Lesson 9: REACTOR PROTECTION

MODULE 1: REACTOR SHUTDOWN SYSTEMS

REACTOR SHUTDOWN SYSTEMS OBJECTIVES:

At the end of this module, you will be able to:

1. **Briefly describe the rationale for having** *two shutdown* **systems.**
2. **Name the two shutdown systems for CANDU and state the preferred system for recovery purposes.**
3. **Explain the difference between an *absolute* and a *conditional* trip**
4. **Define reactor period and state a typical value.**
5. **Sketch a simplified triplicated contact trip circuit and explain the operation.**
6. **Explain why the other reactivity devices would be *interlocked* with the shutdown system.**
7. **Describe how the marginal Shutdown Rod *drop test* can be performed.**
8. **Explain the operation of a typical trip channel circuit.**
9. **Sketch and describe the operation of a triplicated gas injection valve manifold.**
10. **Sketch a channel of SDS2 to show gas injection and interspace valves with solenoid valves**
11. **Describe precautions taken to prevent *ground faults* from disabling a trip channel.**

MODULE 1: REACTOR SHUTDOWN SYSTEMS

Introduction

- **The shutdown systems are designed to shut down the reactor for all plant conditions to prevent a potentially hazardous situation from occurring.**

- **To ensure high shutdown reliability two completely independent systems are provided:**
  - **ShutDown System 1 (SDS1)**
  - **ShutDown System 2 (SDS2)**

- **SDS2 is designed to operate at higher 'trip' setpoints than SDS1 and ensures a reactor shutdown should SDS1 be unavailable.**

- **Both shutdown systems are designed to quickly insert sufficient negative reactivity into the core to reduce the reactor power output to a safe, subcritical, low level.**

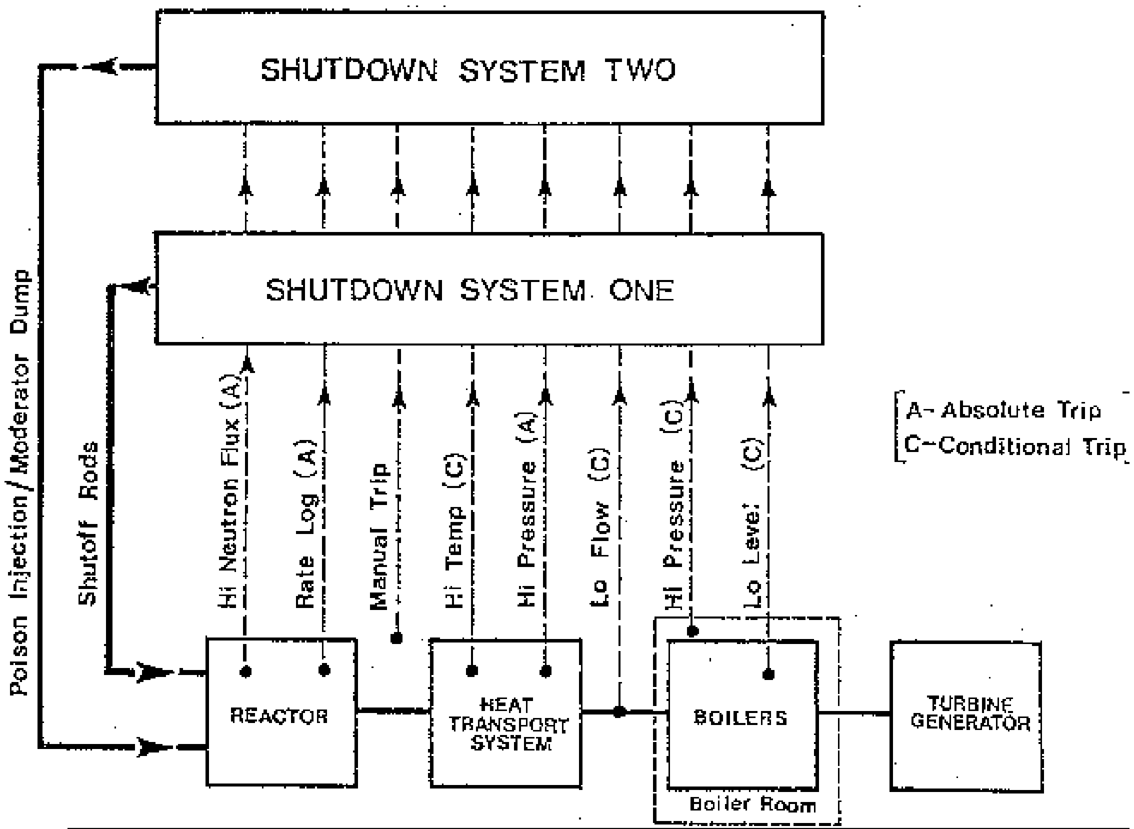


Figure 1: Typical CANDU Shutdown System Block Diagram.

Trip Conditions

- **The conditions which would cause the shutdown systems to come into operation are:**
  - **(1) Loss of reactor regulation.**
  - **(2) Loss of effectiveness of the primary heatsink.**
- **SDS1 is designed to meet all the necessary safety conditions and yet allow the reactor to be quickly restored to operational conditions without exceeding the Xenon poisoning out period of approximately 40 minutes.**
- **SDS2, as a backup method, is more extensive and its operation will mean that a poisoning out due to Xenon will occur as the recovery period from a SDS2 trip will be well in excess of 40 minutes.**
- **To meet shutdown system requirements, it is necessary to monitor several key parameters at all times.**
- **These parameters have "trip" values assigned which are very conservative with respect to the analyzed safety values.**
- **These 'trips' are either absolute (i.e., valid at all states of reactor power), or conditional (i.e., available only above 2% FP).**
- **Loss of reactor regulation is detected by continuously monitoring both gross neutron flux levels and the rate of increase in neutron flux levels by means of neutron detectors.**
- **It is necessary that these detectors should be fast acting so that safety action is initiated as soon as the trip parameter is exceeded.**

Reactor Period

- **Recall that the rate of change of neutron population (and hence, reactor power) essentially follows a logarithmic function.**
- **The time taken for reactor flux to increase by a factor 'e' (base of natural logarithms, equal to 2.718) is defined as the reactor period.**
- **Typical reactor periods for CANDU systems are reasonably long (in the order of 100 sec).**
- **The trip parameter for rate of change of reactor power is set for a reactor period of 10 sec (approximately 10% of usual reactor period) and is defined as the Rate Log Trip.**
- **A ten second reactor period corresponds to a rate of change in power increase of 10% present power per second. (Note: this is 10% of the power level existing at any state of reactor operation).**
- **Rate Log is the abbreviated form of Rate of Change of the Logarithm of Neutron Flux.**
- **The high neutron power limit is a basic design parameter and is set to a level at which fuel bundle maximum over power ratings are not exceeded.**
- **These neutronic trip parameters are absolute.**
- **Another absolute trip parameter is high heat transport system pressure.**
- **This pressure trip value is likely to be exceeded if the heat sink capacity on the steam generator side of the heat transport system is drastically reduced. For example, an in complete tripping of the turbine could cause a high heat transport system pressure condition.**

Reliability Considerations

- Conditional trips are armed automatically at an output level of greater than 2% full power. Below this level of power it has been demonstrated by analysis that these trip parameters are not critical to safe reactor operation. Protection will still be provided at low power levels by the absolute trips.

- In order to meet the requirement of continuous availability, it is essential at least one of the special shutdown systems be available at all times.
- Each SDS should be designed, operated and maintained as closely to 100% reliable as practicable.
- The equipment chosen should therefore be of the highest quality with key items triplicated.
- It is also essential that the system should be available for testing at all times and this, together with any maintenance requirements, implies that each trip system should have more than one channel.
- In fact, each system, SDS1 and SDS2, consists of three separate and independent channels (Channels D, E and F for SDS1 and Channels G, H and J for SDS2) with a requirement that two of the three channels must exceed the setpoints before a reactor trip is initiated. This removes the possibility of spurious trips causing a reactor shutdown.
- It should also be noted that equipment used on shutdown systems is allocated exclusively to reactor shutdown protection and for no other purposes.
- In addition, interlocks are provided such that if a shutdown system has been operated, it is not possible to insert any positive reactivity into the reactor core by, for example, insertion of booster rods or removal of adjuster rods.
- This eliminates the possibility of the reactor power increasing while the original fault condition still exists.

Shutdown System One

- **This system consists of multiple, stainless steel encased, hollow cadmium rods which drop, under gravity, into the reactor core in the event of a trip.**

- **The rods are an effective and distributed neutron absorber which quickly reduce the reactor power to a safe, subcritical, low level.**

- **These rods are retracted on cables which are connected to a winch via an electromagnetic clutch and are normally suspended out of core in the 'poised' state.**

- **Each individual trip channel can be triggered if any trip parameter for that channel is exceeded.**

- **The system must be fail safe so that in the event of an equipment or power failure, the shutdown system will activate and the reactor will be shut down.**

- **The general method of achieving this fail-safe condition is to ensure that the shutdown system operates when consitituent devices are de-energized.**

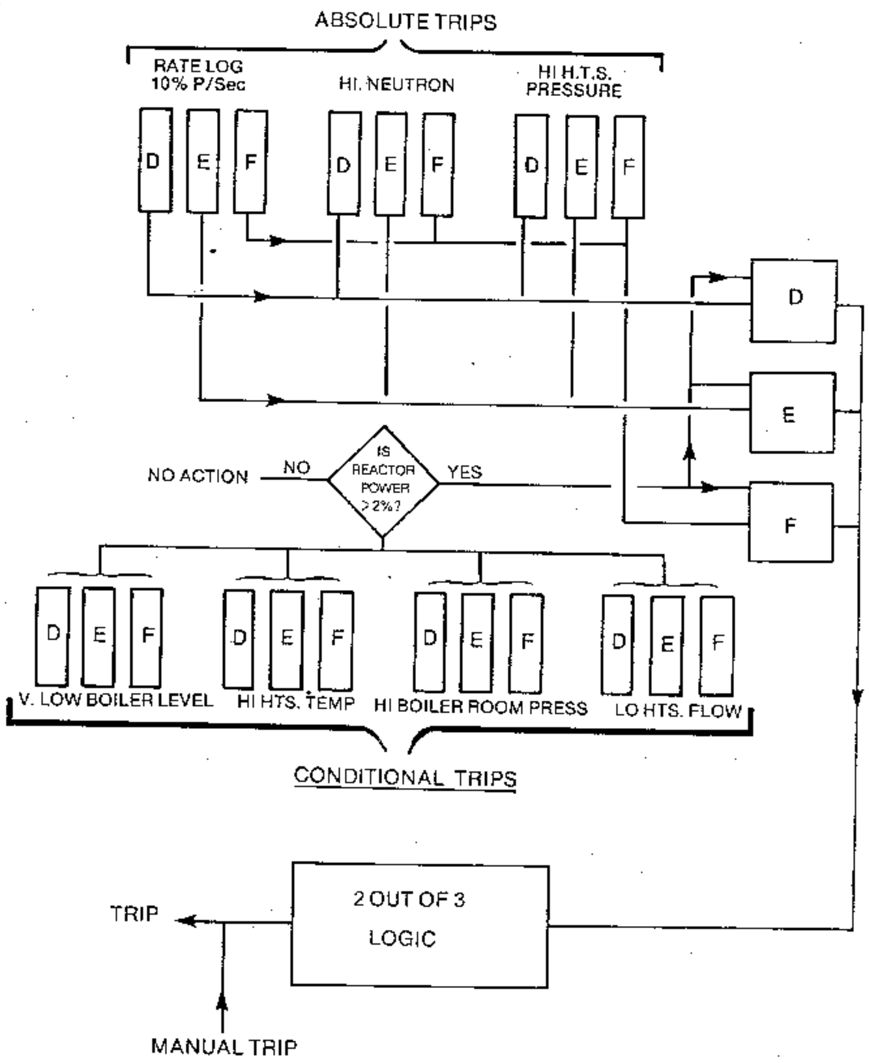


Figure 2: Shutdown System One Schematic.

SDS1….Continued

- **The relays associated with the individual trip parameter detectors (RL1, 2 & 3; HN1, 2 & 3; HTS1, 2 & 3) are energized when the reactor is operating normally (i.e. not tripped).**

- **The individual relay contacts (1RL1 etc.) are therefore closed and thus, relays D, E and F are energized. The contacts D_1, D_2, E_1, E_2, F_1 and F_2 are also closed and a current path exists through the electromagnetic clutch.**

- **This clutch, when energized, holds the shutdown rod, suspended on its cable, out of the reactor core. This arrangement of relay contacts is known as a triplicated contact set. It ensures that the two out of three requirement for tripping is maintained.**

- **In practice, there are multiple sets of Relays D, E and F. Each triplicated contact set controls one pair of shutdown rods.**

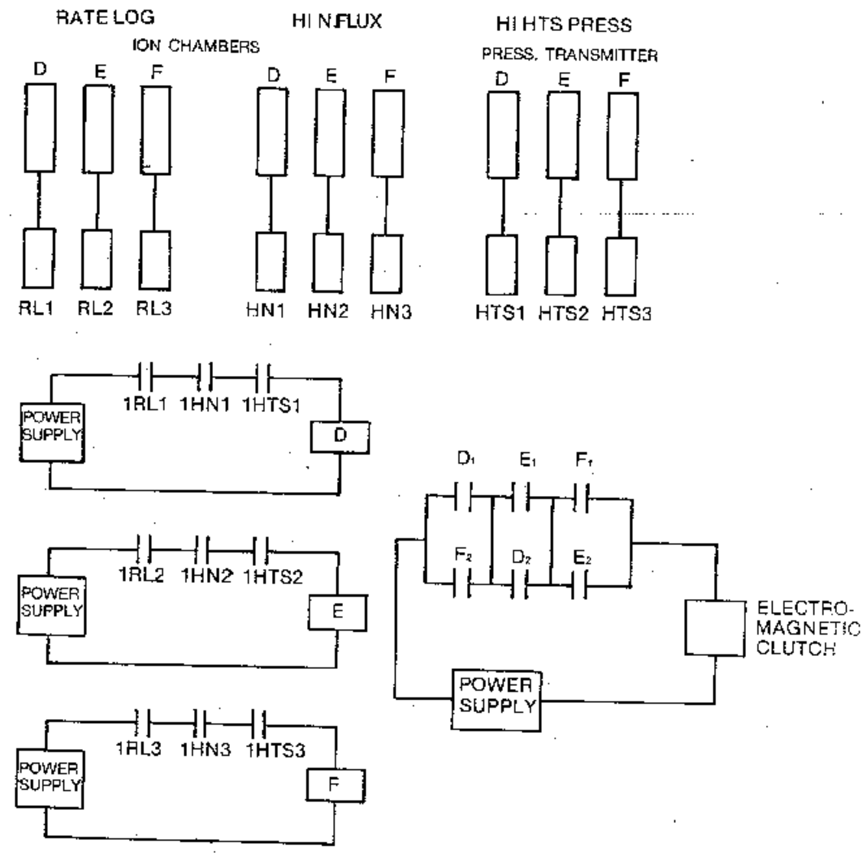


Figure 3: SDS1 Control Schematic

- **Consider, for example, a Rate Log Trip occurring on Channel D. Relay RL1 will de-energize opening contact 1RL1. This will cause relay D to de-energize with the consequent opening of contacts D_1 and D_2.**

SDS1….Continued

- **A current path through the clutch still exists, however via contacts F_2, E_1 and either E_2 or F_2, and the clutch remains energized so that the shutdown rods will not drop into the reactor.**

- **As this trip exists only on channel D, the two out of three criterion has not been met and therefore no reactor trip has occurred and the cause of the incident should quickly be investigated.**

- **Now consider a trip, say High Neutron Flux, on channels D and E. Contacts 1HN1 and 1HN2 would open, de-energizing relays D and E. Contacts D_1, D_2, E_1 and E_2 open, the clutch is de-energized, the shutdown rods will drop and the reactor will be shut down. In actual fact, there is, a bank of shutdown rods to distribute the shutdown action across the reactor core.**

- **It can be seen that the arrangement of equipment as shown in the previous diagram, fulfills all the reliability requirements of the shutdown system:**
  - **System is not susceptible to spurious trips.**
  - **Two out of three channels are required to initiate trip action.**
  - **On-line testing and maintenance of individual channels is possible.**

Marginal Rod Drop Testing and Circuit Status Logic

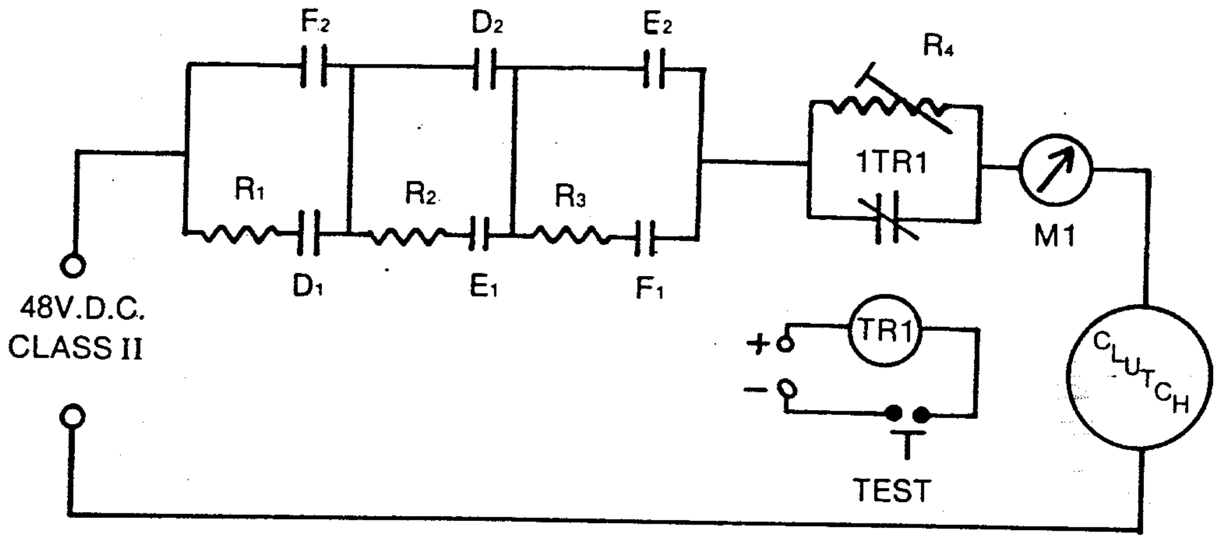


Figure 4. Marginal drop Test Feature in the Clutch Control Circuit

- **To enable testing, and to give further indications to the operator of equipment serviceability, modifications to the triplicated contact set are made by the addition of resistors R_1, R_2 and R_3 in one leg of the contact set.**
- **Under normal operating conditions, with all contacts closed, the preferential (low resistance) current path will be via contacts D_2, E_2 and F_2 with a higher current indication shown on the ammeter M1.**
- **Should a trip occur on just one channel, say D_1 contacts D_1 and D_2 will open and the current path to the clutch will now be via E_2, F_2 and E_1. (Note: only one channel tripped, so no shutdown).**
- **The presence of R_2 will lower the current flowing in the circuit with a new, lower, indication shown on M_1.**
- **This, in addition to a warning annunciation lamp, will indicate to the operator that one channel has tripped.**

System Reliability and Testing

- **To ascertain full system reliability, it is also necessary to check that the shutdown rod is free to drop, when the clutch is de-energized.**
- **It would not be desirable, due to the large local negative flux transient which would occur, to completely drop any one rod for its full distance into the core for testing at power.**
- **If we can arrange to drop the rod for a limited distance only, we can be reasonably certain that the SDR has freedom of motion and that, in the event of a trip, the rod is able to drop to its limit.**
- **This procedure is known as a Marginal Drop Test and is performed, on one rod at a time, by the following means:**
  - **Relay contact 1TR1 is normally closed (Relay TR1 - de-energized).**
  - **Resistor R_4 is adjusted such that, when in circuit, (i.e., when 1TR1 is open) the current flowing through the clutch will be reduced such that the clutch can not quite sustain the load, and the rod will drop.**
  - **Relay TR1 is a timer relay.**
  - **Operation of the Marginal Drop Test switch button will energize the relay for a period of about, 0.2 sec.**
  - **During this time interval 1TR1 will open, R_4 will be in circuit, clutch current will be reduced and the shut off rod will begin to drop.**
  - **The distance dropped (typically 1.2 metre) will be indicated on a rod position meter.**
- **The contact TR1 will then close, the clutch is re-energized and the rod will be retracted to the 'poised' position by means of the motor driven winch.**
- **Operation of the Manual Trip will de-energize all three channels independent of trip logic status.**

Shutdown System Two

- **Shutdown system two is similar to shutdown system one with the following differences:**
  - **Higher trip set points.**
  - **The final negative reactivity device.**

- **All CANDU units built after 1975 have a SDS2 that operates by injecting a suitable neutron absorbing liquid (poison) into the reactor. The poison chosen is Gadolinium Nitrate.**

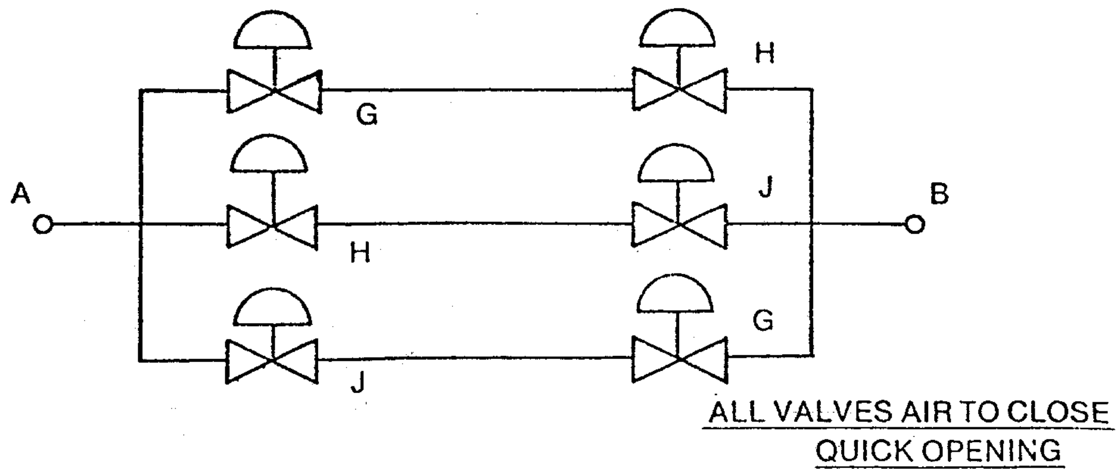


Figure 5: SDS2 Control valve Arrangement.

- **The system has a two out of three trip circuit using control valves to apply the high pressure injection gas instead of relay contacts.**

- **The valves used are air to close style so that following a loss of instrument air, the valves will fail open and a reactor shutdown (fail safe) will occur.**

- **In the event of a trip, the air supply to the valves is dumped via electrically operated solenoid valves.**
- **If any two of the three pairs of valves open, a flow path will be established between A and B, allowing the high pressure cover gas to inject the poison into the moderator.**
- **In actual fact, there are interspace vent valves located between the injection valves to keep the interspace pressure low in case the upstream valve leaks. These interspace vent valves will close on a trip signal to maintain the interspace integrity. However, if the interspace vent valves should fail open, their capacity is restricted so that injection will still occur.**

Poison Injection System

- **Some of the general principles introduced in this lesson can be demonstrated by examining the poison injection system (SDS2) utilized for a CANDU reactor.**

- **The triplicated channels are designated as G, H and J and can be activated manually or by such trip parameters as** rate log, high neutron power, **or** high primary heat transport pressure.

- **The helium storage tank is maintained at approximately 8 MPa.**

- **Trip action must be requested by at least two of the three channels to initiate poison injection.**

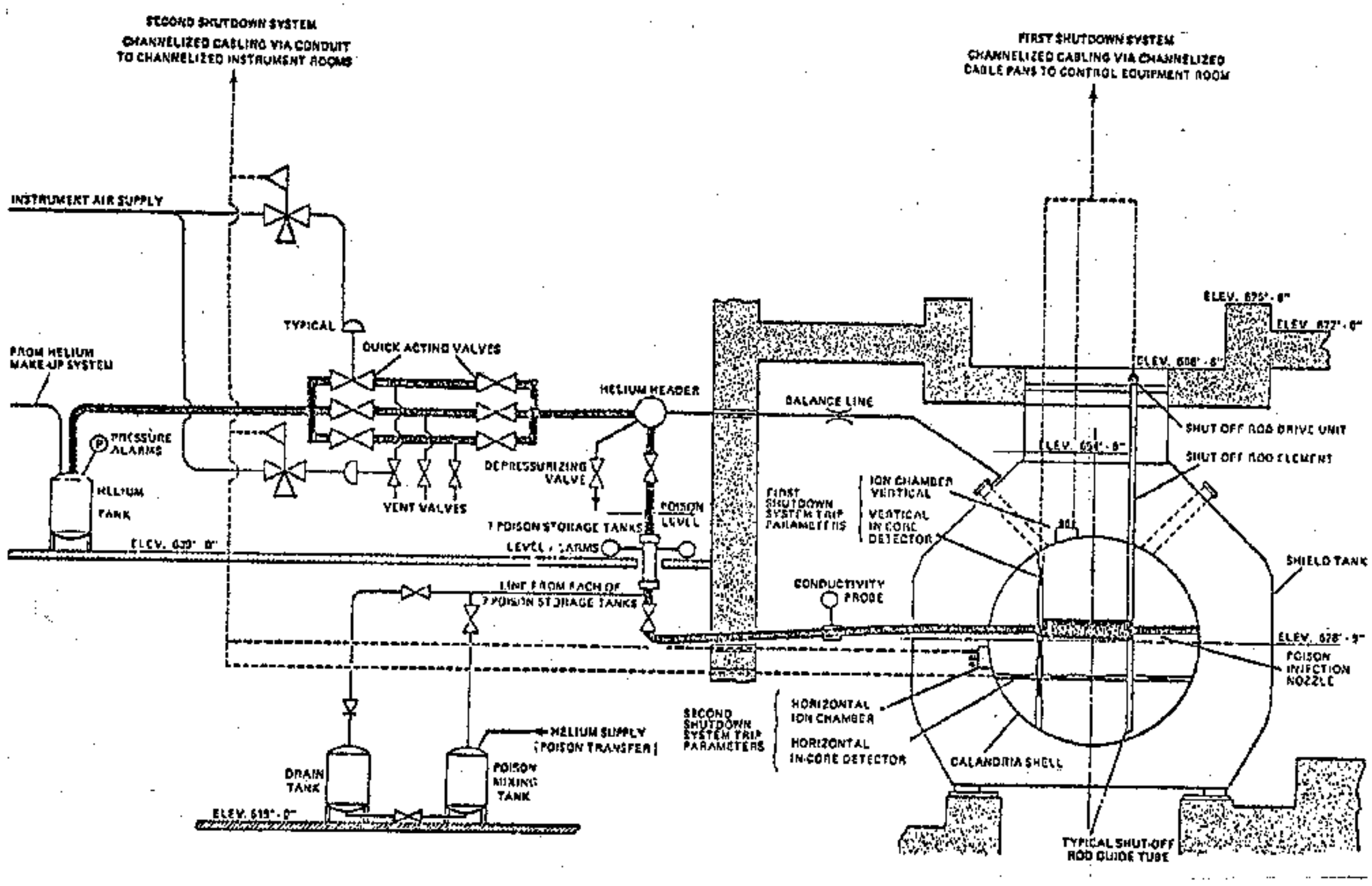


Figure 6: Simplified Schematic for SDS2.

Poison Injection

- **The poison injection valves will open and apply the stored helium pressure to the gadolinium nitrate in the seven storage tanks.**
- **The poison is forced through the seven injection nozzles by the helium pressure so that it is sprayed into the centre of the reactor core.**
- **The poison tanks each contain a polyethylene ball which floats on the surface of the poison. Once the poison is injected, the ball will be forced onto the lower seat in the poison tank which prevents the helium gas from over-pressurizing the calandria.**

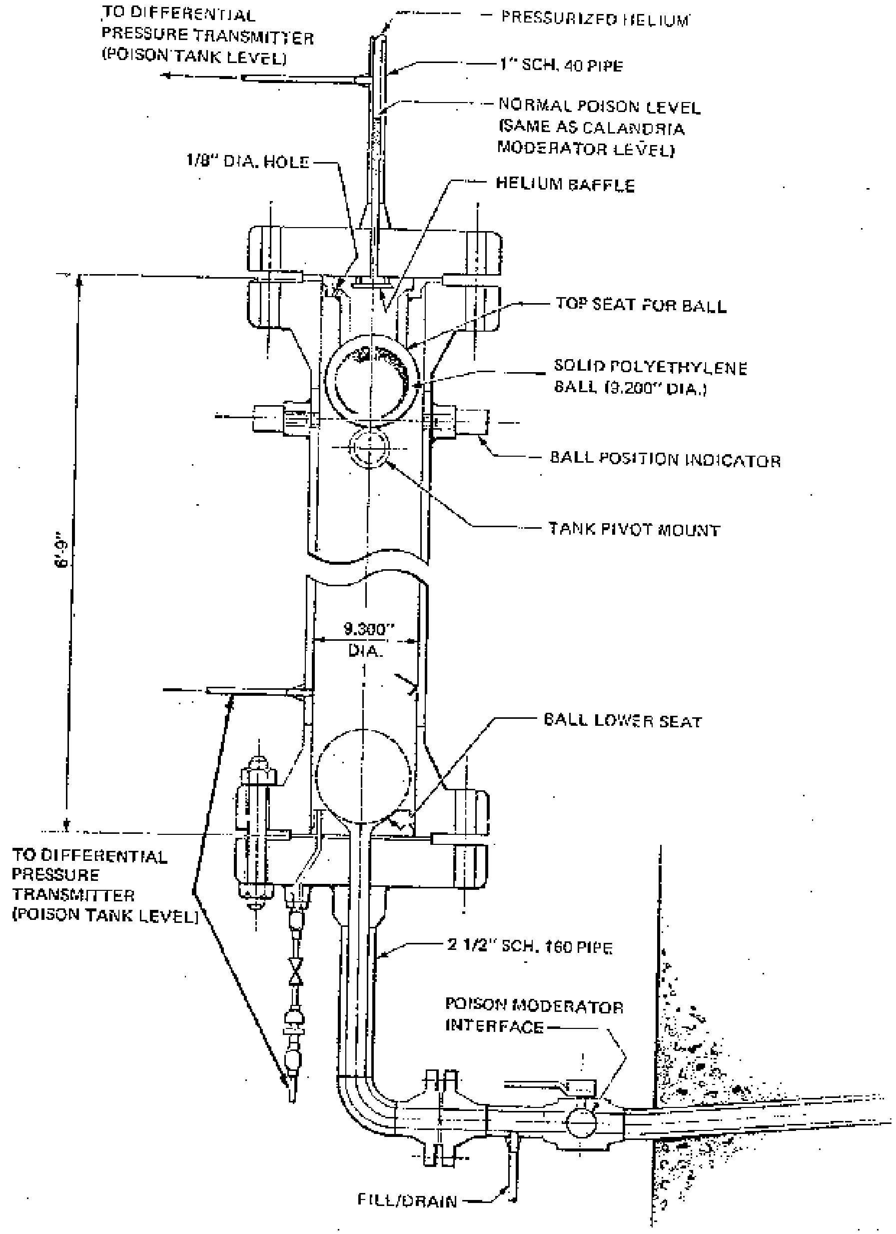


Figure 7: Gadolinium Poison Tank.

Injection Valve Logic

- **The triplicated valve configuration consists of the six injection valves (MV1G, 2G; MV1H, 2H; MV1J, 2J) and the three interspace vent valves (MV3G, MV3H, MV3J).**
- **All nine of these valves are of the air to close, quick-open type.**
- **The poison injection valves will have some** seat leakage **when the valves are closed.**
- **With the large pressure differential across the closed valve, a substantial pressure could build up between the two injection valves.**
- **This would then result in pressure leakage across the second injection valve and the possible accidental injection of the poison into the moderator.**

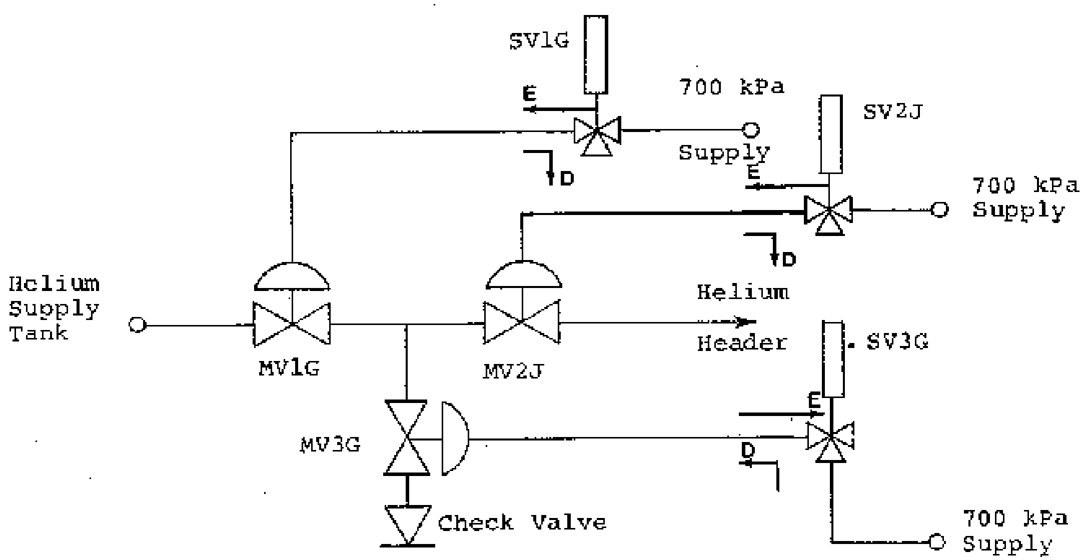


Figure 8: Connections to Injection and Vent Solenoid Valves.

- **To counteract this problem, the interspace vent valves are held open as long as the channel is energized (i.e. not tripped).**

- **These valves will vent the interspace so that the seat leakage of the poison injection valves is not a problem.**

SDS2 System Operation

- **Assume channels G and J are energized (refer to Figure 9).**
- **Then SV1G, SV2J, and SV3G will all be energized so that the 700 kPa(g) signal is applied to MV1G and MV2J holding these injection valves closed.**
- **MV3G is able to vent through SV3G so that the interspace vent valve will be open.**
- **Should a trip occur, these solenoid valves would become de-energized, allowing the injection valves to open and the interspace vent valves to close.**

SDS2 Trip Channel

- **A simplified trip channel (for example G in Figure 9) can be considered for this injection system.**
- **For simplicity, consider only high neutron power or rate log as the parameters which will activate SDS2.**

- **Assume a rate log trip occurs so that relay R_2 is de-energized.**
- **Immediately contacts 2C1 and 2C2 will open and remove power from relay R_3 and the three solenoid valves (1G, 2G & 3G) .**

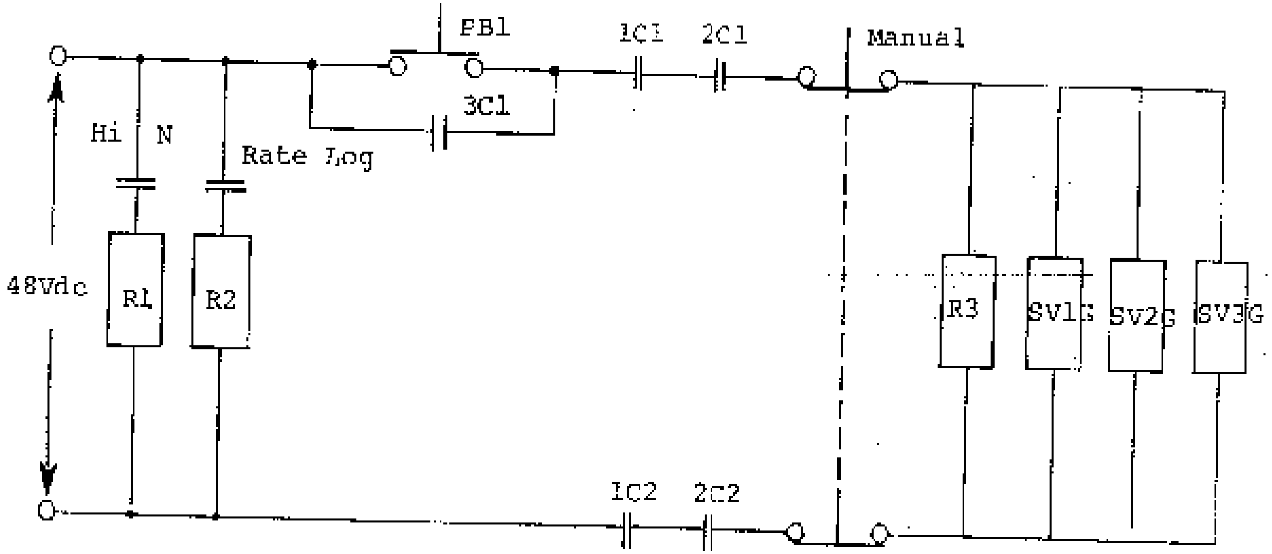


Figure 9: Simplified Trip Channel G for SDS2.

- **The poison injection valves will drive open and the vent valve will close due to the change in status in the solenoid valves. Contact 3C1 will open since relay R_3 was de-energized.**

- **Notice that if only channel G had tripped, a complete injection would not occur.**

- **Two of the channels must request safety action before a trip can occur.**

Operating Considerations

- **If this was a spurious trip, the rate log contacts would reclose, but the channel would not be energized since the trip was latched in by relay R_3.**
- **Operating staff would have to depress pushbutton PB1 to reset the channel.**
- **Pushing PB1 will apply power to R_3, SV1G, SV2G, and SV3G. Contact 3C1 will close so that power is maintained to the equipment mentioned when the pushbutton is released (3C1 bypasses PB1).**
- **The channel has been restored or reset to its pretrip status.**
- **Testing and maintenance can be carried out on individual channels without tripping the reactor. The control valve state, i.e., closed for normal operation, is verified by control room indication.**

Normal System Operation (SDS1 and SDS2)

- **During normal reactor operation, both shutdown systems must be available and operational at all times.**
- **The minimum number of shutdown rods to guarantee reactor shutdown (safety bank) must also be available.**
- **Maintenance of more than one channel at the same time is not allowed.**
- **Testing and maintenance should be performed with the reactor at full power in order to verify the full power trip settings.**
- **Various control room indications are available to ascertain that the systems are fully operable.**
- **The flux detectors can be tested by driving a boron shutter which is a neutron absorber located near the detector.**
- **his should result in a rise in indicated neutron flux readings which has the effect of testing the complete channel for both Rate Log and High Neutron Flux.**
- **Control room indications are also available to verify that supply voltage to the neutron detectors is present and at the correct value.**

Normal System Operation (SDS1 and SDS2)…continued

- **During normal (i.e., safe) conditions, certain changes in reactor operation are necessary which, if not compensated for, could produce situations where trip parameters would be exceeded unnecessarily with shutdown of the reactor occurring.**
- **Consider for example, requests for increases in power output. Power increases are computer controlled, and applied at a rate such that the High N and the Rate Log Trip Setpoints should not be exceeded.**
- **It is necessary at these times that the operator visually checks the neutron instrumentation to ensure that trip settings will not be exceeded.**
- **Refuelling can cause large fluctuations in neutron detector output.**
- **This is due simply to the physical movement of neutron absorbers (metal fuelling ram extension or spent fuel), or neutron producers (fresh fuel) between the detectors and the usual neutron source (reactor core).**
- **Increased instrumentation surveillance is necessary during refuelling to ensure that compensation is present and that the Neutron Flux (High Neutron and Rate Log) trip parameters are not likely to be exceeded.**
- **Use of welding equipment can induce a voltage spike which can trip a channel**
- **Maintenance on trip channel instrumentation cab initiate a quickly changing signal**

Abnormal Operating Conditions

- **Should a single channel trip, the operator must first establish, by instrumentation inspection, whether the trip was genuine or due to equipment malfunction or noise.**
- **In the event of a genuine trip due to a transient condition occurring on just one channel (e.g., during refuelling) the channel may be reset after the transient has subsided.**
- **If the trip was the result of equipment failure, the channel must be rejected, the necessary approval for maintenance must be obtained, and the work carried out.**
- **If for any reason, a single channel has been worked on during an outage, that channel must be rejected (ie placed in the trip condition) until normal operating (full power) conditions have been attained to permit the proper testing under in-service conditions.**
- **Normal testing and maintenance must be carried out at full power.**
- **In the event of a complete reactor trip, it is first necessary for the operator to establish, from the instrumentation and read-out devices, the cause of the trip.**
- **The operator must then decide whether it is possible to diagnose and clear the fault within thirty minutes and thus be able to restore criticality before poisoning out.**
- **Should a shutdown rod become trapped in the core (say faulty marginal drop test), this condition will be indicated by the appropriate shutdown rod position meter.**
- **Severe local flux distortions will result. These local negative reactivity excursions may be partially corrected by other reactivity devices, (e.g., adjuster rods and liquid zone level adjustment).**
- **However, the reactor power output must be reduced to avoid local fuel overrating and possible fuel failure.**
- **Care must also be taken when operating with the heat transport system at reduced pressure. The heat transport system could boil if the pressure is allowed to fall too low.**
- **This will result in cavitation of the main HTS pumps and a low flow condition may develop which could cause a conditional trip.**
- **If boiling were allowed to persist, voiding in the fuel channels could occur. This condition would case the reactivity to increase which could also trigger a neutron trip.**

Grounding Problems

- **Ground faults appearing in a trip channel circuit must not be able to make that channel fail unsafe.**
- **Consider a simplified trip channel consisting of a power supply, a contact set, and a relay. De-energizing the relay by opening the contact set will initiate safety action.**
- **Ground faults can occur in a system as a result of physical abuse or dampness allowing a leakage path. Imagine someone drilling through a tray bracket, and the drill bit nicks the insulation allowing the conductor to contact the bracket screw. On the other hand, the insulation of a flexed portion of the cable may become cracked and split, and allow a current to flow to ground if the cable should become wet.**

- **The ground faults (G1 and G2) shown in Figure 10 can allow a ground current flow (I_g).**
- **This ground current may be sufficiently large to keep the relay energized.**
- **In this case, the trip contacts can be opened or closed, and the relay will remain energized.**
- **This is a potentially hazardous situation where the ground faults have caused the trip channel to fail unsafe - a requested trip would be ignored.**

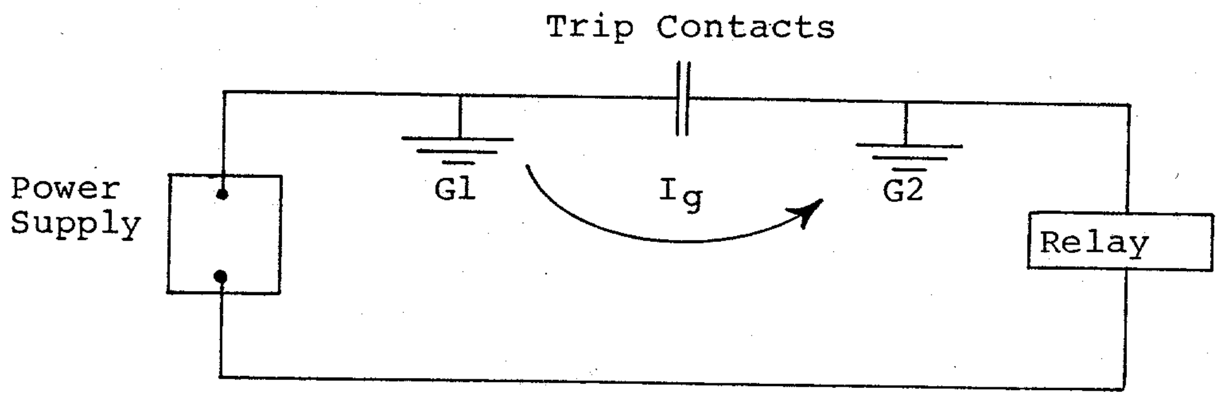


Figure 10: A Simplified Trip Channel with Ground Faults G1 and G2.

One solution to this problem that is employed in conventional instrumentation loops is to apply an intentional ground to the power supply.

Grounding

- **The disadvantage of applying an intentional ground is that one ground fault appearing on the trip channel can now cause a channel trip request.**
- **For example, if ground fault G_1 appears as shown in Figure 11, the ground current effectively shorts out the relay.**
- **The relay is de-energized and safety action is initiated.**
- **This results in ground faults causing an unnecessary channel trip - but the trip channel has failed safe.**

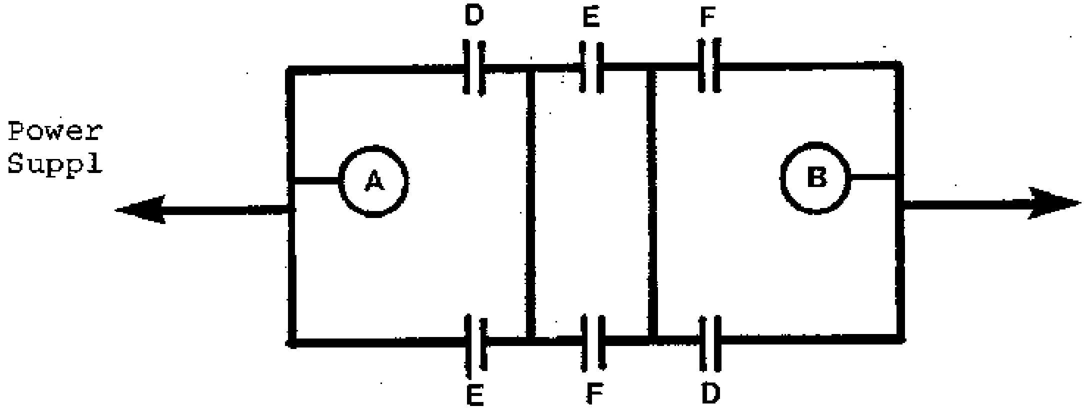


Figure 11: A Trip Channel with an Intentional Ground (G1)

- **An improvement in the trip channel performance with ground faults can be achieved by duplicating the trip contacts on both sides of the trip channel.**

- **If ground faults G1 and G2 should now occur, the top line of the trip channel would appear as a complete circuit regardless of the trip contact status.**
- **However, the second contact on the lower line of the trip channel can still open the circuit if a trip is requested. This channel will now initiate safety action even with two ground faults (G1 and G2) present.**

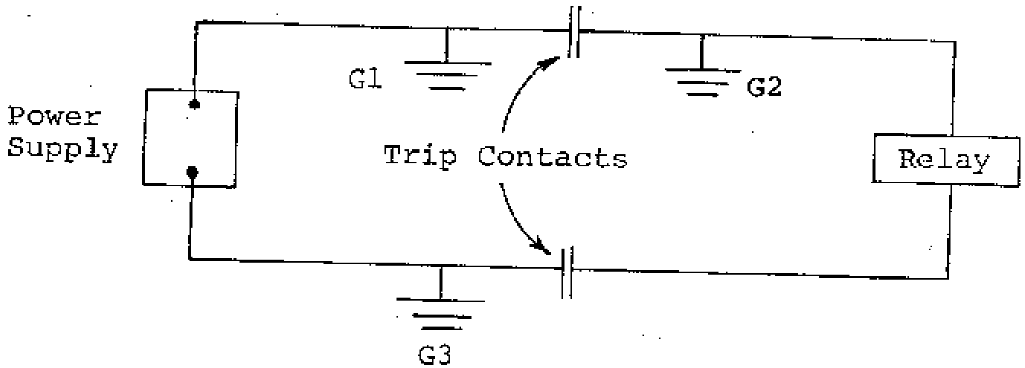


Figure 12: A Trip Channel with Duplicated Trip Contacts.

Ground Faults on the Triplicated Contact Set

- **Note that as the normal circuit is not referenced to ground at any point, the circuit will show complete immunity to a single ground fault.**
- **More than one ground fault could affect the reliability of the circuit depending upon their locations. The worst possible location for multiple ground faults is at position A & B. This could by-pass the contacts and disable one set of shutdown rods.**
- **However ground detection equipment is installed for Class II systems. This will inform staff that a ground fault has occurred, enabling immediate corrective maintenance action to be initiated.**

Summary

- **The complete loss of electrical power to either shutdown system will result in a reactor trip.**
- **Loss of air to the control valves for shutdown system two will result in a reactor trip.**
- **Remember also that operation of SDS2 will automatically result in a poisoning out of the reactor.**

Figure 13: Triplicated Contact Set.

- **An overriding consideration in the design of both shutdown systems is that they must FAIL-SAFE.**
- **In the event of equipment failure an erroneous trip is preferred to the possibility of no trip should a safety parameter be-exceeded.**
- **If the plant is to be in an operational state, the reactor protective system must be in a poised state in order to provide safety action at all times.**

MODULE 1: REACTOR SHUTDOWN SYSTEMS ASSIGNMENT

1. **Briefly describe the rationale for having two shutdown systems.**

2. **Name the two shutdown systems for CANDU and state the preferred system for recovery purposes.**

3. **Distinguish between an absolute and a conditional trip and state two typical parameters for each trip group.**

4. **Sketch a typical relay contact configuration used to actuate SDS1. Explain the advantages of such redundancy in a protective system, and discuss the general operation following a trip condition.**

5. **Refer to Figure 5 and describe how the clutch current indication (M1) will change should channel F be de-energized, opening contacts F1 and F2.**

6. **Explain why the other reactivity devices would be interlocked with the shutdown system.**

7. **Sketch and describe the operation of a triplicated gas injection valve manifold.**

8. **Sketch a channel of SDS2 to show gas injection and interspace valves with solenoid valves and describe the general operation for both poised and tripped states.**

9. **Describe precautions taken to prevent ground faults from disabling a trip channel.**

10. **Refer to Figure 10 and describe the circuit response following a rate log trip. State the purpose of push button PB1. Why are Relay 1 and 2 contacts duplicate on both sides of the trip channel?**

Lesson 9: Computerized Shutdown Systems
Module 2: Trip Computer Systems and Safety Critical Software

- CANDU reactors are designed with ***two shutdown systems*** to provide a combined shutdown system unavailability target of **10^{-6} years/year.**

- These special shutdown systems are **physically** and ***functionally separate*** from the process control systems and from each other.

- Each reactor shutdown system is designed to be fully capable of ***independently shutting down the reactor*** when called upon to do so.

- The special shutdown systems are designed, built and maintained to a very ***high quality assurance*** standard.

- These systems are designed to ***fail-safe*** so that safety action will always be provided, perhaps unnecessarily, in the event of system or device failures (i.e. such as loss of power).

- ***Shutdown System Number One*** (**SDS1**) utilizes neutron absorbing rods (i.e. stainless steel coated cadmium rods) which are poised above the reactor core (i.e. ***vertical*** core access) and ***drop by gravity*** upon a request for shutdown (i.e. ***de-energize*** the clutches holding the rods above the core).
- ***Shutdown System Number Two*** (**SDS2**) injects a liquid chemical neutron absorber into the core through horizontally mounted injection nozzles (i.e. ***horizontal*** core access) upon a request for shutdown (i.e. fail-open control valves are driven open by venting to allow the poison injection to proceed).
- Note the ***diverse*** and ***independent*** reactor shutdown mechanisms.

Computerized Shutdown Systems…..continued

- For ***reliability*** purposes and to ensure that no ***single failure*** prevents a necessary reactor trip, each system consists of ***three*** identical instrumentation and logic trip ***channels*** (i.e. triplicated).

- The system safety action is initiated if ***two of the three*** channels detect a condition requiring reactor shutdown. This is referred to as ***2 out 3*** consensus logic.

- A ***single channel*** tripping will not trip the reactor but has placed only its channel output devices in the ***safe state*** and initiated the corresponding ***alarms***. Now either of the remaining channels can initiate safety action when that second channel also senses and responds to a trip parameter condition.

To date, there are ***four evolutionary phases*** of CANDU shutdown system trip logic design to present times from the traditional ***analog*** approach to the ***fully computerized*** system. It is important to note that the shutdown system itself (i.e. the physical system design) remained a consistent design over the years, but the control logic implementation strategy gradually become more and more computerized.

Typical Trip System Parameters

- It is worth considering typical shutdown system trip parameters for illustrative purposes. In each case the ***trip parameter*** and ***trip level*** is selected by safety analysis to ensure that the licensed fuel temperature limits are not exceeded. The parameters will trip with an ***adequate margin*** to the analyzed safety limit to ensure continual safe performance.

Typical Trip System Parameters

Neutronics

1. Neutron Flux Level High *- reactor power level is too high*

2. Neutron Rate Log *(Rate of Change of Logarithmic Power High) - rate of change in power is too fast*

Process

3. Steam Generator Level Low *- impending loss of principle heat sink*

4. Feedwater Line Pressure Low *- impending loss of principle heat sink*

5. Pressurizer Level Low *- unexpected low heat transport inventory*

6. Heat Transport Pressure High *- energy mismatch, reactor power too high*

7. Heat Transport Pressure Low *- impending heat transfer problems, boiling & cavitation*

8. Heat Transport System Gross Flow Low *- impending heat transfer problems*

9. Reactor Building Pressure High *- possible hot fluid leak in containment or loss of vacuum*
10. Moderator Level Low *- possible overrating of those channels still moderated*

11. Moderator Temperature High *- lower subcooling margin for moderator*

Manual

12. Manual Channelized (i.e. D, E & F) ***Trip Pushbuttons*** (with common or individual capability)

Traditional Shutdown System Trip Logic - original design

- A shutdown system consists of process and neutronic ***sensors***, ***reactivity devices***, ***comparator logic instrumentation***, ***man-machine interfacing*** (MMI) devices as well as ***cabling*** and ***interfacing relays***.

- If any of the trip parameters are sensed to be operating ***beyond the acceptably safe margin*** to the analyzed unsafe state (i.e. power level too high, coolant flow too low, etc), then that parameter in that channel is recognized as being tripped and so the channel is ***de-energized*** in an attempt to trip the system (and thereby the reactor).

- As mentioned before, ***2 out of 3*** majority voting must occur before the system is tripped.

- Once ***two channels*** are tripped or de-energized, the final reactivity device control circuits are de-energized and safety action is initiated (e.g. shut down rods drop into core or liquid poison is injected) to shutdown the reactor.

- Even if one shutdown system does trip the reactor, the alternate Special Safety System remains poised in readiness to also trip the reactor, if called upon to do so, independent of the actions of the other shutdown system.

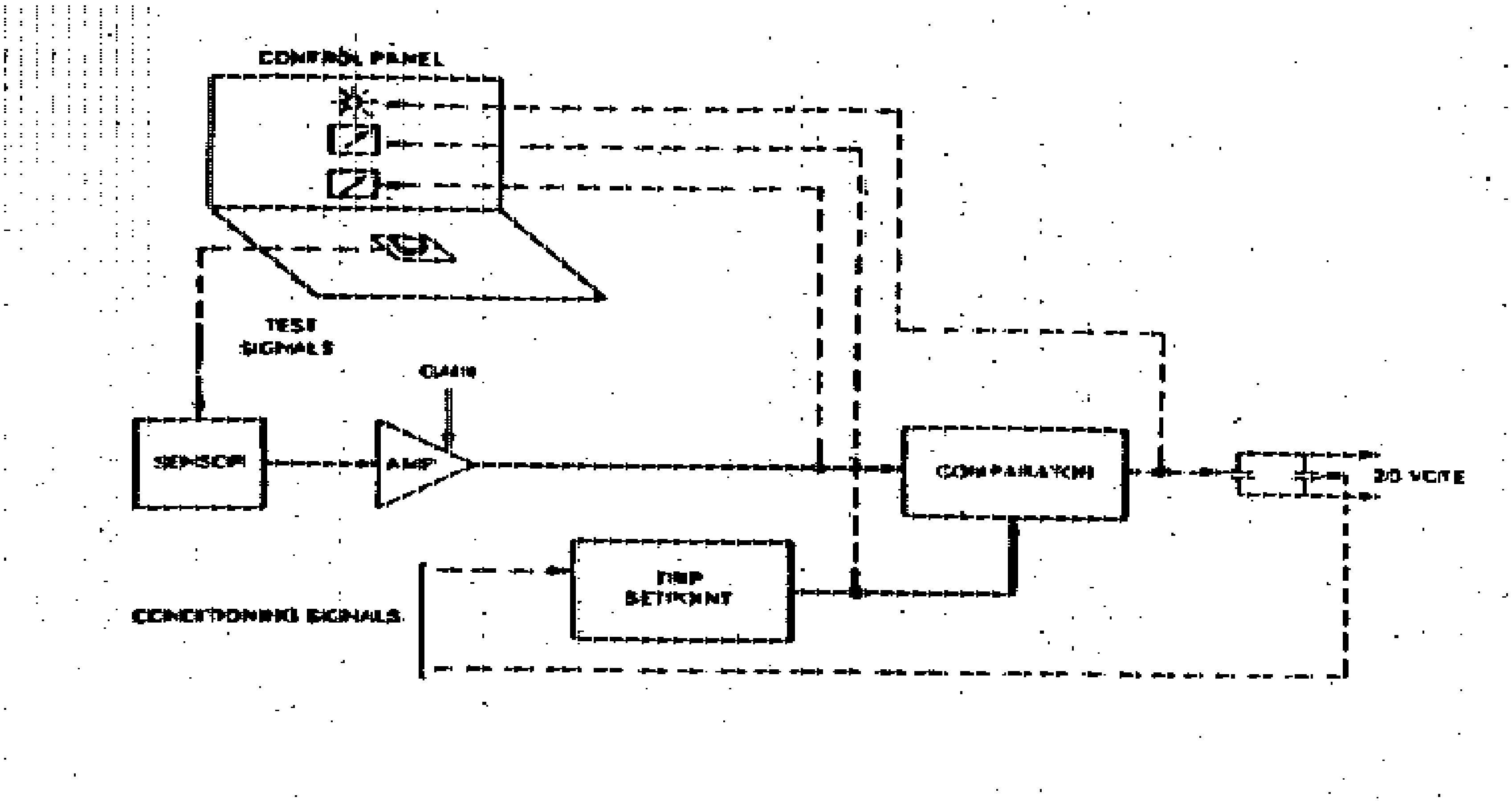


Figure 1. Conventional Shutdown System Analog Instrumentation Configuration

Traditional Shutdown System Trip Logic - Operator Interfacing

- All ***trip signals***, ***trip setpoints*** and ***trip status information*** are ***continuously displayed*** in the main control room via conventional instrumentation devices.

- Manual controls allow the operators to ***periodically test*** the shutdown system channel devices from sensor to final reactivity device.

- Note that an ***entire channel can be tripped*** during a test condition but that the reactor would still be operated at power since ***a second channel has not tripped***.

- The ***unavailability target*** for each special shutdown system of 10^{-3} yrs/yr (i.e. 8.76 hours per year) must be verified by an on-going shift-based ***testing program*** to ***demonstrate the availability*** of the system to function if called upon to do so.

- The trip logic in this design is implemented by ***relay logic*** , operator displays are small ***panel mounted meters and lights*** while operator controls are pushbuttons or handswitches.
- Data monitoring and logging was quite limited for this design and usually consisted of multipen trend recorders.
- This was an obvious area for human-system interfacing improvements.

Traditional Shutdown System Trip Logic with Monitoring Computers - First Evolution

- The shutdown system ***trip circuitry*** and ***man-machine interfaces*** remained fundamentally unchanged from the traditional design - so that ***signal comparisons***, ***decision making***, ***trip initiation*** and ***man-machine interfacing*** remained as originally designed and proven functional.
- However, a new ***monitoring computer*** system was connected to the trip channels by rigorously ***buffered*** unidirectional interfaces.
- These ***one-way*** buffered interfaces were designed and tested to ensure that no faults in the computer system could possibly be ***propagated*** back into the trip circuit to ***degrade*** or ***disable*** the trip circuit functions.
- Once the safety system data was available within a computer system, then unlimited ***data manipulations***, ***statistical checks*** and ***comparisons*** could be made along with flexible and informative display capabilities.
- The monitoring computer system improvement consisted of a ***remote multiplexor***, located in each shutdown system channel, which obtains the channel parametric data.
- This data can then be ***displayed*** on convenient bar chart or analog trend displays on a selected CRT in the main control room.
- The computer can also give the operator an ***early warning*** if a variable is detected too close to a setpoint (i.e. impose an operating margin threshold) , or for failed signals or signal discrepancies among similar signals.
- This manner of providing information to the operator can be much more ***user friendly*** while making it easier for the operator to be aware of ***small changes*** or to be alerted early of an ***impending upset*** condition.

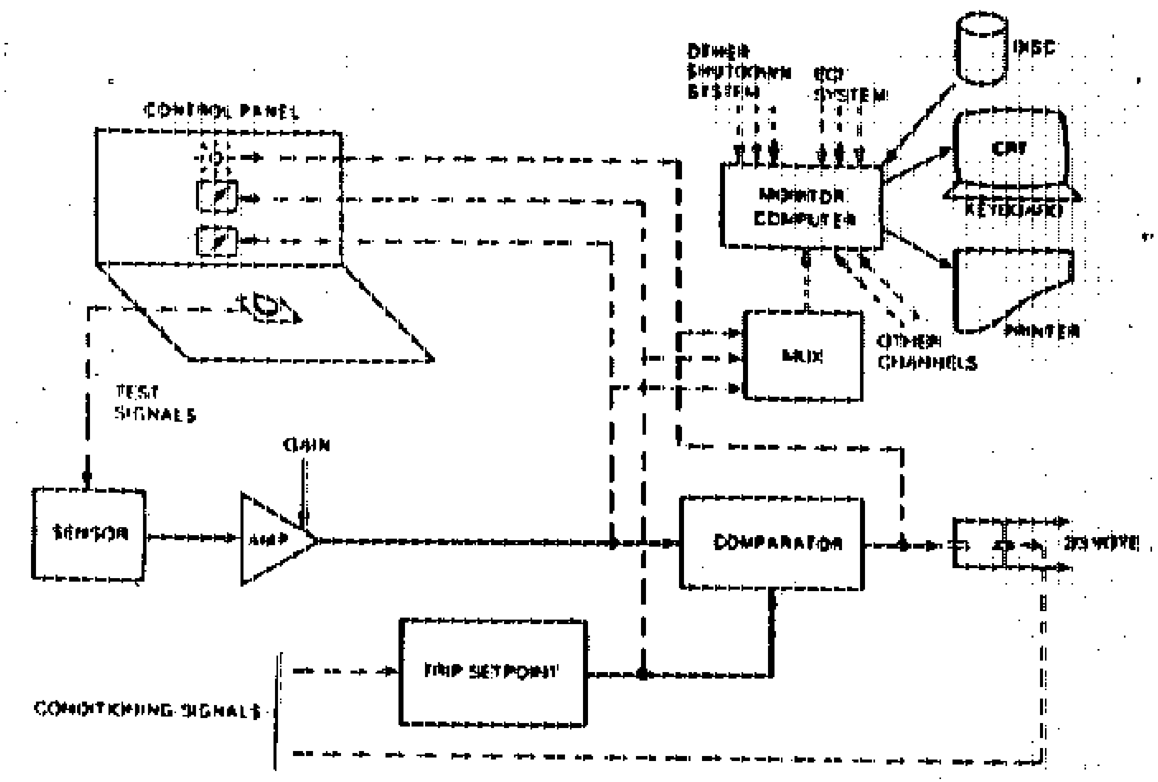


Figure #2 – Traditional Analog Shutdown System with a Computerized Monitoring Interface

Programmable Digital Comparators (PDC's) - second evolution

- The next step taken in the evolution of the trip channel implementation was to digitize the process ***trip comparison*** and ***logic circuitry*** by using digital ***microcomputers*** or programmable digital comparators (**PDC's**) in place of the analog devices.

- The shutdown system instrumentation still consists of the analog process and neutronic ***sensors*** and ***man-machine interfacing*** devices as well as ***cabling*** and ***interfacing relays*** which remained fundamentally unchanged from the traditional design. The ***neutronic trips*** were still implemented by the traditional analog circuits.

- But the ***process signal comparisons***, ***decision making***, and ***trip initiation*** were now moved to a ***programmable logic base***.

- Note that this method of design change is ***quite conservative*** so that the condition of the final configuration is ***always known***. As well, an additional diversity was provided in that the ***neutronic trips*** were provided by ***analog logic*** while the ***process trips*** were initiated by ***digital logic*** - a gradual design progress was accomplished.

- The PDC's are provided with a ***simple hardware timer***-like device called a ***watchdog*** timer. The watchdog monitors the ***performance of the PDC*** to ensure that expected operations are performed within an ***expected time interval*** to avoid such traps as an infinite loop execution or a stalled sequence.

Programmable Digital Comparators (PDC's) ……continued

- If the conditions for the ***watchdog timer are not satisfied***, that trip channel is ***de-energized*** independent of the trip parameter conditions (i.e the PDC has failed-safe)

- The logic in the PDC's could now be programmed to implement ***conditioning logic***, ***signal spread checks***, ***rationality checks*** and ***calculate power dependent trip setpoints***.

- In addition to outputting the trip channel signals, the PDC drives analog and digital outputs which ***drive conventional indicating devices*** on the main control room panels.

- Correct operation of the PDC analog and digital outputs can be ***dynamically verified*** by wiring the ***outputs*** (analog & digital) back to ***special inputs*** (analog & digital).

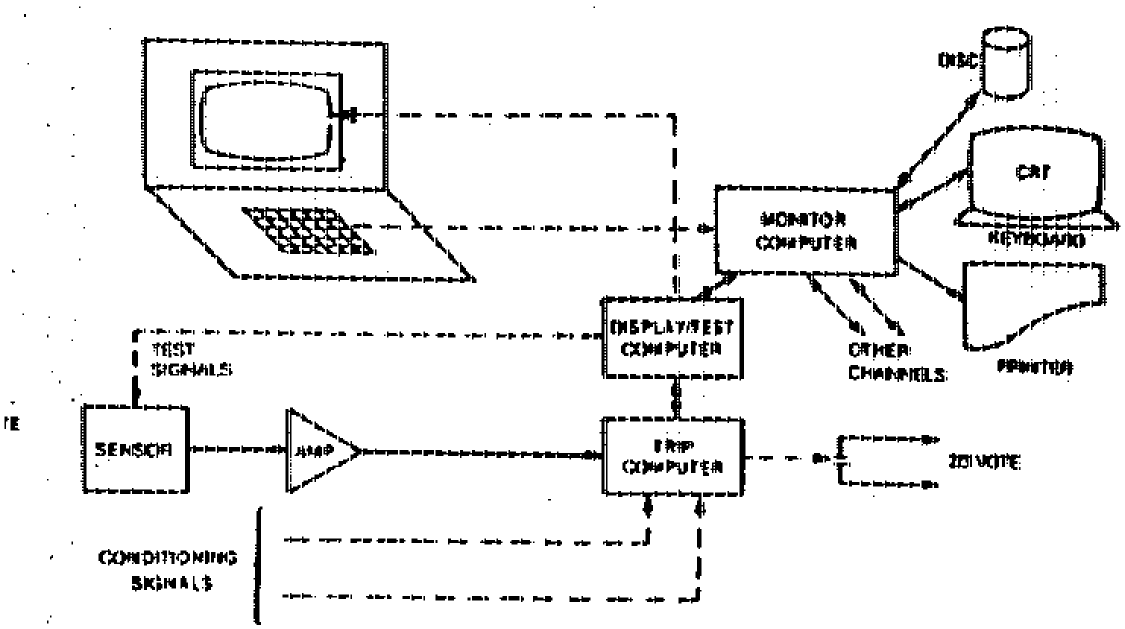
- Periodic programs can then be executed to ***test drive the outputs*** and ***read back*** the corresponding field value developed by the output system. If significant discrepancies are recognized on the read-back, then an appropriate alarm can be annunciated to prompt operator or maintainer intervention.

Fully Computerized Shutdown System - third evolution

- The fully computerized shutdown system is a combination and extension of the PDC trip computers and the Monitoring computers strategies. The four special safety system functions (i.e. ***trip logic***, ***testing***, ***monitoring*** & ***display***) are implemented in a fully computerized design.

- The shutdown system parameter sensing instrumentation still consists of the analog process and neutronic sensors, but ***all*** of the ***comparison*** and ***trip logic***, ***testing*** and ***monitoring*** functions along with the ***man-machine interfacing*** is now ***computer based.***

- The two shutdown systems (SDS1 and SDS2) have a ***similar computer configuration*** but each system is designed by a ***separate team of designers*** as one measure to ensure functional independence and each design team ***specifies equipment from a different manufacturer***.



<u>Figure#4 – Fully Computerized Shutdown System for Process and Neutronic Trip Parameters</u>

Fully Computerized Shutdown System….continued

- ***Fifteen computers*** (15) , organized into a ***three level hierarchy***, are used in the total computerized shutdown system design for each reactor. The three hierarchy level functions are ***trip***, ***display/test*** and ***monitoring***

- There are channelized ***trip*** and ***display/test computers*** for each safety system channel.

- There are ***seven*** (7) SDS1 related computers.There are ***three SDS1 trip*** computers (i.e. Channels D,E & F) ***and three SDS1 Display/Test*** (i.e. D/T) computers. A ***SDS1 monitoring*** computer is also provided to track and assess the trip parameters for this system by interfacing with the D/T computers so that seven computers are dedicated for SDS1.

- Similarly, there are ***seven*** (7) SDS2 related computers.There are ***three SDS2 trip*** computers (i.e. Channels G,H & J) and ***three SDS2 Display/Test*** (i.e. D/T) computers. A ***SDS2 monitoring*** computer is also provided to track and assess the trip parameters for this system by interfacing with the D/T computers so that seven computers are also dedicated for SDS2.

- Finally, a unit ***monitoring*** computer (common to SDS1 & SDS2) is provided to coordinate the system performance and testing data as well as administrative information from both the SDS1 and SDS2 monitoring computers.

- The data links to the unit monitoring computer are ***uni-directional*** from the individual safety system monitoring computer and are interlocked to allow data transmission from ***only one safety system at a time***.

<u>Fully Computerized Shutdown System….continued</u>

- This results in 7 SDS1, 7 SDS2 and 1 unit computer for a total of 15 safety system computers.

- This configuration provides a central interface between the operator and the special safety systems while preserving the ***required separation*** between different safety systems and among different channels of the same system by using uni-directional fibre-optic links and hardware interlocks. External hardware interlocks on both inputs and outputs ensures that ***only one channel can be tested*** or calibrated at a time.

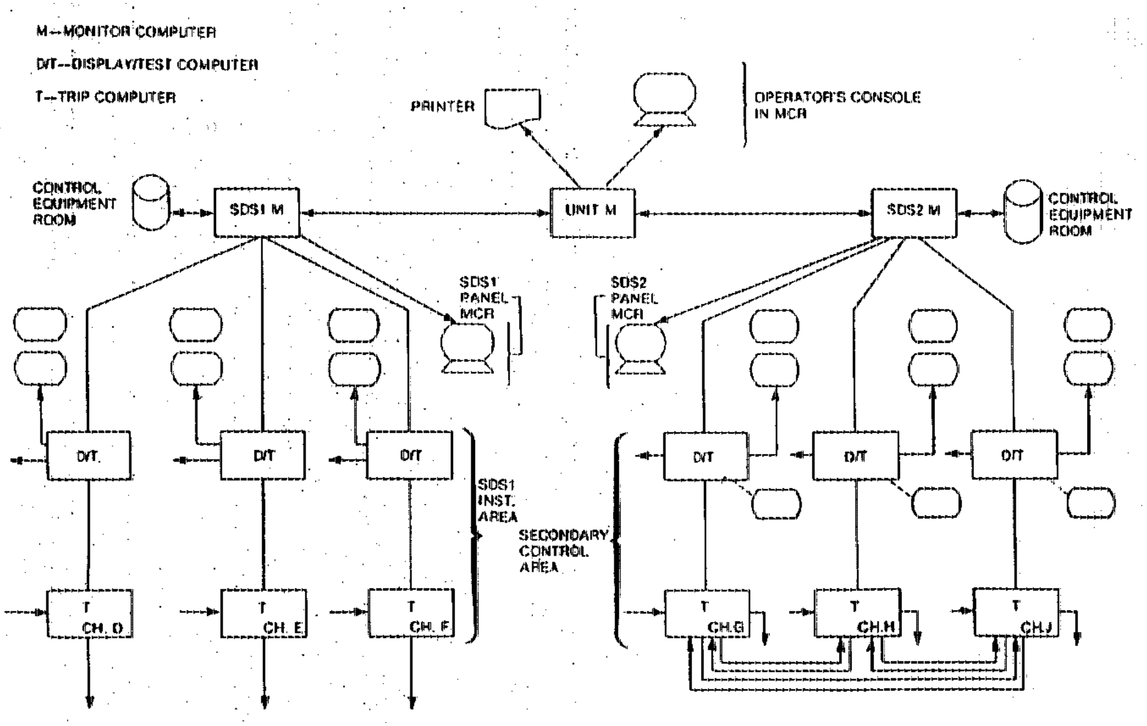


Figure #5 – Configuration of Trip, Display/Test and Monitoring Computers for SDS1 & SDS2

CANDU NPP S/W Categorization Methodology for Safety, Plant Control, Monitoring & Testing Systems

This lecture will provide some insight into the ***S/W categorization*** methodology that should be taken into consideration ***for safety critical software*** for computerized shutdown system applications.

Lecture topic summary includes:

- Purpose of the S/W Categorization Process
- Necessary Definitions
- Specification of the S/W Categories
- Categorization Process Basis
- Determining Plant System Safety Significance
- Determining S/W Failure Impact Type
- Final S/W Category Determination
- S/W Categorization 1-4 Summary

Purpose of the S/W Categorization Process

- ***Minimize any unnecessary reliance on S/W*** or S/W controlled systems for nuclear safety
- Ensure that S/W necessary for nuclear safety is ***clearly identified, understood*** and ***achieved***
- Select the S/W Engineering practices which ***assures the reliability*** and ***safety*** of the S/W
- Categorize S/W with respect to its ***failure effect on nuclear safety***

S/W Categorization Process - Necessary Definitions

- ***Safety-Related System*** - A plant system that, upon failure, has the potential to impact the ***radiological safety*** of the public or plant personnel due to the operation of the nuclear power plant (NPP).
- ***Nuclear Safety-Related Functional Requirements*** - those functional requirements which ensure that the system will fulfill its role in ***achieving acceptable levels of radiological safety*** with respect to the public and plant personnel
- ***Process System*** - a safety-related plant system whose role is to contribute, directly or indirectly to the ***production of electricity***.
- ***Initiating Event*** - a malfunction of a plant system that would, ***in the absence of Special Safety System actions***, lead to a release of radioactivity which could result in doses exceeding the most restrictive regulatory dose limit for that station.
- ***Mitigating System*** - a safety-related system that has nuclear safety-related functional requirements ***to reduce the consequences of an initiating event***.
- ***Safety System Significance*** - a classification (into ***High***, ***Medium*** or ***Low***) of the plant system in terms of its ***importance to nuclear safety***
- ***Software Failure Impact*** - the impact of the failure of that S/W ***with respect to the nuclear safety related functional requirements*** of the host plant system
- ***Minimum Performance Requirements*** - the minimum amount of equipment, and the minimum functional and performance characteristics of that equipment, ***necessary to achieve the plant system performance*** specified in the safety analysis completed in support of that station operating license.

<u>Specification of the S/W Categories</u>

- The Software Category - is represented by a ***number*** from ***1*** to ***4***
- ***Category 1*** S/W considered the **most important** to Nuclear Safety
- ***<u>S/W Nuclear Safety Category 1</u>*** - is also referred to as ***Safety Critical Software***
- ***Failure of Safety Critical S/W*** can result in a system with a high safety-related reliability requirement ***not meeting its minimum performance requirements*** or can result in a ***serious initiating event*** (low frequency limit) in a process system.
- ***Category 4*** S/W must have ***no importance*** to Nuclear Safety
- ***<u>S/W Nuclear Safety Category 2</u>*** - Failure of this S/W can result in a ***serious process failure***, or a ***degradation in the performance of a mitigating system***.
- There is a distinct ***reduction in safety significance*** from Category 1 S/W since the ***consequences of the Category 2 S/W failure*** can still be ***mitigated*** by special safety system action.

<u>S/W Nuclear Safety Category 3</u> - Failure of this S/W ***does not prevent*** the affected plant system from ***meeting its Nuclear Safety-related design intent*** or the affected plant system has a ***low safety significance***.

Categorization Process Basis

Fundamental Basis - As the ***Safety Significance of the S/W decreases***, ***less effort is required*** to be expended to demonstrate that the S/W meets its requirements.

Risk Based Approach to Nuclear Safety - The risk associated with the failure of a system to perform is a function of the ***Probability of the failure*** and the ***Consequences of the failure*** (the ***higher the risk*** associated with a failure, ***the higher the assurance*** must be that the S/W will not contribute to that failure).

Acceptable levels of plant risk are achieved by - Designing the plant to have a ***low probability of serious process failures***, and by ***providing redundant mitigating systems*** that minimize the consequences of serious process failures, should they occur.

Two Phases for the Categorization Process

- **Phase I**; Determine the System's ***Safety Significance***
- **Phase II**; Determine the ***S/W Failure Impact***

Two Phases for the Categorization Process

- **Phase I**; Determine the ***System's Safety Significance*** - This involves identifying the safety significance (as ***High***, ***Medium*** or ***Low***) for the plant system of which the S/W to be categorized is a part.

- ***Safety Significance Determination*** - The safety significance is obtained by determining the ***system type*** (safety-related, mitigating, process, etc) and ***quantifying*** the ***systems reliability*** requirements

- It is also important to note that ***more stringent reliability requirements*** due to factors other than nuclear safety may be used (i.e. ***engineering judgment***, experience, etc.) to justify selecting a ***more restrictive*** S/W category

- **Phase II*; Determine the S/W Failure Impact*** - This involves ***identifying*** and ***classifying*** the ***worst possible S/W failure modes*** and effects in terms of ***impairment of plant safety functions***.

- **Failure Impact Types** - The Failure Impact Type is identified as ***Type I***, ***II*** or **III** with ***Type I*** representing a failure with the ***greatest consequences*** with respect to Nuclear Safety.

Failure Impact Type Analysis - The determination of the failure impact type is based on an analysis of the **role of the S/W** with respect to the ***safety-related function*** of the system and on the ***independent mitigating provisions*** within the plant system which can mitigate the consequences of the S/W failure.

Failure Impact Type Considerations

- ***Assess all Failure Impacts*** - Within a plant system there can be sub-systems that perform ***multiple functions***.
- The Failure Impact Assessment must ***identify all possible safety-related impacts*** of S/W failure on a plant system.
- For the purposes of categorization, ***the most severe S/W failure impact type should be used***.
- ***Failure Impact*** - If the ***worst-case*** S/W failure is ***not Type I***, then it is possible to ***reduce the stringency*** of the S/W category because **the** ***role of the S/W within the plant system is less significant from a safety perspective than the role of the overall plant system***.

Definition of Plant Safety Significance values for Phase I use

- ***Safety & Mitigating Systems Safety Significance***

High Significance: **Q .LE. 10^{-3} yr/yr** (Q= unavailability req'mt)

Medium Significance: **10^{-3} .LT. Q .LT. 10^{-1} yr/yr**

Low Significance: **Q .GE. 10^{-1} yr/yr**

- ***Process Systems Safety Significance - Process System Failures***

High Significance: **f .LE. 10^{-3} occ/yr** (f= event frequency limit)

Medium Significance: **10^{-3} .LT. f .LE. 10^{-2} occ/yr**

Low Significance: **f .GT. 10^{-2} occ/yr**

- ***Monitoring/Testing Systems Safety Significance***

High Significance: **Q .LE. 10^{-3} yr/yr** (Q= unavailability req'mt)

Medium Significance: **10^{-3} .LT. Q .LT. 10^{-1} yr/yr**

Low Significance: **Q .GE. 10^{-1} yr/yr**

Four Steps to follow for Phase I - Determining Plant Safety Significance

- ***1. Identify the Plant System*** or Systems Involved -***Determine which plant systems the S/W*** or S/W controlled systems ***are a part of*** or ***interact with*** and determining ***the role of the S/W***.

- ***2. Determine the Plant System Type*** - Identify ***each role*** for the plant system's ***nuclear safety functions*** and determine if it is a ***special safety***, ***mitigating***, ***process*** or ***monitoring/testing*** system

- ***3. Establish a Suitable Plant System Boundary*** -Selection of the ***boundary*** influenced by data availability by using either ***system unavailability requirements*** or the ***initiating event frequency limit***

- 4. Determine the Plant System ***Safety Significance -High***, ***Medium*** or ***Low***

<u>Four Steps for Phase II - Determining S/W Failure Impact Type</u>

- ***5. Identify All S/W Failure Modes & Effects*** -Assess the interactions of the relevant sub-systems that comprise the plant system to determine the ***possible failure impacts of the S/W*** for all conceivable failure modes.

- ***Credit S/W and Computer System Design Attributes*** - This is ***an optional additional step*** to consider the possibility of S/W and computer system ***design attributes for preventing or minimizing specific failure modes***.

- ***6. Determine the limiting S/W Failure Impact Type*** - Apply the classification criteria to determine ***Type 1-3 Failure Impact Type***

- ***7. Determine the S/W Category -***Use the determined plant system S/W failure ***Safety Significance (***Column data) and the **S/W** ***Failure Impact Type (***Row Data) to find the S/W Category Matrix intersection value which is the ***S/W Category*** value.

- ***8. Determine the limiting S/W Category*** - This step is necessary when the application involves ***more than one system*** or can be ***classified as more than one type***. The ***most restrictive*** category should be used.

- Note that S/W categorization ***can be Iterative*** - An initial category may be determined and then further analysis may be initiated in order to resolve any issues which arise during the design process

The *Criteria* for *S/W Failure Impact for Safety* or Mitigating Systems

- ***Type I*** - The designed ***nuclear safety functions will not be available*** or the **minimum performance requirements of the plant system will not be met** for some or all process system failures.

- ***Type II*** - The system's functional performance is ***degraded*** for some or all ***process system failures*** but the ***minimum performance requirements of the plant system will be met***. Or the system's ***redundancy is reduced*** such that the ***probability*** of not meeting the minimum performance is ***increased***

- ***Type III*** - The S/W failure has ***no impact*** on the nuclear safety functions of the plant system

The Criteria for S/W Failure Impact Type for Process Systems

- ***Type I*** - The S/W failure can***, in the absence of safety or mitigating system actions***, directly or indirectly ***cause systematic fuel failures*** or ***release of radioactivity*** which ***could result in doses exceeding*** the most restrictive ***regulatory dose limit*** for the station

- ***Type II*** - The S/W failure can directly or indirectly ***raise the temperature of the fuel*** but ***not lead to systematic fuel failures***. Or the S/W failure leads to an ***increase in probability*** of the ***Type I consequences*** of systematic fuel failure or releases (probability)

- ***Type III*** - The S/W failure has ***no impact*** on ***the nuclear safety-related reliability performance*** of the nuclear safety functions of the plant system

The Criteria for S/W Failure Impact Type for Testing Systems

- **Type I** - The S/W failure can cause the ***designed nuclear safety functions*** under test ***not to be available*** or **the *minimum performance requirements of the plant system under test not to be met*** for some or all process system failures

- **Type II** - The S/W failure causes an ***inaccurate test result*** or ***degrades the functional performance*** of the system under test or causes a ***redundancy reduction*** in the system under test but the ***minimum performance requirements*** of the plant system ***will still be met*** for all process system failures.

- **Type III** - The S/W failure has ***no impact*** on the ***test*** or on the ***safety-related performance of the system*** under test.

S/W Categorization tabled as a function of Safety Significance & Failure Impact Type

System Safety Significance	Impact Type I	Impact Type II	Impact Type III
High	Cat. 1	Cat. 2	Cat. 4
Medium	Cat. 2	Cat. 3	Cat. 4
Low	Cat. 3	Cat. 3	Cat. 4

S/W Categorization 1-4 Process Summary

- Safety Critical S/W - The standard for Safety Critical S/W **will *only be applied to Special Safety Systems*** and to ***Process Systems*** for which the initiating event frequency is ***less than 10^{-3} occ/yr*.**

- Reduction in S/W Rigour from ***Category 1 to 2 Occurs***: If the system ***Safety Significance*** decreases from ***High*** to ***Medium*** while the S/W Failure Impact remains Type I, or if the Safety Significance remains High but the ***Failure Impact Type*** is reduced from ***Type I*** to ***Type II.***

- ***Category III Application*** - If the system ***Safety Significance*** is ***Low*** and the ***S/W Failure Impact Type is I*** or the system has a ***Medium or Low Safety Significance*** and the ***Failure Impact Type is II***.

- ***Category IV Application***: - If the ***S/W Failure Impact Type is III*** (no Safety- Related Impact) the S/W is assessed as Category IV.