from the “hot” leg #1 is normally to and from the Pressurizer via a short connecting pipe, a negative flow (kg/sec) indicating flow out of the pressurizer. Pressurizer Pressure (kPa), Temperature(°C) and Level (m) are displayed.
- Pressurizer pressure is maintained by one variable and five on-off heaters which turn ON if the pressure falls, and by Pressure Relief Valves CV22 and CV23 if the pressure is too high. As well, coolant is drawn from connecting lines with the two cold legs (CL1 & CL2) via control valves for the purpose of spraying to depressurize the pressurizer.
- Parameters displayed for the Letdown Condenser are: Pressure (kPa), Temperature(°C) and Level (m).
- There is Bleed (letdown) flow (kg/sec) from “cold” leg #3 via Bleed (letdown) Control valves CV5, CV6 and MV8, which helps maintain coolant inventory in the main coolant circuit, if the inventory becomes too high, as sensed by high pressurizer level.
- The outflow from the Letdown Condenser goes to the Coolant Purification System. From it, the coolant goes to the Coolant Makeup Storage Tank.
- PRESSURIZER LEVEL SETPOINT and REACTOR OUTLET PRESSURE SETPOINT are also shown.
- The Parameters shown for the core are: average fuel temperature (°C); average coolant temperature (°C); core pressure at upper plenum (kPa); average core flow (kg/s)
- The following time trends are displayed:

  ⇒ Pressurizer pressure (kPa); Reactor core outlet pressure (kPa)

  ⇒ Letdown condenser level (m); letdown condenser pressure (kPa)

  ⇒ Pressurizer level (m) and setpoint (m)

  ⇒ Pressurizer spray flow (kg/s)

  ⇒ Coolant bleed (letdown) flow (kg/s); coolant feed (charging) flow (kg/s)

7.2.12 PWR Coolant Inventory Control

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

⇒ Inventory control is achieved by controlling Pressurizer Level.

- Pressurizer Level is normally under computer control, with the setpoint being ramped as a function of reactor power and the expected shrink and swell resulting from the corresponding temperature changes. Level control may be transferred to MANUAL and the SETPOINT can then be controlled manually.
- The amount of feed (charging) and bleed (letdown) is controlled about a bias value that is set to provide a steady flow of bleed to the Purification System. The amount of flow may be adjusted by changing the value of the BIAS. The positions of feed (charging) and bleed (letdown) valves are normally under AUTO control, but may be changed to MANUAL using the pop-up menus.

- The current Reactor Outlet Pressure is shown and the Reactor Outlet Pressure Setpoint (kPa) may be controlled manually via the pop-up menu.
- The following time trends are displayed:
  - ⇒ Reactor neutron power (%); reactor thermal power (%)
  - ⇒ Reactor coolant pressure (kPa) & Setpoint (kPa)
  - ⇒ Pressurizer level (m) & Setpoint (m)
  - ⇒ Reactor coolant makeup feed (charging) valve position (%); Reactor coolant bleed (letdown) valve position (%)

7.2.13 PWR Coolant Pressure Control

This screen is designed to for Reactor Coolant Pressure Control:

- The six HEATERS are normally in AUTO, with the variable Heater (#1) modulating. The other five heaters are either ON or OFF, and under AUTO control. Via the pop-up menus MANUAL operation can be selected, and each heater may be selected to START, STOP or RESET.
  - ⇒ Note: in order to control the variable Heater (#1) MANUALLY, one must use the pop-up menu to switch the control mode from AUTO to MANUAL first, then the control signal to the Heater #1 will be "frozen", as shown in the numeric value display. Observe the display message above the Heater control. If it says: "MAN O/P OK", that means Heater # 1 can now be controlled by the "MAN" pop-up menu. If it says: "MAN O/P NOT OK", that means the MANUAL control signal from the "MAN" pop-up, and the "frozen" control signal to the Heater does not match. One must then use the "MAN" pop-up menu to enter a value equal to the "frozen" numeric value display, then the message will say "MAN O/P OK".
- PRESSURIZER RELIEF VALVES CONTROL is via CV22 and CV23. These are normally in AUTO mode, but may be placed on MANUAL and the valve opening can be controlled manually via pop-up menus.
- PRESSURIZER SPRAY VALVES CONTROL is via SCV1 and SCV2. These are normally in AUTO mode, but may be placed on MANUAL and the valve opening can be controlled manually via pop-up menus.
  - ⇒ NOTE: in order to control the Pressurizer Relief Valves or Pressurizer Spray Valves MANUALLY, one must use the pop-up menu to switch the control mode from AUTO to MANUAL first, then the control signal to the control valve will be "frozen", as shown in the numeric value display. Observe the display message above the valve control. If it says: "MAN O/P OK", that means the control valve can now be controlled by the "MAN" pop-up menu. If it says: "MAN O/P NOT OK", that means the MANUAL control signal from the "MAN" pop-up, and the "frozen" control signal to the control valve does not match. One must then use the "MAN" pop-up menu to enter a value equal to the "frozen" numeric value display, then the message will say "MAN O/P OK".
- The current Reactor Outlet Pressure is shown, and the Reactor Outlet Pressure Setpoint (kPa) may be controlled manually via the pop-up menu.
- The following time trends are displayed:

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

⇒ Reactor outlet pressure (kPa) & Setpoint (kPa)

⇒ Pressurizer level (m) & Setpoint (m)

⇒ Pressurizer Relief Valve position (%)

7.2.14 PWR Turbine Generator

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

- Main steam pressure (kPa) and main steam flow (kg/s) ; Main Steam Stop Valve (MSV) status
- Main steam header pressure (kPa)
- Status of Main Steam Safety Relief Valves (SRVs)
- Control status (Auto/Manual), opening (%) and flow (kg/s) through the Steam Bypass Valves
- Steam Flow to the Turbine (kg/sec)
- Governor Control Valve Position (CV) (% open)
- Generator Output (MW); Station Services (MW)
- Turbine/Generator Speed of Rotation (rpm)
- Generator Breaker Trip Status
- Turbine Trip Status (Tripped or Reset)
- Turbine Control Status - Auto (by computer) or Manual
- The trend displays are:

⇒ Reactor neutron & thermal power (%)

⇒ Generator output (MW)

⇒ Turbine steam flow (kg/s); steam BYPASS flow (kg/s)

⇒ Turbine speed (RPM)

⇒ Turbine governor position (%)

⇒ Main steam Stop Valve (MSV) inlet pressure (kPa)

The following pop-up menus are provided:

- TURBINE RUNBACK - sets Target (%) and Rate (%/sec) of runback when ‘Accept’ is selected
- TURBINE TRIP STATUS - Trip or Reset
- Steam Bypass Valve ‘AUTO/MANUAL’ Control - AUTO Select allows transfer to MANUAL control, following which the Manual Position of the valve may be set.
- Computer or Manual Control of the speeder gear.
- Turbine Runup/Speedup Controls

7.2.15 PWR Feedwater and Extraction Steam

This Screen shows the portion of the feedwater system that includes the condenser, low pressure heater, deaerator, the boiler feed pumps, the high pressure heaters and associated valves, with the feedwater going to the Steam Generator Level Control Valves, after leaving the HP Heaters.

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

- Main steam header pressure (kPa), steam flow through the Turbine Governor Valve and the Bypass Valve (kg/s).
- Deaerator level (m) and deaerator pressure (kPa); extraction steam motorized valve status and controls from turbine extraction, as well pressure controller controls for main steam extraction to deaerator. The extraction steam flows (kg/s) are shown respectively for turbine extraction as well as for main steam extraction to the deaerator.
- Main Feedwater Pump and Auxiliary Feedwater Pump status with associated pop-up menus for 'ON/OFF' controls.
- HP Heater motorized valves MV2 and MV3 and pop-up menus for open and close controls for controlling extraction steam flow to the HP heaters.
- Feedwater flow rate (kg/s) at Boiler Level Control Valve (LCV1 & LCV2) outlet and feedwater temperature (°C).
- Pop-up controls for "Auto/Manual" for Boiler Level Control valves LCV1 & LCV2
- Pop-up controls for changing Boiler Level Setpoint Control from "Computer SP" to "Manual SP", or vice versa.
  - ⇒ NOTE: in order to change the Boiler Setpoint Control from "Computer SP" to "Manual SP", one must use the pop-up menu to switch the control mode from COMPUTER SP to MANUAL SP first, then the "Steam Generator Level SP" value will be "frozen", as shown in the numeric value display. Observe the display message next to SP control status. If it says: "MAN SP OK", that means the boiler level SP can now be controlled by the "MAN SP" pop-up menu. If it says: "MAN SP NOT OK", that means the MANUAL SP value from the "MAN SP" pop-up, and the "frozen" SP value (as displayed) do not match. One must then use the "MAN SP" pop-up menu to enter a value equal to the "frozen" numeric value display, then the message will say "MAN SP OK".
  - The following trends are displayed:
    - ⇒ Reactor neutron power (%); reactor thermal power (%); turbine power (%)
    - ⇒ Steam flow to Deaerator (kg/s)
    - ⇒ Deaerator pressure (kPa) & Setpoint (kPa)
    - ⇒ Main steam header pressure (kPa)
    - ⇒ High Pressure Heaters HX5A, HX5B extraction steam flows (kg/s)
    - ⇒ Steam Generator level (m)

7.2.16 PWR MW Demand Set point (SP) and Steam Generator Pressure Control (SGPC)

- This screen permits control of station load setpoint and its rate of change while under “TURBINE LEADING” Control Mode. Control of the Main Steam Header Pressure is also through this screen, but this is not usually changed under normal operating conditions.
- PWR PLANT CONTROL MODE can be changed from “REACTOR LEADING” to “TURBINE LEADING”.
- TARGET LOAD - on selection Station Load (%) and Rate of Change (%/sec) can be specified; change becomes effective when ‘Accept’ is selected.
  - ⇒ The OPERATOR INPUT TARGET is the desired setpoint inserted by the operator; the CURRENT TARGET will be changed at a TARGET and POWER RATE specified by the operator.
  - ⇒ Note that the RANGE is only an advisory comment, numbers outside the indicated range of values may be input on the Simulator.
- STEAM GENERATOR PRESSURE SETPOINT CONTROL - alters the setpoint of the Steam Generator Pressure Controller, which is rarely done during power operation*. Caution must be exercised when using this feature on the Simulator*. However, this feature can be used for educational study of PWR plant responses under different secondary pressure conditions.
  - ⇒ To change SG pressure setpoint, first use the “SP Mode” pop-up to change the SP mode from “HOLD” to “INCREASE” or “DECREASE”, depending on new pressure setpoint target. After that, use the “Pressure SP Change Rate” pop-up to enter new values for “Pressure SP TARGET” (in MPa), and the “Pressure SP change Rate” (in MPa /minute). Observe that the SP value changes immediately, after the new SP target and Rate are “accepted”. As well, the Main Steam Header Pressure shown in the display will be changed. At any time, if one wants to return the original Pressure Setpoint, just press the button “SP Recovery” once. It can observe that the Pressure SP will recover to 5740 kPa, and the Main Steam Header Pressure will follow accordingly.
- The following trends are provided:
  - ⇒ Reactor neutron power (%); reactor thermal power (%)
  - ⇒ Main steam header pressure (kPa) & Setpoint (kPa)
  - ⇒ Current Target Load (%), and Turbine Power (%)
  - ⇒ Steam Generator 1 & 2 level (m)

7.2.17 PWR Passive Core Cooling

This screen shows the Passive Core Cooling System in an advanced PWR. The Passive Core Cooling System requires no operator actions to mitigate design basis events like Loss of Coolant Accident (LOCA). The system relies on natural forces such as gravity, natural circulation, compressed gas. There are no pumps, fans, diesels, chillers used. Only few valves are used in the System, supported by reliable power sources.

- The System uses three sources of water to maintain core cooling:
  - ⇒ Two Core Makeup Tanks (CMT)
  - ⇒ Two Accumulators
  - ⇒ In-Containment Refueling Water Storage Tank (IRWST)

  Press the button "Passive Core Cooling 3D Diagram" to see the 3 dimensional layout of the Passive Core Cooling System which shows the above three cooling water injection sources.

- All these injection sources are connected directly to two nozzles on the Reactor Pressure Vessel.

⇒ For SMALL LEAKS following transients, or whenever the normal reactor coolant makeup system is unavailable, the water level in the Pressurizer will reach a low-low level, the reactor will be tripped on low-low Pressurizer level. As well, the Reactor Coolant Pumps will be tripped. When that happens, the Discharge Isolation Valves at the two Core Makeup Tanks (CMT), filled with borated water, will open automatically. Because the CMTs are located above the Reactor Coolant System (RCS) loop piping, if the Pressurizer level continues to drop, the borated water in the CMTs would drain into the Reactor Vessel. This will be the initial cooling injection following a small LOCA.

⇒ In the event of large LOCA, where RCS piping may be severed, leading to blowdown of Reactor Coolant System, the initial injection from CMTs will not be able to rapidly refill the Reactor Vessel. Following coolant blowdown due to large LOCA, RCS pressure will drop rapidly, even with CMTs injection. When the RCS pressure drops to such a low pressure that the gas pressure at the Accumulators forces open the check valves that normally isolate the Accumulators from the RCS. The Accumulators will deliver cooling injection, sufficient to respond to the complete severance of the largest RCS pipe by rapidly refilling the Reactor Vessel downcomer and lower plenum.

⇒ With CMTs draining and rapid refilling of RCS by Accumulators taking place, the Passive Cooling System provides approximately 200 °C margin to the maximum peak fuel clad temperature limit (generally designed not to exceed 1000 °C), even with a severe LOCA such as the double-ended rupture of a main reactor coolant pipe.

⇒ LONG TERM INJECTION water is provided by gravity from the Injection Containment Refueling Water Storage Tank (IRWST). IRWST is located in the Containment just above the RCS loops. Normally, the IRWST (at atmospheric pressure) is isolated from the RCS by self-actuating check valves. Therefore in order for the long term injection water to be deployed by IRWST, RCS must be depressurized. This is achieved by the Automatic Depressurization System (ADS), which is made up of four stages of valves to permit a relatively slow, controlled RCS pressure reduction to 180 kPa (note atmospheric pressure is 101 kPa). At this point, the head of water in IRWST is sufficient to overcome the small RCS pressure and the pressure loss in the injection lines, and forces open the check valves to allow injection flow to the reactor core.

  ⇒ The first three stages of ADS are connected to the Pressurizer via set of valves, so that reactor coolant (2 phase fluid) is discharged through spargers into the IRWST.

   - ⇒ The fourth stage of ADS is connected to a hot leg, and discharges through redundant isolation valves to the Containment.
   - ⇒ The various ADS stages are actuated by CMT level, which will be decreasing as a result of draining in response to LOCA.

- ⇒ PASSIVE RESIDUAL HEAT REMOVAL - A passive residual heat removal heat exchanger “PRHR HX” is provided and resides inside the IRWST tank. The PRHR HX inlet is connected to one hot leg exit at the Reactor Vessel, and its outlet is connected to one cold leg entry at the Reactor Vessel. Therefore the PRHR HX serves as a backup residual heat removal source, in addition to the Steam Generator feedwater and steam system. Because PRHR HX resides inside the IRWST tank, the IRWST water volume is sufficient to absorb decay heat for more than 1 hour before the water begins to boil. Once boiling starts, the steam passes to the Containment, condenses on the steel lining in the Containment. After collection, the condensate drains by gravity back to the IRWST, thus conserving the long term cooling inventory for the IRWST.

7.2.18 PASSIVE CONTAINMENT COOLING SYSTEM

The Passive Containment Cooling System (PCS) as shown in the Screen consists of:

- ⇒ The steel containment vessel that encloses the NSSS - Reactor Pressure Vessel, the Reactor Coolant Loops piping and pumps, the Pressurizer, the Steam Generators. It provides the barrier between the NSSS system and the outside atmosphere. As well, it provides the steel heat transfer surface that removes heat inside containment and rejects it to the atmosphere.
- ⇒ Natural air circulation system for heat removal - heat from inside of containment is removed by continuous natural circulation flow of air (drawn through Outside Air Cooling Intakes) over the outside surface of the Containment steel lining.
- ⇒ In case of severe LOCA accident, there may be large amount of steam collected inside Containment Vessel. The natural air cooling is supplemented by evaporation of water. This is achieved by water drains by gravity from a water tank located on top of the containment shield building

- The Screen shows the following parameters:
   - ⇒ Inside Containment steel vessel - pressure (kPa); Temperature (°C)
   - ⇒ IRWST water temperature (°C), IRWST air space pressure (kPa)
   - ⇒ Pressurizer pressure (kPa) and level (m), and animated level shown in blue colour.
   - ⇒ Animated water level for CMTs and Accumulators.
   - ⇒ Average core flow (kg/s); average fuel temperature (°C); average reactor coolant temperature (°C)
- The Screen also shows the various cooling injection flow paths during the various phases of core emergency injection, in the course of a LOCA.

⇒ The cooling injection flow paths during various injection phases are shown by "thick" blue colour lines on the flow diagram displayed by the Screen.

⇒ The injection phases shown include:

(a) Cooling Injection with CMTs

(b) Accumulators in action

(c) RCS Depressurization with ADS

(d) IRWST in action with long term injection

(e) Sump Recovery started

(f) Reactor decay heat removal via PRHR HX

(g) Spray from Dousing Tank in support of containment heat removal

7.3 PWR Operations & Transient Recovery Exercises

7.3.1 Plant Load Maneuvering - Reactor Lead

POWER MANEUVER: 10 % Power Reduction and Return to Full Power

(1) initialize the Simulator to 100%FP.

(2) select "Reactor Power Control" Screen.

(3) run the simulator by pressing the "run" button.

(4) select the Plant Mode to be "REACTOR LEAD".

(5) Record down the following parameters in the "Full Power" column, before power maneuvering.

(5) reduce power using "Reactor Power Setpoint" pop-up.
⇒ press the "Reactor Power Setpoint" pop-up button at the bottom left corner of the screen
⇒ enter "Reactor Power SP target" = 90 %; enter "Power Rate" = 0.5 %/sec, and press "accept"
⇒ observe parameter changes during transient and record comments
⇒ freeze simulator as soon as Reactor Neutron Power just reaches 90% and record parameter values in the column (2) for "90%" power just reached.
⇒ unfreeze simulator and let parameters stabilize, record parameter values in the column (3) for "90%" power stabilized.

(6) explain the responses for -
⇒ Steam generator pressure
⇒ Primary coolant pressure
⇒ Average coolant temperature
⇒ "Grey" Rods and "Dark" Rods movement

(7) return reactor power to 100% FP at 0.5 %/sec by using the "Reactor Power Setpoint" pop-up

(8) when reactor power has returned to 100 % and the parameters have stabilized, unfreeze, record parameter values in the column (4) "return to 100 % stabilized"

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

Table 7.2

Parameter	Unit	(1) Full Power ____%	(2) 90 % just reached	(3) 90 % stabilized	(4) return to 100 % stabilized	Comments
Reactor Neutron Power	%					
Reactor Thermal Power	%					
Reactor Power SP	%					
Actual Setpoint	%					
Demanded Power Setpoint	%					
Demanded Rate Setpoint	%/sec					
Current Reactor Power	%					
Power Error	%					
Average Coolant Temperature - T_{avg}	°C					
Coolant Temperature Reference - T_{ref}	°C					
Grey rods average position in core	%					
Core average top flux	%					
Core average bottom flux	%					
Dark rods average position in core	%					
Boron Concentration	ppm					

7.3.2 Plant Load Maneuvering - Turbine Lead

POWER MANEUVER: 10% Power Reduction and Return to Full Power

- initialize Simulator to 100% full power
- verify that all parameters are consistent with full power operation.

- select the MW Demand SP & SGPC page

  ⇒ change the scale on the "Reactor Pwr & Thermal Pwr" and "Current Target Load & Turbine Pwr" graphs to be between 80 and 110 percent; the "Main Steam Hdr Pressure & SP" to 5000 and 6500 kPa, "Boiler Level" to 10 and 15 meters, and set "Resolution" to "Max Out".

  ⇒ Record down the following parameters in the "Full Power" column (1) of Table 7.3, before power manoeuvring.

Table 7.3

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

  ⇒ Go to "Reactor Power Control Screen", and record down the following parameters in the "Full Power" column (1), before power manoeuvring. (TABLE 7.4)

- go back to "MW Demand Setpoint & SGPC" screen
- reduce unit power in the 'Turbine Lead' mode, i.e.

  ⇒ select the plant mode to be "Turbine Lead"

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

  ⇒ in pop-up menu lower 'target' to 90.00% at a 'Rate' of 1.0 %/sec

  ⇒ 'Accept' and 'Return'

Table 7.4

Parameter	Unit	(1) Full Power ____%	(2) 90 % stabilized	(4) return to 100 % stabilized	Comments
Reactor Neutron P	%				

Power					
Reactor Thermal Power	%				
Reactor Power SP	%				
Actual Setpoint	%				
Demanded Power Setpoint	%				
Demanded Rate Setpoint	%/sec				
Current Reactor Power	%				
Power Error	%				
Average Coolant Temperature - T_{avg}	°C				
Coolant Temperature Reference - T_{ref}	°C				
Gray rods average position in core	%				
Core average top flux	%				
Core average bottom flux	%				
Dark rods average position in core	%				

- observe the response of the displayed parameters until the transients in Reactor Power and Steam Pressure are completed without freezing the Simulator and/or stopping Labview.
- when the parameters have stabilized, freeze the simulator and record the parameter values in column (2) 90 % stabilized of Table 3.2.1. Go to "Reactor Power Control" screen, and record parameter values in column (2) of Table 7.4.
- explain the main changes.

⇒ Why main steam header pressure rises first then drops back to the Steam Pressure Setpoint value, although the Steam Pressure Setpoint value is unchanged?

⇒ Why Steam Generator's Level drops initially and then recovers?

⇒ Turbine Power (%) lags target load (%), but follows it nicely. However, the Reactor Neutron & Thermal Power overshoot beyond 90 % power, but recover later. But their values drift up and down for sometime before they stabilize. Recall previous power maneuvering in "Reactor Leading" mode, the Reactor Neutron & Thermal Power decrease orderly and do not drift as much during power changes. Can you explain why this occurs in this power Maneuvering in Turbine Lead mode? What is the difference in the way reactor power is controlled in “Reactor Lead” mode, versus “Turbine Lead” mode?

- continuing the above operation, raise “UNIT POWER” to 100% at a rate of 1.0%FP/sec.
- when reactor power has returned to 100 %, and the parameters have stabilized, freeze the simulator and record the parameter values in column (3) 100 % stabilized of Table 1. Go to “Reactor Power Control” screen, and record parameters in column (3) of Table 2.
- note any major difference in parameter values between column (3) and column (1). Can you explain why the differences in parameter values, if any?

7.3.3 RC Hot Leg #1 LOCA Break (Loss of Coolant Accident)

This malfunction event causes a “crack” opening at the Reactor Pressure Vessel (RPV) outlet nozzle that connects the RPV Upper Plenum and the Hot Leg #1 piping. This break causes a Loss of Coolant Accident (LOCA) event. Before the malfunction is inserted, it is recommended that the simulator user should be familiar with the design of the Passive Core Injection System as described in Section 2.17 “PWR Passive Core Cooling” screen, before performing this exercise.

⇒ First load the full power Initial Condition (IC) and “Run” the simulator.

⇒ Go to “Reactor Coolant System” screen, and select the malfunction “RC Hot Leg #1 LOCA Break”, then press “Insert MF”, and press “Return”.

⇒ Observe that the “Malfunction Active” alarm is “on”.

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

⇒ Record the RC coolant pressure when the reactor is scrammed.

⇒ After the reactor is scrammed, go to “PWR Passive Core Cooling” screen. On this screen, the injection flow path by the Passive Core Cooling System will be shown in “thick” blue lines, during the various stages of injection cooling, which will be announced by messages displayed on the right side of the screen.

⇒ Record the parameters in the following table 7.5 during the various stages of injection:

Table 7.5

Stages of Injection	CMT in service	ACC in Service	RC Depress Starts	IRWST in service, PRHR HX in service, Sump Recovery starts
Time elapsed after Break[2]	_____ sec after Break	_____ sec after Break	______ sec after Break	_____ sec after Break
Reactor Power (%)				

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

Turbine power (%)				
Reactor Thermal Power (%)				
Break Flow (kg/s)				
Total Injection Flow (kg/s)				
Core Flow (kg/s)				
Tavg (°C)				
Fuel Temp (°C)				
PRZR level (m)				
PRZR Pressure (kPa)				
Coolant Pressure at Cold Legs (kPa)				
Containment Pressure (kPa)				
Containment Temp (°C)				
CMT Level (% full)				
ACC Level (% full)				
IRWST tank temp (°C)				

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

⇒ Explain why the Accumulator is necessary? Can the accumulator be eliminated if we have a very large Core Makeup Tank (CMT) instead?

⇒ Explain why the RC Depressurization is necessary - to serve what purpose?

8. Verification of the radiation level during reactor operation

8.1 Introduction

The TRIGA Mark II reactor will be operated at a constant power level of 250 kW. During this operation, different points in the reactor hall, the reactor platform and the control room will be measured. The main interest in this exercise focuses on the gamma and neutron radiation.

In the protocol the various measuring points and the corresponding dose rates should be recorded in the horizontal and vertical cross section of the reactor. Points with a dose rate higher than 10 µS/h should be highlighted.

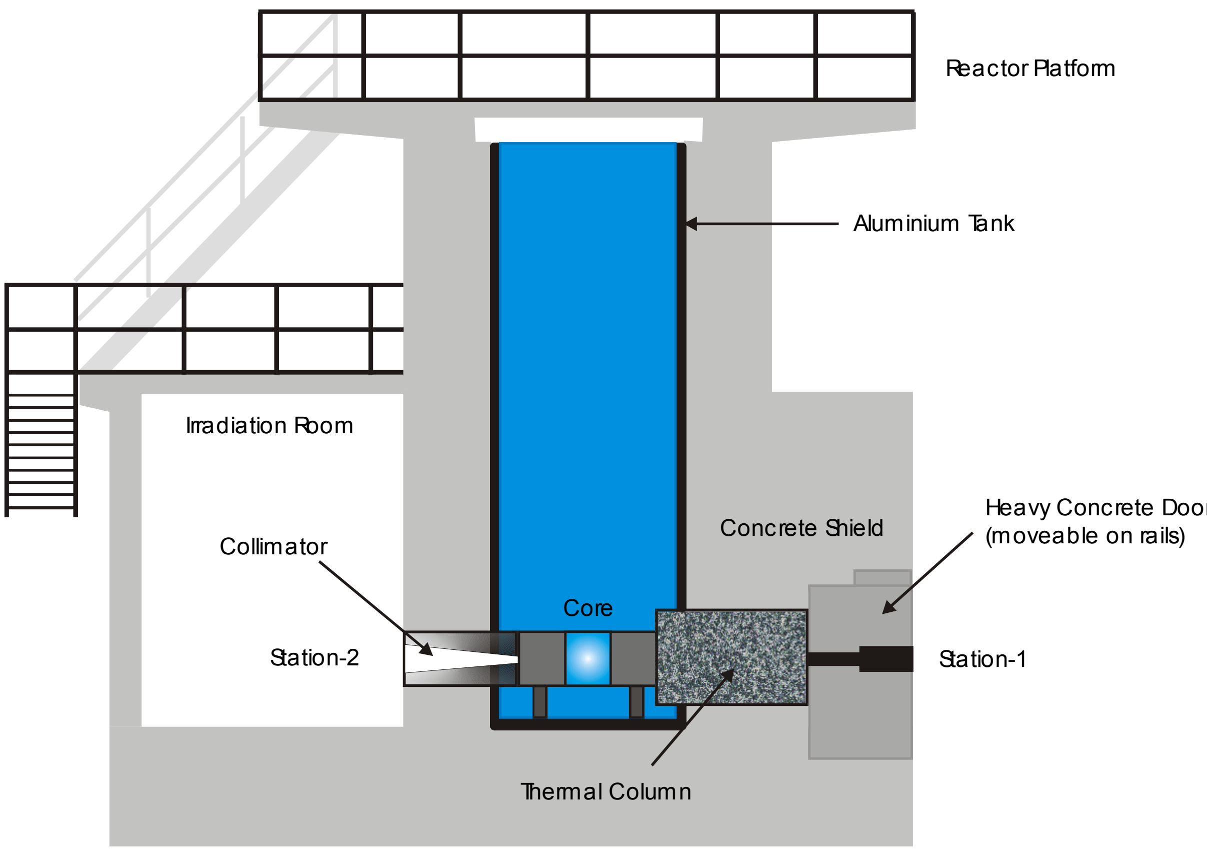
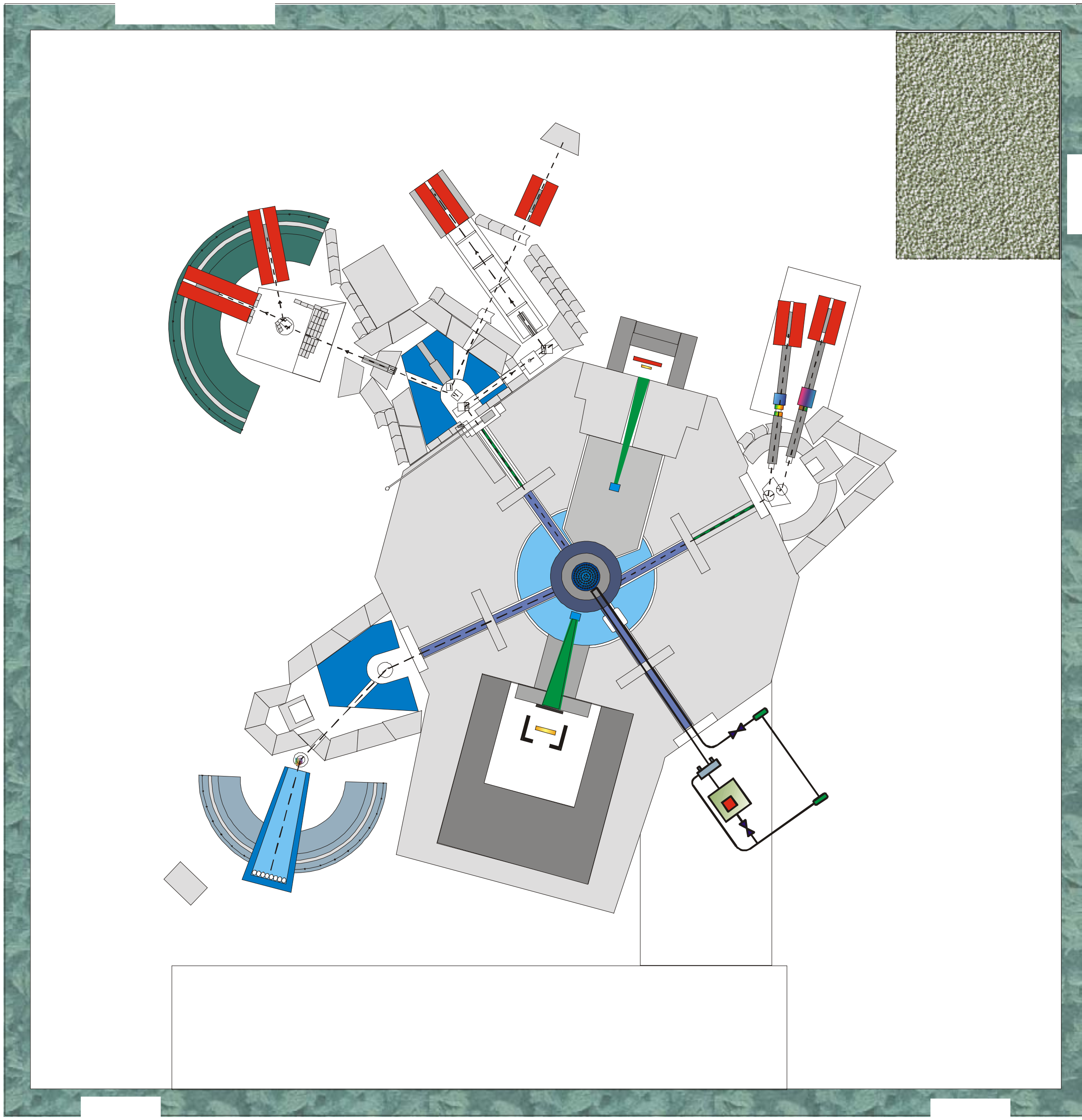


Fig. 8.1: Vertical cross section



STORAGE ROOM FOR
LOW ACTIVE WASTE
1st Floor

STORAGE ROOM FOR
LOW ACTIVE WASTE
2nd Floor

Fig. 8.2 Horizontal cross section

Appendix. Neutron-sensitive thermocouples

A.1 Introduction

The method of measuring the neutron flux based on thermal effects resulting from the interaction of neutrons with materials with high neutron absorption cross-section was developed as early as 1949. Because of the high amount of thermal energy released during reaction with neutrons only the materials ^{10}B and natural and enriched uranium are of considerable importance. Both the $^{10}B(n,\alpha)^{7}Li$ and the U(n,f) reactions result in reaction products which dissipate enough energy during their slowing-down process that α temperature increase of the surrounding material can be measured. In one special case also the $^{14}N(n,p)^{14}C$ reaction has been used.

The main characteristics of neutron-sensitive thermocouples are
(a) Range of operation between 10^9 to 10^{14} $cm^{-2}s^{-1}$
(b) Sensitivity of about 1 mV at 10^{12} $cm^{-2}s^{-1}$
(c) Response time of about 1 s
(d) Proportional behaviour over several decades
(e) Limited lifetime due to the detector burn-up.

A.2 Specific detectors

Generally there must be a distinction drawn between thermopiles and thermocouples. The more complicated thermopiles were developed in the beginning because of the demand to measure a rather low neutron flux (about 10^{12} $cm^{-2}s^{-1}$) with an acceptable signal strength. The sensitivity was between 10^{-13} to 10^{-14} $V.cm^2.s$ for systems composed of up to 42 thermocouples. The response times were between 4 to 20 s. Other detector systems with a shorter response time (about 250×10^{-3} s) were only designed for an operation time of only about 100 hours in the core. As the neutron flux and the power density in the core increased, high detector sensitivity was no longer the most important aim in detector design. On the contrary, because of the limited space in the reactor core individual, miniature neutron-sensitive thermocouples were developed. More emphasis was now put on good proportional behaviour, low gamma-sensitivity for high temperature application and on low detector burn-up. Neutron-sensitive thermocouples have also been developed for application in nuclear power plants (Fig. A.1 and A.2).

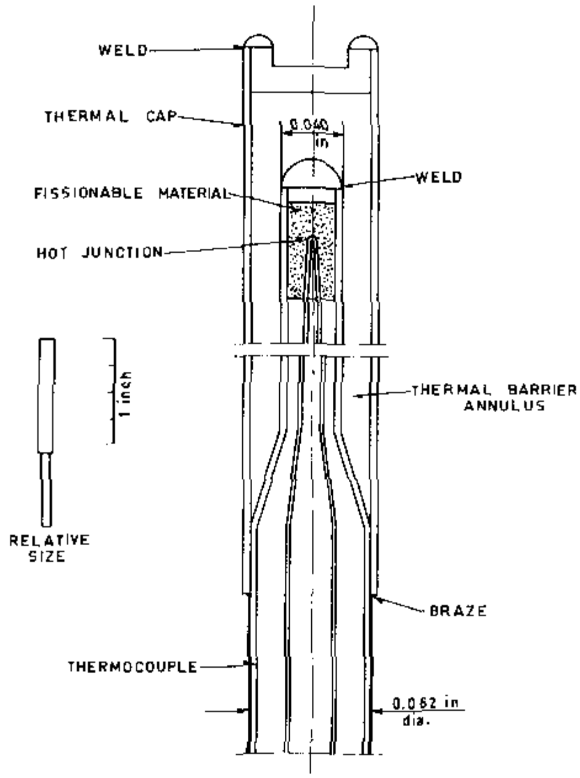


Fig. A.1: Neutron sensitive thermocouple model 1

In one case the sensor is designed as a difference thermocouple: One thermocouple is composed of the Ni-NiCr junction. The first junction is heated by the released fission energy from UO_2, which is dissipated from NiCr via a thermal connection to the outer casing and to the coolant. The temperature difference between the two thermocouples is independent of the absolute temperature if radiation losses and conduction is negligible and only proportional to the fission energy generated in the uranium. In this case the temperature difference is also proportional to the local neutron flux. In another design the thermocouples installed in a German NPP had a sensitivity of 1.5×10^{-16} V.cm^2.s, revealed good operational history and were installed especially to measure the long-term behaviour and the burn-up. Originally it was intended at a later stage to connect the in-core detectors to the reactor protection system. However, miniature fission chambers have been successfully developed in parallel and, therefore, it was decided to install in future only in-core fission chambers because of their dual-purpose properties. This is the main reason why neutron-sensitive thermocouples did not gain the same importance as other in-core detector types in spite of their excellent in-core behaviour.

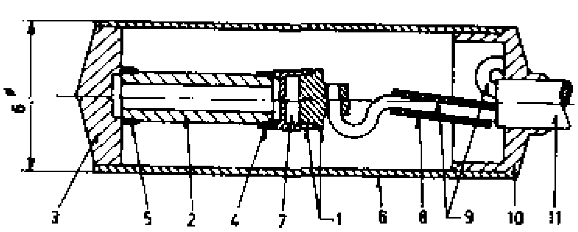


Fig. A.2: Neutron sensitive thermocouple model

1-Ni capsule; 2-CrNi thermal bridge; 3-Ni end plug; 4-hot junction: 5-cold junction; 6-Ni mantle; 7-UO_2 powder; 8-Al_2O_3 tube; 9-Ni cable; 10-Ni end plug; 11-Inconel cable.