ATOMIC ENERGY OF CANADA LIMITED
Power Projects, Sheridan Park, Ontario

Lecture 9

Nuclear power symposium

REACTOR AND STATION CONTROL

E.M. Yaremy

HEAD, REACTOR AND STATION CONTROL BRANCH

ATOMIC ENERGY OF CANADA LIMITED
Power Projects

NUCLEAR POWER SYMPOSIUM

LECTURE NO. 9: REACTOR AND STATION CONTROL

by

E. M. Yaremy

1. INTRODUCTION

To some people, nuclear reactors may seem like very complex pieces of equipment that require very specialized skills, especially in the design of the control systems associated with the reactor itself.

However, this is not completely the case. The same type of skills are required in the design of controls for nuclear power plants as are required in the design of controls for conventional processes such as oil-fired power plants, chemical plants, etc. In our design groups, we have graduates in mechanical engineering, electrical engineering, and engineering physics. Most of the individuals have worked in other industries not associated with the nuclear power field before coming to AECL. They have been able to apply this previous experience while at the same time learning the necessary skills unique to the nuclear field.

Generally speaking the type of instrumentation and control equipment which is used throughout the nuclear plant is standard commercially available equipment similar to the type used in other industries. For example, the total value of the instrumentation and control equipment associated with the nuclear steam supply system of a single unit plant is approximately $5, 000, 000. Of this, less than $500, 000 could be interpreted as being for purely nuclear type instrumentation and control equipment which is only manufactured for the nuclear industry and not used elsewhere.

However, due to the large number of measurements involved in a nuclear power plant there has been a strong incentive towards the use of computers in these plants, and as a result I would say that we have pioneered in the use of computers in the direct control of processes.

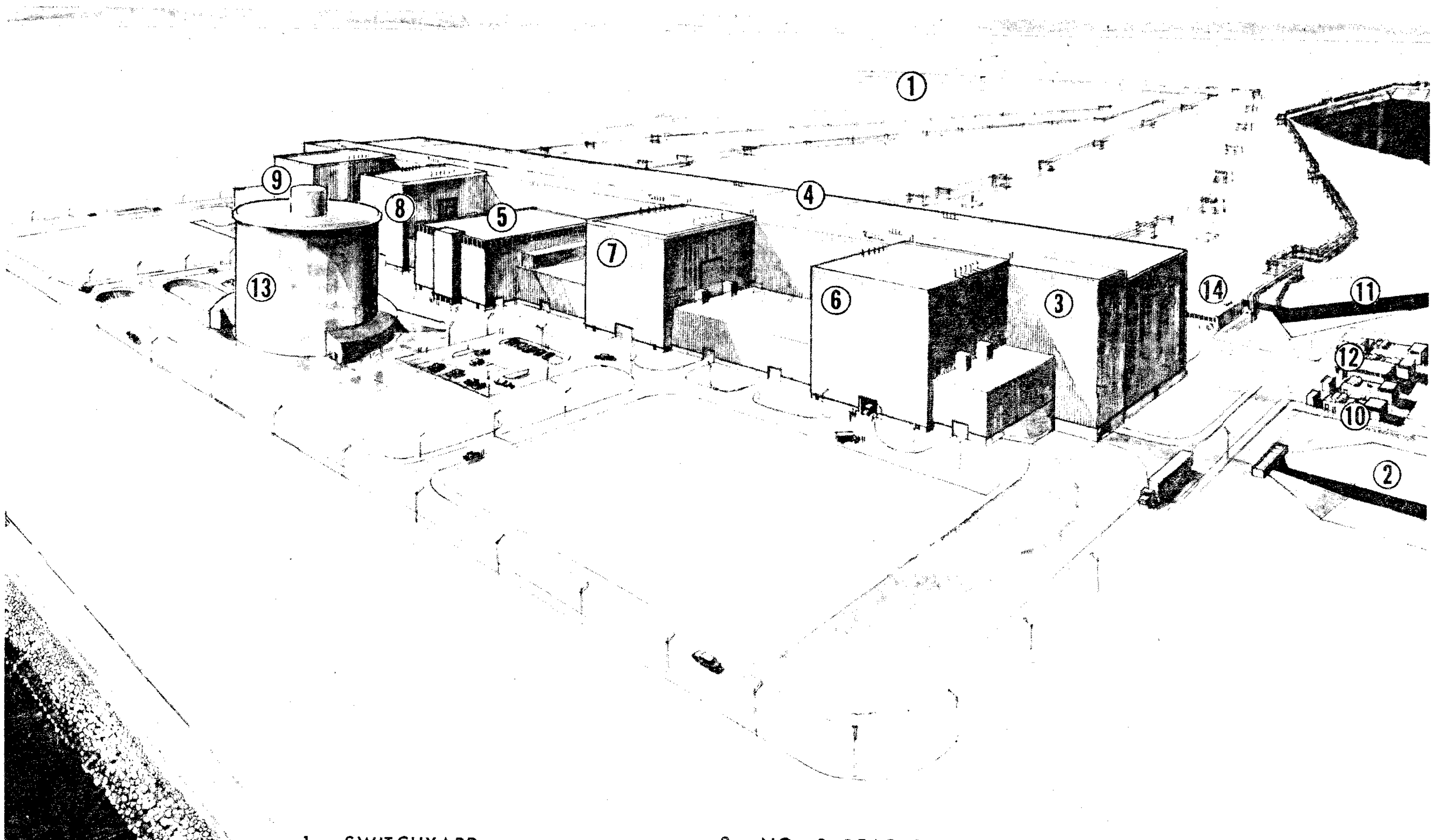
2. GENERAL DESIGN CONSIDERATIONS

General considerations which have strongly influenced the design of the control systems in a nuclear plant are (1) the large number of variables involved, (2) the unfriendly environment, (3) heavy water costs, and (4) reliability.

Figure 1 shows the general view of a four-unit plant such as Bruce. The control centre is located under the number 5 which is 500 feet away from Reactor Unit 4 (number 6 on the figure). Figure 2 shows an elevation view of the reactor containment vault. Most of the main processes and equipment are inside this vault area, which is inaccessible during operation due to the radiation fields. At the same time, most of the measurements must be made within this area and transmitted to the control centre. The fact that the process equipment is inaccessible also influences the amount and type of instrumentation necessary to enable the plant operator to know what is going on at all times.

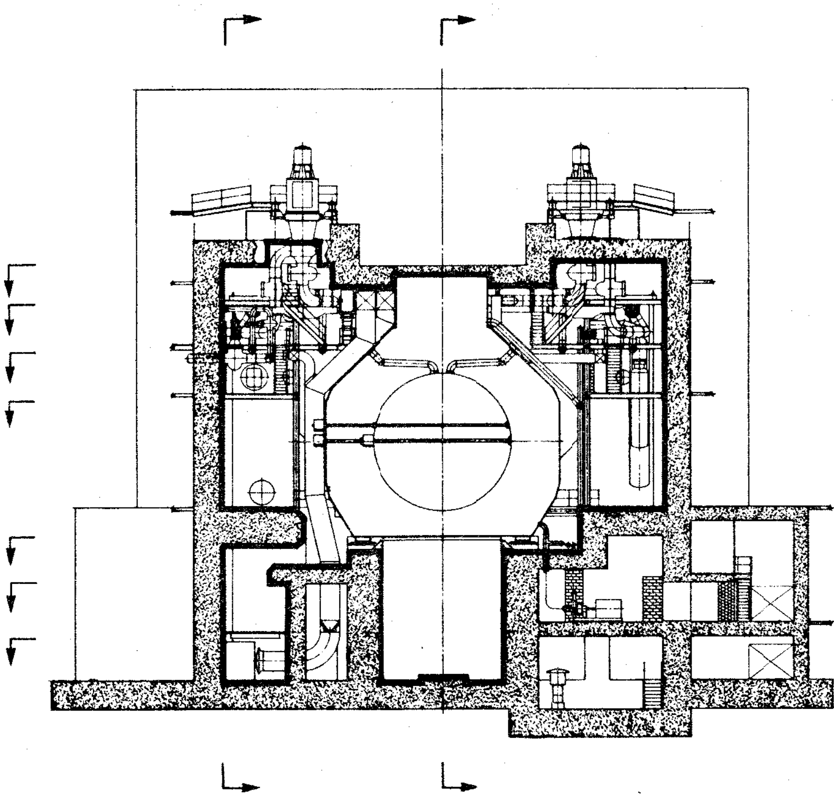
An example to illustrate the numbers of variables which must be processed can be made by referring to Figure 3. This figure shows the end view of a reactor and the 400 coolant channels with the individual channel feeder pipes which transfer the heat from the reactor to the boilers. There could be up to two temperature measurements per feeder pipe required, giving a total of 800 measurements from this source alone.

To minimize the heavy water hold-up in instrument impulse lines and especially to keep the number of physical joints to a minimum (the problem is not just minimizing the amount of heavy water in the lines costing $27 per pound, but minimizing the number of potential leak points from which the heavy water can escape), it would be desirable to put the process type instruments adjacent to the process itself (which is normal practice in other industries). If all instruments had a reliability of 100%, putting them inside the inaccessible vault would be alright. However, since we do not have such perfection it is necessary to locate these instruments outside the vault connected to the process by 100 feet impulse lines. Great effort is made to use a minimum number of valves and fittings, using welded construction wherever possible. Where it is not possible to locate instrumentation remotely we attempt to use the best reliable equipment that can be made, or use redundant instrumentation such that a single failure will not cause a forced shutdown of the plant.



1	SWITCHYARD	8	NO. 3 REACTOR BUILDING
2	DISCHARGE DUCT	9	NO. 4 REACTOR BUILDING
3	POWER HOUSE	10	STANDBY AUXILIARY GENERATOR
4	TURBINE HALL	11	INTAKE
5	SERVICE BUILDING	12	FUEL OIL PUMPHOUSE
6	NO. 1 REACTOR BUILDING	13	VACUUM BUILDING
7	NO. 2 REACTOR BUILDING	14	PUMPHOUSE UNIT 1

Figure 1 Bruce Generating Station - Looking South



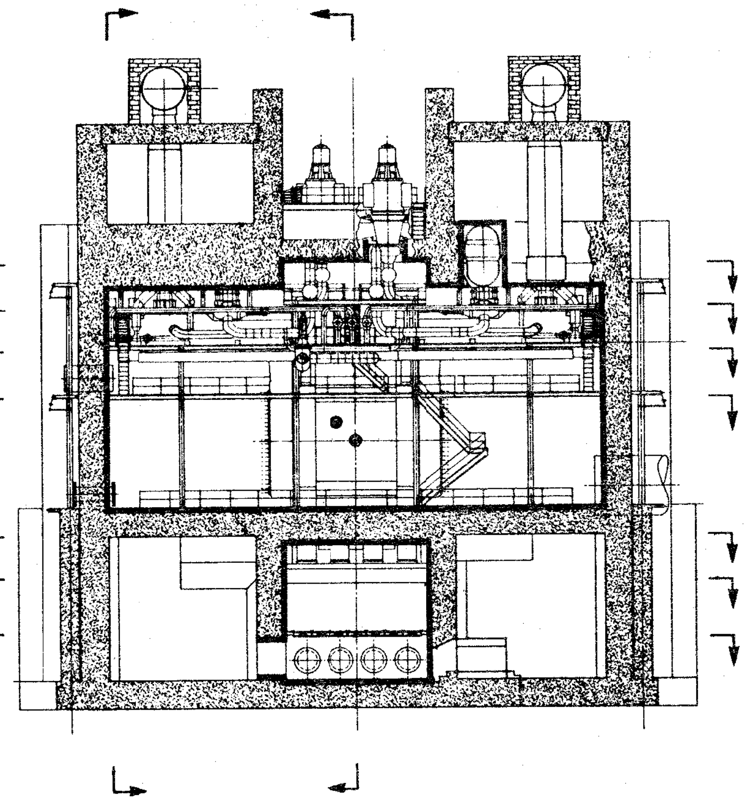


Figure 2 Bruce G. S. Reactor Building, Plans and Elevations

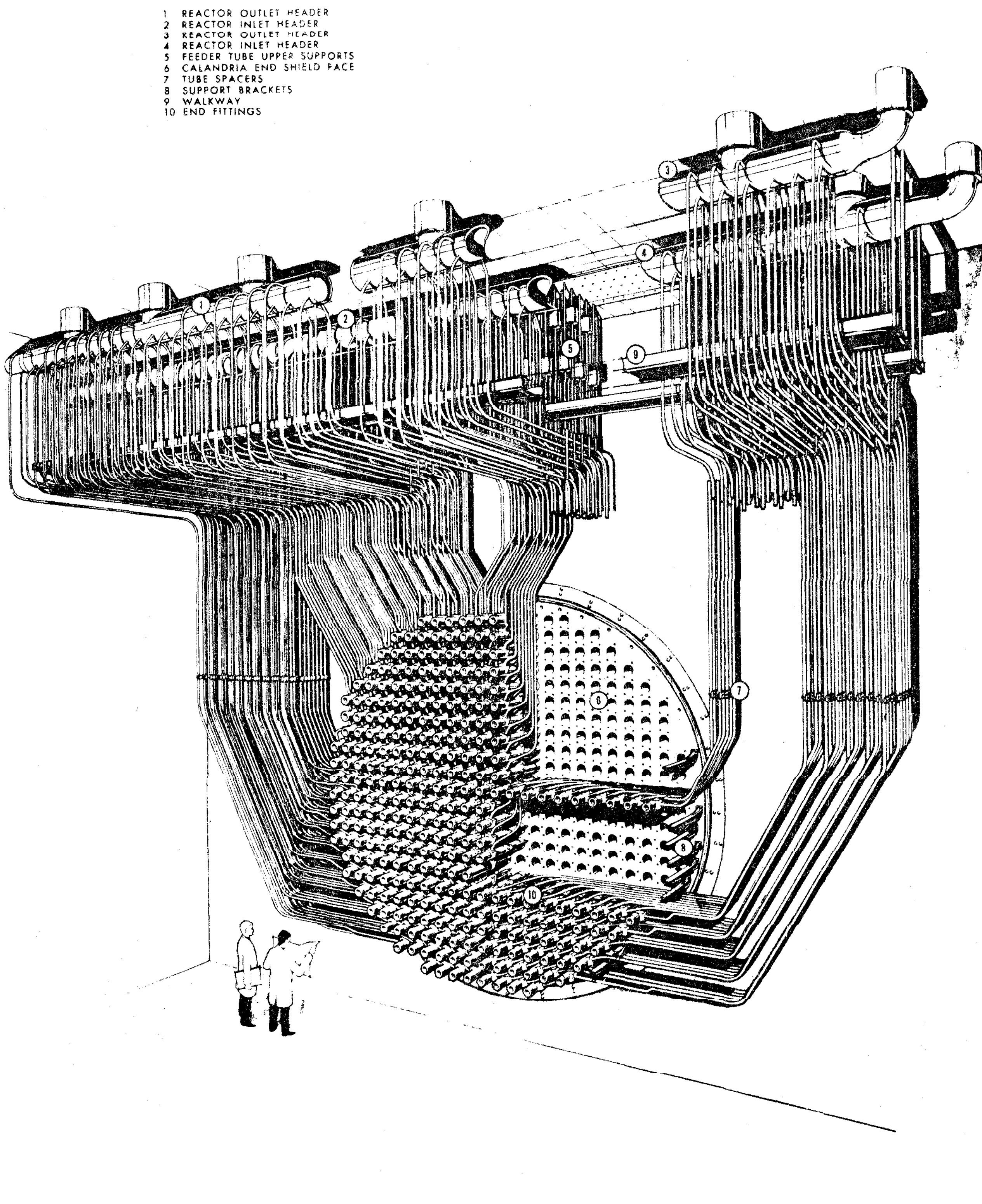


Figure 3 Pickering G. S. - Feeder Tube Arrangement

Redundancy is not really the complete answer since the failed device must be replaced. Even if this is done during a shutdown, there is a certain radiation burnup exposure of maintenance personnel involved, since the radiation fields inside the vault do not go to zero.

Actually there is one horrible example of how not to design instrumentation systems which we use to illustrate this point to our own designers. For a similar layout as shown on Figure 3, we located resistance temperature detectors in the reactor inlet and outlet headers of the Douglas Point plant. Because of the desired speed of response these detectors were installed without thermowells using compression type fittings. Not only did they leak heavy water, but they soon failed completely. To replace six detectors caused a radiation dose of 70 man-rems. A maintenance employee is normally limited to 3 rem exposure per year. Therefore the replacement of these detectors used up the equivalent of over 20 men who could not be used on maintenance of nuclear components for a year. A figure usually used is that 1 rem of radiation exposure costs up to $5,000. You can figure out yourself how much the replacement cost.

3. REACTOR CONTROL THEORY

3.1 Reactor Kinetics

As discussed in Lecture 4 (Reactor Physics), the reactor is an inherently logarithmic device, since its power changes are based on neutron multiplication (rather than addition, which would lead to a linear device).

The heat generated within the reactor is directly proportional to the number of fissions per second, which in turn is proportional to the neutron flux.

The whole subject of reactor kinetics is concerned with one parameter, the multiplication factor. The multiplication factor determines the basic state of the reactor (i.e., whether reactor power is increasing, decreasing or holding steady).

The multiplication factor k_{eff} is the ratio of the number of neutrons in any one generation to the number of neutrons in the immediately preceding generation (in a finite sized reactor). If k_{eff} is unity, the reactor is critical and at steady power. If k_{eff} is greater than unity the power level increases exponentially; if k_{eff} is less than unity the chain reaction cannot persist, and the power must ultimately die down to a very low level.

For control studies, we talk of reactivity, which in fact is excess reactivity, and is defined as follows:

$$\Delta k = \text{excess reactivity} = k_{eff} - 1$$

usually expressed in milli-k (one thousandth of a "k"). For a critical reactor at steady power the reactivity $\Delta k = 0$ (the reactor could be at any power from 1 watt to 750 MW).

Reactor operation is defined by equations of the following form:

$$\frac{d\phi}{dt} = \frac{(\Delta k - \beta)\phi}{L^*} + \sum_{i=1}^{6} \lambda_i C_i \qquad \text{........(1)}$$

$$\frac{dC_i}{dt} = \frac{\beta_i \phi}{L^*} - \lambda_i C_i \qquad \text{........(2)}$$

where ϕ = neutron flux
- k = reactivity
- β = total delayed neutron fraction $= \Sigma\beta_i$
- L^* = effective mean neutron lifetime
- λ_i = inverse time constant of i'th delayed neutron group
- β_i = delayed neutron fraction of the i'th group
- C_i = concentration of neutrons in the i'th delayed neutron group.

As discussed in Lecture 4 the first part of equation (1) represents the prompt component of neutron flux response, and the second part represents the effect of the delayed neutron groups. The above equations lump the delayed neutron effects into m groups. In the simulation of the reactor for our overall plant control behaviour studies we generally use 6 groups.

The effect of the delayed neutron groups can be shown by a simple calculation of a +1 mk step input with and without the delayed neutron effect ($L^* = 10^{-3}$ sec, $\beta = 0$ and .0060, $\lambda = 0.1$/sec)

With $\beta = 0$ $$\frac{d\phi}{dt} = \frac{\Delta k \phi}{L^*}$$

solving $$\frac{\phi(t)}{\phi(0)} = e^{\frac{\Delta k}{L^*} t} \qquad \text{........(3)}$$

With $\beta = .0060$ with one delayed neutron group

$$\frac{d\phi}{dt} = \frac{(\Delta k - \beta)\phi}{L^*} + \lambda C$$

$$\frac{dC}{dt} = \frac{\beta\phi}{L^*} - \lambda C$$

solving $$\frac{\phi(t)}{\phi(0)} = \frac{\beta}{\beta - \Delta k} e^{\left(\frac{\lambda \Delta k}{\beta - \Delta k}\right)t} - \frac{\Delta k}{\beta - \Delta k} e^{-\left(\frac{\beta - \Delta k}{L^*}\right)t} \quad \ldots\ldots(4)$$

solving (3), for $t = 3$ sec, $\frac{\phi(3)}{\phi(0)} = 20$

solving (4), for $t = 3$ sec, $\frac{\phi(3)}{\phi(0)} = 1.3$

This example illustrates the effect of 0.60% fraction of delayed neutrons on the reactor response.

The above time responses are plotted in linear fashion in Figure 4 and in logarithmic fashion in Figure 5. The example illustrates that a +1 mk change in reactivity causes power to increase at 2%/second (a good rough rule of thumb).

Figure 4 illustrates the fact that there is a prompt flux change due to the prompt neutron effect followed by the long term exponential change. This prompt jump has the value of:

$$\frac{\beta}{\beta - \Delta k}$$

Because of the exponential nature of the reactor response rates of change of power are expressed in "Log Rate" terms, where

$$\text{Log Rate (\%/sec)} = \frac{100}{T}$$

Where T is the time to change the power level by a factor e (2.718).

The Rate Log is a measure of how fast the <u>existing</u> power level is changing.

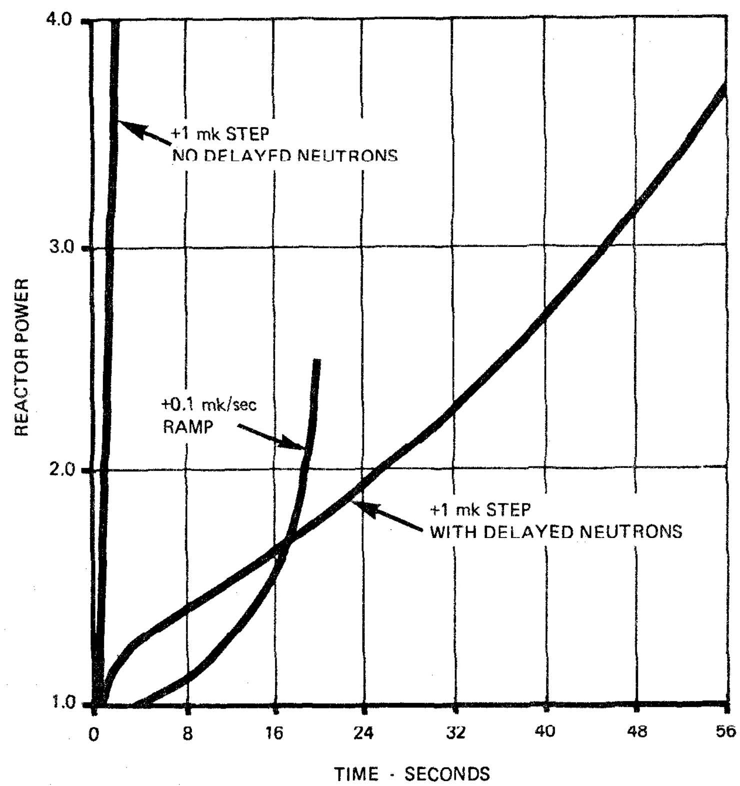


Figure 4 Reactor Response - Plotted Linearly

In an actual reactor step changes in reactivity cannot occur. The usual type of reactivity insertion is a ramp change. In Figure 4 we also illustrate a typical response of a 0.1 mk/sec ramp input such as when a booster rod or absorber rod is moved within the reactor.

Since the reactor is a logarithmic device and the absolute power response to a perturbation is a function of power level, the controls associated with the reactor must have gain terms which are also functions of power, in order to achieve stable control over all levels of power operation. For this task a digital computer is also very well suited.

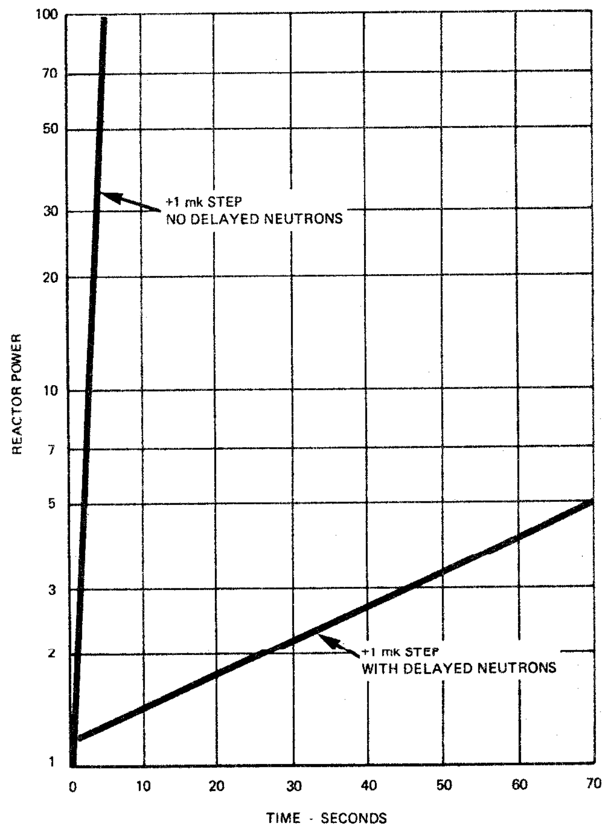


Figure 5 Reactor Response - Plotted Logarithmically

3.2 Reactivity Feedbacks

To this point we have neglected the reactivity feedbacks due to coefficients changing with power and due to the poisoning of fission products. The complete reactivity feedback loops are as shown in the block diagram of Figure 6. Typical values for these feedback effects for a pressurized heavy water reactor for a 0-100% power change are as follows:

	Fresh Fuel 0-100%	Equilibrium Fuel 0-100%
Fuel Temperature	-11 mk	-3 mk
Coolant Temperature	+0.2 mk	+0.9 mk
Coolant Density	0	0
Moderator Temperature	-1.1 mk	+2.0 mk

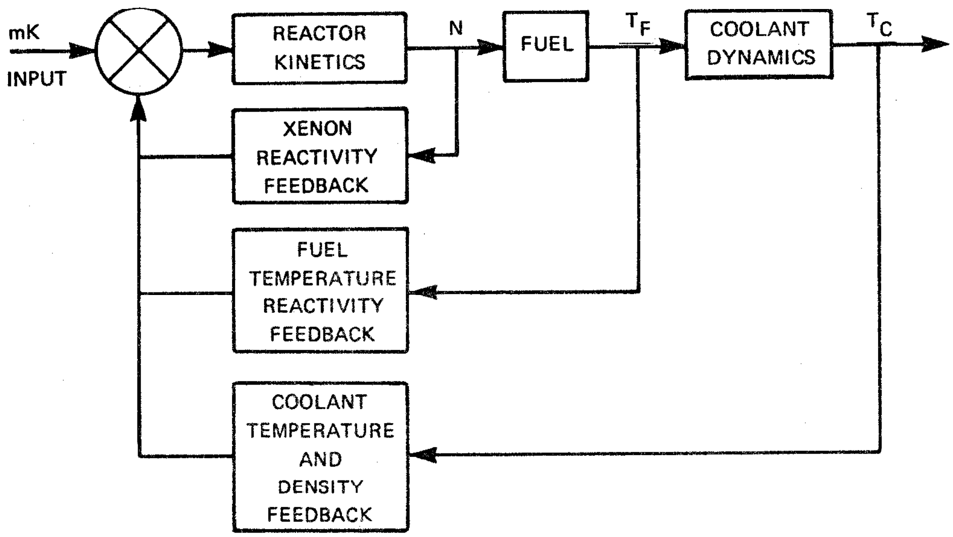


Figure 6 Reactor Feedback

The remaining feedback effect in the block diagram is one due to absorption of neutrons by fission products. The most notorious of these fission product poisons is Xenon-135. There are other fission product poisons but they are usually treated as a second order effect upon the Xenon poisoning.

Xe-135 is formed by the decay of I-135 (6.7 hour half-life). Xe-135 has a very high neutron absorption cross section, and a natural decay half life of 9.2 hours. Xe-135 is destroyed by neutron capture and by decay. In a high flux power reactor the neutron capture term is the dominant one incurring a steady state reactivity loss of 25-30 mk.

The behaviour of Xenon is shown in Figures 7 through 10:

(1) Figure 7 shows the buildup of Xenon after startup.

(2) Figure 8 shows the buildup and decay of Xenon after a complete shutdown from various power levels. The initial buildup of Xenon is due to the Xenon production remaining the same (6.7 hour half life of I-135) but the destruction of Xenon having decreased (the only destruction mechanism is natural decay).

(3) Figure 9 shows the Xenon buildup and decay due to a step decrease in power. The results are similar to those of Figure 8.

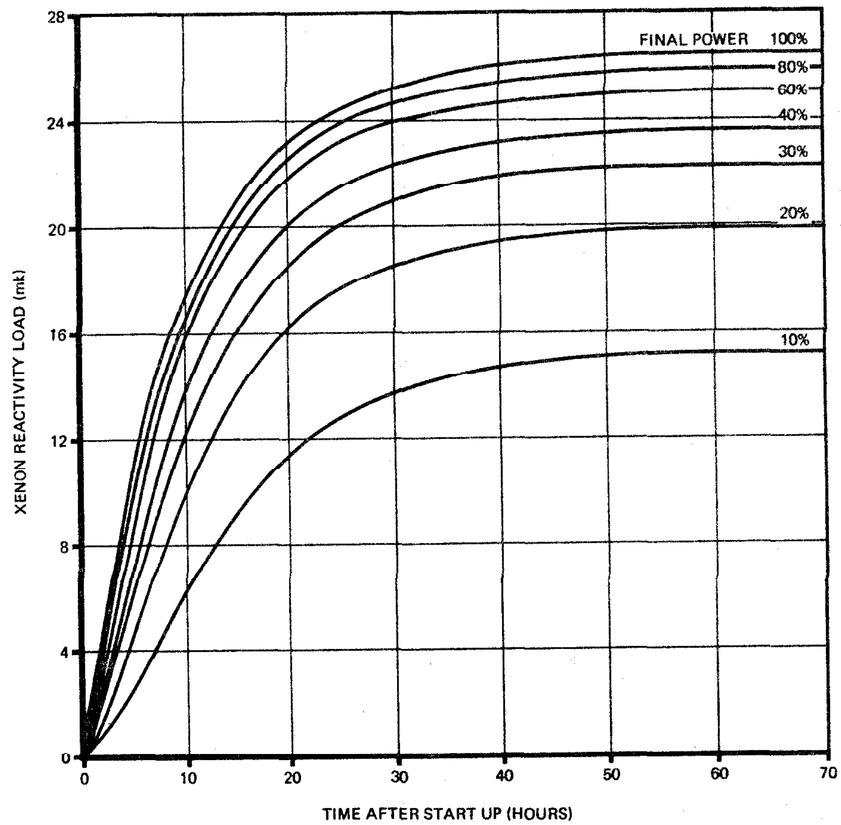


Figure 7 Buildup of Xenon After Reactor Startup

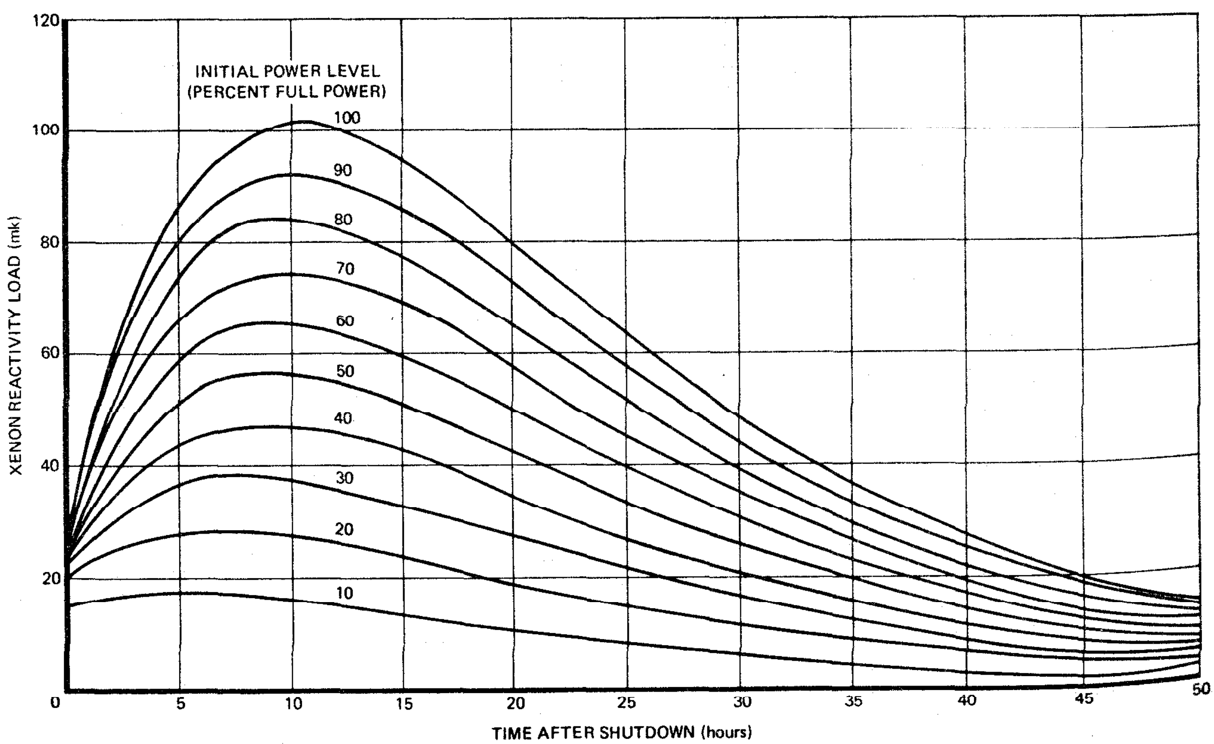


Figure 8 Xenon Transient Following Reactor Shutdown
From Various Initial Powers

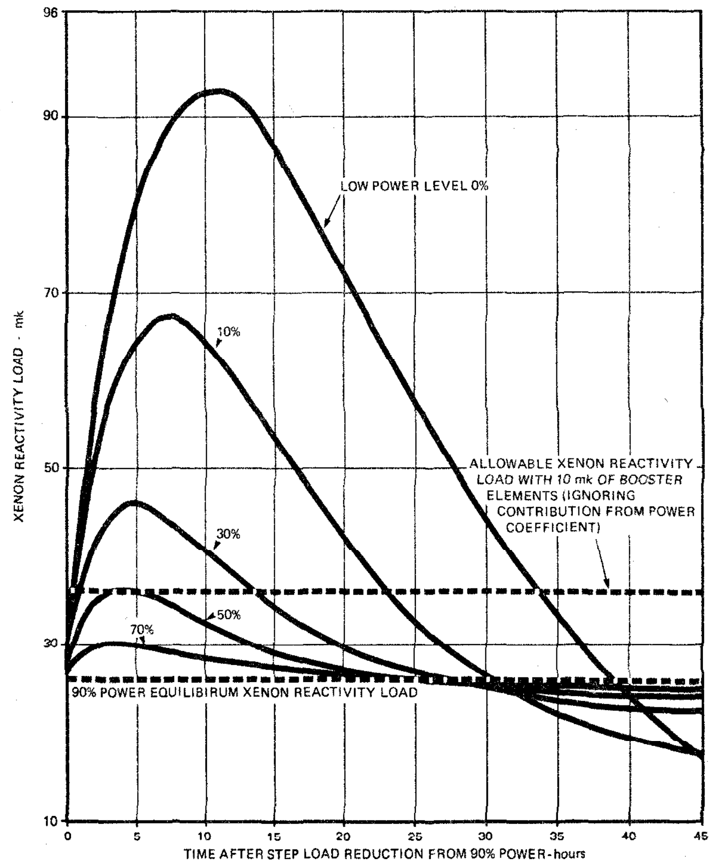


Figure 9 Effect of Step-Power Changes on Reactivity

(4) Figure 10 shows the Xenon transient following step increases in power. The response is the reverse of that described under Figure 8. Due to the neutron flux increase the Xenon destruction rate increases while production initially remains constant.

The Xenon reactivity therefore decreases initially, then builds up to its equilibrium level.

Figure 11 illustrates a simple balance sheet for reactivity. This is based on a similar core to the Bruce station with an excess reactivity of 110 mk (as discussed in Section 8.1 of Lecture 4). In order to maintain the reactor critical at any power level, Available Reactivity must equal Reactivity Usage.

During reactor startup after a prolonged shutdown (with the Xenon decayed away), the equivalent of 30 mk of neutron absorber must be added to the reactor and removed to compensate for the buildup of Xenon poison (refer to Figure 7).

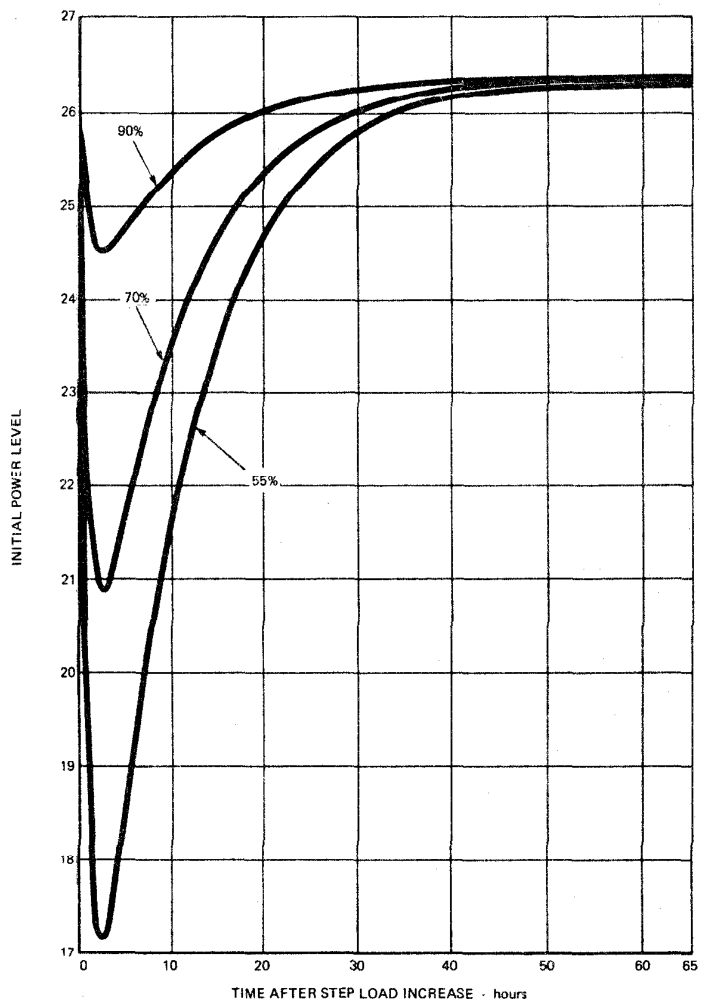


Figure 10 Xenon Transient Following Step Load Increase From Various Initial Powers to 100% Final Power

During initial reactor startup with fresh fuel the equivalent of 60 mk of neutron absorber must be added to the reactor and removed as the fuel burns down.

After decreasing reactor power to a lower value, the Xenon load builds up (Figures 8 and 9). When this buildup exceeds the amount of other absorption which can be removed from the reactor (e.g. 3 mk in the control elements) the reactor will "poison out" and go to zero power. The only other alternative is to add to the "Available Reactivity" by inserting booster elements (containing additional fuel) to balance the actual reactivity usage during the Xenon transient.

After increasing reactor power to a higher value, the Xenon load transiently decreases (Figure 10). Additional neutron absorber must be added to the reactor and then removed to compensate for this transient.

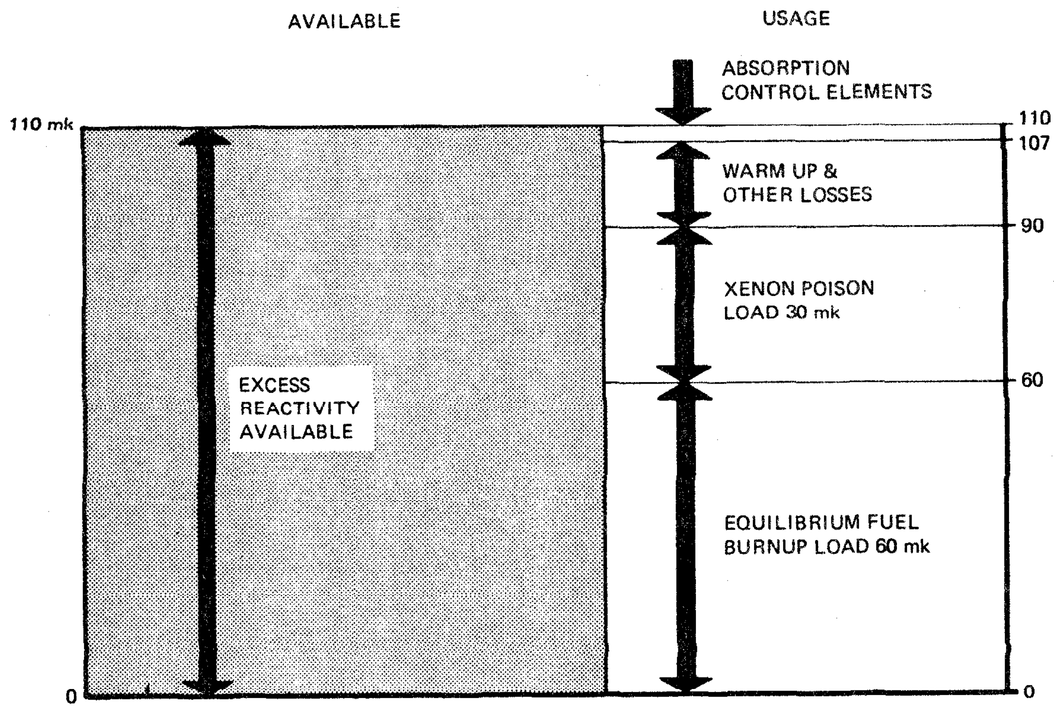


Figure 11 Reactivity Balance Sheet

3.3 Reactor Control Characteristics

A typical reactor startup at maximum rate is illustrated in Figure 12. During the shutdown the reactor would be maintained at approximately 0.1% of full power. Up to 25% full power the startup would proceed at a log rate of 4%/second. Above 25% power the rate would be maintained at a linear rate of 1% full power/second.

Figure 13 illustrates a typical maximum shutdown rate (reactor trip).

Figure 14 illustrates a typical recovery after a fast shutdown. The curve portrays a reactor with 18 mk worth of boosters. Using the boosters it is necessary to reach 60% full power within 40 minutes after the shutdown in order to prevent the Xenon transient becoming so large as to "poison out" the reactor and prevent startup until the Xenon load decays back down again (refer to Figure 8). This poison-out period would last approximately 40 hours.

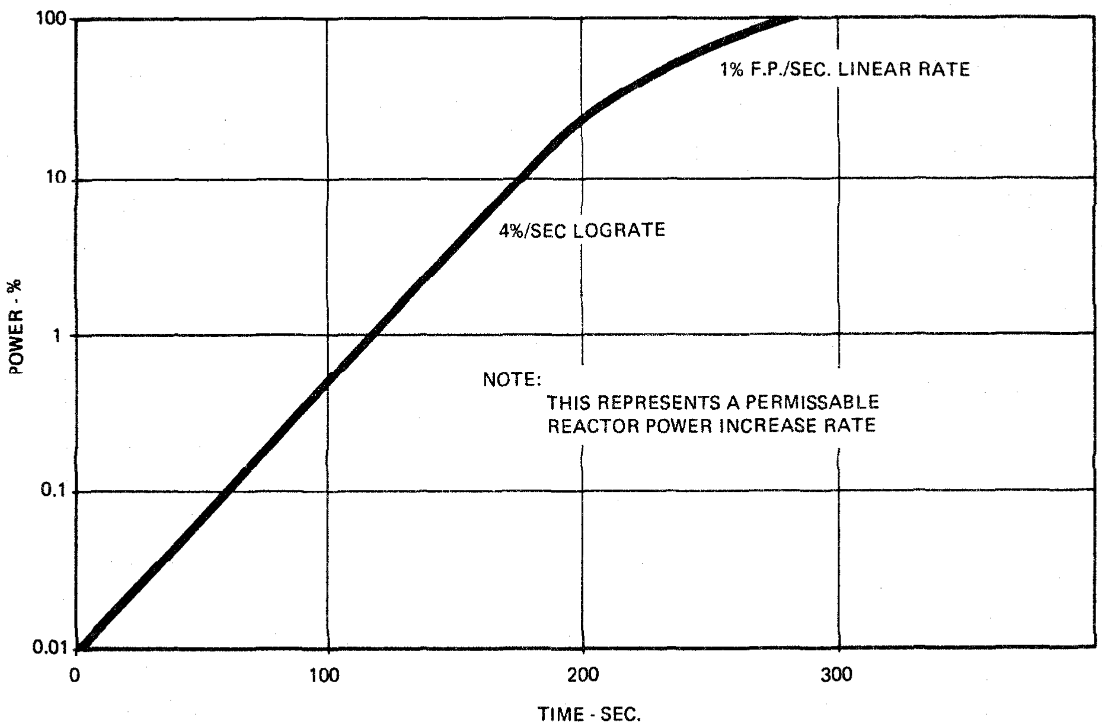


Figure 12 Reactor Startup Curve

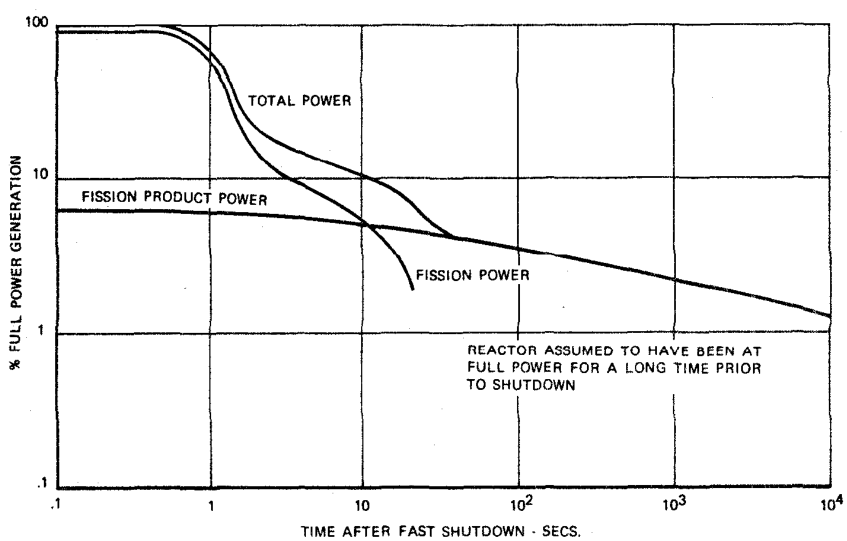


Figure 13 Fission and Fission Product Power - Pickering G. S.

The same boosters mentioned above would also be used when reactor power is manoeuvered to follow the load requirements of the electrical grid. Boosters with 18 mk would typically permit manoeuvering power between 50 and 100% of full power.

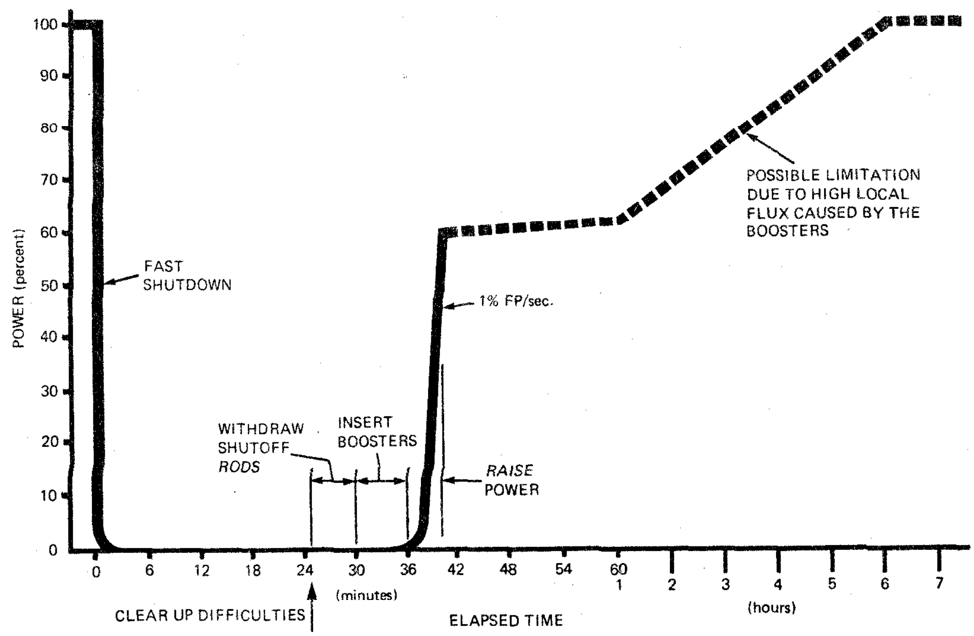


Figure 14 Recovery After Fast Shutdown

4. MEASUREMENT TECHNIQUES

In controlling a reactor, as in controlling any other process, we require accurate measurement of the variables to be controlled. We also require that the measurement signals follow the process variables with acceptably small time lags. The measurements must also span the entire operating range of the reactor from very low power to rated full power, so that control and protective actions are available under all possible operating conditions.

Figure 15 shows the power range of interest. At the top end we are talking about full rated power, with a typical overrange capacity of 50%, full power neutron flux being in the order of 2×10^{14} n/cm^2/sec. At the bottom end we are talking about the spontaneous fission rate of the natural uranium in a clean reactor core, before the moderator has been pumped up. The flux is in the order of 1 n/cm^2/sec. The chart shows the range of various instrument systems necessary to cover this range. It should be stressed that only the top 7 decades are of interest after a new reactor core has been made critical and operated at significant power levels. The bottom 7 decades are of interest during the initial startup.

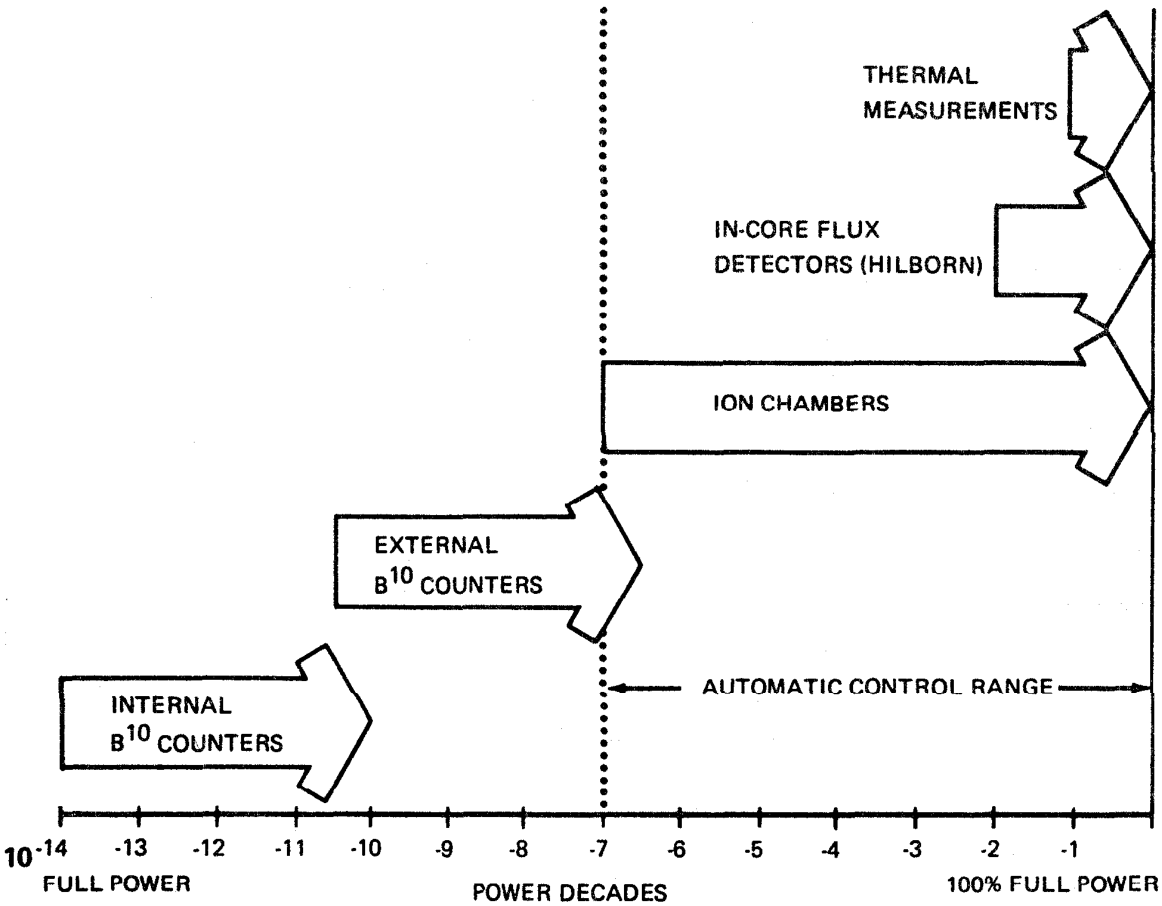


Figure 15 Power Ranges and Devices Used

We can dispose of the thermal measurements first, since they are in a sense the most conventional and they are made with devices identical to those used in fossil plants and other industrial processes.

Power out of the generator is readily measured electrically. Power into the turbine is readily computed from measurements of steam flow, steam pressure and feedwater temperature. Power output of a single fuel channel is similarly computed from measurements of coolant flow and temperature rise (Figure 16). It could be noted at this point that the thermal measurements meet the accuracy requirements but are too slow to act directly in the reactor control loop. We therefore use

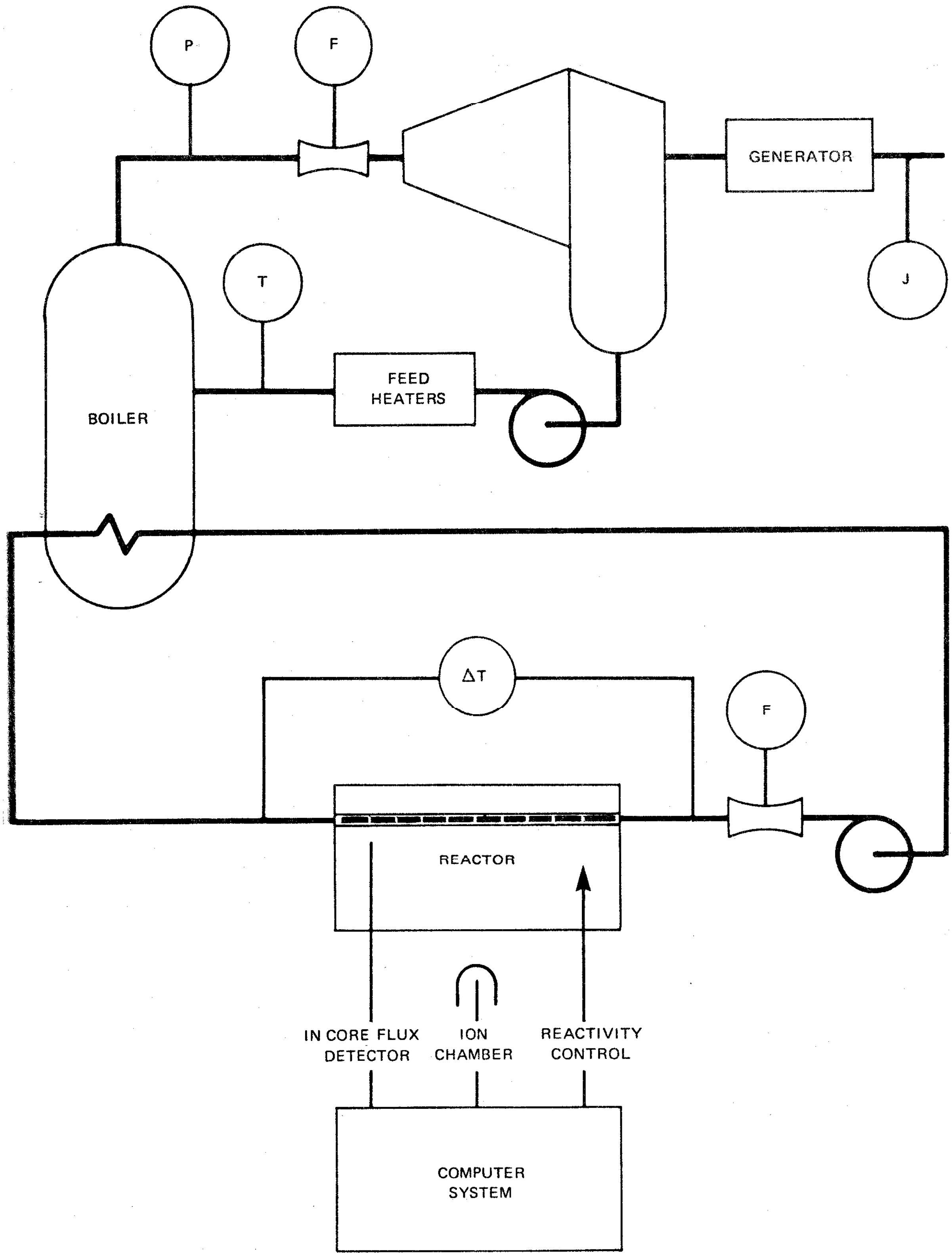


Figure 16 Power Measurement

neutron flux measurements which, while difficult to relate accurately to the thermal power output, respond very rapidly to any change in reactor power. It will be apparent then that the method of control is to close the neutron flux control loop with neutron flux measurements and adjust this loop for the required output by means of more accurate thermal measurements. We can turn our attention now to the nuclear or "neutronic" measurements, as these are specific to nuclear power stations.

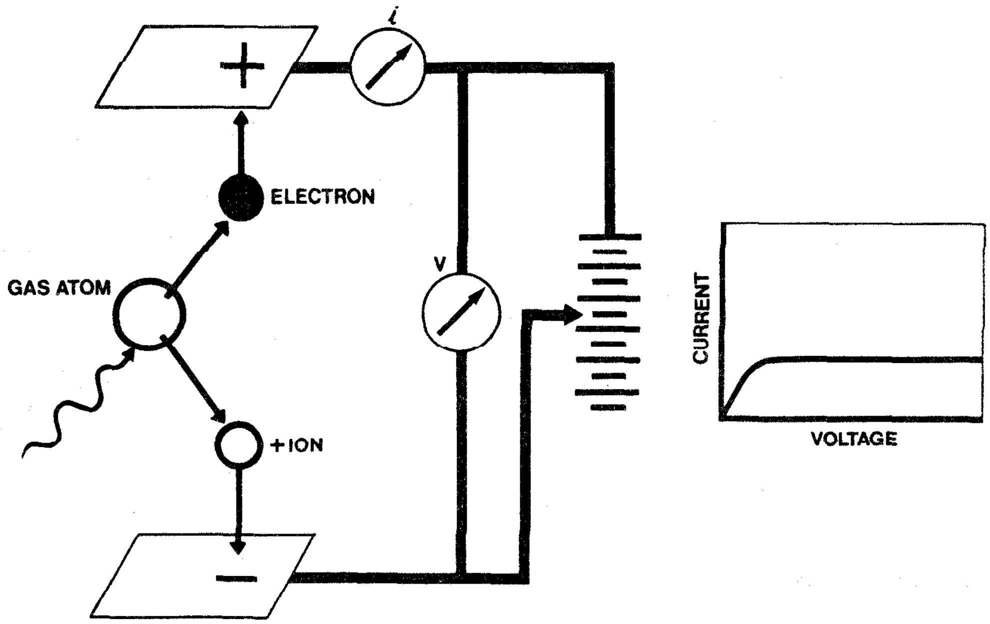


Figure 17 Ion Chamber Operating Principles

If we put a pair of metal plates in a radiation field and connect them to a battery and microammeter (Figure 17) a current will flow proportional to the number of ion pairs produced by the radiation. Provided certain technical details are satisfied, the current will be proportional to the intensity of the radiation. Two of the more important technical details are:

(1) There must be a high enough voltage on the plates to collect all the ions produced before they can recombine.

(2) The ionizing and recombination properties of the gas should remain constant throughout the useful life of the device.

We can readily check that the first requirement is met by putting the device in a constant radiation field and increasing the voltage until the current no longer increases, i.e., all ions produced are being collected. Plotted, the current vs voltage curve is as shown in Figure 18. This is known as a saturation curve. As the radiation intensity is increased a family of curves results, with higher radiation intensities requiring more voltage to saturate.

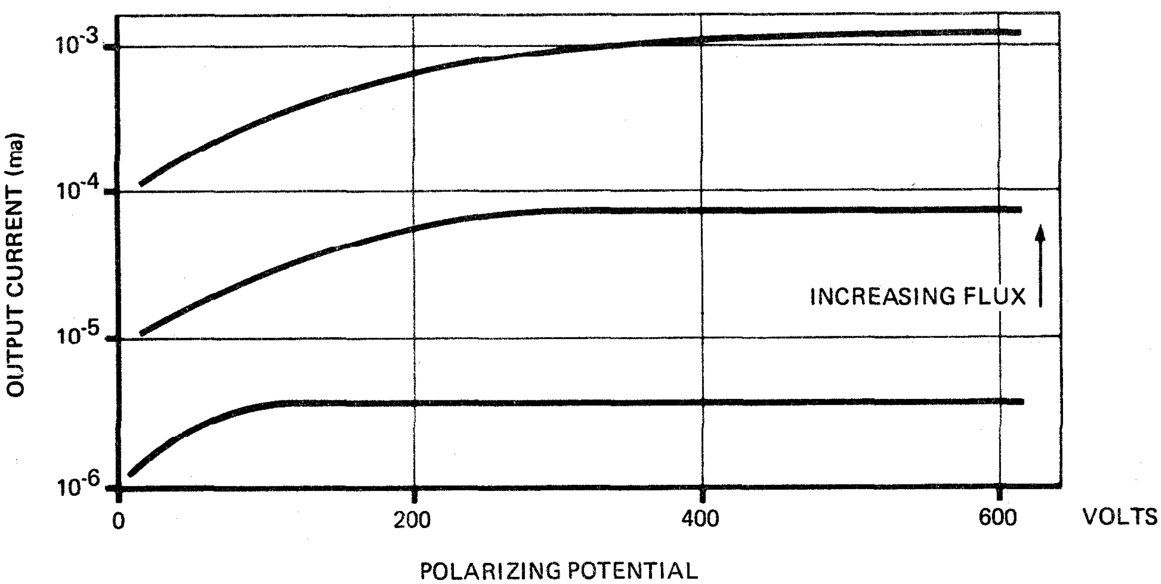


Figure 18 Ion Chamber Output Characteristics

The second requirement is met by enclosing the electrodes in a sealed can, filled with an inert gas such as helium or argon. The assembly is called an ionization chamber, or more commonly, an ion chamber. In commercial ion chambers, the electrodes are usually concentric cylinders (Figure 19).

To make an ion chamber preferentially sensitive to neutrons we coat the inside with boron which contains the isotope boron 10, which has a high capture cross section for thermal neutrons. ^{10}B captures neutrons and yields energetic alpha particles, which create dense ionization inside the chamber (Figure 20).

The neutron capture reaction is:

$$n + B^{10} \longrightarrow Li^7 + He^4$$

The use of boron in a chamber greatly increases its sensitivity to neutrons, relative to its gamma sensitivity. This is important because gamma radiation is always present in measurements of neutron flux from a reactor.

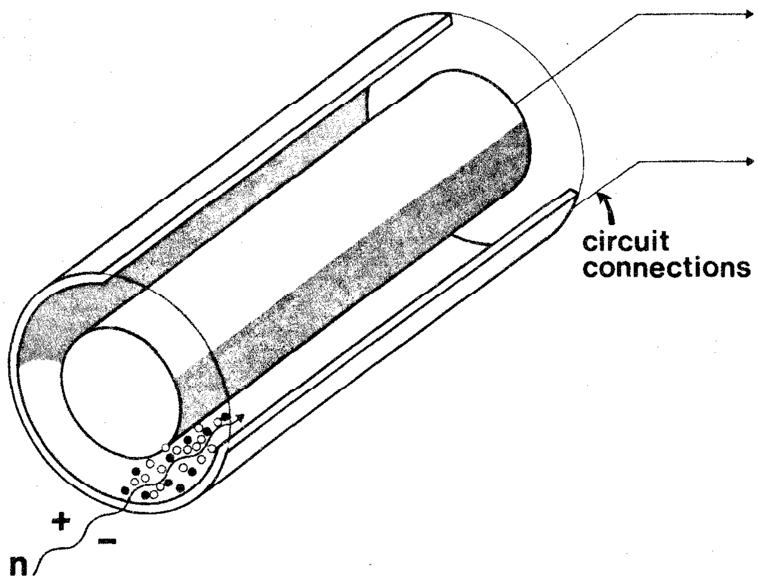


Figure 19 Ion Chamber Construction

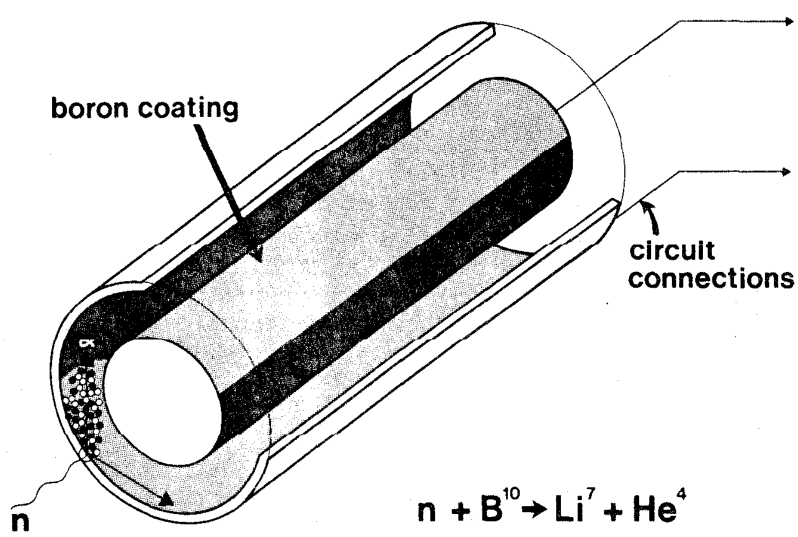


Figure 20 Ion Chamber Construction With Boron Coating

To measure the very low levels of flux that can exist after a prolonged shutdown, a further reduction in gamma sensitivity can be obtained. This is accomplished by adding a third electrode, without a boron coating and subtracting its current from that of the neutron-sensitive electrode, thereby cancelling out the gamma component of the neutron-sensitive portion. This is known as a gamma-compensated ion chamber (Figure 21). Chambers of this type, in properly designed enclosures, measure the neutron flux from full power (typically 100 μA) down to 10^{-7} x full power (10^{-5} μA), and are highly reliable over this range.

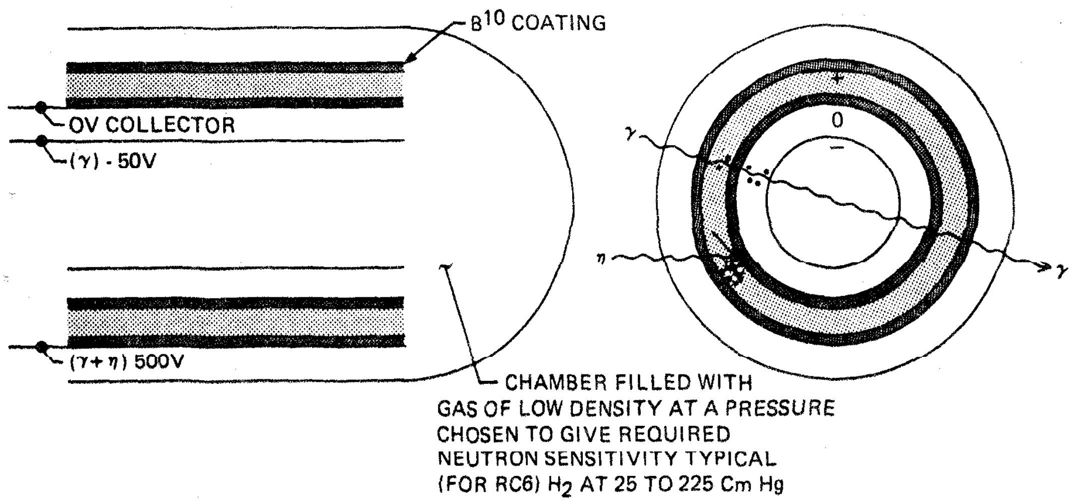


Figure 21 Ion Chamber (Compensated)

If we take a simple ion chamber and replace the central cylinder with a thin wire, we create a very non-uniform potential gradient, with the steepest part around the central wire. If this device is exposed to a constant radiation field and the voltage increased, we find that it passes through three distinct modes of operation (Figure 22). At low voltages, it behaves like an ion chamber. At higher voltages the electrons from the ionized gas are accelerated sufficiently to ionize other gas atoms, thereby producing a "shower" of ions for each of the original ions produced. This, in effect, amplifies the original event and makes it possible to count each ionizing event individually. This is known as gas amplification, and is capable typically of multiplying the original signal by 10^4. Further, in this region, the size of the output voltage pulse is proportional to the energy of the ionizing radiation.

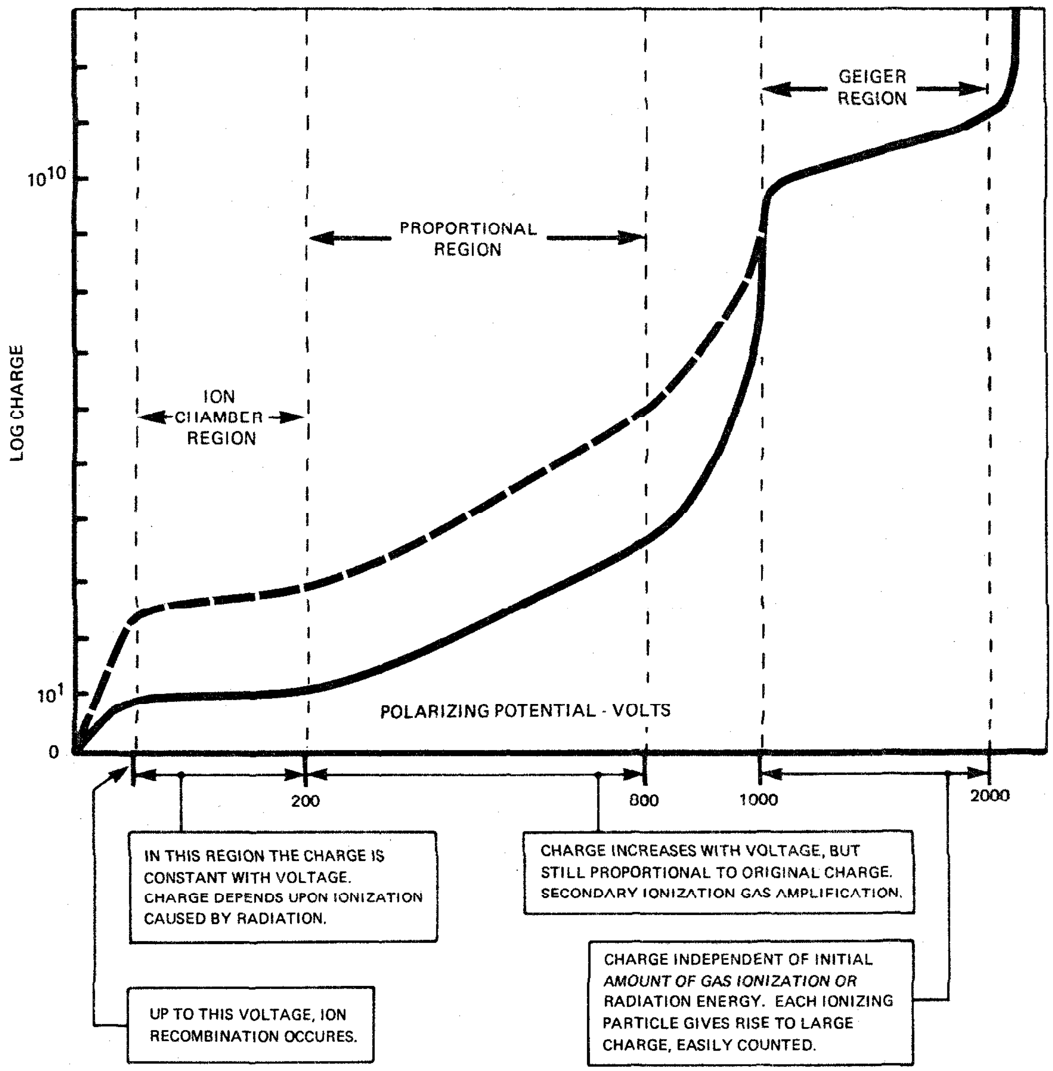


Figure 22 Gas Ionization

Counters operating in this region are known as proportional counters. Advantage is taken of this effect to make a counter sensitive to neutrons only. The inside is coated with boron or filled with a gas containing boron (typically BF_3).

The dense ionization from the neutron-alpha reaction produces very large voltage pulses, which can be electrically separated from the smaller pulses produced by gamma rays. These neutron-sensitive proportional counters are used in the initial startup of a reactor. They are extremely sensitive and can readily detect fluxes of less than one neutron per cm^2 per second. If we increase the voltage still further, we enter a region where any incident radiation, no matter how small, causes an avalanche of secondary electrons, and a large voltage pulse that is independent of the initial ionizing energy. This is known as the geiger region, and has applications in radiation monitoring but not in reactor control.

A third neutron sensitive device commonly used for reactor power measurement is the Hilborn flux detector. It consists of a piece of mineral-insulated wire with a co-axial metal sheath. The central conductor or "emitter" is made of vanadium, platinum or cobalt, and the outer sheath of inconel. The materials are chosen so as to emit different numbers of electrons when they capture thermal neutrons. The net difference in electron emission creates a potential which will cause current to flow in an external circuit. The current is a function of the neutron flux. Usually several of these detecting elements are wound around a support rod to form a flux detecting assembly (Figure 23).

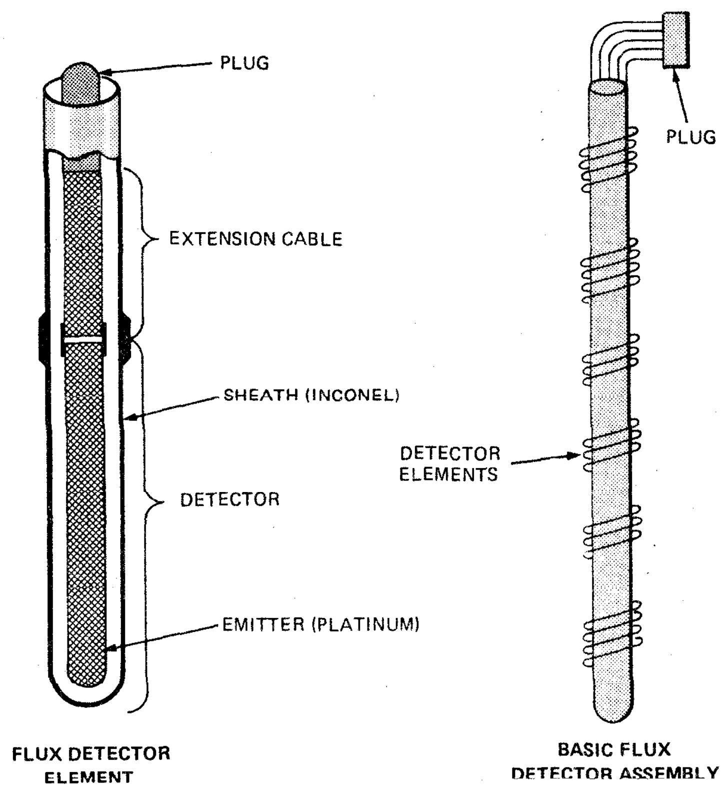


Figure 23 Hilborn Flux Detector

Several such assemblies are used in a power reactor to map the neutron flux and power distribution. These instruments are sometimes referred to as "self-powered detectors" because they require no external source of potential. In this respect they are simpler than ion chambers.

Each of the three instruments mentioned has limitations that must be considered in designing a control system. In general, while they give very rapid and reliable indications of power increments, their absolute accuracy, in terms of thermal power, are not sufficient for plant control. Thus, flux measurements are backed up by thermal measurements, which although too slow for reactor control, have the necessary absolute accuracy.

5. CONTROL DEVICES

To manoeuvre reactor power we need to either add or remove reactivity in the reactor core.

Long term reactivity control is achieved by replacing the spent fuel with new fresh fuel.

Short and medium term control is usually achieved by the use of booster elements and by a combination of a number of neutron absorber elements.

The booster element is a mechanical control rod made up typically of zirconium-uranium (U-235) alloy. The rod is driven into the core to increase reactivity. A typical rod is shown in Figure 24. The rod may have a constant speed or variable speed drive dependent upon the overall control dynamic requirements and could be driven in or out of the core in approximately 1 minute.

Medium term type of neutron absorption is the addition of a neutron absorber to the moderator. Commonly used is natural boron in the form of B_2O_3 dissolved in the heavy water moderator.

One ppm of boron is equivalent to approximately -8 mk of reactivity in Canadian reactors. Although a liquid absorber dissolved in the moderator can be added in a period of seconds, it can only be removed by ion exchange resulting in a removal time constant of several hours. This method of control is therefore generally used for longer term reactivity control such as compensating for the excess reactivity present in the core during the initial operating period with fresh fuel,

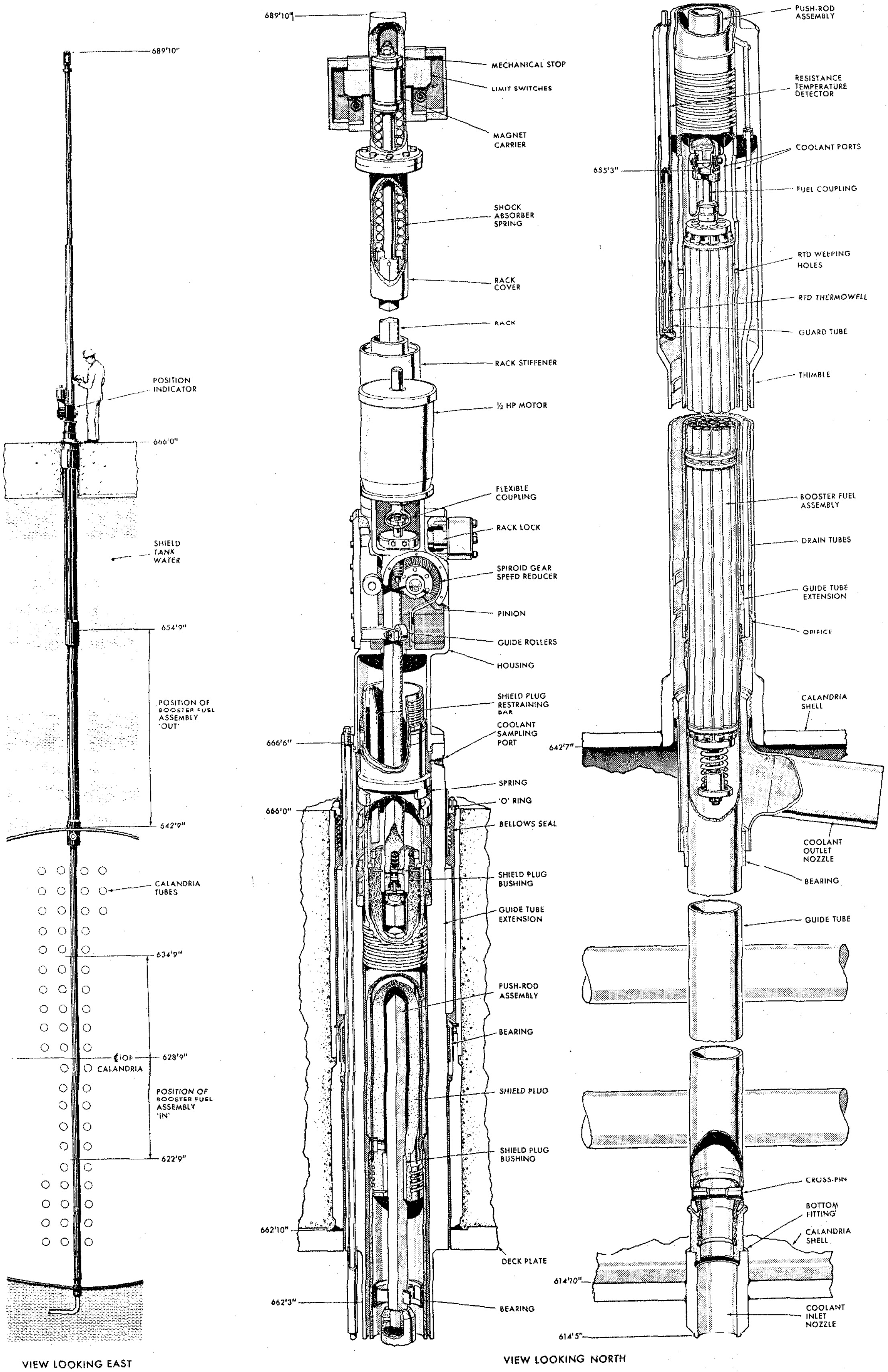


Figure 24 Bruce Booster Assembly

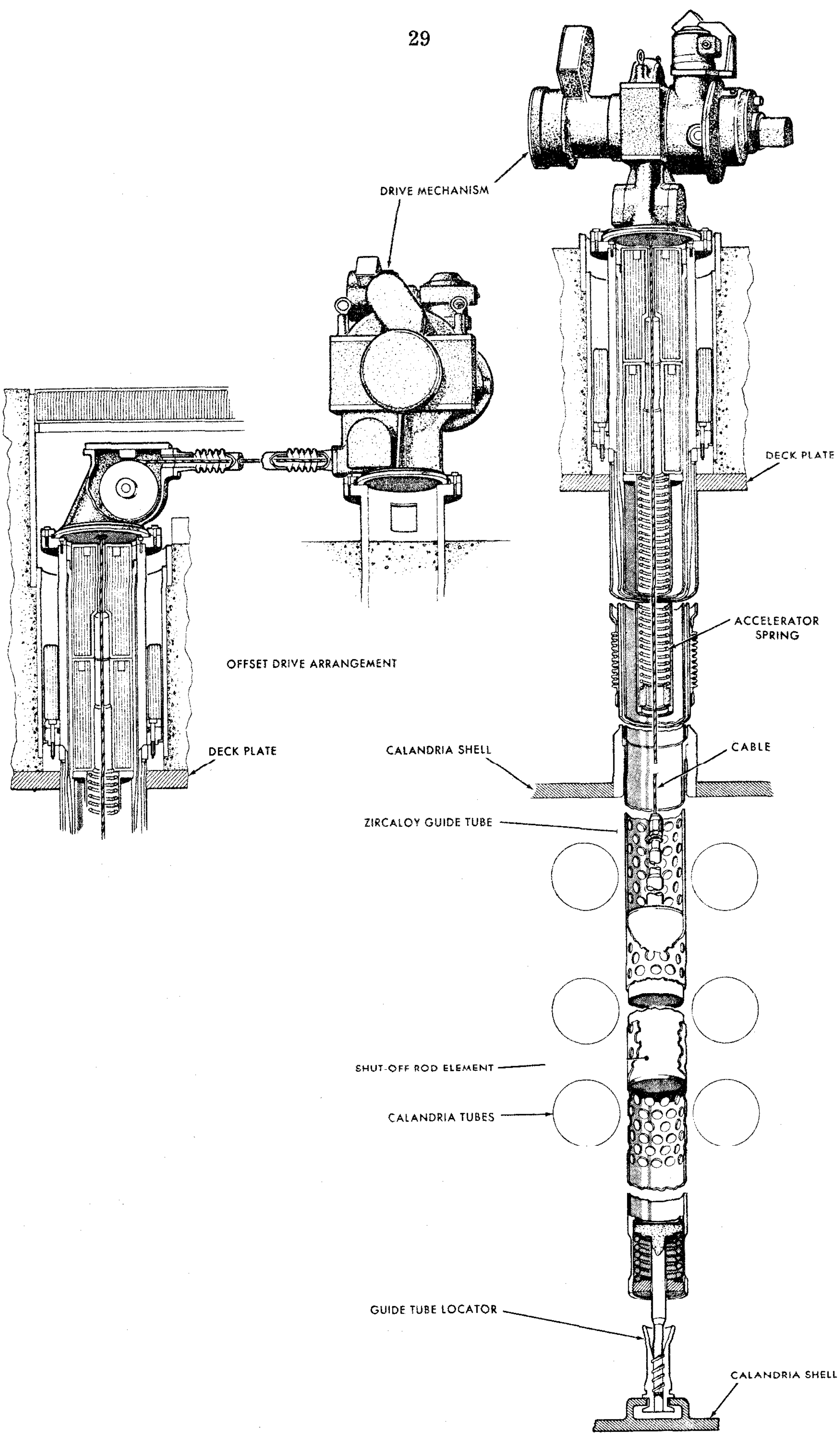


Figure 25 Bruce Shutoff Rod

compensating for the absence of Xenon poison after a shutdown, and for a fast shutdown of the reactor itself. For reactor control purposes the addition and removal of dissolved poison are manual operations and not connected to the automatic reactor regulating system.

The most common short term automatic regulating device is a neutron absorbing control rod which is driven into the reactor core. This rod can be either a mechanical rod or a liquid one. A mechanical rod is shown in Figure 25. In this case, the rod illustrated is one used for fast shut-off purposes. Various materials could be used, with a typical absorber element being made of a stainless steel-cadmium-stainless steel sandwich in the form of a tube 4-1/2 inches in diameter. The element is pulled out of the core by means of a cable and winch arrangement and held out by means of an electromagnetic clutch. De-energizing the clutch releases the cable and the element falls into the core under the infuence of gravity or assisted by a spring. A typical insertion time would be 2 seconds. Similar type rods can also be driven in and out at slower rates for normal control purposes. (in a similar manner to the boosters).

At the present time liquid control rods are being extensively used in the control of Canadian nuclear power plants. The rod is essentially a container within the core which is filled and drained with ordinary water. The amount of neutron absorption varies directly with the amount of water in the container. A typical liquid rod container would be 4 inches in diameter and 8 feet long. A simplified flow sheet of such a system is illustrated in Figure 26. There is a constant outflow of water from the rod container. The reactivity is changed by increasing or decreasing the water inflow until the desired liquid level is obtained in the container. Level in the container is measured by the conventional bubble pipe technique. The container can be filled or emptied in approximately 1 minute. The liquid rod has the advantage that you can insert the reactivity in the location and in the amount that will give the desired overall results.

Referring back to the discussion on Reactor Control theory, it will be recalled that one of the reactivity feedback terms was "coolant density". Varying the coolant density is used as one of the reactor control mechanisms on both the American boiling water reactors and in Canada's Boiling Light Water (BLW) design. The circulation flow of coolant is adjusted by flow control valves in the coolant loops. Starting with a fixed reactor power output, closing the control valve will decrease the cooling water flow, thereby causing more boiling in the

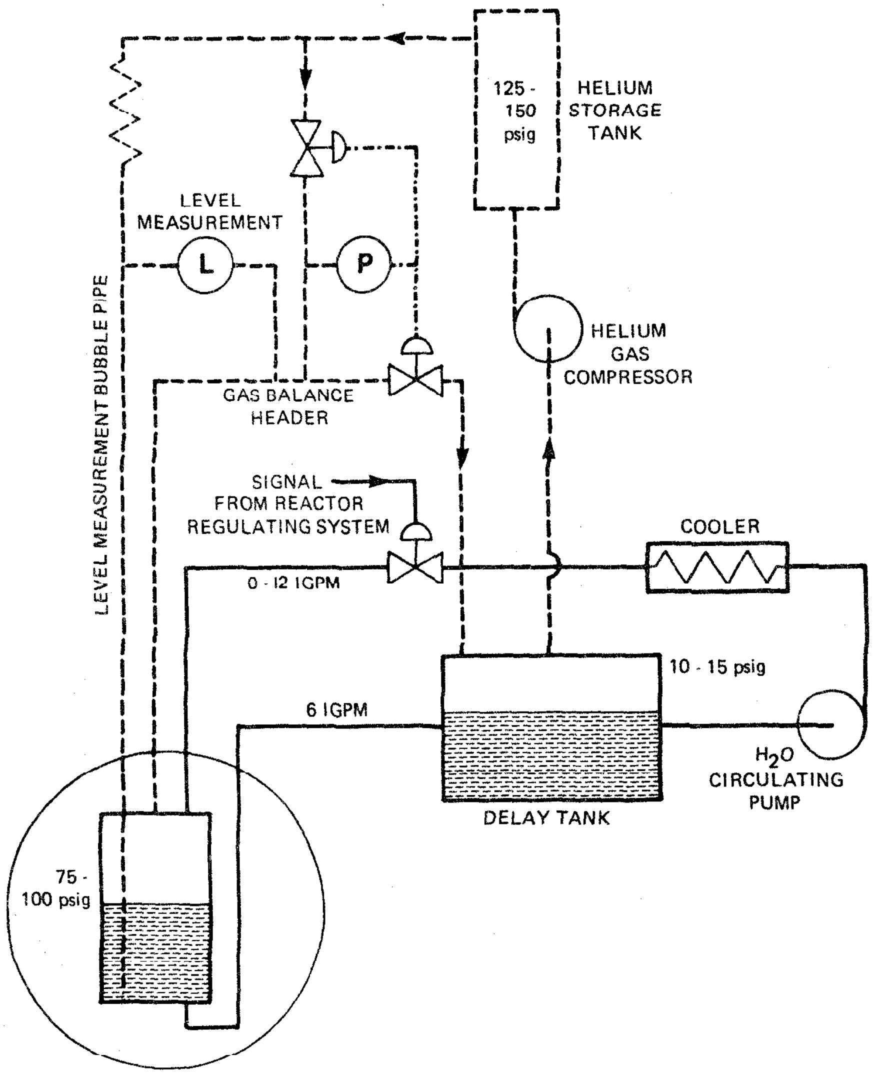


Figure 26 Liquid Control Rod

reactor core. This increased boiling produces more steam voids, thereby, in effect, decreasing the effective coolant density. This coolant density change produces a reactivity change, thereby increasing or decreasing reactor power dependent upon the reactor core characteristic.

The final control device to be discussed and a uniquely Canadian development (as are the liquid control rods discussed above) is the adjustment of moderator level in the reactor core to adjust reactivity.

This method of reactivity control is used on every operating nuclear power plant of Canadian design. However, the trend towards larger and larger units probably means this method will not be used again, due to the economics involved and the inadequate speed of response of such a method. The method is illustrated in Figure 27. Moderator level is adjusted employing the U-tube principle and maintaining a differential gas pressure between the two legs of the U. Moderator height in the calandria is a direct function of the differential pressure

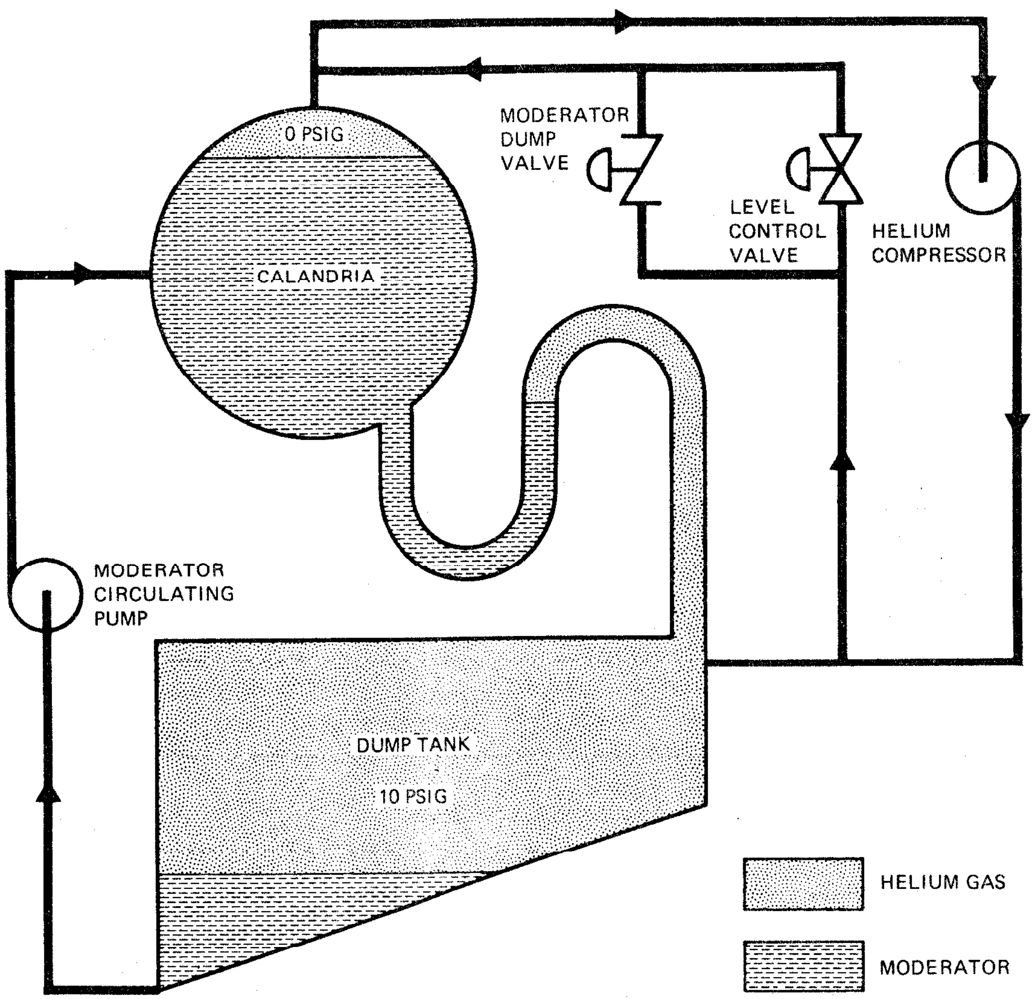


Figure 27 Moderator Gas Balance System

across the gas balance control valves. Fast reactor shutdown is achieved by opening large butterfly type dump valves in parallel with the gas balance control valves. This action very quickly equalizes the pressure on both legs of the U and the moderator is dumped out of the reactor core into the dump tank beneath. To reverse the process and raise moderator level the gas valves are closed and the moderator is pumped out of the dump tank into the calandria. Typically, half of the moderator would be dumped in 10 seconds and it would require 35 minutes to pump all the moderator back into the calandria. Referring back to Figure 14, one of the strikes against the moderator dump can be seen. If a reactor trip initiates a moderator dump and it takes 35 minutes to pump the moderator back up there is no time in which to diagnose and correct the cause of the reactor trip, and a reactor Xenon poison-out will automatically follow (shutting the reactor down for a further 40 hours). A second strike against the moderator level adjustment method for normal regulation is that it causes neutron flux distortions in the core (refer back to the discussion on liquid rods).

6. CLOSING THE LOOP

Now that we have discussed the individual elements, let us put them all together in a typical reactor control scheme.

The selection of the types of devices to be used in a particular reactor along with the operating characteristics of these devices is influenced by a number of factors as follows:

(1) Necessary controllability and manoeuverability. For generally the reactor is not the limiting process in the nuclear power plant for these factors. The reactor is able to respond more quickly than the steam cycle and the turbine generator.

(2) Economics. It is desired to minimize the parasitic neutron absorption of the control devices.

(3) Safety. An unnecessarily responsive control system places more stringent requirements on the safety systems. They must be able to protect against the control system failing and driving in the wrong direction.

(4) Physical restraints. There are only so many tubes and holes that can be put in a reactor vessel and still leave room for fuel and heat transfer processes.

DEVICE	NUMBER	FUNCTION	RANGE
Platinum Incore Detectors	28	total and zone control in power production range	10 - 120 % FP
Top Ion Chambers	3	log power / log rate / linear power — in startup and shutdown range	10^{-7} - 1.5 FP -20 to +20 % / sec 0 - 150 % FP
Spare Ion Chamber Positions for Fission Counters	3	initial startup and after long shutdown — normally disconnected	
Coolant Channel Flow Measurement	22	signals are combined to give a measure of total reactor thermal power	0 - 120 % normal
Coolant Channel Temperature Detectors	88 (2 pairs per coolant channel)	signals are combined to give a measure of total reactor thermal power	0 - 55°C
Coolant Channel Outlet Quality	22	signals are combined to give a measure of total reactor thermal power	0 - 10%

Figure 28A Typical Reactor Measurement Devices

DEVICE	NUMBER	FUNCTION	CAPABILITY	
			DEPTH mk	RATE mk/sec max
Liquid Control Rods	14	total & zone control	6	± 0.115
Mechanical Control Rods	4	backup to liquid control rods for large plant upsets	8.7 8.7	± 0.115 driven −4.0 dropped
Booster Rods	16	xenon poison override	18	± 0.05
Dissolved Poison in Moderator (Boron)	1 into moderator circulating system	compensation for fresh fuel excess reactivity & absence of xenon	as needed	manual − in minutes + in hours
Shutoff Rods	30	reactor trip	40	− 25 dropped + 0.8 driven
Poison Injection into Moderator (Gadolinium Nitrate)	9 injection tubes directly into calandria	reactor trip	300	−28 mk in 1½ seconds + in hours

Figure 28B Typical Reactor Control Devices

For a nuclear reactor such as Bruce G. S., the detection devices and their uses are listed in Figure 28A. The control devices and their capabilities are listed in Figure 28B. The normal control in the electrical power production range is via the Hilborn platinum in-core detectors through the 14 liquid control rods.

The 22 coolant channel measurements (flow, temperature and quality) are used to compute the absolute reactor power and to adjust the in-core detector readings accordingly (on a long term basis). The four mechanical control absorbers are not normally in the reactor core but are only driven or dropped into the core when there is a major plant upset (such as a turbine trip, or a sudden trip of a main coolant circulating pump). The liquid rods do not have the reactivity depth or speed necessary to maintain the nuclear steam supply systems in their normal operating ranges under such conditions. The booster rods are used to compensate for the Xenon poison buildup during a sudden lowering in reactor power (refer to the discussion on Reactor Control theory and Figure 14).

The operating ranges of the automatic control devices based on power error (actual reactor power minus desired reactor power) are shown in Figure 29. Under normal operating conditions (with equilibrium Xenon in the core and no dissolved poison in the moderator) the liquid rod levels will drop as the fuel is burned up. Low level is a signal that refuelling is necessary. At Bruce, with no refuelling the reactivity will drop at the rate of 0.4 mk per day. Refuelling 8 of 12 fuel bundles in one coolant channel in the central area of the core will increase the reactivity by 0.5 mk.

During initial operation after a shutdown high liquid rod level indicates a need for the manual addition of moderator poison; low liquid rod level indicates a need for the removal of dissolved moderator poison by the use of the ion exchange purification system.

Figure 30 indicates the location of the various control devices. It is noted on the elevation view that the reactor is divided into 7 zones with a liquid zone rod in each. There are actually 7 more zones located immediately behind the seven shown, giving a total of 14 zones in the reactor core. In addition to adjusting the liquid rod levels up and down to maintain the desired total reactor output, the Hilborn detectors are used to compute the reactor's power in each zone. The liquid rod levels are then differentially adjusted to maintain equal output power from each zone and correct any Xenon instabilities that may occur.

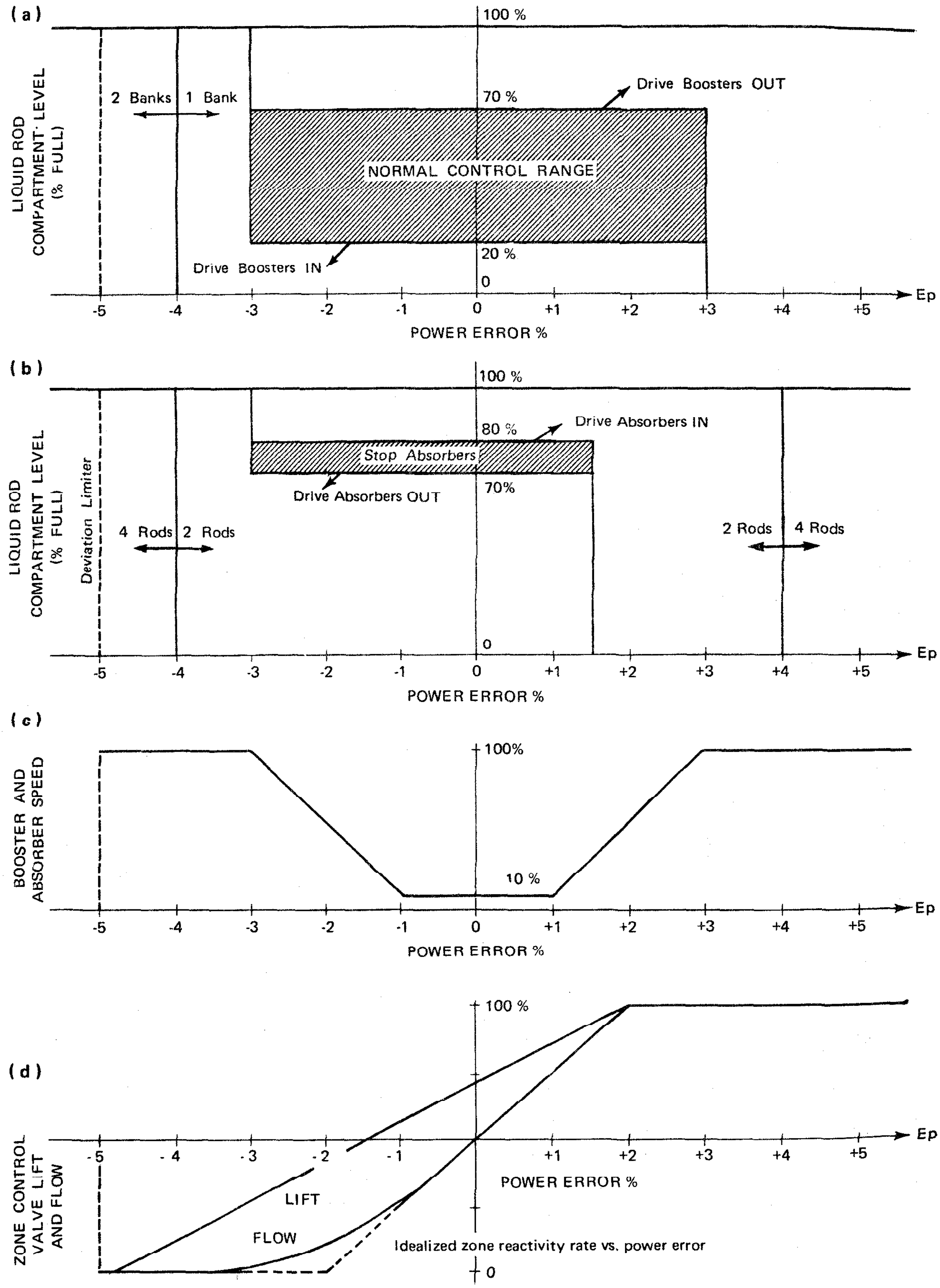


Figure 29 Reactivity Mechanism Limit Control

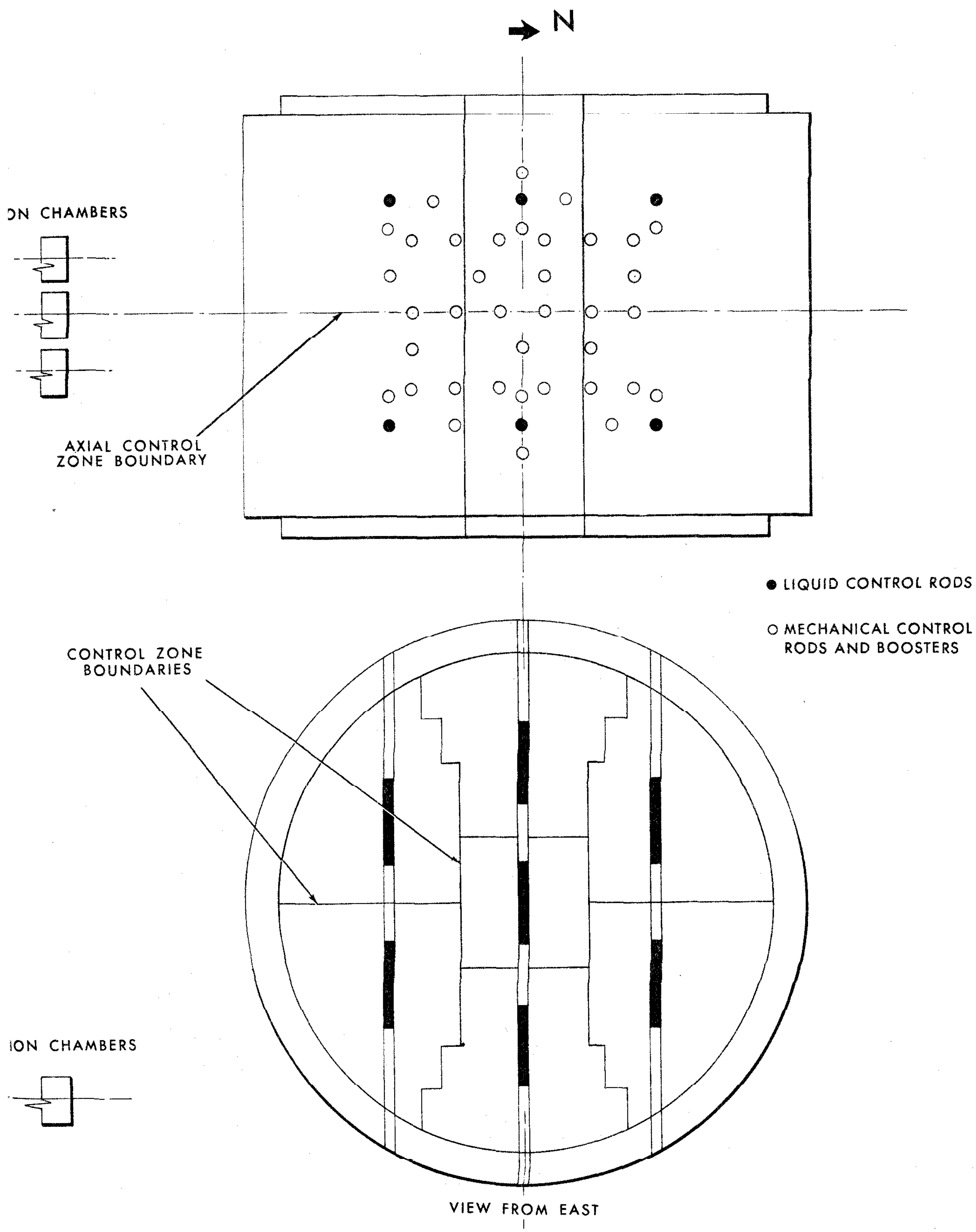


Figure 30 Zone Control Absorbers

All measurement signals are fed into the computer system. It in turn computes the necessary control action necessary from the control algorithms in the control programs. The computer then directly adjusts the positions of the various control devices to maintain the desired reactor output power.

7. OVERALL POWER OUTPUT CONTROL

Figure 31 is a block diagram of the plant control system illustrating the reactor control interconnections discussed above as well as the overall plant control loops to achieve the desired power output control.

There are essentially two methods of overall plant control, both of which are being used in Canadian and American designed nuclear power plants. These can be described as (1) turbine power following the reactor power, and (2) reactor power following the turbine power.

In the first type of control scheme reactor power is set at the desired value and the boiler pressure controller adjusts the set point of the turbine governor valve speeder gear to maintain boiler pressure at a constant value. This is a base load type of power regulation, since the turbine output remains constant no matter the frequency changes on the electrical grid system. As the power producing units get larger and larger, representing a larger percentage of the total power production, it is desirable to have these units share in the grid frequency correction task as well as share in the daily load following task brought on by the varying grid power demands.

To enable these units to share in these tasks, the second type of control scheme (reactor power following the turbine power), as illustrated in Figure 31, is being adapted. The nominal desired electrical power is obtained by adjusting the set point of the governor valve speeder gear via the unit power regulator program. The boiler pressure is controlled at a constant value by adjusting the set point of the reactor regulating program via the Demand Power Routine. The governor valve compensates for grid frequency changes by its governing action. The changes in turbine power cause a drop or rise in boiler pressure. The reactor power is automatically adjusted to return the boiler pressure to its proper value.

The above scheme is quite a conventional one and the only variations are the presence of atmospheric steam discharge valves and steam discharge valves directly to the turbine condensers under the control of the Boiler Pressure Controller.

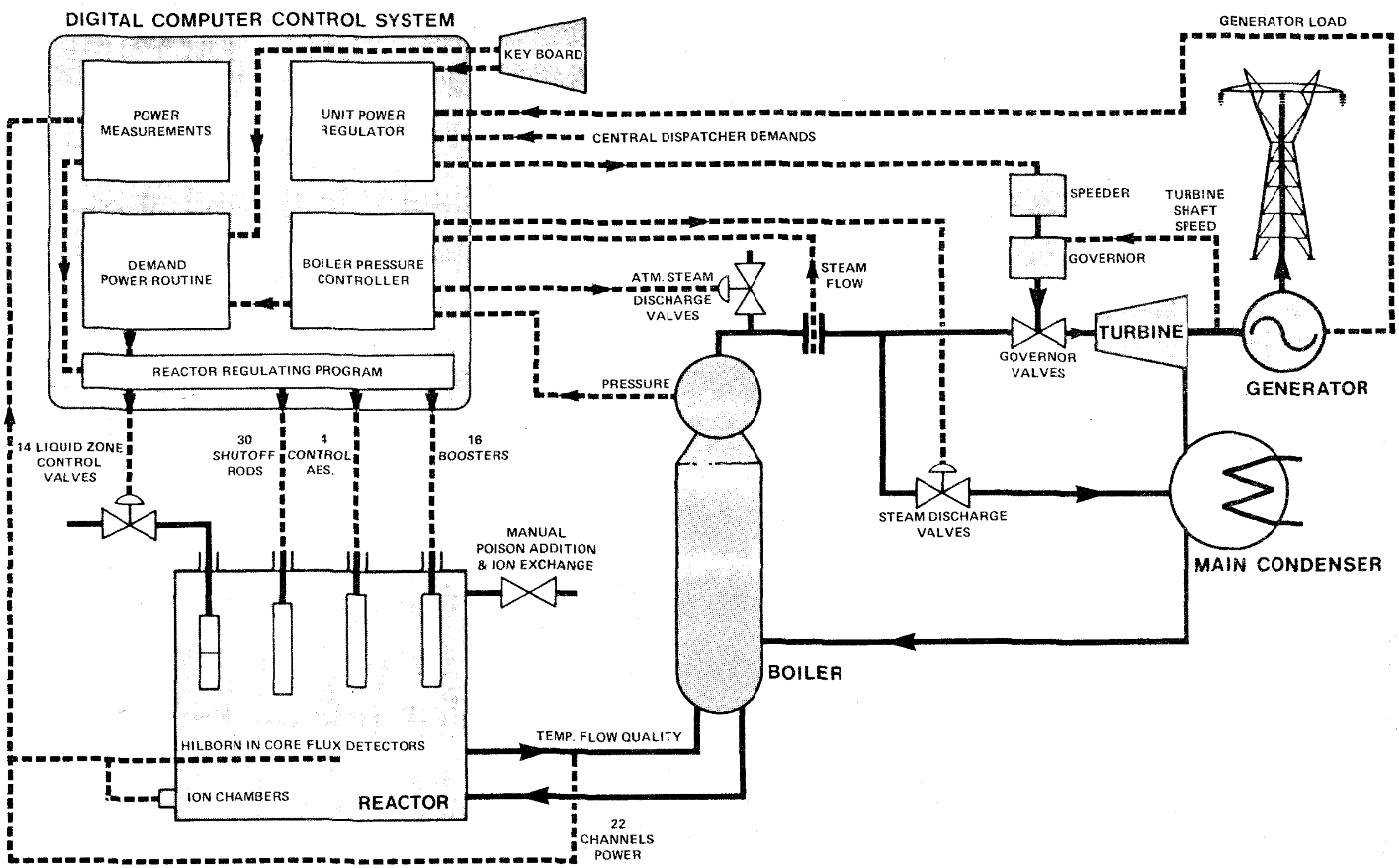


Figure 31 Unit Control System

The purpose of these steam discharge valves is related to the Xenon poison phenomenon and the required 90% availability factor of electrical generating units. Refer back to the discussion on Reactor Control theory and particularly to Figure 14. After a reactor shutdown it is necessary to restart within 40 minutes with the booster capability listed in order to prevent a 40 hour reactor poison-out. It is therefore necessary to minimize the effects of turbine trips and losses of electrical grid connections upon the reactor. Supplying an alternate heat sink for the reactor power is therefore economically necessary. This heat sink may take the form of direct atmospheric steam discharge or steam dump to a condenser system, or a combination of both. With one of these arrangements there would be no plant unavailability caused by turbine or grid system faults.

The condenser steam dump system can also be used to accomplish the load following requirements of the plant. Load following can be accomplished by the use of boosters in the reactor as discussed previously, or alternatively the reactor can be run continuously at 100% thermal power, any unwanted steam being bypassed to the steam condenser. The choice becomes a matter of economics of how much load following is to be done, the cost of enriched booster fuel and its replacement, and the cost of the steam being wasted using the normal reactor fuel.

A comment should be made here about the operating characteristics of a nuclear power plant. The reactor power can be normally manoeuvred at $\pm$ 1% full power per second in the power range. The steam supply system can also manoeuvre at this rate but with an approximate 20 second time delay with respect to the reactor. Maximum turbine manoeuvering rates are of the order of 8% full power per minute. Therefore the conventional turbine is the limiting device in power manoeuvering as the steam supply system could accommodate faster rates.

The nuclear power plants are being designed for a plant capability factor of 90% and a capacity factor of 80% on a yearly basis at plant maturity. Early operating experience at Pickering G. S. shows these to be quite achievable goals.

8. REACTOR SAFETY SHUTDOWN

To this point we have been discussing "control" in the generic sense. In fact, in the design of the reactor control systems there are two distinct requirements:

(1) Reactor Regulation - This is the system which adjusts reactor power over all plant operating, shutdown and startup modes, and caters to recognizable abnormal events such as turbine trips, trips of circulating pumps, loss of electrical line, etc. We have, in fact, been essentially discussing the Reactor Regulating System up to this point.

(2) Reactor Safety Shutdown - This is the system which shuts down the reactor very quickly when the reactor power is apparently outside the bounds of safe operation. In Canada we call this action Reactor Trip; Americans refer to this action as Reactor Scram.

Reactor Trip is only one of the safety systems in a nuclear power plant. All of these systems will be discussed in Lecture 11 - Accident Analysis. We will only discuss reactor trip here.

The object of the reactor trip system is to detect those fault conditions in which reactor heat production could exceed the heat removal capabilities of the reactor system, and to quickly shut down the nuclear reactor. A thermal imbalance as mentioned above leads to increasing heat storage in the fuel itself, which in turn could lead to a rupture of the fuel cladding and the release of harmful radionuclides into the surrounding environment. This heat imbalance can be caused by a failure of the Reactor Regulating System which permits the reactor power to increase beyond the capabilities of the coolant system, or with the reactor maintained at a constant power output the heat removal capability can be lost because of a break in the coolant system. The Reactor Trip system is designed to cater to the worst such accident situation with adequate safety margins.

The requirements placed upon a Reactor Trip system are as follows:

(1) The safety reliability must be 0.999 or better. That is, the system may be faulty or unavailable to shut down the reactor only 0.001 of the time, or about 1/3 of a day per year.

(2) The trip system must be functionally and physically independent of the reactor regulating system and all other reactor safety systems, so that no common fault could disable more than one line of defense.

(3) Where possible, each potential process failure should be guarded against in two different ways, i.e., using two different parameters as initiating devices, so as to minimize the risk of trip system failure due to design error or common faults.

(4) It should be possible to test the trip functions in such a way that each series chain from instrument to final tripping device is exercised, preferably in a single test.

(5) All trip equipment must be designed to fail safe, where possible.

(6) Trip systems must be immune to spurious or unnecessary trips. Abnormal process conditions should where possible be dealt with by the regulating system or other systems, without invoking any of the safety systems.

It is requirement (6) above, which gives the designer his greatest concern. To satisfy the safety requirements of the trip systems is a relatively simple matter compared with the problem of making sure the system does not actuate when it is not needed. The production reliability goal to which these plants are being designed limits the number of sudden forced outages, i.e., reactor trips, to one per year due to all causes. This stringent requirement forces a careful review of the type of initiating parameters used, and their settings, to ensure that a reactor trip is not initiated needlessly during normal plant manoeuvering.

The high safety reliability requirements as well as the need to demonstrate this reliability has led to the use of redundant channels of instrumentation. In Canada we basically use triplicated systems with 2 out of 3 channels required to be actuated to complete the trip. American reactors use similar 2 out of 3 systems or 2 out of 4 systems. Such arrangements permit on-line testing of complete instrumentation channels to prove to the licencing authorities that the trip system does in fact have the required safety reliability.

A simplified schematic of a triplicated trip system is illustrated in Figure 32. The operating of any two of the trip channels d, e, f, will open a flow path between X and Y. The spurious operation of any one channel will not complete a flow path. Figure 33 illustrates a simplified safety reliability calculation for a triplicated system.

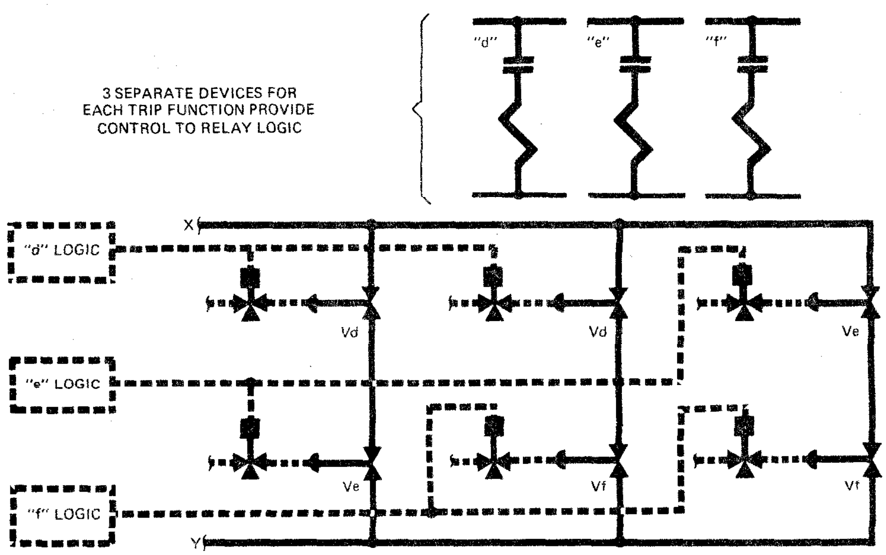


Figure 32 Reactor Trip System

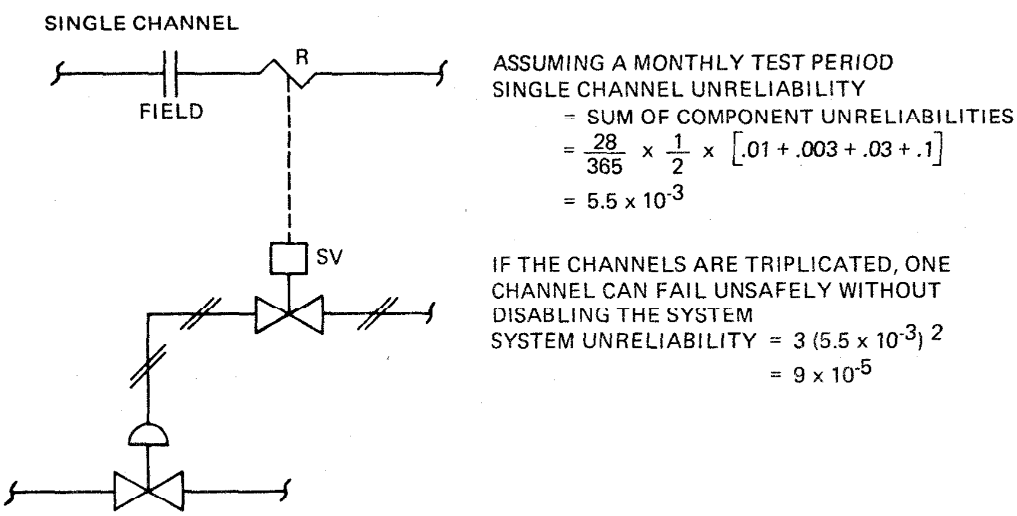


Figure 33 System Reliabilities

In fact there could be two separate reactor trip systems. Referring to Figure 28B you will note that there are two control devices that we have not discussed to this point, i.e., the shutoff rods and the direct poison injection. Each of these is actuated by a separate trip system. Figure 34 is a simplified schematic of these two systems. The first trip system uses the gravity drop shutoff rods. The second trip system injects liquid poison from the 9 poison storage tanks directly into the moderator when the quick acting gas valves are actuated.

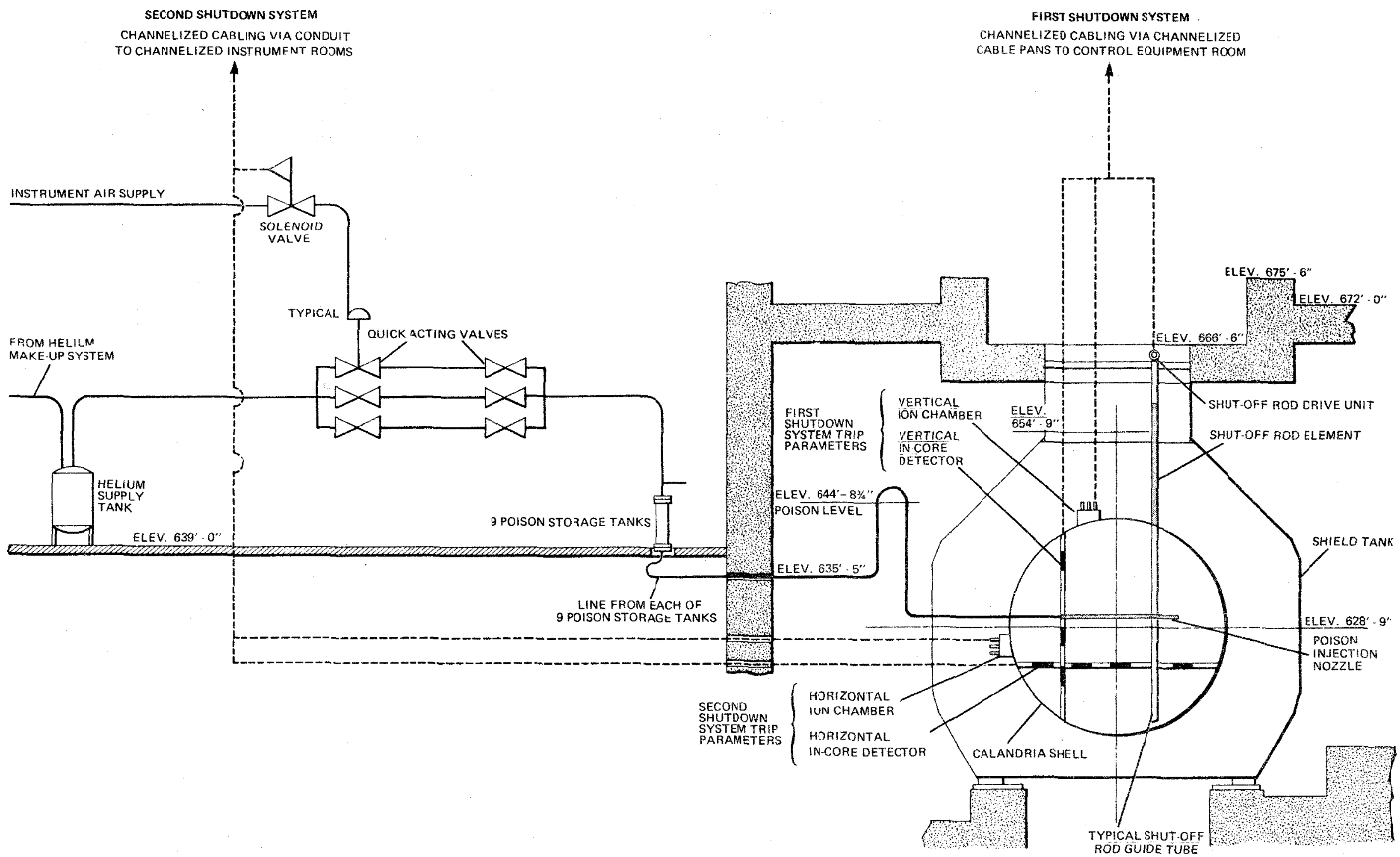


Figure 34 First and Second Shutdown Systems

Typical testing frequencies would be once per week per trip channel and once per month for each shutoff rod. Typical trip initiating parameters could be high neutron power, high rate of change of neutron power, high primary coolant pressure, low primary coolant flow, high primary coolant temperature, etc.

9. DIGITAL COMPUTER CONTROL

Because of the large number of parameters involved and the many system inter-relationships, the control of a nuclear power plant seems to be a natural job for a computer. In fact, the Canadian nuclear power program has been a pioneer in the application of direct digital control to process systems.

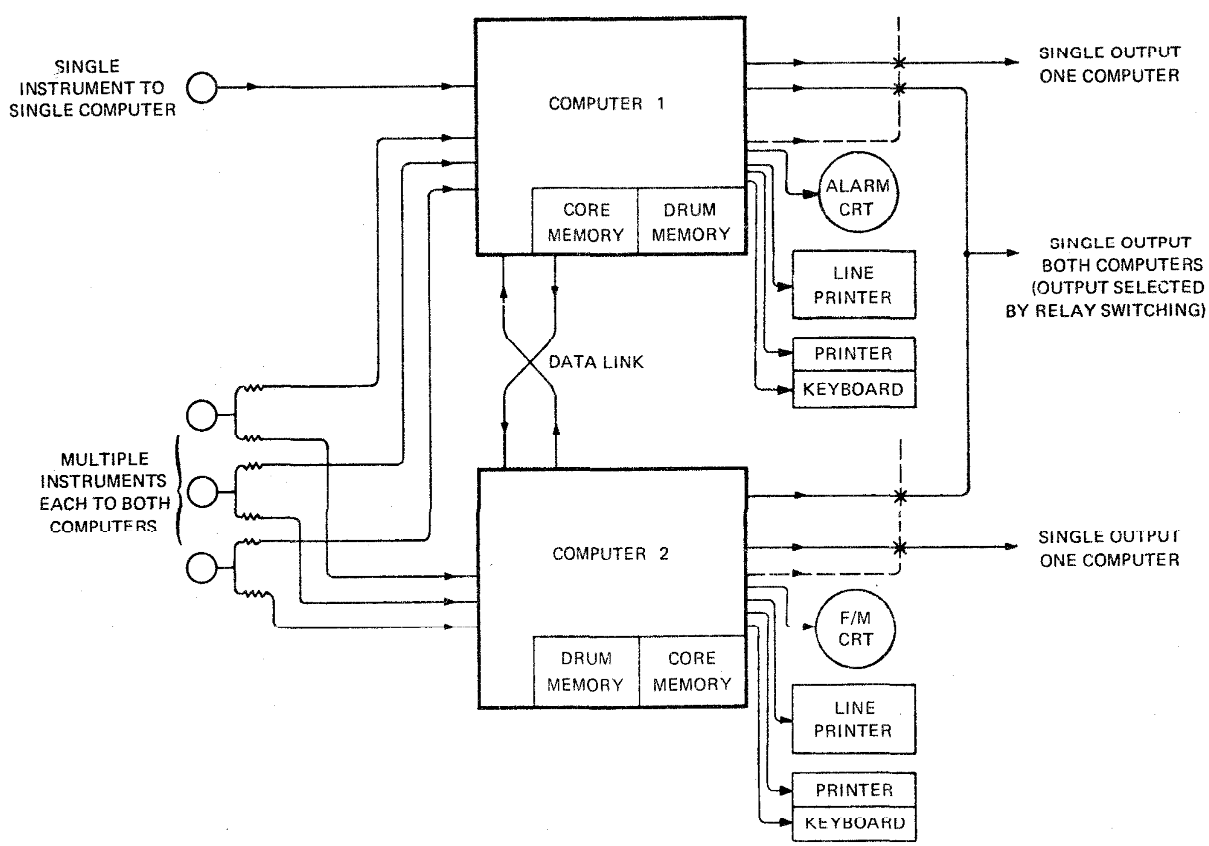


Figure 35 Arrangement of Computers with Peripheral Equipment

The main functions of the computer system were shown in Figure 31. However, in addition to the main control loops illustrated, the computer system may perform such tasks as: fuelling machine control, alarm scanning, sequence of events monitoring, turbine run-up, and data logging. However, in order to maintain independence between reactor regulating functions and reactor safety systems, no safety functions are performed by the control computers.

Since manual operation of a nuclear power plant is virtually impossible, an outage of the computer system means a plant shutdown. Therefore, the concept of a dual computer system for each reactor unit has been adopted. Figure 35 illustrates a typical dual computer arrangement; Figure 36 shows the inputs and outputs of this arrangement. For those functions which are essential for plant operation, e.g. Reactor Regulation, the computers operate in a master-slave fashion. The failure of the master computer transfers control to the slave computer. Other less important functions are divided between the two computers, such of which can be manually transferred between the two computers, others which cannot be transferred and are therefore lost if that computer is out of service. For example, in the arrangement shown the control of the fuelling machines rests in one computer only. Unavailability of that computer means that no refuelling operations can be carried out.

Typical Digital Computer Controller (DCC) characteristics are as follows:

(1) Access time 0.75 micro sec, add time 1.5 micro sec.

(2) Core storage 32 k words, fixed head drum storage 256 k words, moving arm disc storage 585 k words.

(3) Analogue multiplexing capability 10,000 points per second, alarm scanning 2,000 points in 4 milliseconds.

(4) Analogue input levels 0-1 volts, 0-5 volts.

(5) Line printer capability 3200 lines per minute.

(6) Frequency of program executive variable up to 10 times per second.

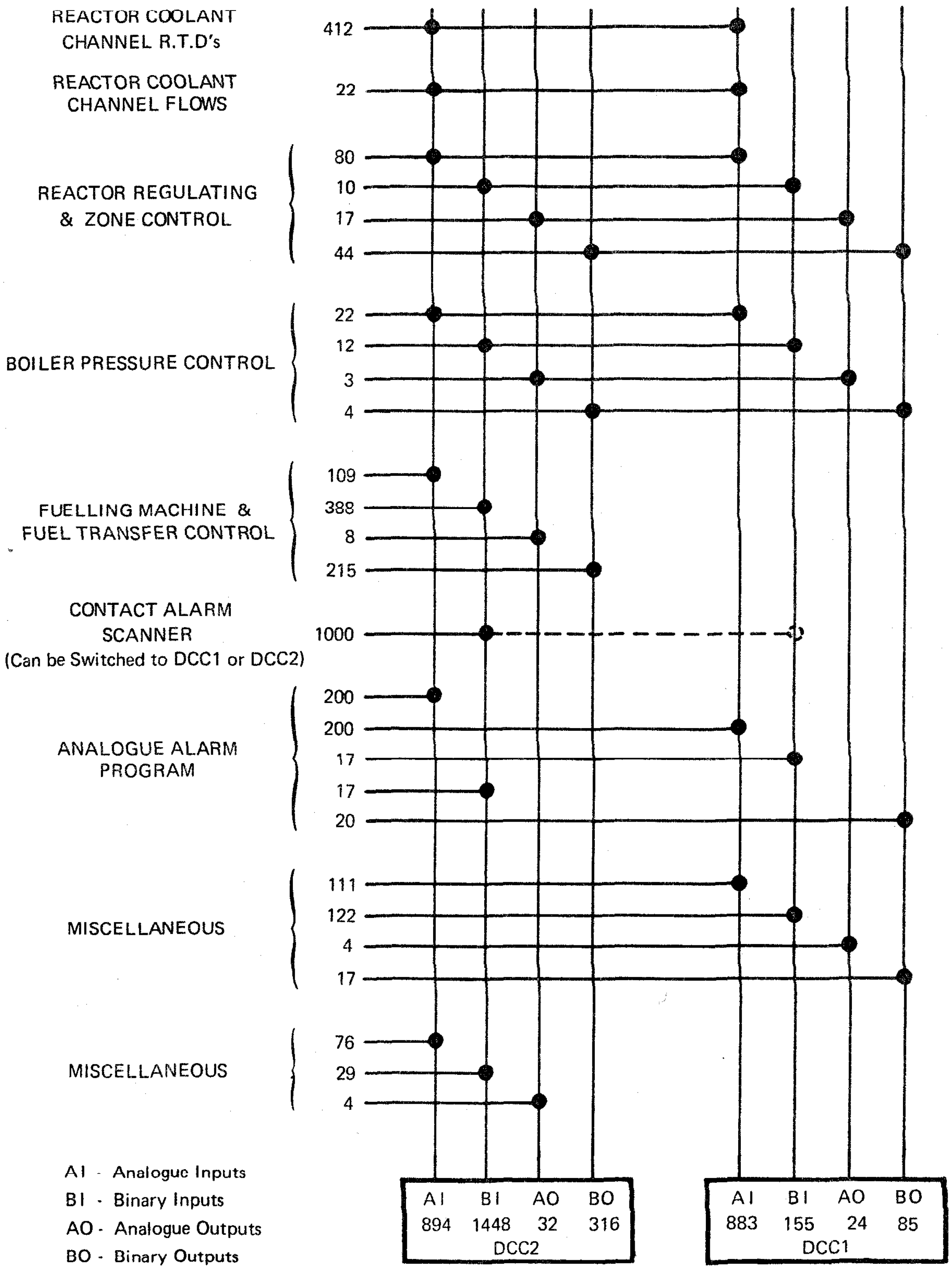


Figure 36 Computer Inputs and Outputs

The computer executive program processes the various control tasks in a normal sequence. Input/output devices and internal devices communicate with the computers via an interrupt process controlled by this executive program. The interrupts have various priority levels assigned to them and they interrupt the normal control sequence based on these priority levels; a low priority interrupt cannot seize control of the machine if a high level program is being run.

There are a number of check routines built into the computers to detect faults. These are as follows:

(1) Memory Parity error - on transfer of information the number of bits in the word are checked. A parity check failure after a second try results in complete DCC failure and transfer of control, or transfer of that program only.

(2) Core Memory protect - safeguards are built into the hardware/software system to prevent writing in a protected area.

(3) Operations Monitor - this timer must be reset every 5 seconds by the executive program to prevent it from timing out. If some functional program goes into a loop and hangs up, the timer will transfer control to the other computer.

(4) Count-down Registers - these are timers, one of which is assigned to each functional program. A failure of a program to execute in its prescribed period (after a second try) will transfer that program.

(5) Availability check - this is a separate program which outputs a number of signals which are in turn read back into the computer and manipulated. A failure of the routine to give the required answers is annunciated.

(6) Power Loss Restart - a power interruption is recognized and the DCC restarted when power is returned.

The operating experience with the dual computer arrangement has been extremely successful. During the first full reactor year of operation at Pickering G.S., there were two sudden forced outages caused by complete failure of the dual computer system. Neither caused a reactor poison out. Considering that this was during the immature stage of the station life during which the bugs were being worked out of both the computer systems and all the plant control systems themselves, this is a very good record. The dual computer concept has a particular attribute during this immature period that is very valuable. Major control changes can be made without shutting down the plant. At Pickering, early commissioning experience showed that fairly extensive changes were required in the reactor regulating control system. These were designed, input into the slave computer and tested out. Control was then transferred to the slave unit and the process repeated in the master unit. The ease with which changes can be made without causing any plant derating whatsoever was never fully recognized before.

While our efforts in the past have concentrated on the control aspects of computers, the design emphasis is now shifting towards a better communication link between plant and operator. On the Bruce plant extensive use of CRT (Cathode Ray Tube) displays is planned - ten monitors per reactor unit. These CRT displays are displacing the conventional indicators and recorders used on previous plants. This approach is giving the designers a good scope in using their imagination as to how to convey the message to the operator - alphanumerias, graphics, symbolics, etc.

10. CONTROL SERVICES

The Canadian nuclear power plants to date have been designed on the central control centre principle. All reactor units in that plant are controlled from this one area.

All controls associated with the main power production train as well as any auxiliary controls which require operator action within 15 minutes of an event are located in this control centre. There are local control areas for those processes on which slower operator response is satisfactory. Figure 37 illustrates a typical control centre area with the control auxiliaries associated with the main control panel of each reactor unit located immediately behind that panel. There are additional control rooms located adjacent to the reactor areas in which amplifiers and other auxiliary equipment is located.

Figure 38 shows a view of the main control panel of the Pickering station.

All instrumentation electrical power supplies originate from 250 V dc batteries via invertors and convertors. Generally speaking, safety system controls are designed to fail safe, i.e., the loss of electrical power de-energizes the instrumentation circuits, which in turn cause the safety action to proceed. However, because of production reliability requirements, instrumentation power failure is unacceptable. The power systems are therefore being designed with a separate battery supply for each channel of instrumentation. Since triplication is the rule in the Canadian program this means that there are three separate battery supplies with no automatic switching between these channels. The loss of one supply affects neither safety operation nor power production.

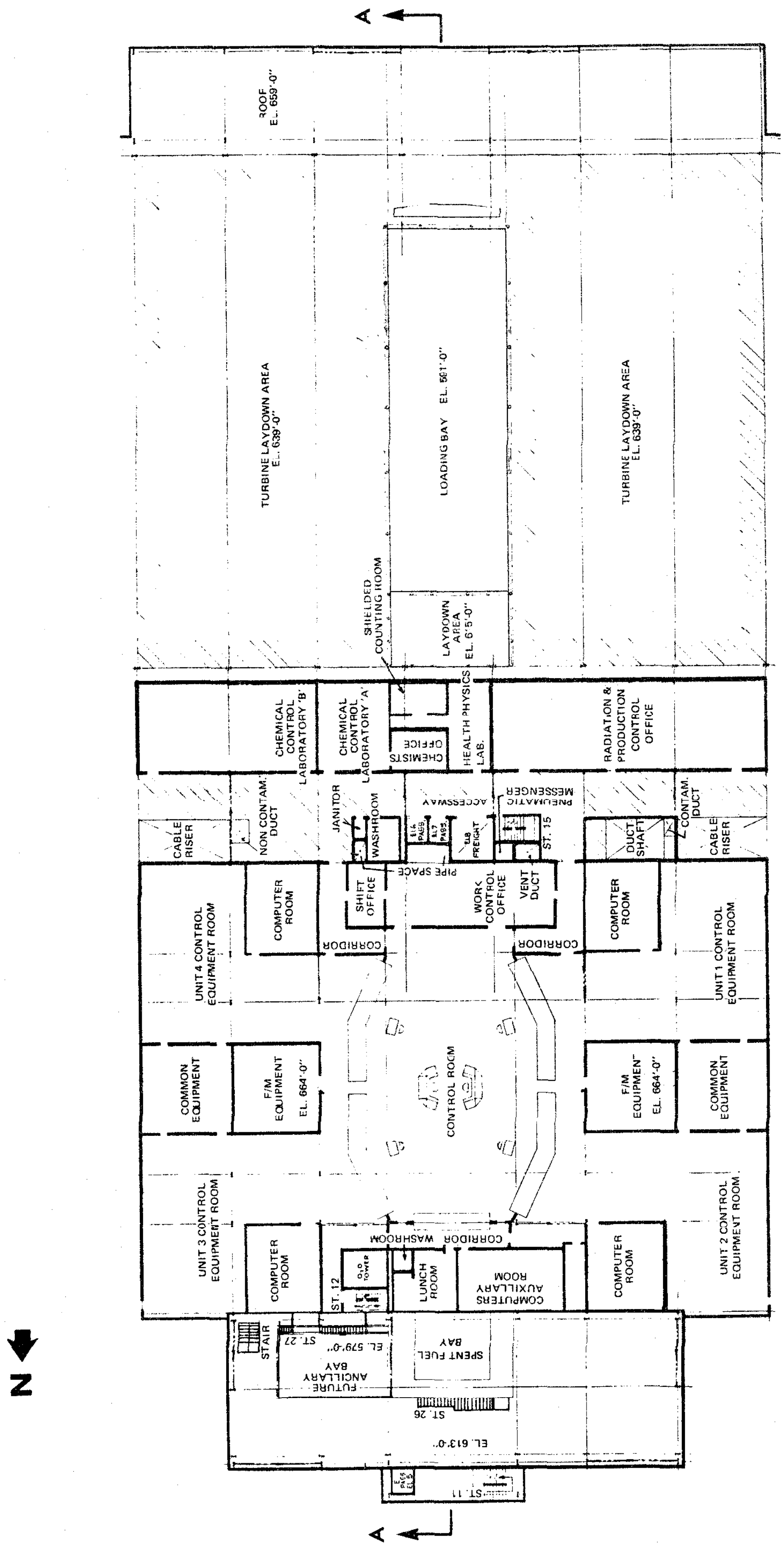
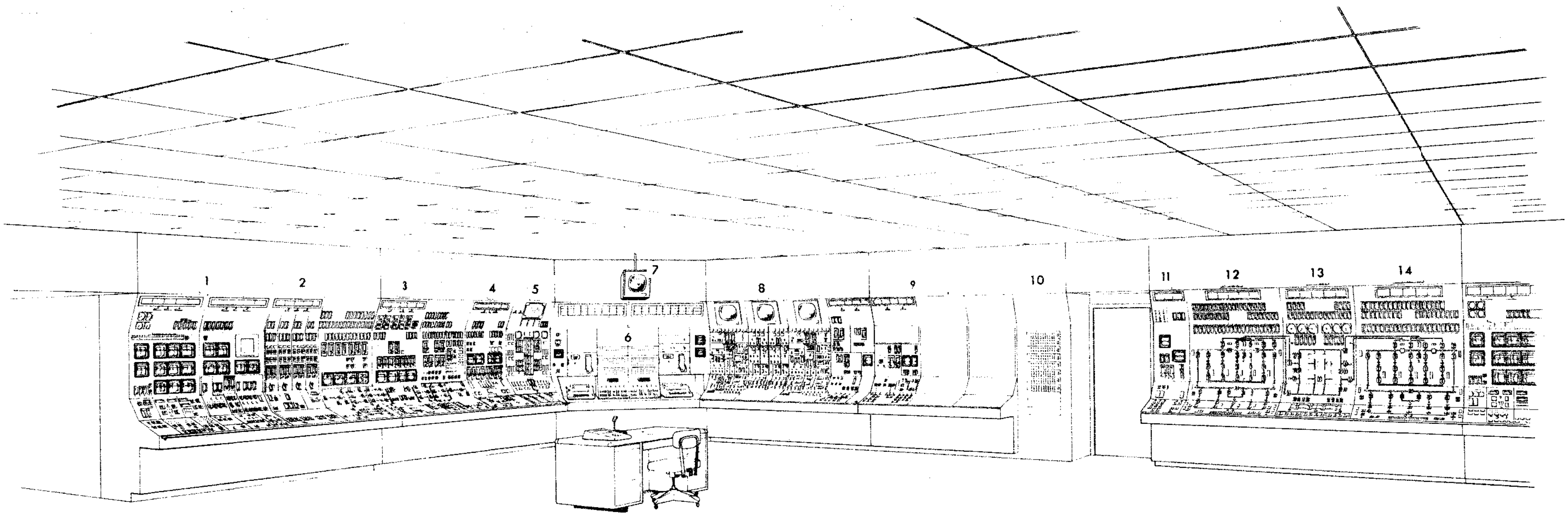


Figure 37 Typical Control Centre Area



1 TURBINE-GENERATOR AND AUXILIARIES

2 BOILERS AND STEAM

3 PRIMARY HEAT TRANSPORT

4 MODERATOR

5 REACTOR CONTROL

6 32 COLUMN PRINTERS
PRINTER KEYBOARD
COMPUTER CONTROLS NO. 1 & NO. 2
CLOSED CIRCUIT T.V. CONTROLS

7 C.R.T. DATA DISPLAY

8 CLOSED CIRCUIT TV
FUELLING MACHINES AND FUEL TRANSFER CONTROLS

9 PLANT AUXILIARIES
SHIELD COOLING
VENTILATION
HEALTH MONITOR AND MISCELLANEOUS

10 D_2O LEAK DETECTION

11 COMMON EQUIPMENT

12 230 KV & STANDBY GENERATORS

13 UNIT 1 STATION SERVICES

14 UNIT 2 STATION SERVICES

Figure 38 Pickering Control Panel Layout