Pressurised water reactor cold shutdown transient analysis

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ABSTRACT

The term “shutdown” as it applies to Nuclear Power Plants cannot be construed as meaning either dormant or inherently safe. Once a nuclear reactor is brought to the cold shutdown condition there are still decay process going on inside the reactor with the accompanying generation of heat in the megawatt range. There is a vast body of literature available investigating the occurrence of transients in reactors at varying levels of power and even hot standby, but comparatively little is available investigating transients in the cold shutdown condition.

The objective of this investigation was to simulate the insertion of selected transients into a Westinghouse 3 loop pressurised water reactor – CPR 1000 - and carry out an analysis of said transient insertions in terms of the behaviour of the reactor. The transients that were analysed were:

- Complete Residual Heat Removal failure without replenishment of Reactor Coolant System inventory
- Loss of coolant accident (Hot and cold legs)
- Inadvertent control rod withdrawal
- Moderator dilution
- Fuel clad failure

The analysis revealed that by far the most severe repercussions – in fact meltdown – came about as a result of Residual Heat Removal failure without Reactor Coolant System replenishment. If steps are not taken to re-establish Residual Heat Removal in 5 hours and 10 minutes, the situation rapidly deteriorates to the point where meltdown occurs a further 2 hours and 29 minutes later. Large break Loss of Coolant Accidents were easily handled by the Residual Heat Removal System and held no real danger. Inadvertent rod withdrawal, moderator dilution and fuel clad failure either had to have key variables shifted so far away from the cold shutdown condition or the extent of the transient raised far beyond design based accident conditions, or both, yet did not hold any real threat as the available systems kept the reactor safe.

It is recommended that selected combinations of transients are injected into the reactor and analysed and that Residual Heat Removal failure without replenishment is re-simulated with commercial code such as CORYs or MAPPS and analysed further.

Keywords: Pressurised Water Reactor. Cold Shutdown. Transient analysis. CPR1000. Complete residual heat removal failure without replenishment. Loss of coolant accident. Inadvertent control rod withdrawal. Moderator dilution. Fuel clad failure. Beyond design based accident conditions.

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LIST of DEFINITIONS, ABBREVIATIONS and ACRONYMS

The following is a list of all the definitions, abbreviations and acronyms used in this document in alphabetical order, for quick reference:

Definitions:

Cold Shutdown[1] – A Reactor state which is characterised by the Reactor coolant system (RCS) being depressurised to atmospheric pressure of 101.325 kPa and being at a temperature of 93.3 °c or below.

Fuel temperature coefficient of reactivity – It is the change in reactivity of the nuclear fuel per degree change in the fuel temperature.

Mid – loop – An infrequently used refuelling outage procedure in which, after shutdown and a cooling period, reactor coolant is lowered below the hot and cold leg shell piercings, permitting work to be performed in a relatively dry environment.

Moderator temperature coefficient of reactivity – It is the change in reactivity per degree change in moderator (water) temperature.

Probabilistic Risk Assessment – A systematic and comprehensive methodology to evaluate risks associated with complex technological entities.

SCRAM (Shutdown Control Rod Axe Man) – Emergency shutdown of a nuclear reactor where the rod control clusters are driven into the reactor to shut it down.

Soluble Boron reactivity coefficient – It is the change in reactivity as a result of the increase of soluble Boron or the decrease of soluble Boron in the reactor coolant system.

Steady State – A system in which the conditions at each point are not changing.

Transient – Non-steady condition encountered by a system when going from one state of equilibrium to another.

Abbreviations:

AC	–	Alternating current
ASP	–	Accident Sequence Precursor
BDSA	–	Beyond Design Basis Accident
BNL	–	Brookhaven National Laboratories
BWR	–	Boiling Water Reactor

CDF	–	Core Damage Frequency
CPR 1000	–	The simulations are based on this reactor operating in China with an output of 1080MWe
DG	–	Diesel Generator
DNBR	–	Departure from nucleate boiling
ECCS	–	Emergency Core Cooling System
FSAR	–	Final Safety Assessment Report
GUI	–	Graphic user interface
LOCA	–	Loss of Coolant Accident
NNR	–	National Nuclear Regulator (South Africa)
NPP	–	Nuclear Power Plant
NSSS	–	Nuclear Steam Supply System
NUREG	–	The Nuclear Regulatory Commission (United States) publish the Nuclear Regulations as and when required in the regulation of the nuclear environment in the United States
OECD	–	Organisation for economic co-operation and development
PORV	–	Power Operated Relief Valves
POS	–	Plant Operating State
ppm	–	Parts per million
PRA	–	Probabilistic Risk Assessment
PWR	–	Pressurised Water Reactor
RCC	–	Rod Control Cluster
RCP	–	Reactor Coolant Pump
RCS	–	Reactor Coolant System
RHR	–	Residual Heat Removal
RHRV	–	Residual Heat Removal Valve
RHRS	–	Residual Heat Removal System
RPV	–	Reactor Pressure Vessel
SAMG	–	Severe Accident Management Guidelines
SDC	–	Shutdown cooling
SG	–	Steam Generator
STS	–	Standard Technical Specifications

USNRC – United States Nuclear Regulatory Commission

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CHAPTER 1: INTRODUCTION

1.1 Introduction:

"Shutdown" is inherently safe, or is it? The term shutdown is a generic term used across many different industries and sectors to describe a condition where whatever has been shut down is no longer in an operational state. In the case of a nuclear reactor the fact that the plant may be shut down – or cold shutdown – in this case, most certainly does not mean that it is inherently safe. The fuel is still present, decay heat is still being released and as such need to be removed. Refer to table 1 below outlining the various modes the reactor can be in, the applicable mode being number 5.

Table 1.1 Pressurised Water Reactor modes.[2]

Mode	Title	Reactivity Condition (K_{eff})	% Rated Thermal Power(a)	Average Reactor Coolant Temperature °c
1	Power Operation	≥0.99	>5	NA
2	Startup	≥0.99	≤5	NA
3	Hot Standby	<0.99	NA	≥176.6
4	Hot Shutdown(b)	<0.99	NA	$176.6 \geq T_{avg} \geq 93.3$
5	Cold Shutdown(b)	<0.99	NA	≤93.3
6	Refuelling(c)	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts not fully tensioned.

All pressurised water reactors (PWR's) require periods of reactor shutdown for refuelling and maintenance, also referred to as planned outages. Unplanned outages are typically the result of mechanical breakdown or deviation from operating parameters. While there is more than one type of shutdown state, this dissertation will focus on what is referred to as the "Cold Shutdown" condition. This condition which is characterised by the Reactor coolant system (RCS) being depressurised to atmospheric pressure of 101.325 kPa and a temperature of 93.3°c[1] will be the starting point for carrying out malfunction simulations on a CPR1000 pressurised water reactor.

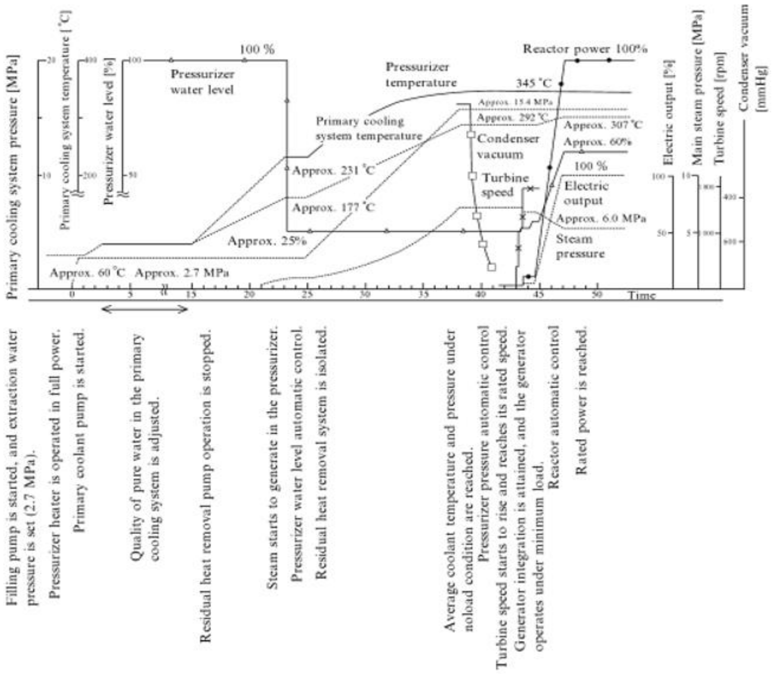


Figure 1.1 Graphical representation of starting a reactor. Reverse for cold shutdown

The simulation software of choice in carrying out the stated objective was PCTran PWR 3LP version 6.0.1 licensed to North West University by Micro-Simulation Technology. While PCTran comes with a choice of 20 malfunctions, most of which can be applied as singular events or combinations thereof, the initial challenge lies in getting to the correct initial condition. PCTran comes with 22 initial conditions already preloaded but none of them are representative of a cold shutdown condition.

As a result the correct system variables have to be established that adequately represents the conditions, not only in the reactor core, but also the various auxiliaries which adheres to the definition of a cold shutdown state. Once that is achieved it can then be saved as an initial condition.

With the transient start established, the various transients can be applied singularly or in combination with one another during the shutdown analysis. Not all malfunctions will be applied as some will have little to no repercussions in terms of the safety or stability of the plant.

1.2 Problem Statement:

Comparatively little research is available related to transients being injected into a 3 loop Westinghouse PWR in a cold shutdown condition, when compared to the body of work investigating the injection of transients into pressurised water reactors that are at varying power levels, or even in hot standby or hot shutdown conditions. A probable reason for this is due to the fact that most experience in relation to NPP behaviour under the influence of transients, stems from actual operations over time, and the most severe ramifications have been witnessed in PWRs during power operations or in a hot condition. This however does not preclude the possibility of severe corollaries, in the case of carefully selected transients in reactors in a cold shutdown condition.

1.3 Methodology:

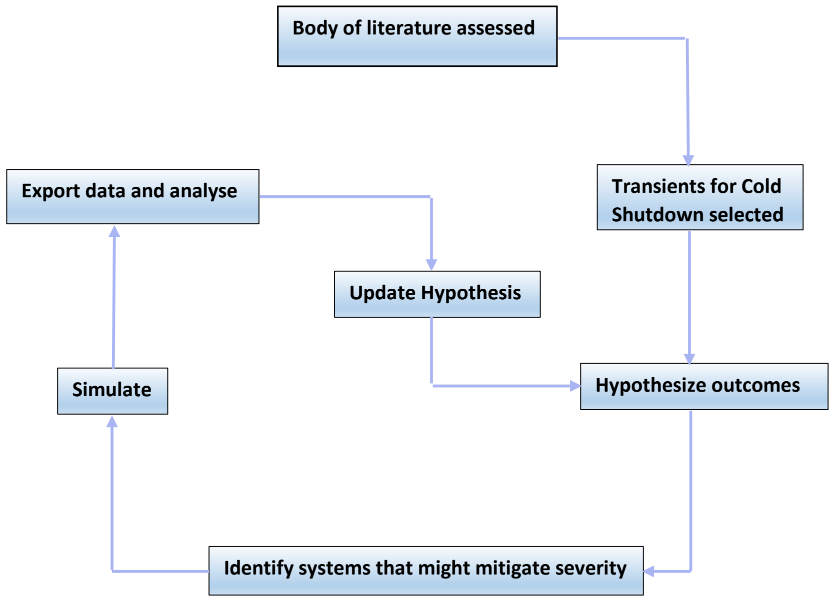
The methodology employed in analysing the problem is stated as the following: Literature informs the choice of transients to be selected for insertion into the cold shutdown pressurised water reactor. The hypothesis follows from this with the likely outcomes of the selected transients, including identifying systems that might mitigate the severity of these outcomes.

Following on this, the simulation is carried out and the planned transients are started. The data is then exported for manipulation and analysis, following which observations are made in comparison to the original hypothesis. The latter were then explained and adapted as the analysis supported it.

1.4 Objectives:

While comparatively little is known about the behaviour of Pressurised water Reactors subjected to the insertion of transients vis a vis reactors during power operations and other shutdown conditions, this dissertation aims to add to the body of knowledge during the stated condition. Thus, the objective is to analyse the behaviour of a 3 Loop Westinghouse Pressurised Water reactor – in a cold shutdown condition - subjected to specific chosen transients.

Flowchart 1.1 Process flow utilised in carrying out the analysis



1.4 Scope

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The scope of this body of work will within the broader frame work, be limited to the cold shutdown condition as defined. Thus no transients will be investigated as they may occur – irrespective of likelihood – during power operations or during hot shutdown conditions. Nor will transients be investigated during cold start-up, hot start up or hot standby.

The scope of this dissertation will not be limited purely to the available transients – or combinations thereof - as presented in PCTran but will also investigate other occurrences not pre-loaded into PCTran. The latter will however only be assessed based on the likelihood of occurring, coupled to the expected severity of such an event occurring.

1.5 Limitations

Although “cold shutdown” is defined[1] as establishing atmospheric pressure and a moderator temperature smaller than 93.3°c in the primary system, notwithstanding all attempts, the PCTran model of the CPR1000 could be brought to atmospheric pressure but the moderator temperature was 7.09°c above the defined limit. In operating PWR’s the last step in the establishment of a cold shutdown condition is the replacement of the steam bubble by Nitrogen. The PCTran model of the CPR1000 does not have the ability to do that.

CHAPTER2: LITERATURE REVIEW

2.1 Transients – Historical Perspective

In September 1993 the United States Nuclear Regulatory Commission published a report titled "Shutdown and Low Power operation at commercial Nuclear power plants in the United States" or NUREG 1449[3]. Various parties used the "Accident Sequence Precursor" method (ASP) to evaluate 10 samples of actual shutdown events that could be significant. At the time of publication of said report, the ASP method mostly concerned itself with operational events that occurred at power or during hot shutdown.

The analytical approach used during power and hot shutdown ASP application, was also applied to the cold shutdown condition. However, accident sequence models describing failure combinations had to be developed with little previous work to base it on.

The intent of the analysis was to gain greater understanding as to the types of events that have occurred during cold shutdown, and to determine which characteristics of these events were important in terms of risk. Table 1.1 contains a list of the actual events that occurred, their location, and via the application of the ASP program, quantified their probability of causing core damage.

Table 1.2[4] List of actual NPP transient events and their core damage probabilities

Location	Date	Description	Core damage probability
Vermont Yankee	3/9/1989	39.7kL of RCS inventory was transferred to the torus when maintenance stroked-tested the SDC valves in the but-of-service loop of the RHR with the minimum flow valve already open	$1x10^{-6}$
Fort Calhoun	2/26/1990	Loss of offsite power with the emergency diesel generators not immediately available. Breaker failure relay operated to strip loads, but EDG design feature prevented auto loading	$4x10^{-4}$
Oconee 3	9/11/1988	Loss of AC power and loss of RHR during mid-loop operation with vessel head on. Testing errors caused a loss of power to feeder buses resulting in loss of SDC with no reactor temperature or level indication	$2x10^{-6}$
Location	**Date**	**Description**	**Core damage probability**

Crystal River 3	2/2/1986	RHR pump shaft broke during mid-loop operation. Pump had been in continuous operation for 30 days. Tripped circuit breaker delayed getting second train online.	$1x10^{-6}$
Diablo Canyon 2	4/10/1987	Loss of RHR during mid-loop operation. Loss of RCS inventory and air entrainment in both RHR pumps causing loss of SDC. Extended boiling occurred	$5x10^{-5}$
Waterford 3	7/14/1986	Loss of RHR during mid-loop operation. Complications in restoring RHR due to steam binding and RHR suction line design. Extended boiling occurred	$2x10^{-4}$
Susquehanna 1	2/3/1990	Electrical fault caused isolation of SDC suction supply to RHR system. Alternate RHR provided using suppression pool	$3x10^{-5}$
WNP - 2	5/1/1988	RHR suppression pool suction and SDC suction valves were open simultaneously and 39.7kL of reactor water was transferred to the suppression pool	$5x10^{-5}$
Braidwood 1	12/1/1989	RHR suction valve stuck open and drained 254kL of water from the RCS	$1x10^{-6}$
River Bend	4/19/1989	57.8kL of service water flooded the auxiliary building when a freeze seal failed. One RHR train, normal spent fuel cooling and auxiliary and reactor building lighting lost	$1x10^{-6}$

From the list of preceding actual events, Vermont Yankee, Susquehanna 1, WNP – 2 and River bend will not be further analysed as they occurred in BWR reactors, which does not form part of the scope of this dissertation. For the remainder of the events, occurrences with conditional probabilities below $1x10^{-4}$ are considered minor in relation to causing core damage. Hence, the two most serious events were Fort Calhoun and Waterford 3. It is worthwhile to look at these events in more detail:

2.1.1 Fort Calhoun - 26 February 1990

During a refuelling outage a spurious relay actuation resulted in the isolation of offsite power supply to the facility. One diesel generator (DG) was out of service for maintenance reasons, a second DG started, but failed to connect to its engineered safety features bus by a shutdown cooling pump interlock. Operators identified and rectified the problem and the DG was brought online and restored power to the plant. Importantly, the dominant sequence here was the failure to recover AC power.

2.1.2 Waterford 3 – 14 July 1986

In this instance, a non-proceduralised drain path was not isolated when the RCS level was reduced to mid-loop. Draining continued undetected and resulted in the cavitation of the working RHR pump. Re-establishment of shutdown cooling (SDC) took 3 hours during which time boiling occurred in the core region. Both RHR pump suction lines were steam bound (likely as a result of the suction loop seal design feature). RCS inventory was restored using a low pressure safety injection pump (LPSI) and taking suction from the refuelling water storage pool. Shutdown cooling was eventually restored using the pump warm up lines in collaboration with repeated pump jogging. The latter being a non-proceduralised action.

The procedural method for restoration of RHR pump suction would in any event not have worked as the hot leg temperature exceeded 100°c. The dominant core damage sequence for this event included an assumed failure to recover RHR and an assumed unavailability of the steam generators as an alternative means of removing decay heat.

2.2 Operational issues important to risk during shutdown

Plant operators and their differences in behaviour in detecting and handling an event are apparent in many of the 10 events previously listed. Several events were actually ongoing for some time before someone realised that it was actually occurring, and initiated the necessary corrective action. As an example in the case of the Braidwood event, operators quickly realised that an RHR suction relief valve had lifted, but it took a further 2.5 hours for the valve to be located. In the latter event SDC was not lost, and as a result operators had ample time to systematically handle each event. Hence the availability of a long time period prior to boiling lowers the probability for recovery of a faulted system.

By contrast, the Waterford event – where SDC was lost during a mid-loop event – boiling started 45 minutes after losing SDC. It is a very short period of time to institute reliable recovery actions from the control room.

2.3 Factors that strongly influence the significance of an event

Nureg 1449 found the following factors affecting the significance of an event based on the 10 events listed above:[5]

2.3.1 High decay heat load

A high decay heat load significantly reduces the time available for SDC recovery prior to boiling occurring. This increases the risk of failing to recover SDC or implementing alternative cooling strategies. At high decay heat load the number of systems that could be used to recover SDC is also lower than at low decay heat load.

2.3.2 RCS inventory

Having the refuelling cavity filled with water to a level as high as possible with the upper equipment removed, increases the time available for SDC recovery and operator action. The converse is patently true, during mid-loop operation the RCS inventory is at a minimum and hence the reliability of RHR is reduced.

2.3.3 Reactor vessel head position

In general, transients which occurs with the vessel head removed is less severe than with the vessel head in place. The reason being that RPV make-up and core region boiling will provide RHR.

2.3.4 SDC – Diverse system availability

Systems that might be available that can operate independently of the components of the RHR, reduces risk as it is not RHR recovery dependant.

2.4 Probabilistic Risk Assessments

In general, risks associated with shutdown and refuelling are poorly understood when compared to those associated with power operations. However, there were some studies conducted, and here specifically it is worthwhile to look at the findings of NUREG/CR-5015[6] which looked at the loss of RHR in PWR's during cold shutdown. It was found that the CDF (core damage frequency) was $5.2\text{x}10^{-5}$ with the breakdown by initiating event being:

- Loss of RHR 82%
- Loss of offsite power 10%
- Loss-of-coolant accident 8%

The failure of the operator to diagnose loss of cooling while at reduced inventory, and the failure to restore it was the cause of 64% of the CDF cases.

2.4.1 Seabrook Probabilistic risk assessment

The study at Seabrook is of particular significance as it is also a pressurized water reactor and among others, it specifically looked at the likelihood of core damage during cold shutdown (mode 5). Interestingly the shutdown CDF was found to be 4.5×10^{-5} per reactor year while the power operation CDF was 1.1×10^{-4} per reactor year, or a factor of 2.4 times higher than during shutdown operations. It was found that 82% of CDF was due to loss of RHR initiators, and 71% of CDF was in the condition where the RCS was vented and partially drained. RHR failure was mainly due to:

- Hardware failure of an operating RHR pump due to long duty cycles
- Loss of RHR suction due to inadvertent closure of RHR suction valves
- Low level cavitation when the RCS was drained

The latter two events were caused by operator error.

2.4.2 Sequoyah: LOCA in cold shutdown

In this study a core melt accident during cold shutdown was initiated as a result of a postulated LOCA. Two LOCA initiating events were considered:

- Safe shutdown earthquake
- Operator error

It is important to note that RHR induced LOCA's were not considered. It was found that the core melt frequency fell in the range 7.53×10^{-5} to 8.5×10^{-7} per reactor year. The major contributing factors to the preceding were:

- Operator induced LOCA's
- Power availability to feed equipment
- Maintenance
- Operator errors during response (Lack of procedures, RCS monitoring equipment)
- Failure of an air bound RHR pump
- RHR suction failure

2.4.3 Surry Probabilistic risk assessment

This study was conducted by Brookhaven National Laboratories (BNL) during low power and - more applicable – during shutdown mode of operation. Moreover, the study followed a two pronged approach with, a first phase where screening was conducted in order to determine which accident sequences needed to be studied further, and a second phase which consisted of a detailed analysis of those dominant sequences identified in the first phase. This PRA first determined the accident frequency analysis and subsequently the accident progression and consequence analysis. As part of the approach of this assessment various plant operating states (POSs) were defined, and of special interest were those defined in relation to mid-loop operation. Of the three POSs, the largest contributing factor – or POS – to the CDF were the operating state where the reactor was in a drained maintenance state. The CDF for this POS was calculated as $3.0x10^{-5}$ per reactor year and was characterised by high decay heat levels and a relatively short time for operator intervention.

2.5 Probabilistic Risk Assessment Studies – Conclusion

In the preceding sections various PRA based studies were looked at, as a matter of scope confine, BWR's were excluded and only those studies investigating real PWR's in the cold shutdown mode were considered. When the findings of the various studies are analysed, it can be clearly seen that there is a recurring theme in terms of those events being more significant in relation to cold shutdown risk, and those are:

- Failures during mid-loop operation (PWRs)
- Operator error
  - Especially the failure to determine the proper actions to restore shutdown cooling (especially during mid-loop conditions)
  - Procedural deficiencies
- Loss of RHR shutdown cooling, especially
  - Operator induced error
  - Suction valve trips
  - Cavitation due to over draining of the RCS
- Loss of offsite power
- LOCA's, especially
  - Operator induced
  - Stuck open RHR relief valves
  - Ruptured RHR pump seals
  - Ruptured temporary seals

Operator error will not be investigated, but it is worthwhile to look at loss of RHR shutdown cooling, even in combination with mid-loop conditions. Firstly though, the RHR system has the following purposes:

- Decay heat removal and reactor coolant system temperature reduction during the second phase of plant shutdown

- During a loss of coolant accident, it functions as the low pressure injection part of the emergency core cooling system (ECCS)
- This system transfers refuelling water between the refuelling storage tank and the refuelling cavity, preceding and subsequent to refuelling

From a study done by Watanabe and Masashi[7], the loss of RHR transient was investigated based on whether the RCS was filled or whether it had reduced inventory. It was found that with RHR loss in the reduced RCS inventory state (due to air entrainment) it took substantially longer to re-establish RHR than was the case with RHR loss with a full RCS inventory.

2.6 Shutdown and low –power operations: Regulatory Requirements

Requirements for the stated conditions were compiled by the Organisation for Economic Co-operation and Development (OECD) on Nuclear Regulatory Activities led by the NRC and addresses two types of requirements, these being design requirements and operational requirements.

Operational requirements to control low-power and shut down operations are informed by the technical specifications of the individual plants. At present the Standard Technical Specifications [8] (STS) address the following areas, fluctuating in degree of coverage and allowable limits:

- Reactivity control
- Inventory control
- Residual heat removal
- Containment integrity

2.6.1 Reactivity Control

In the case of PWR's during cold shut down operation, the shutdown margin has to be reduced from 1.6% Δ k/k to 1.0% Δk/k. Flux monitors must be operable if control rods can be moved, otherwise the reactor protection systems are not required to be operable once the reactor is shutdown. Furthermore, the Boron injection tank is not required to be operable during cold shutdown and refuelling, but sources of un-borated water need to be isolated from the primary system.

2.6.2 Inventory Control

During cold shut down it is not required that leakage limits and leakage systems to be operable. One pump train for emergency coolant injection is required to be operable during hot shutdown, but not required during cold shutdown and refuelling. The refuelling water storage tank is not required to be operable during cold shutdown and refuelling. Finally, for the case of PWR's, the low temperature overpressure protection system is required to operate during hot shutdown, cold shutdown and refuelling. The preceding takes the guise of two power operated relief valves (PORV) being operable, or 2 residual heat removal valves (RHRV) being operable and only one high pressure injection pump train can be operable.

2.6.3 Residual Heat Removal

In terms of residual heat removal during cold shutdown, it is required that 2 RHR loops are available unless the steam generators are filled to at least 17% of the normal level for said generators. In that case two steam generators and one RHR loop are deemed sufficient.

The residual heat removal system, as shown in Figure 2.1, consists of two heat exchangers, two residual heat removal pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet line to the residual heat removal system for the second phase of cool down, is connected to the hot leg of reactor coolant loop 4, and the return lines are connected to each cold leg of the reactor coolant system. These return lines also function as the emergency core cooling system low pressure injection lines.

2.7 South African Regulatory body

The National Nuclear Regulator (NNR) in South Africa is a government body which is mandated to provide and maintain an effective and efficient national regulatory framework for the protection of persons, property and the environment against nuclear damage. As such, the NNR is mandated to monitor and enforce regulatory safety standards for the achievement of safe operating conditions, prevention of nuclear accidents or mitigation of nuclear accident consequences, resulting in the protection of workers, public, property and the environment against the potential harmful effects of ionizing radiation or radioactive material.

Unlike in the United States where the Nuclear Regulatory Commission fulfils the same role, the NNR does not issue “Generic Letters” on an as needed basis to licensees. It issues an annual national report for the convention of nuclear safety in which it addresses ongoing issues which falls within its mandate, the latest of which refers to Severe Accident Management guidelines (SAMGs) [9] which were drawn up by Westinghouse. The latter has been augmented to include guidance on severe accidents initiating during shutdown conditions. The SAMGs would have been valuable in terms of the analysis conducted as part of the dissertation, by focusing on local regulatory requirements, but they are not publicly available.

2.8 USNRC

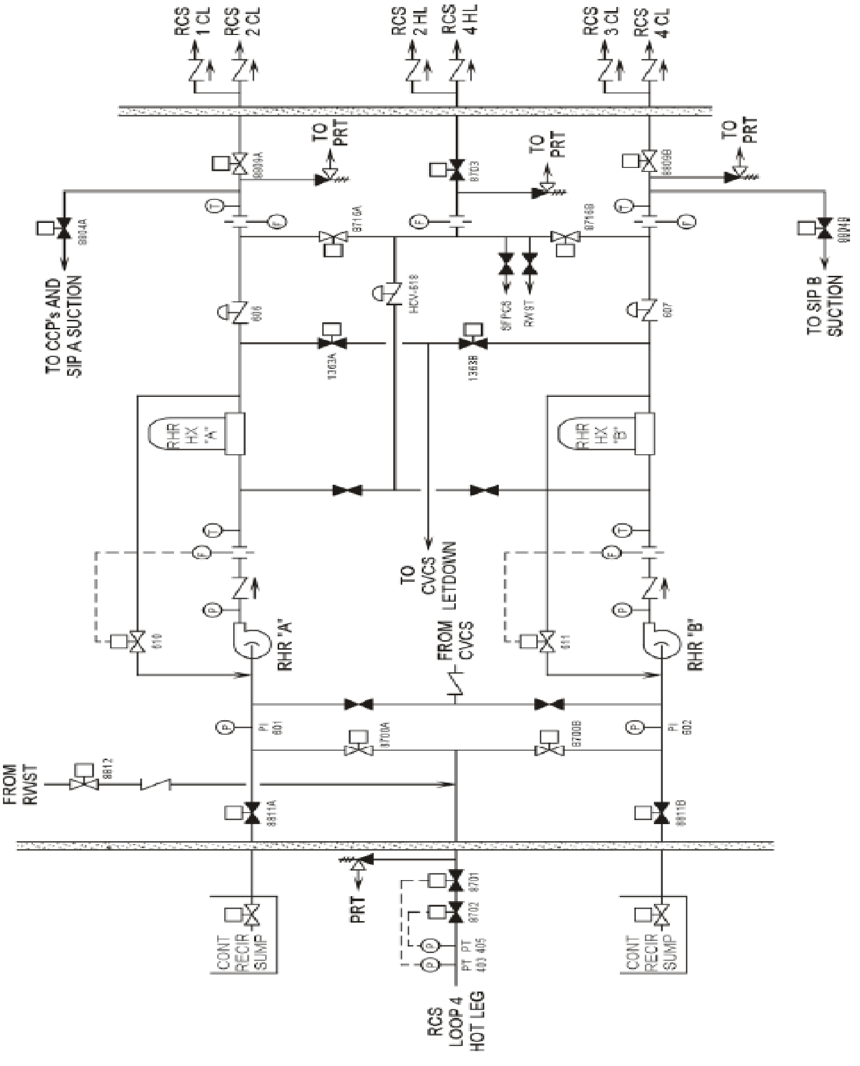
The United States Nuclear Regulatory Commission issues "Generic Letters" [10] on an ongoing basis to licensees in terms of issues that is deemed to be addressed by NPP operators and licensees. An investigation into generic letters issued by the NRC yields the following information:

- Since 1994, 33 Generic letters have been issued by USNRC
- No Generic letters were issued from 2000 to 2002, 2005 and 2009 to 2014
- Of all 33 letters only GL98-02 related to some anomaly while a PWR was in a shutdown condition

GL98-02 [11] dealt expressly with "Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition". The purpose of the letter was to assess the susceptibility of residual heat removal (RHR) and emergency core cooling systems (ECCS) to common-cause failure, as a result of reactor coolant system (RCS) drain down while in a shutdown condition.

Evident from the preceding, the lone Generic Letter issued related to an anomaly during shutdown, more specifically hot shutdown and not cold shutdown. Hence the analysis carried out under cold shutdown conditions might shed more light on an area of interest otherwise not that well documented.

Figure 2.1 Residual Heat Removal System



2.9 Containment Integrity

Containment integrity requirements are not applicable to PWR's during cold shutdown and refuelling, including the operability of the containment spray system.[12]

2.10 Conclusion

In terms of regulatory requirements Nureg 1449 concluded that current standard technical specifications (STS) for pressurized-water reactors (PWRs) were not detailed enough to address the number and risk significance of reactor coolant system configurations used during cold shutdown and refuelling operations. That was particularly true for PWR technical specifications.

CHAPTER 3: SIMULATION METHODOLOGY

3.1 Methodology: Cold shutdown establishment [13]

In order to establish the cold shutdown condition in a 3 loop Westinghouse PWR there are a number of clear steps to be taken, from normal full power operation and they are:

- The turbine/Generator must be taken offload from a point where the reactor power level is in excess of 15%.
- Once the reactor power level reaches the "house" load, reactor control and steam generator control (SG) must be switched from automatic to manual.
    - The electrical loads required by the power station is then transferred to the off station grid.
- The control rods are inserted and the secondary systems placed in a hot standby condition.
- Two reactor coolant pumps (RCPs) are taken off-line and the remaining RCP's ensure sufficient loop flow and loop temperatures are maintained, until such time as residual heat removal system (RHRS) takes over.
- Steam dump is utilized via the pressure operated remote valves (PORV's) in order to shed excess heat.
- The Boron concentration is increased to bring the reactor to the cold shutdown condition.
- The pressuriser level is raised in anticipation of coolant contraction during the process of cooling down.
- Makeup water at the requisite cold shutdown Boron concentration is added to keep the pressuriser level above the no load condition during cool down.
- Steam dumps are adjusted to reduce the temperature at a rate of 28°C/hr.
- The pressuriser heaters are turned off, and the spray valves switched to manual control to cool the pressuriser at a sufficiently slow rate.
- Prior to reaching the pressure where the safety injection accumulators would discharge, the accumulator isolation valves are closed.
- At a reactor coolant system temperature and pressure less than 177°C and 2.76MPa the RHRS starts.
- As the steam generator pressure approaches atmospheric pressure the steam dump valves are closed.
- At RCS temperature of 100°C the sides of the steam generators are filled to the vents.
- All control rods are inserted.
- RCP's only run to provide spray water for the pressuriser cool down.

- Pressuriser heaters are controlled manually to control pressure at the desired level.

The preceding are the steps employed to establish an initial condition of having the NPP in the cold shutdown condition. From this point forward the methodology for effecting the various transients will be discussed.

3.2 Methodology: Transient Analysis - Non executable transients

As stated previously, PCTran has 20 possible malfunctions which can be applied separately or in combination with each other, some of them can also be adjusted in terms of the severity of the particular transient, for example, the size of the loss of coolant accident can be adjusted. Five of the listed malfunctions are not in use and hence can't be applied either singularly or in combination with other transients and these are:

- Loss of main feed pumps
- Loss of AC power
- Loss of flow (Locked rotor)
- Anticipated transient without scram
- Turbine trip

In terms of being in the a cold shutdown condition at the time of the transient, four of the preceding five malfunctions are likely to have a very limited effect on the reactor safety and working parameters should they become a reality.

In cold shutdown state the main feed pumps are in any event not functioning, similarly the turbine trip would have no effect, as in said condition the turbine has been unloaded. Loss of flow is likely to have a very limited effect, as the RCP's are not running and the heat load is very small and handled by the RHR pumps. Similarly for the case of an anticipated transient without scram, the repercussions are expected to be limited as the reactor power level is vastly reduced compared to when operating at full power.

The only likely transient of those listed above which could have a significant impact would be the loss of AC power. Even though the transient can't be applied – in order to gain insight into what can transpire – we will take a look at what happened at Fukishima Daichi [14] when exactly such a scenario became a reality.

3.2.1 Loss of AC power – Fukushima Daichi:

On the 11th of March 2011 at 2:46 pm a magnitude 9 earthquake struck 130km offshore of the city of Sendai, off Honshu Island, Japan. At the time 11 reactors at 4 nuclear plants in the region were operational and all of them shut down automatically when the earthquake struck. Thus they all proved seismically robust and 8 of the 11 achieved cold shutdown 4 days after the event. The 3 units in operation at Fukushima Daichi (Units 1-3) lost power at 3:42pm, when the site was flooded by the 15m Tsunami. Twelve of the thirteen backup generators on site were disabled as well as the heat exchangers dumping waste heat to the sea.

Without the RHR removing heat to an outside heat exchanger a lot of steam was produced in the RPV's which was released to the dry primary containment through safety valves. Water injection into the RPVs were done by various systems and ultimately the ECCS. They all failed and finally water was injected with fire pumps.

The fuel was fully exposed by 7:30pm and the temperature rose to 2800°C, 16 hours after the SCRAM. Melting started in the centre of the core and the molten mass fell into the water at the bottom of the RPV. Over the following days Units 1, 3 and 4 experienced Hydrogen explosions and units 1, 2, 3 and 4 are written off and will be decommissioned.

Thus from the experiences – regardless of the trigger – at Fukushima, the loss of AC power with no readily available backup can have dire consequences.

3.3 Methodology: Executable transients

The various transients available in PCTran will be applied as single mode failure events, before various transients will be paired together on the basis of the likelihood of the actual multiple mode transient occurring. Again, the actual application of an available transient will be based on whether it would be possible considering that the PWR is in a cold shutdown condition. The single transients that will be applied are:

- Loss of coolant accident (LOCA)
- RHR failure without replenishment
- Inadvertent rod withdrawal
- Moderator dilution
- Fuel clad failure

3.3.1 Loss of coolant accident (LOCA)

Due to the fact that the PWR is in a cold shutdown condition, and hence at the opposite end of the scale compared to power generation conditions, it is not expected that there will be any difference between a hot and a cold leg LOCA. For the sake of completeness and confirmation though, both will be simulated and compared.

3.3.2 Residual Heat Removal failure without replenishment

During cold shutdown the plant is in mode 5 and as such, many of the automatic functions otherwise protecting the reactor and equipment is either not functioning or in manual mode. Notwithstanding, the limiting conditions [15] for operation during cold shutdown are:

- 2 Pump trains have to remain operable in respect to the Residual Heat Removal system
- 1 Offsite source of AC power has to remain operable
- Emergency core cooling is not required
- 1 Onsite source of AC power has to remain operable
- Primary containment integrity is not required

3.3.3 Inadvertent rod withdrawal

This transient will be of special interest as the main RCPs will be offline and the RHRS will be handling the loop flow.

3.3.4 Moderator Dilution

As was noted in the shutdown procedure, Boron injection was initiated as part of the process to get the reactor into the required cold shutdown condition in the first instance. Hence dilution of the moderator would be of special interest.

3.3.5 Fuel clad failure

The likelihood of this transient occurring can be debated, especially in the cold shutdown condition. However it would most certainly be of interest to investigate such an incident due to the possible radiological severity of the consequences.

CHAPTER 4: EXPERIMENTAL RESULTS AND FINDINGS

4.1 PCTran Benchmarking

PCTran is a reactor transient and accident simulation software program that was first released in 1985, and has been constantly upgraded in terms of performance and capabilities. Micro simulation technologies who is the author and owner of the simulator, claims that extensive benchmarking have been carried out against other software simulation packages. [16]

This specific simulator - and the simulations done - were based on the CPR-1000 reactor which was built as phase II of Ling Ao unit 1[17]. The simulator and how it reacts is based on the FSAR (Final Safety Assessment Report) for the plant.

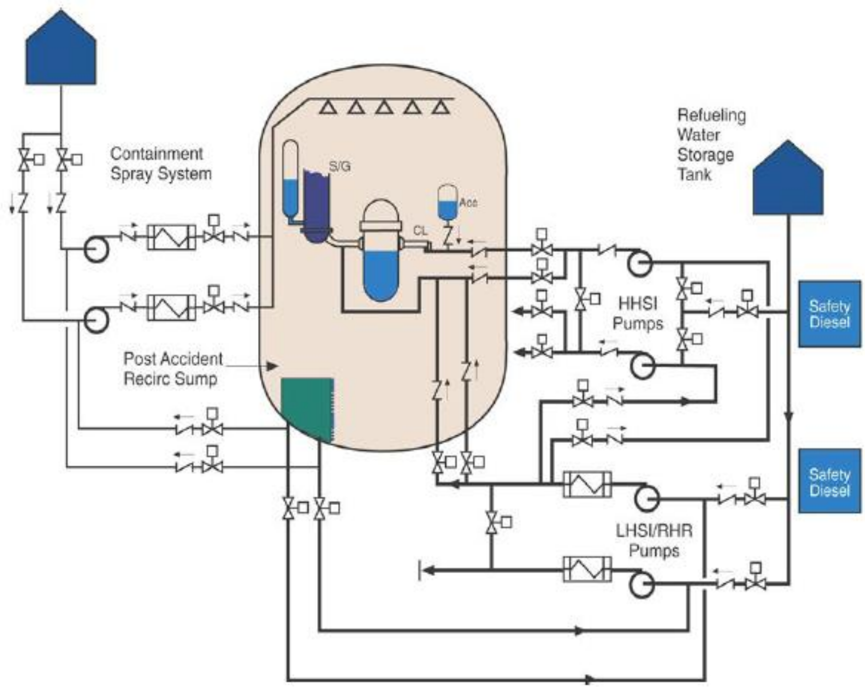


Figure 4.1 Schematic of CPR-1000

The CPR-1000 is a 3 loop PWR based on the French 900MW class PWR reactors with a design life of 60 years, and a rated NSSS thermal power of 2905MWt and a gross power of 1086MWe. In terms of further benchmarking before proceeding with the analysis, the performance of the simulator will be measured against a body of work done by Watanabe and Masashi (Analysis of Operating Experience Involving Loss of Decay Heat Removal during Reactor Shutdown in Pressurized Water Reactors, Japan

Atomic Research Institute). In the work done by the aforesaid they looked at RHR failure with the RCS filled and at mid-loop. They then also looked at the rate of temperature rise and time to core boiling. In this case it will not be able to have a comparison to the latter, as it was not possible to get the average coolant temperature below boiling point. In reference to Fig4.2 below, Watanabe and Masashi defined the mid-loop as the centre line of the hot leg piping. For the simulator that equated to 23% voids. It is at this point that the mid loop simulation was started. Refer to figure 4.2 below, it is the GUI at the end of the mid-loop run. Note SG's and Pressuriser are empty. The core is completely drained.

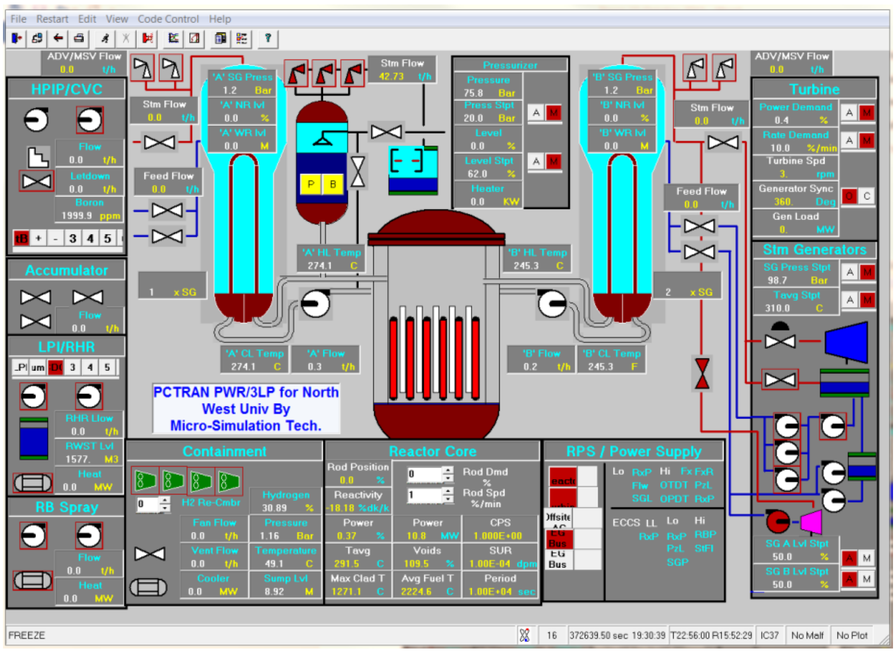
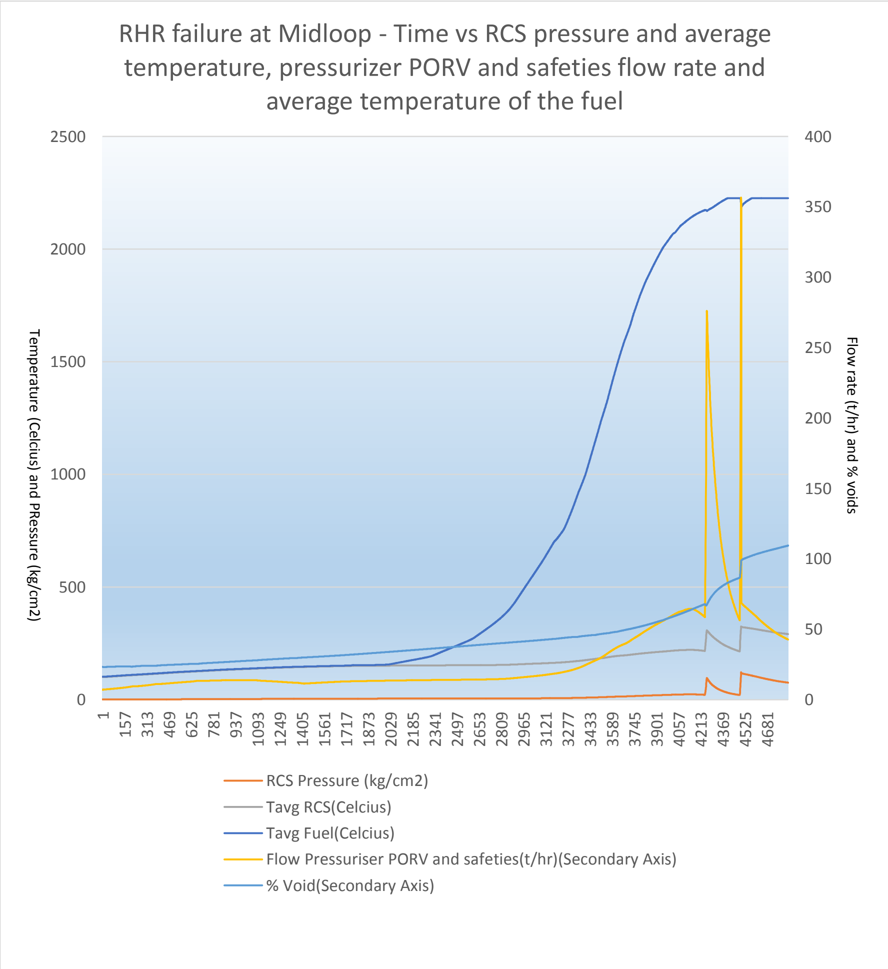


Figure 4.2 GUI RHR failure at mid-loop – End State

In reference to graph 4.1 below, the RHR pumps are stopped at an average RCS temperature of 100.345°C and 1.03 bar – as a result - the RCS content is barely boiling, concomitantly the average temperature of the fuel is at 102.05°C. After 165 minutes the average RCS temperature and the average fuel temperature is still tracking very closely together circa 152°C and a ∆T of 3.9°C. Barely 29 minutes later the average RCS temperature is still 152.5°C, but the average fuel temperature is now 200.1°C or a ∆T of 47.6°C. Ultimately though the average rate of increase in RCS average temperature is in the order of 0.47°C/min compared to the Beaver Valley 3 loop PWR which experienced RHR failure at mid-loop operation in 1981 which had an average rate of coolant rise of 0.68°C/min.



Graph 4.1 RHR failure at mid-loop

4.2 Cold Shutdown Establishment

In terms of simulating the planned transients, it was postulated that the reactor needed to be brought to a condition of cold shutdown. Definitions of a cold shutdown condition vary according to the literature, with one definition requiring the average temperature in the core coolant to be smaller than 100° C and the Pressuriser at atmospheric pressure.

Another definition requires the Pressuriser to be at atmospheric pressure with a concomitant average coolant temperature in the core of smaller than or equal to 60°C. In either case the following requirements have all been established:

- RHR system operating
- Both RCP's non-functioning
- Steam generators filled to vents
- Pressuriser filled
- All critical equipment are on manual control

In the simulation, in establishing cold shutdown nearly 3 days after initiating shutdown and the preceding conditions having been met – by definition – cold shutdown has not officially been established. The average temperature in the core coolant is 100.34°C and the Pressuriser pressure is atmospheric. The author has spent many days following the correct shutdown procedure, as per the Westinghouse manual in order to establish cold shutdown but the stated conditions are the closest achieved. The rate of change of the key variables converge to zero, and it seems as if the PCTran simulator is not capable of simulating a cold shutdown condition, albeit getting very close.

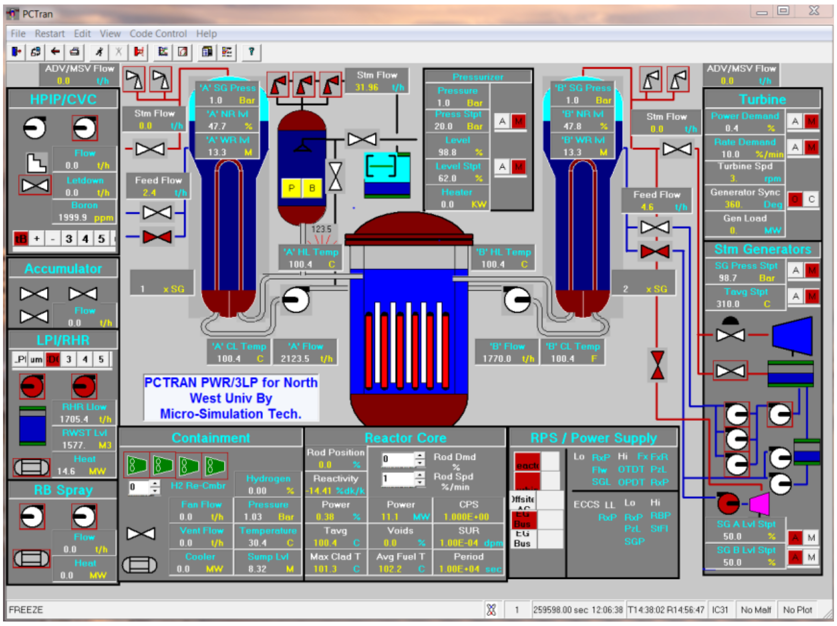
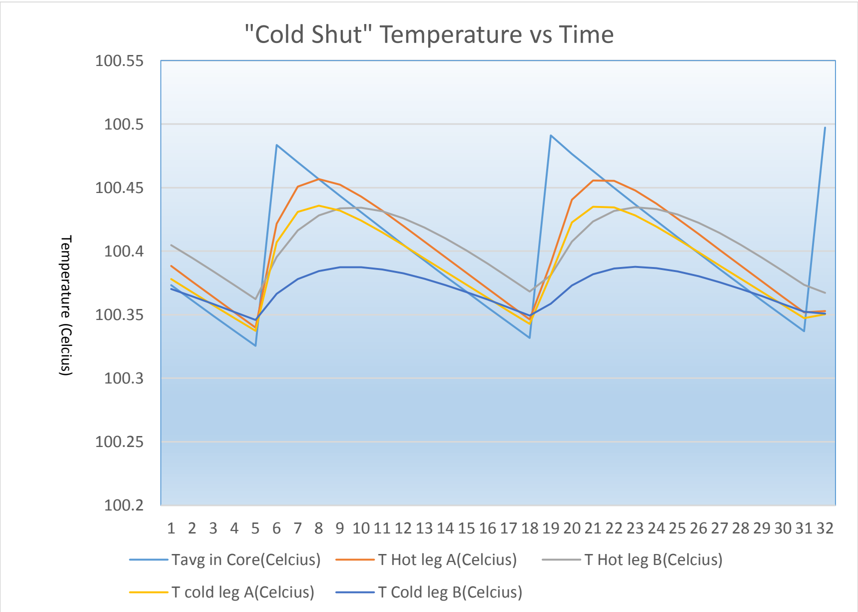


Figure 4.3 Cold Shutdown Mimic

In real world PWR's roughly at the condition as represented by the mimic above, the steam bubble in the Pressuriser would be replaced with Nitrogen, however PCTran does not have that functionality.[18]

Hence, after spending a vast amount of time in trying to establish a defined cold shutdown condition, the actual condition reached will be deemed as sufficient to insert the transients planned, and to carry out the analysis. Please note that the presence of a void in the core as well as a purported hot leg LOCA are erroneous as the actual void percentage is 0 and there is no malfunction present.

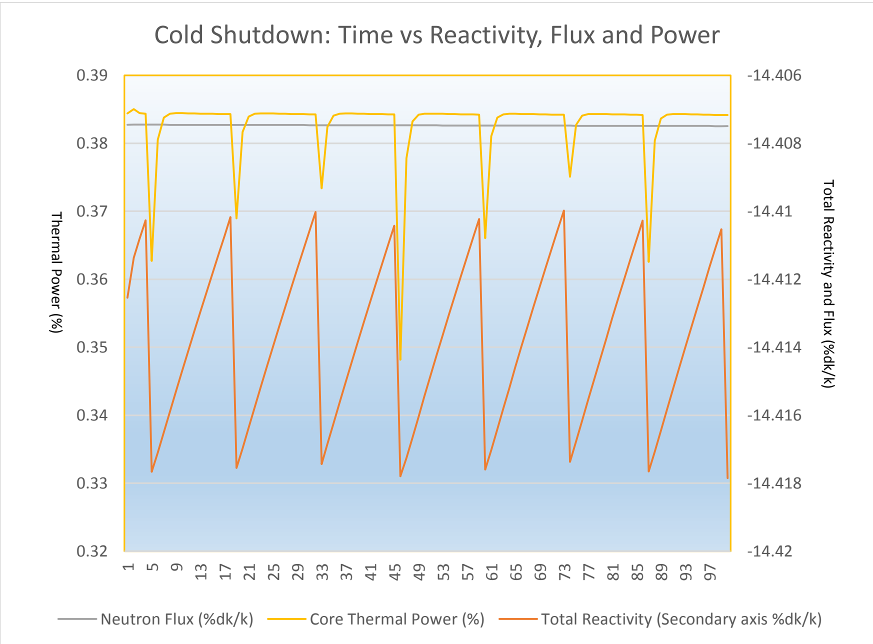
The following graphs then show the condition as attained for inserting the transients into the cold shutdown condition. Please refer to Graph 4.2 below. This is just a reflection of the average temperature and leg temperatures transferring heat to the secondary system. What is especially notable is the cyclical nature of the temperatures of over time. Every 65s there is a spike of roughly 0.159°C in the average core coolant temperature where after it decreases over 60s by 0.152°C. This can only be as a result of some anomaly unique to the PCTran simulator.



Graph 4.2. Cold shut – Temperature vs Time

Also, related to establishment of the cold shutdown condition, please refer to graph 4.3 below which show the relationship between reactivity, flux and power over time for the stated condition. Interesting to note here is that every 65s we see a negative spike in the core thermal power lasting 20s, the magnitude of which varies in a repeating pattern. The neutron flux – concurrently - is very slowly trending lower, but without any cyclical behaviour. Meanwhile, the total reactivity coefficient is showing cyclical behaviour whereby it peaks at a value and 5s later it reaches its maximum negative value, where after it takes a further 60s to peak again before the cycle repeats itself.

What is further quite apparent is that the core thermal power and the total reactivity is perfectly synchronized. As soon as there is a negative reactivity insertion – for whatever reason – the core power shows a decrease. The recovery time to values before the negative spikes differ greatly. Whereas the core thermal power takes about 15s to recover to pre-spike levels the total reactivity takes about 60s to recover, before the process repeats itself. Obviously the trigger for the decrease in the core thermal power is the increased negative total reactivity. But the question is, what is the cause of the increase in the negative reactivity?

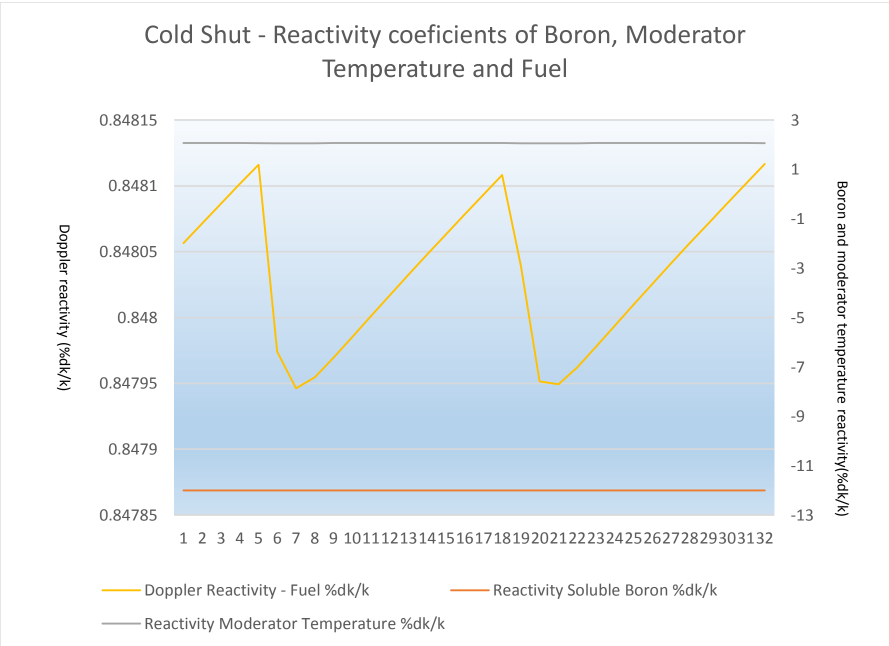


Graph 4.3 Cold Shut – Time vs Reactivity, Flux and Power

In answering the preceding question, please refer to graph 4.4 below. In said graph, the moderator reactivity for the same period of time as well as the reactivity of the soluble Boron are for all intent and purposes stable. In contrast though, the Doppler reactivity (Fuel temperature reactivity) is also cyclical in response to the core power being cyclical, and shows the most variance of all the reactivity coefficients.

When the core thermal power in graph 4.3 is examined in conjunction with the Doppler coefficient in graph 4.4, it is noted that once the core power bottoms out and starts to rise, 5s later, the Doppler reactivity bottoms out and starts to rise. The dominant cause is the Doppler broadening of the resonance cross section of U-238. In a PWR the Doppler reactivity is also called the "prompt reactivity" and is even more important during transients than the moderator reactivity.

The cold shut condition analysed above was done as a check on the behaviour of the various systems and to check that their behaviour is in-line with expectations.



Graph 4.4 Cold Shutdown – Time vs Reactivity coefficients of soluble Boron, Moderator temperature and fuel

4.3 Complete RHR failure without replenishment of the Pressuriser

Nureg 1449 investigated the occurrence of 16 loss of RHR events in PWR's over a period of 9 years and hence, it can be reasoned that this specific transient is a very real risk to the safe operation of PWRs. It must be noted that 60% of the events were due to human error and 16 % due to equipment problems. [18]

The loss of RHR can be initiated by a myriad of causes, but they all essentially relate to a loss of flow, be that as a result of pump shaft failure (Crystal River 3) or the RHR pumps becoming steam bound (Waterford 3) or a general loss of offsite AC power (Fort Calhoun)[19], among others.

For the simulation of this transient both RHR pump trains will be taken offline from a cold shutdown condition, and from that point forward the simulator will be allowed to run its course and the analysis carried out. During this transient the Pressuriser will not be replenished, as during a real cold shutdown the systems that can replenish it will be in manual operation.

4.3.1 Hypothesis and experimental results

It is hypothesized, that if corrective action is taken in restoring RHR before the situation becomes too critical, be that either through the ECCS or any other systems available then fuel and core damage can be averted. As is clearly shown below, the experimental results behaves as hypothesized but also quantifies the critical time frame within which corrective action needs to be taken.

Figure 4.4 below captures the key reactor and system variables at the end of the injection of the RHR loss simulation. Firstly note that there are some display problems with the mimic as even though the core look to be filled with coolant it is in fact empty, same applies to the pressuriser. The data confirms that. Furthermore both RHR's pump trains have been stopped.

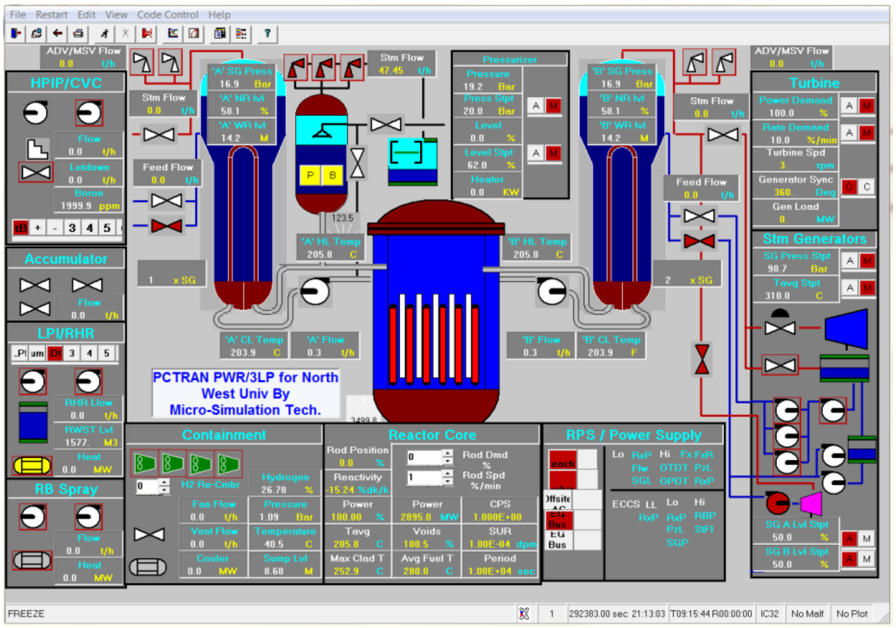


Figure 4.4 Mimic of end state with RHR failure.

In reference to graph 4.5 below, where a RHR failure was injected at time t=0 we note that there is a very gradual increase in the average core coolant temperature starting at 100.38°C and rising to 205.75°C in slightly over 9 hours. More importantly the fuel peak temperature also shows a gradual increase in roughly 5 hours, tracking at the same rate of rise as the coolant temperature. However after 310 minutes the peak fuel temperature increases from roughly 151.87°C to 2211.17° C over a period of 2hours and 27 minutes, or at a rate of 14 degrees per minute. Thus if corrective action has not been forthcoming at this juncture, rapidly escalating events going forward is very likely to negate any mitigating actions taken.

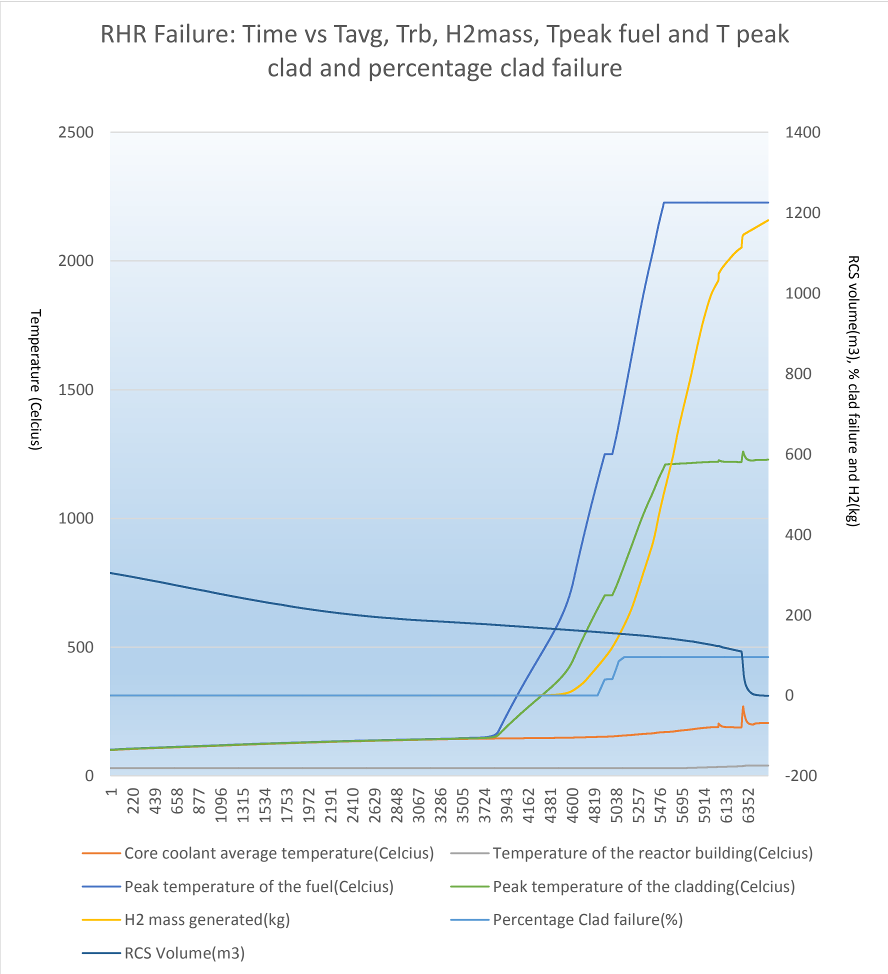
Further, another critical variable to keep track of is the production of Hydrogen as the Zirconium in the cladding reacts with the water to form Hydrogen gas. After 4 hours and 49 minutes of initiating the transient the H_2 production is still zero. A further 1 hour and 16 minutes later hydrogen gas is starting to form but only slightly over 1kg and a further 3 hours later the total production of H_2g gas is 1181.31kg. The formation of the hydrogen gas is subject to the following reaction. [20]

$$Zr + 2\ H_2O \rightarrow ZrO_2 + 2\ H_2 + \Delta Q_1$$

Unless this build-up of gas is vented, there is a real risk that should there be some source of ignition this mass of hydrogen can explode, with the associated catastrophic results.

In relation to the preceding, what remains a bit puzzling is the fact that the fuel clad temperature at which the first kilogram of Hydrogen gas is produced is a relatively lowly one of 343.675°C while clad failure only sets in at a peak clad temperature of 648.968°c.

As postulated there is a certain period of time within which damage can be averted under the situation created by this transient. The simulation showed that the plant operators essentially has 3 hours from the time the transient starts, to the time RHR is restored, to avert major damage.



Graph 4.5 RHR failure. Time versus core average coolant temperature, Temperature of reactor building, hydrogen mass generated, peak temperature of the fuel, peak temperature of the cladding and percentage clad failure

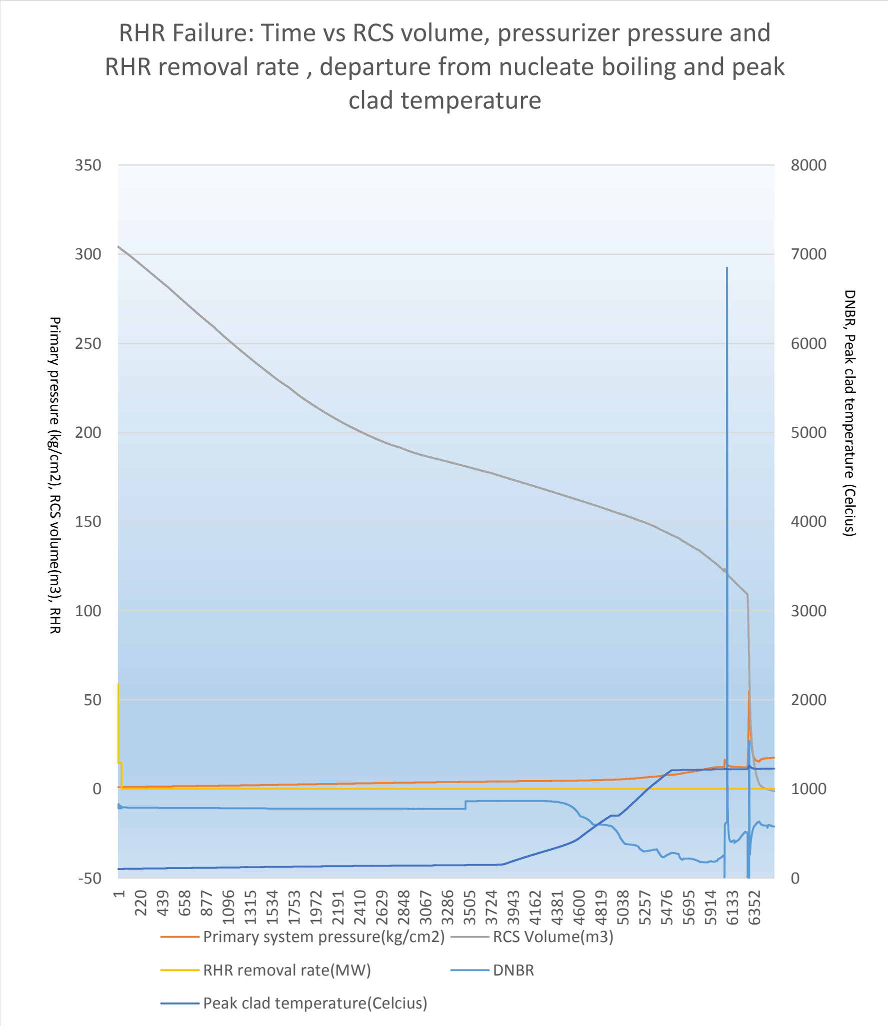
4.3.2 Analysis

Ultimately, the fact of the matter is that 7 hours and 39 minutes after the transient of RHR failure with full RCS inventory was introduced, the reactor experienced meltdown with the fuel temperature reaching 2226.67°C.

Fuel clad failure starts at a peak clad temperature of 648.22°C and 404 minutes and 34 seconds after the simulation started, and rapidly progresses to a condition where 96% of the cladding has failed only 42 minutes after clad failure started. Commensurately the radiation levels in the air increases as well as the levels of I131, I135 and Xe135.

Please refer to Graph 4.6 below. As can be seen 165s after the start of the simulation the RHR rate drops from about 14.5MW to 0. The volume of coolant in the RCS is steadily decreasing and the primary system pressure increasing as can be expected. What is of particular interest is that with the RCS volume about 42% reduced, the rate of temperature rise outpaces the rate of heat transfer and the peak cladding temperature rise accelerates. This is also the point at which the top of the fuel rods are being uncovered, hence it correlates perfectly with the reduced heat transfer capacity.

Slightly lagging this, the departure from nucleate boiling ratio (DNBR) starts to rapidly reduce further inhibiting heat transfer causing the peak cladding temperature to rise even further. Departure from nucleate boiling ratio – according to the USNRC - can be defined as the ratio of the heat flux needed to cause departure from nucleate boiling to the actual local heat flux of the fuel rod. Needless to say, this is also inextricably tied to the primary system pressure, which also shows an accelerated rise at about the same time as the departure from nucleate boiling ratio increases.



Graph 4.6 RHR failure

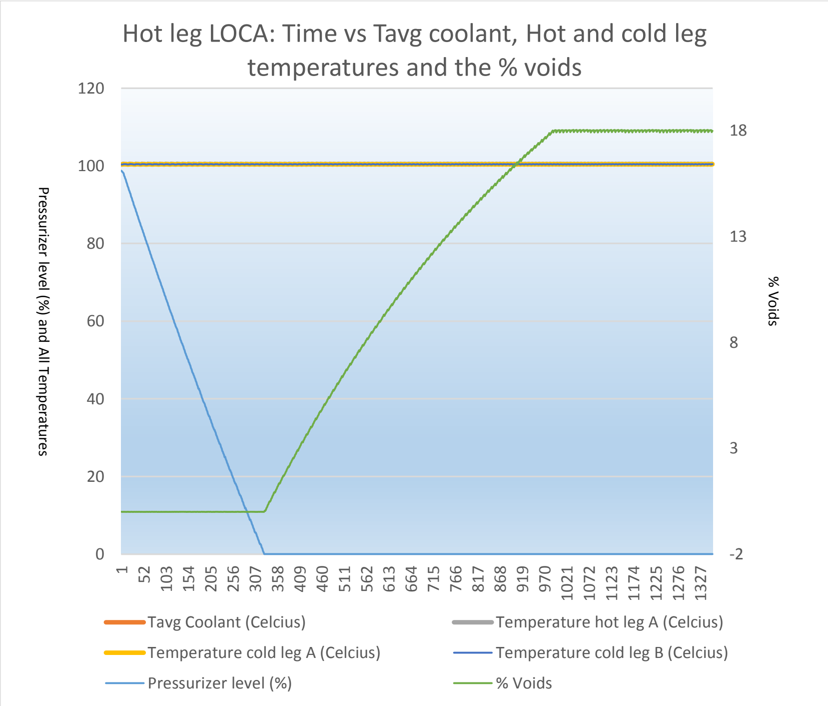
4.4 Loss of coolant accident (LOCA)

This transient is probably one of the most researched and understood transients – as it might occur during power operations - of all. During the cold shutdown condition there is effectively no temperature difference between the hot and cold legs of the primary system.

4.4.1 Hypothesis and experimental results

It is hypothesized that even with a large break LOCA (as simulated) during the cold shutdown condition there will be a loss of RCS inventory, but the loss of inventory will be limited to the height of the location of the LOCA in the RPV. At that level the rate of RCS inventory loss through the failed pipe will essentially diminish to zero and the RHR system will have sufficient capacity to keep the level of the coolant stable.

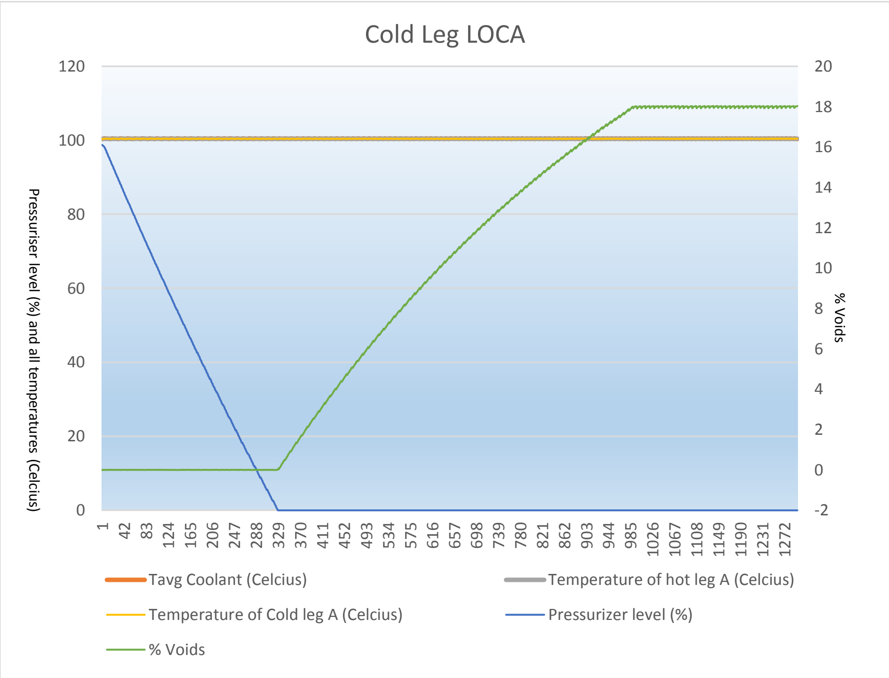
It is further hypothesized, that due to the fact that the temperature in either the hot or cold legs are the same, a failure in either leg would result in identical results. The experimental results below correlates perfectly with the hypothesis.



Graph 4.7 Hot leg LOCA

In the preceding graph, after 20 seconds a malfunction was initiated taking the form of a 100% LOCA in hot leg A, with a ramp time of 10 seconds. The residual heat removal system was functioning as planned. After 26 minutes and 55 seconds the Pressuriser – which was filled at the start of the transient – has completely drained which then started the formation of voids in the pressure vessel itself. A further 56 minutes and 40 seconds later the void formation peaks at 18% and stays at that level without further changes. This is just at the level where the hot leg pierces the reactor vessel. The fuel temperature and the coolant temperature are still the same as during the cold shutdown condition. All other vital variables remained stable and it shows that in the case of a hot leg LOCA the void formation will stabilise by itself and the RHR will have sufficient capacity to cool the core as long as it is functioning; which aligns with the hypothesized expectations.

In graph 4.8 below the same variables have been plotted as for the hot leg LOCA as in graph 4.5 and the results were identical, as expected.



Graph 4.8 Cold leg LOCA

In figure 4.5 a 4 loop Westinghouse PWR pressure vessel and steam generator arrangement is shown. Notice that the hot and cold legs of the steam generators are located in exactly the same vertical elevation. The same is of course true for the 3 loop reactor pressure vessel. That is the reason why exactly the same result were obtained for the hot and cold leg LOCA's. If the cold legs pierced the RPV shell at a lower elevation than the voids would have been commensurately more.

4.4.2 Analysis

What this particular simulation does prove, is that in the event of either a hot leg LOCA or a cold leg LOCA in the cold shutdown condition with the RCS inventory filled, the RHR system has sufficient capacity to deal with the decay heat generated within the core. The coolant average temperature stays at the exact same level following the transient when compared to before the transient. This is not to say that the outcome might not be far more severe if more than one transient is inserted simultaneously, but the probability rapidly diminishes. The voids in the core – once the Pressuriser has been drained - starts to increase, however it stabilises 1 hour and 33 minutes after the start of the simulation. Regardless of the initiating reasons both the Three Mile Island accident and the Fukushima Daichi accident experienced LOCA's and subsequent partial core melt downs, neither were anywhere near the cold shutdown conditions simulated here at the time of the accident.

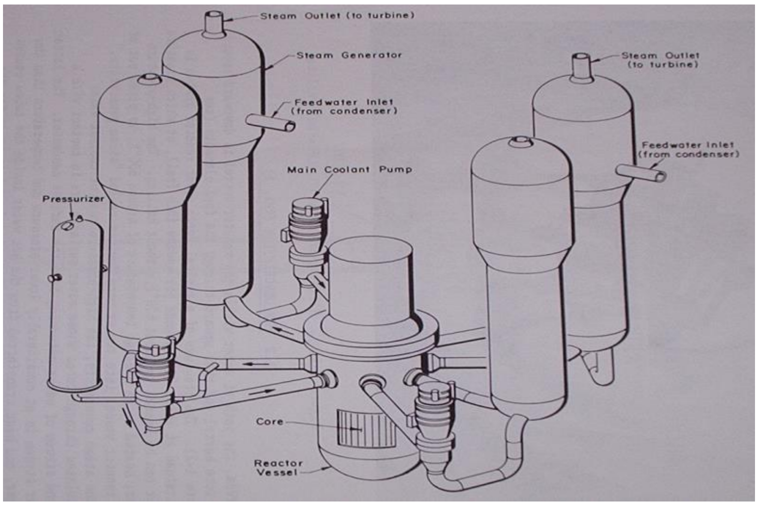


Figure 4.5 4 Loop Westinghouse PWR showing piercings of the RPV by the SG hot and cold legs all at the same vertical elevation.

4.5 Inadvertent control rod withdrawal

As stated previously, a typical 3 loop Pressurized Water Reactor requires refuelling every 18 to 24 months. It is for this reason that the reactor is brought to a cold shutdown condition. Sometime after the reactor has been in cold shutdown, normal operations call for the removal of the RPV head, at which stage this transient would be of no interest. But there is a period of time where the PWR is in cold shutdown with the RPV head in position and the rod cluster control assemblies inserted. It is here that this transient will be inserted in order to analyse the results of such a transient.

4.5.1 Hypothesis and experimental results

It is hypothesized that the inadvertent ejection of the control rods will decrease the total negative reactivity coefficient of the core. The magnitude of the decrease, and whether the total reactivity coefficient will become positive will be a function of the number of assemblies ejected, the fuel rod worth and the levels of soluble Boron in the coolant. It also expected that a neutron flux spike will be observed as the reactivity coefficient becomes positive, and then a dampening response on the flux as the Doppler Effect recurs.

The hypothesis and the experimental results shows a good correlation, highlighting the dependence of the ejected rod assemblies and the soluble Boron concentration, as well as the flux behaviour and dampening.

An accident of this nature is defined as an uncontrolled addition of reactivity to the core caused by the withdrawal of rod cluster control assemblies(s). This could be caused by a malfunction of the control rod control system. Positioning of rod cluster control assemblies are done by latch-type magnetic jack control rod drive mechanisms mounted on the RPV head see figure 4.6 below. It has 4 major subassemblies: pressure vessel, internal latch assembly, drive rod operating coil stack and seismic assembly.

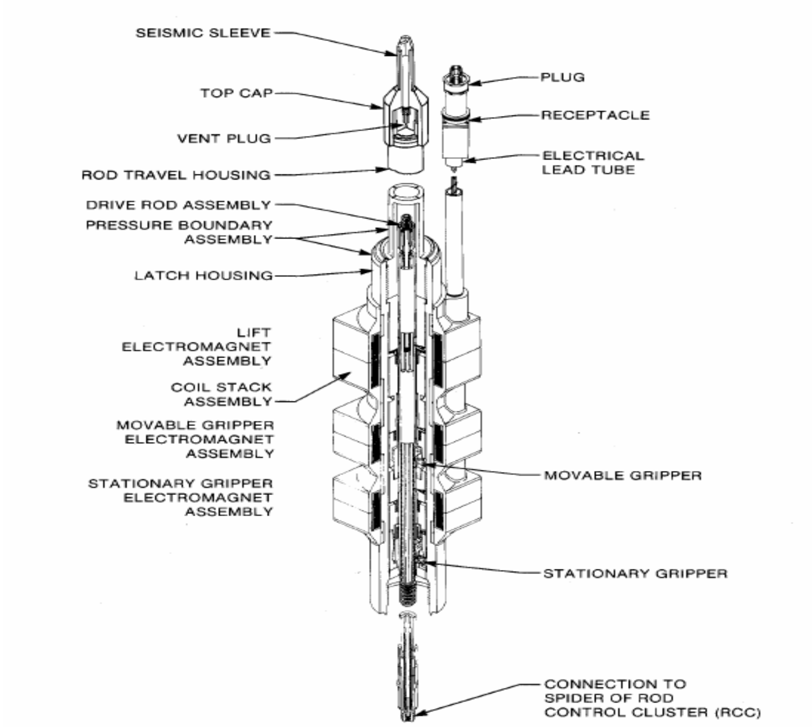
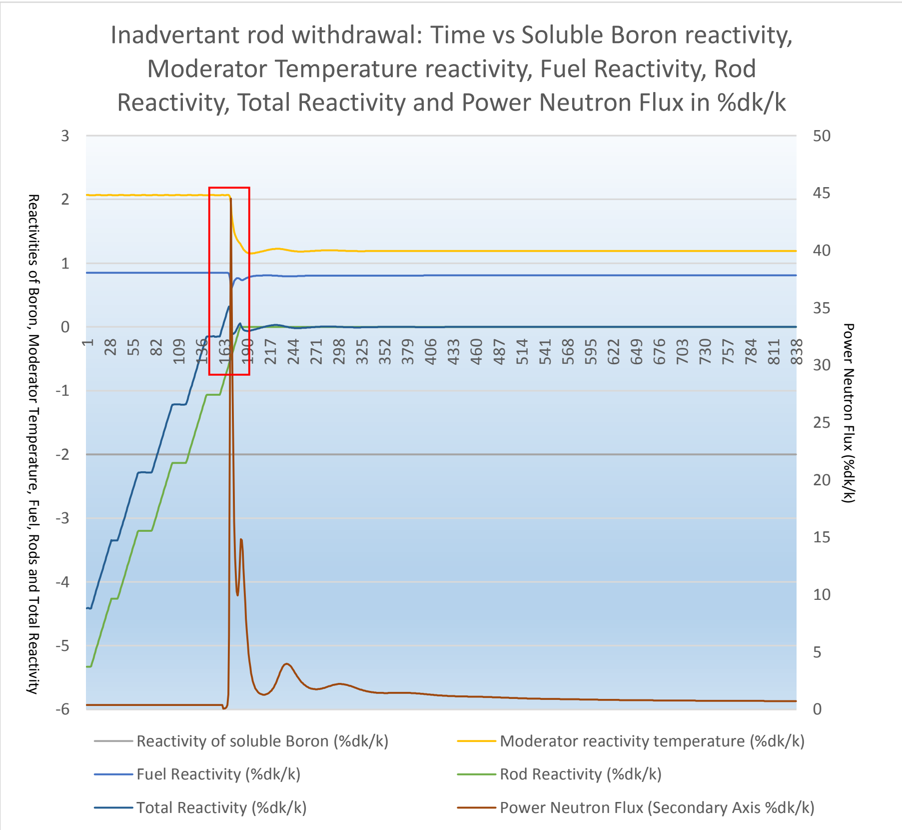


Figure 4.6 Cutaway of control rod drive mechanism

At the start of this simulation, the 3 RCP’s are off-line, the RHR system is in operation, primary and secondary systems are at 1 bar pressure, HPIPs are disabled and the RCS inventory is 100%.



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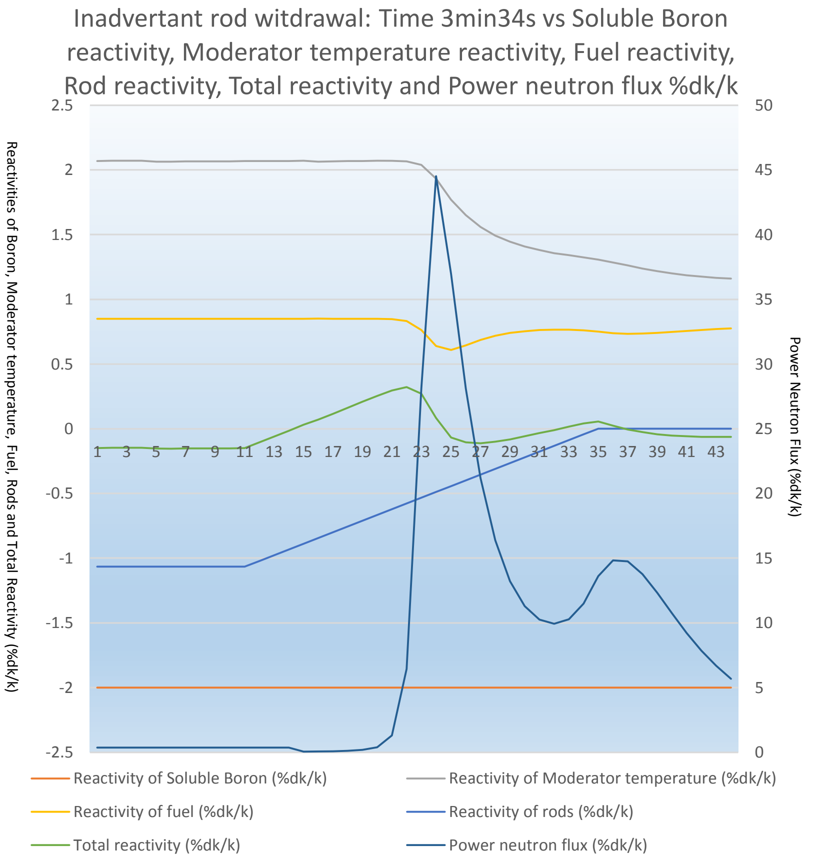
Graph 4.9 Inadvertent rod withdrawal: Time vs Soluble Boron reactivity, Moderator temperature reactivity, Fuel reactivity, Rod Reactivity, Total Reactivity and Power Neutron flux in %dk/k.

In reference to graph 4.9, the relationship between the various reactivity coefficients and the neutron flux for a constant Boron concentration of 1000ppm is shown. The rod ejection transient was also injected for constant Boron concentrations of 2000 ppm and 1200ppm, the results are not plotted though, because in both instances the overall reactivity remained negative and the primary system pressure and RCS coolant temperature showed no change. The RHR system succeeded in keeping the reactor stable and removed the decay heat.

However, with the Boron concentration reduced to 1000ppm and a step wise ejection - in 20% increments - at maximum rod ejection rate, it is clear that when the final 20% rod ejection is nearly complete there is a surge in neutron flux. Please refer to graph 4.10 below. It is a graph of the same parameters but it covers only the time period highlighted in graph 4.9 above (Red rectangle).

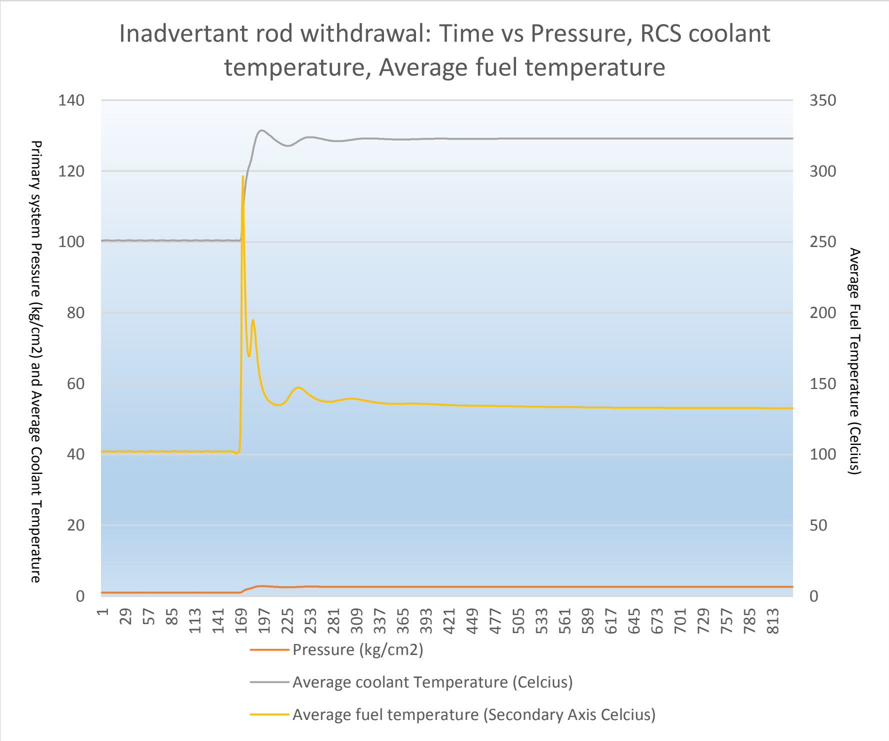


Graph 4.10 Inadvertent rod withdrawal: Time vs Soluble Boron reactivity, Moderator temperature reactivity, Fuel reactivity, Rod Reactivity, Total Reactivity and Power Neutron flux in %dk/k.

4.5.2 Analysis

It is clear in graph 4.10 that the soluble reactivity of Boron is kept constant at a nett negative value of -2%dk/k. Simultaneous to the rod reactivity increasing, (last 20% being removed) the total reactivity starts increasing and becomes positive after 13min and 25s. It then becomes negative, and 1min and 29s later it becomes positive again, the cycle is repeated, and 2min and 36s later it becomes positive again. The trend repeats itself 5min and 10s later and again 7min and 40s later. The total reactivity is thus being dampened by the Doppler coefficient in an inversely correlated way. As the Doppler coefficient increases the total reactivity coefficient decreases and conversely. After 14 minutes and 10 seconds, the neutron flux (secondary axis) reaches a peak value of 44 %dk/k or nearly 12000% of what it was prior to the last rods being ejected. However the Doppler Effect is the first responder in mitigating the neutron flux surge and the moderator coefficient later adds to bringing the neutron flux power levels back to pre-transient levels.

In normal cold shutdown conditions the Boron concentration would be slightly over 2000ppm [21] and it would be very unlikely that all control rod assemblies would be ejected. Hence the likelihood of a critical failure or meltdown occurring under normal conditions is extremely remote. Even under the conditions simulated, the power excursion was limited and very unlikely to cause any damage to the core or fuel.



Graph 4.11 Inadvertent rod withdrawal: Time vs Pressure, RCS coolant temperature and Average fuel temperature

In reference to the preceding graph the primary system pressure increases from 1bar to 2.83 bar in 16 minutes and 10 seconds, due to a rapid rise in fuel temperature during the neutron flux excursion and the coolant temperature reaches a peak of 131.46°C. Similarly the average fuel temperature also peaks at 296.46°C, but 1 minute and 55 seconds before the coolant temperature and the pressure. That is as a result of the moderator coefficient responding much slower than the fuel coefficient.

4.6 Moderator Dilution

In a study of the literature related to the transient where moderator dilution is the anomaly under scrutiny, all previous work found, investigated the occurrence where a large slug of un-borated water were injected into the core due to some malfunction. Furthermore, most of those studies looked at the situation when the reactor was at hot zero power or hot full power.

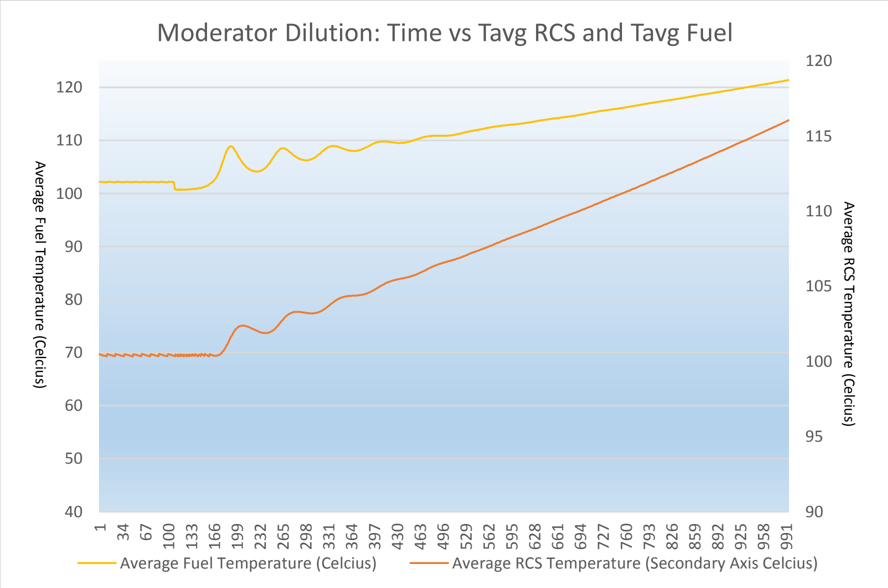
Essentially, what happens under said circumstances is that the reactor sees a reactivity insertion[22], the magnitude of which depends on the volume, rate and location of the insertion. With the PCtran simulator the same simulation – but at cold shut conditions – is done by changing the chemical shim or by reducing the soluble Boron content when the reactor is operating.

4.6.1 Hypothesis and experimental results

It is hypothesized that the reduction in the soluble Boron will decrease the total negative reactivity in the core and the temperatures of first the fuel and then the coolant will start to rise. Meanwhile the total negative reactivity will continue to decrease as the Boron concentration decreases and temperatures and pressures will continue to rise, along with the neutron flux, as more neutrons are absorbed in the fuel. The process should be dampened though as a result of the Doppler Effect.

The hypothesis and experimental results shows a good correlation as the reduction in the Boron concentration continues, the total reactivity increases, and the flux spikes as the reactivity coefficient becomes positive. The latter occurrence is short lived as the Doppler Effect dampens the flux.

Please refer to Graph 4.12 and 4.13 below. In the former case it plots the relationship between the average RCS temperature and the average fuel temperature over time, and in the latter case it plots the relationship over time between the various reactivity coefficients and the neutron flux in the core as the concentration of soluble Boron is decreased.



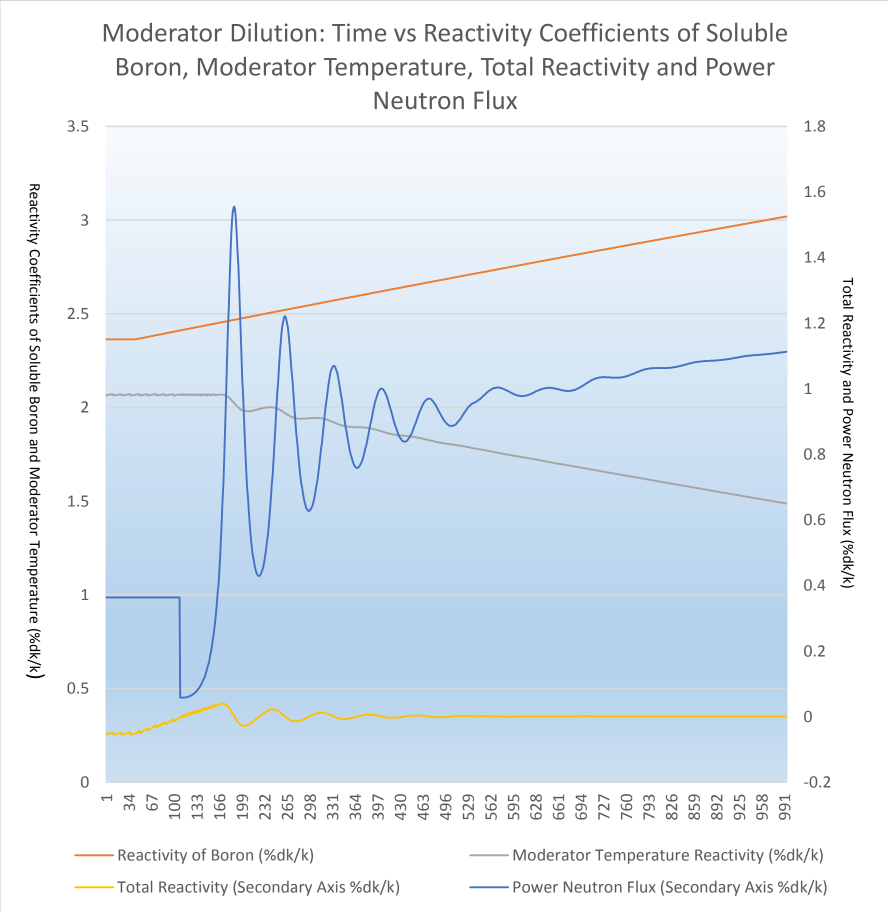
Graph 4.12 Moderator Dilution: Time vs Tavg RCS and Tavg Fuel

What is clear from graph 4.12 is that the average fuel temperature changes before the moderator temperature, as water is thermally more stable than the fuel. As the total reactivity coefficient turns positive the neutron flux increases by 2577% in 6 minutes and 30 seconds, the temperature of first the fuel and then the moderator temperature increases, and the reactivity becomes negative and the flux reduces by 257% in 3 minutes and 5 seconds. The process repeats itself as the total reactivity coefficient turns positive, the flux increases and the temperature and RCS pressure follows suit.

4.6.2 Analysis

In analysing the result, it is noticeable that with each cycle the extremities are dampened until the total reactivity stabilises at very near to zero after 53 minutes and 20 seconds, of the simulation starting. The average fuel and moderator temperatures continue to trend upwards, as do the neutron flux at rates of 0.26 and 0.27°C per minute and the flux by 0.003% per minute.

It must be borne in mind that generally the soluble Boron concentration in the cold shutdown condition is in excess of 2000 ppm. Thus, as in this scenario, if moderator dilution does occur there are hours to take action prior to observing the results of this simulation, and even then, as the temperature rises, negative reactivity is added and the process becomes self-correcting in controlling the flux.



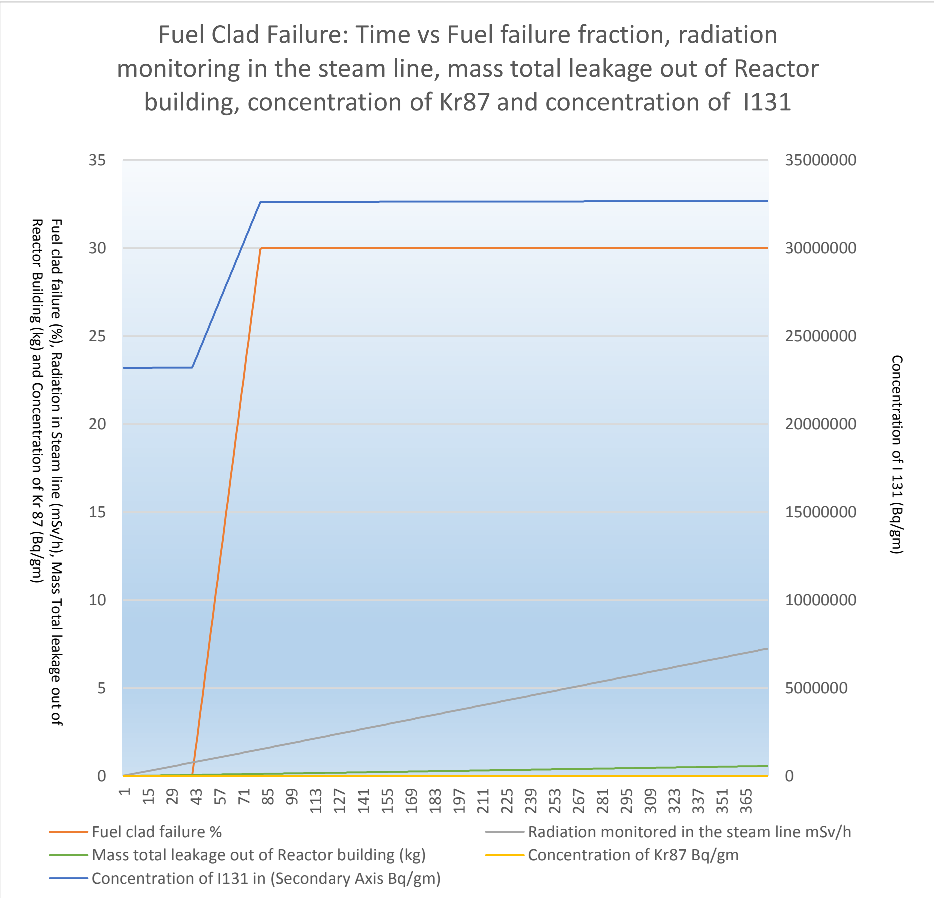
Graph 4.13 Moderator Dilution: Time vs Reactivity Coefficient of Soluble Boron, Moderator Temperature Reactivity Coefficient, Total Reactivity Coefficient and Power Neutron Flux

4.7 Fuel Clad Failure

The final transient to be inserted into the PWR reactor core in cold shutdown condition, is the failure of the Zirconium alloy fuel cladding. As far as this is concerned the failure rate injected is completely beyond design base at 30% or 12434 rods out of 41448 rods. The reason for this extreme choice is that the effect of this transients is very small compared to other transients in other operating modes.

4.7.1 Hypothesis and experimental results

It is hypothesized that this transient will have little to no effect on the primary or secondary system pressures and temperatures. It is expected though that there will be an increase in the levels of radioactive isotopes, like I131 and Kr87 in the steam generator feed water and the mass total leakage out of the reactor building.

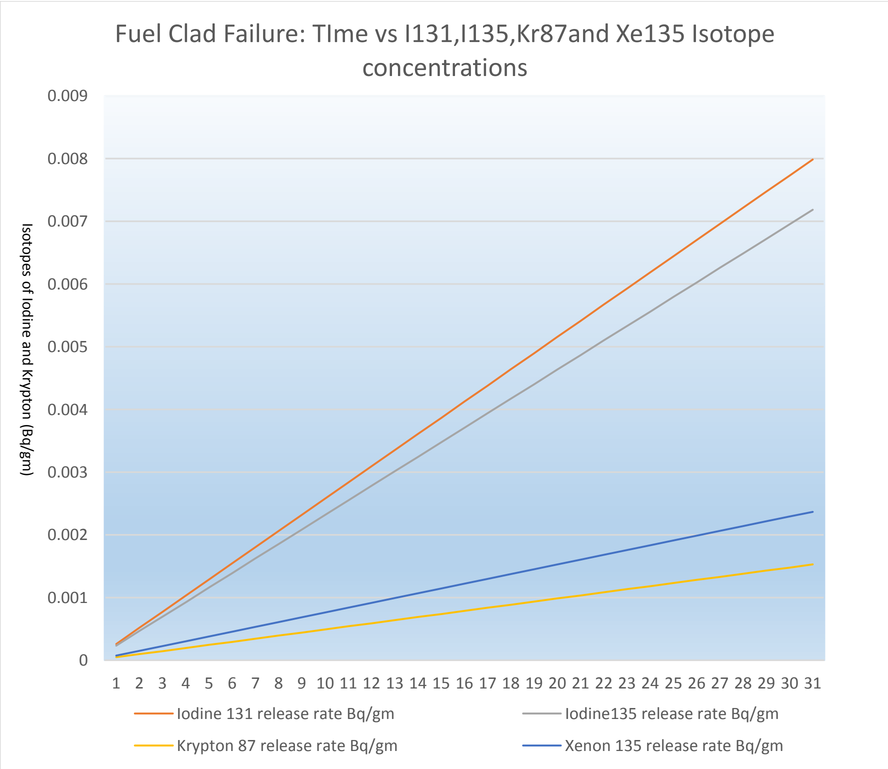


Graph 4.14 Fuel clad failure: Time vs Fuel failure fraction, radiation monitoring in the steam line, mass total leakage out of Reactor building, concentration of Kr87 and concentration of I131.

In the preceding graph a 30 percent fuel cladding failure is injected into the core over after a period of 200s, ramping from 0 to 30% failure in the following 200s. The concentration of I131 (secondary axis) increases linearly as the fuel continues to fail and once failure is complete the I131 concentration has increased by 9.46e^{6}Bq/gm in 3 minutes and 20 seconds.

4.7.2 Analysis

The mass total leakage out of the Reactor building increases from 0 to 0.57Kg over the time period of the simulation and is continuing to rise linearly as the decay process continues. The Thyroid and whole body dose rates shows no change from start to finish and hence holds no danger. Concomitantly, the radiation in the steam line increases from 0.76mS/hr to 7.2mS/hr or by nearly a factor of 10 during the simulation period. It continues to rise linearly at the end of the simulation. Regardless, said radiation rise is contained in the primary system and holds no threat as such.



Graph 4.15 Fuel clad failure: Time vs I131, I135, Kr87 and Xe 135 isotope concentrations

Graph 4.15 shows the increase in the concentrations of some Isotopes of interes; all follow a linear trend with the two Iodine isotope concentrations increasing at 3.25 times the rate of the Xe135 and Kr87.

Regardless of the preceding, the levels of radiation holds no threat to humans. Again, it must be borne in mind that the number of rods with assumed clad failure are inordinate, and in reality is likely to be orders of magnitude smaller with a commensurate reduction in radiation.

CHAPTER 5: CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions:

The following transients were executed in the 3 loop pressurized water reactor in a cold shutdown condition and the following conclusions can be drawn:

- Loss of coolant accident (LOCA).
- RHR failure without replenishment.
- Inadvertent rod withdrawal.
- Moderator dilution.
- Fuel clad failure.

5.1.1 Loss of coolant accident

The nett outcome in terms of severity was of an academic nature, as the cold and hot legs are at the same temperature and the vertical height of the piercings where the legs penetrate the RPV were also at the same height. Thus identical results were produced regardless of whether it was a hot or cold leg LOCA.

Once the Pressuriser was completely drained, void formation in the core started and increased to the level of the actual RPV piercing, representing 18% of voids. During this time the RHR was able to maintain the RCS average temperature at pre-transient levels. Thus, a large break LOCA in the cold shutdown condition holds no threat to the safety of the reactor, the plant or personnel.

5.1.2 RHR failure without Pressuriser replenishment

There were various transients inserted into the reactor in the cold shutdown condition, but of all of them residual heat removal without replenishment has the most severe repercussions. If the RHR system fails for whatever reason, plant operators will have about 5 hours to re-establish decay heat removal before the laws of physics will render them superfluous to requirements.

Possible reasons for failure of the RHR system are [23]:

- The RHR pumps have become air-bound. Essentially a cavitation condition that could occur if the RCS level was dropped to far.
- Complete AC failure both onsite, offsite and emergency power (Fukushima).
- RHR pump shaft failure with the second train unavailable.

From the analysis, peak clad temperatures slowly follows the peak fuel temperatures as the percentage voids in the RPV increases, and they increase at an identical rate of rise of 0.156°C/min. Once the fuel is uncovered, the peak fuel temperature increases at 25.32°C/min and the peak cladding temperature at 12.84°C/min, hence once the step change occurs, events unfold very rapidly.

By the time meltdown has occurred, the debris in the Core is 34.5% and the containment debris is 433.3t. Hence the reactor has experienced catastrophic failure.

The only course of action to avert a meltdown is to keep the fuel flooded, and to remove the heat at a rate at least equal to decay heat production.

Residual heat removal failure without replenishment, is the transient holding the single biggest risk as to possible meltdown of all the simulated transients, in the cold shutdown condition.

5.1.3 Inadvertent Rod withdrawal

The simulation were carried out with the control rods being withdrawn in 20% batches – from fully in to fully out - and at varying concentrations of soluble Boron in the coolant. With the Boron concentration above 1000ppm, the reactor remains completely stable as the rods are withdrawn with no changes in the overall reactivity and the primary and secondary system pressures and temperatures.

However once the simulation is repeated with the soluble Boron concentration at 1000ppm, once the last rods are withdrawn the neutron flux spikes to 12000% it's levels prior to final rod ejection. Simultaneously the average fuel temperature spikes and the primary pressure rises. The flux spike is short lived though, as the Doppler-effect virtually instantaneously brings the flux under control, after which the process repeats itself as witnessed through the dampening behaviour observed.

This transient is very low risk in terms of possible damage to plant or personnel as the nominal soluble Boron concentration during cold shutdown is 2000ppm and higher. Furthermore, it is very unlikely that all control rods will be withdrawn from the reactor without corrective action being taken.

5.1.4 Moderator Dilution

During this transient simulation all the control rods remain fully inserted into the core. Prior to the start of the simulation the soluble Boron concentration is reduced to 563.7ppm. Once the simulation starts the Boron concentration is further reduced at a constant rate, with a commensurate increase in the reactivity coefficient of the Boron. Once the total reactivity turns positive, the neutron flux spikes to 2577% in 6 and a half minutes. First the fuel and then the moderator temperature increases, and similar to the previous transient, the Doppler-effect has a dampening effect on the neutron flux.

Also similar to the previous transient, the likelihood of moderator dilution in a reactor in cold shutdown being damaging to the plant or being dangerous to personnel, is remote in the extreme. The typical soluble Boron concentration in the cold shutdown condition is in excess of 2000ppm, and as a result there is ample time to detect moderator dilution and to correct the situation. Finally, if detection is delayed due to some reason or the other, the Doppler-effect mitigates the severity of the reaction.

5.1.5 Fuel Clad Failure

The simulation was done completely beyond the design base in order to obtain some interpretable results. Yet, even with an assumption of 30% fuel rod cladding failure, the main reactor parameters remained completely unaffected. The fuel temperature remained constant as well as the pressure. Radioactive release of I131, I135, Xe 135 increases markedly, but into the primary containment. The Thyroid and whole body dose rates did not change at all, hence there is no risk to personnel under the simulated conditions.

5.1.6 Summary of Conclusions

It is clear that – as opposed to most other industries – the term shutdown does not imply completely inert in the case of a pressurised water reactor. Even if it is - as in the preceding - extended to the cold shutdown condition there is a very substantial amount of decay heat produced in the reactor, and it has to be removed and disposed of. Even under the seemingly relatively docile conditions of a cold shutdown, reaction rates can change many orders of magnitude in a matter of seconds.

Of all the transients simulated it is clear that residual heat removal failure without replenishment is likely to have the most severe outcome of any of the transients. LOCAs, usually one of the more severe accidents during power operations, are easily handled by the RHR system. Fuel clad failure holds no real inherent danger, and both moderator dilution and inadvertent control rod withdrawal had to be shifted quite far away from actual cold shutdown conditions, prior to really manifesting negative system responses.

5.2 Recommendations

Based on the preceding conclusions for the 5 transients analysed, it can be recommended that further investigation into the following transients are not warranted:

- Loss of coolant accidents, hot or cold leg, large or small break
- Inadvertent rod withdrawal
- Moderator dilution
- Fuel clad failure

The systems available during such transients are more than capable of keeping the reactor conditions safe and functioning.

However, RHR failure without replenishment does warrant further analyses and it is recommended that all the possible causes of RHR failure are identified and rated in terms of the probability of the mode of failure occurring. Once that is done, the transient should be simulated again using commercial code such as CORYS or MAPPS, where the Reactor can be brought to the defined cold shutdown condition, inclusive of the steam bubble being replaced with Nitrogen. The latter is not possible with PCTran.

Again based on the relative probabilities, it is further recommended that combination transients are injected based on the likelihood of occurring.

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