Effects of thermal stresses on Pressurised Water Reactor nuclear containment vessels following a Loss of Coolant Accident with assimilated containment filtered venting system.

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Abstract

In a nuclear power plant, the last barrier under normal and accident operations is the containment building. This is normally constructed from concrete reinforced with steel bars, which are pre-stressed to enhance the overall capability to withstand thermodynamic stresses like over-pressurisation and high temperatures. The failure of this final barrier will lead to the release of radioactivity to the surrounding environment. To examine the effects of thermo-hydraulic stresses on PWR containment following a LOCA, a model is proposed with simulated scenarios performed at the Koeberg Nuclear Power Station as a case study. The accidents were simulated using the Koeberg engineering simulator to obtain the output data. The scenario for the proposed model correlates the critical mass flow from a double-ended guillotine break to the containment pressure and temperature increase. Different containment filtered venting systems (CFVS) are also investigated in this study as severe accident management systems. CFVS have historically been included in boiling water reactor (BWR) designs, but following the Fukushima Daiichi nuclear accident, they are being introduced as severe accident management systems to manage the threat of containment over-pressurisation in pressurised water reactors (PWR). Finally, the rate of change in containment pressure and temperature is analysed and compared to literature, with the incorporation of a simulated filtered venting system to the PWR containment building.

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General symbols

A	Cross-sectional area, (m^2)
bar	bar, unit of pressure
Bq	Becquerel, unit of dose/radiation level
Cp	specific heat capacity at constant pressure (kJ/kgK)
°C	degrees Celsius, unit of temperature
g	gravitational acceleration (m/s^2)
H	Enthalpy (kJ)
KE	kinetic energy (kJ/kg)
kPa	kilopascal, unit of pressure
kW	kilowatt, unit of power
m	meter, unit of length
M	mass (gases in containment)
MPa	megapascal, unit of pressure
MW	megawatt, unit of pressure
P	pressure (kPa)
ρ	Density (kg/m^3)
Q	heat energy (kJ)
s	second
μSv/h	micro Sievert per hour, unit of dose rate
T	temperature, (°C)
u	water velocity (m/s)
v	vapour velocity (m/s)
W	mass (of flow through break/nozzle, in time increments) (kg)

Acronyms and Abbreviations

ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Based Accident
BWR	Boiling Water Reactor
CFVS	Containment Filtered Venting System
DBA	Design Based Accident
DID	Defence In Depth
IAEA	International Atomic Energy Agency
INES	International Nuclear Event Scale
KNPS	Koeberg Nuclear Power Station
LOCA	Loss Of Coolant Accident
LBLOCA	Large Break LOCA
MVSS	Multi Venturi Scrubber System
NPP	Nuclear Power Plant
OE	Operational Experience
PAR	Passive Auto-catalytic Recombiner
PWR	Pressurised Water Reactor
RCP	Reactor Coolant System, Koeberg trigram
SBO	Station Blackout
SCRAM	Safety Control Rod Axe Man (emergency shutdown of a nuclear reactor)
SSC	Structure, System, Component
TMI	Three Mile Island

1. Introduction

Nuclear power stations generate electricity by using the fission of nuclear fuel, uranium-235 and plutonium-239, to generate heat and in turn produce steam which is directed to the turbine. Radioactivity is the emission of ionizing radiation or particles caused by the spontaneous disintegration of atomic nuclei and therefore a by-product of the fission process. This radioactivity must be contained to protect the personnel, public and the environment from radioactive exposure [1]. In a PWR, similar to Koeberg Nuclear Power Station (KNPS) this radioactivity is contained by means of three barriers namely the fuel cladding, Reactor Coolant System (RCP) which is the primary system boundary and the containment building vessel.

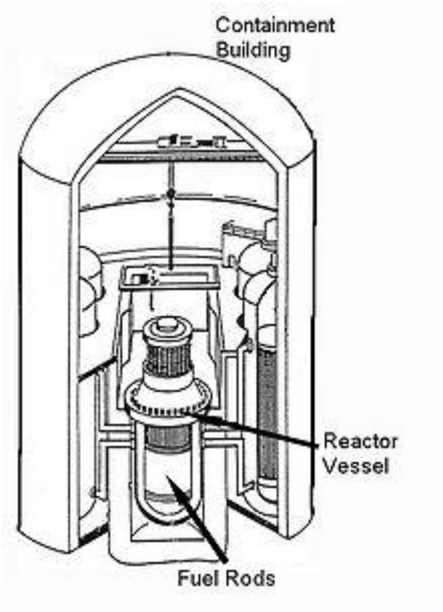

Figure 1: Illustration of the three radiological barriers at a PWR NPP *[2]*

The first barrier is the fuel cladding which is made of zirconium alloy and contains the fission products and gases within the cladding during normal plant operation. It can withstand temperatures of up to 1850 ^{o}C and a pressure of 241 MPa [3].

The second barrier is the RCP primary system boundary. This includes the RCP coolant which is chemically controlled water used to transfer heat from the fuel to the steam generators, and is maintained within the piping and components. If there is a leak on the first barrier, the fuel cladding, the radioactivity can be contained or confined to only the second barrier, the RCP primary system boundary.

The third barrier, the containment building, is considered to be a sealed vessel which can contain its contents up to a specified temperature and pressure. This refers to the containment structure made from reinforced concrete that is one meter thick and designed to contain the radioactivity in the event of an accident. The containment building is designed to withstand the impact of an airplane crash/missile. There are different designs of containment building structures with the main purpose of containing the radioactivity during normal operation and accident conditions.

The general design of the containment building includes a cylindrical, steel lined, concrete pressure vessel with a hemispherical top closure and stepped flat bottom. The building encloses the reactor coolant system with the principal function of containing the mass and energy of the reactor coolant in the postulated event of a rupture in the reactor coolant piping (i.e., an assumed Loss of Coolant Accident (LOCA)).

The design of the containment structure reflects a trade-off between internal volume, design pressure and cost. Current designs contain approximately two million cubic feet (56634 m^3) of free volume and are designed for an internal gauge pressure of 4 bar [4]. Peak pressures for the maximum assumed LOCA can reach up to 4 bar and can occur within 1 minute. [1]

Past operational experiences (OE) and lessons learned, in particular, with the recent accident at the Fukushima-Daiichi Nuclear Power Plant, generated the need to evaluate and reflect on current and developing nuclear power plant designs. Some of these considerations are, but not limited to, the efficacy and adequacy of current acceptance criteria to safety related systems concerning the Structures, Systems and Components (SSC).

During accidents, the last barrier to the radioactive product release into the atmosphere is the reactor containment building. All nuclear facilities are designed to manage design based accidents which are sometimes referred to as severe postulated accidents with, safety design systems like containment sprays, coolers, safety injection and Passive Auto-catalytic Recombiners (PAR), etc. When these above-mentioned safety design systems fail or malfunction during a severe accident or beyond design events, like with Fukushima-Daiichi, core degradation or damage is inevitable. Hydrogen would form in the reactor pressure vessel mainly due to high temperature zirconium-steam reactions and can be released together with steam into the containment building where it will mix with the containment atmosphere. The residual heat of the radioactive isotopes combined with the hydrogen mixture continues to increase pressure and temperature for extended periods. This hydrogen mixture poses a big safety concern as it can lead to an explosive atmosphere and the containment building may lose its structural integrity due to over-pressurisation which will cause uncontrolled radioactive releases to the environment.

Containment building structures undergo many stresses like thermal and pressure stresses. This dissertation will be evaluating the effects of thermal stresses on a containment building in the event of a Large Break LOCA in a PWR. The value of an added containment filtered venting system (CFVS) during this simulated scenario will also be evaluated.

1.1 Problem Statement

There has been a shift in the focus of nuclear safety following the Fukushima-Daiichi nuclear power station incident in 2011 especially with respect to beyond design based accidents and being prepared for them by adding additional measures like containment filtered venting systems. This motivated the investigation into ensuring the containment integrity following a postulated severe accident, i.e. LOCA.

With a delay or failure of the designed safeguard system auto-actions, can the containment building heat and pressure absorption load in a severe postulated accident, like a Large Break Loss of Coolant Accident (LBLOCA) in a PWR system withstand and maintain its design purpose?

This will be analysed using the Koeberg Nuclear Power Station as a case study.

This dissertation will investigate the value of including a CFVS during these LBLOCA simulations in a PWR containment system. Different CFVS designs will also be evaluated and compared.

1.2 Research Objectives

The aim of the dissertation is to evaluate the performance of a PWR nuclear containment building and offer a deeper understanding of the response to a LBLOCA. This will be attained by pursuing the main objectives of this study, which are to:

1. Understand the basic construction of a PWR nuclear containment building and its purpose.
2. Develop a thermodynamic model following a LOCA to simulate the pressure and temperature changes in the containment building.
3. Evaluate the types of CFVS and their impact on the model.
4. Examine the advantages and disadvantages of CFVS options.

This dissertation will not be a justification for the use of a CFVS at the KNPS, nor will it cover the criteria for the selection of any CFVS, but rather investigate the inclusion to a PWR system as a severe accident management tool and to inform of the options available and in use around the world at present.

1.3 Background

Both PWR and BWR type reactors employ the use of containment structures even though the designs differ between the two. In all cases, the containments are designed to sustain any static or dynamic loads that may be caused. This was not always the industry norm especially looking back to the Chernobyl incident of April 1986 with an RBMK type reactor which did not have a containment building or structure. The role of containment structures in the accident that happened at TMI (Three Mile Island) in March 1979 showed that the radioactivity can be contained within these containment structures limiting the harmful effects of radioactivity releases to the public and the environment.

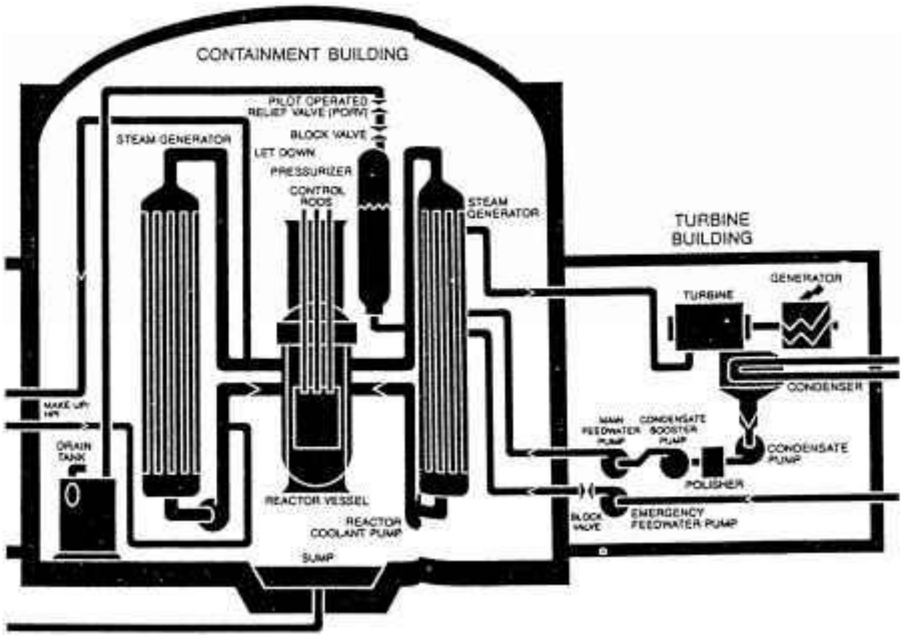

Figure 2: TMI containment vessel *[5]*

Some containment structures are:

- Steel spheres;
- Double wall reinforced, unlined or with a steel or epoxy liner;
- Double wall reinforced and pre-stressed concrete, unlined or with a steel or epoxy liner.

The design pressure and temperature are the two primary parameters used for determining the size of the containment structure. Thus, the design must be able to successfully manage the release of material and energy in excess of the design limits. This peak pressure should be determined on the basis of conservative assumptions in relation to the thermo-hydraulic characteristics.

1.3.1 Containment structural design

Numerous types of containment structure designs are in operation to date used by various utilities worldwide. The designs commonly used incorporate steel vessels or concrete vessels lined with steel plates. These steel vessels can be cylindrical or spherical in shape. Whereas, reinforced concrete vessels may occasionally be post-tensioned cylindrical shaped with hemispherical domes. The type of structure chosen by a utility is dependent on plant layout, site characteristics, and economic feasibility to alternate technology [6].

A typical Westinghouse pre-stressed concrete containment design is shown in Figure 3.

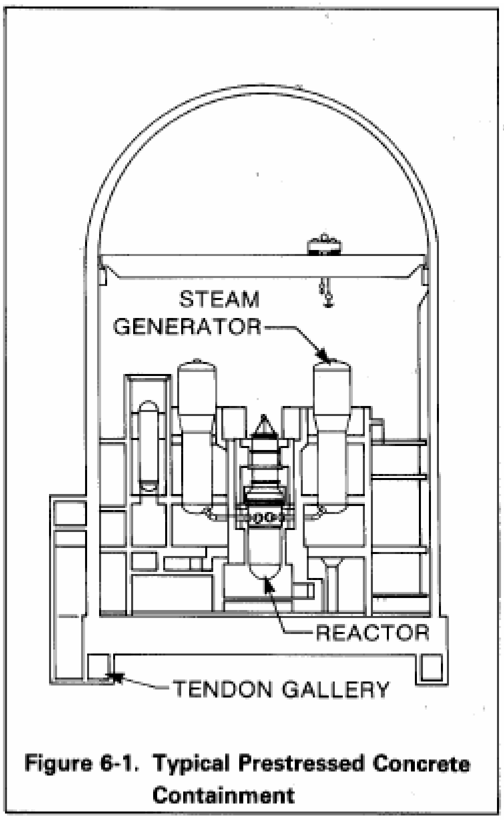

Figure 3: A typical pre-stressed concrete containment *[6]*

Figure 4 shows a typical cylindrical steel containment.

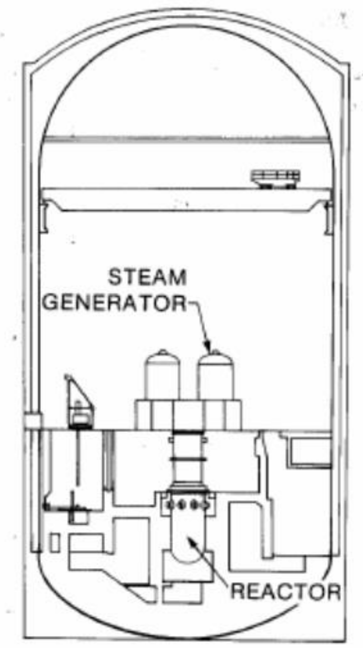

Figure 6-2. Typical Steel Cylindrical Containment

Figure 4: A typical cylindrical steel containment *[6]*

The inside face of the concrete shell, dome, and floor are steel plate lined to ensure a high degree of leak tightness. The cylindrical shell is pre-stressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a two-way post-tensioning system. There are three buttresses equally spaced around the containment. Hoop tendons are anchored at buttresses 240 degrees apart, bypassing the intermediate buttress. Each successive hoop is progressively offset 120 degrees from the one beneath it. Another possible tendon arrangement includes U-tendons and hoops. A third system includes helicoidal tendons in opposite patterns. The foundation base slab is a concrete structure conventionally reinforced with the high-strength reinforcing steel. A continuous access gallery is provided beneath the base slab for the installation and inspection of vertical tendons *[6]*. The base liner of this design is installed on top of the structural slab covered with concrete for post-tension. The containment completely encloses the entire reactor and primary system and certain auxiliary safety systems.

A pre-stressed concrete containment having a cylindrical shell, a hemispherical dome, and a flat base slab (with a pit) can be seen in Figure 5.

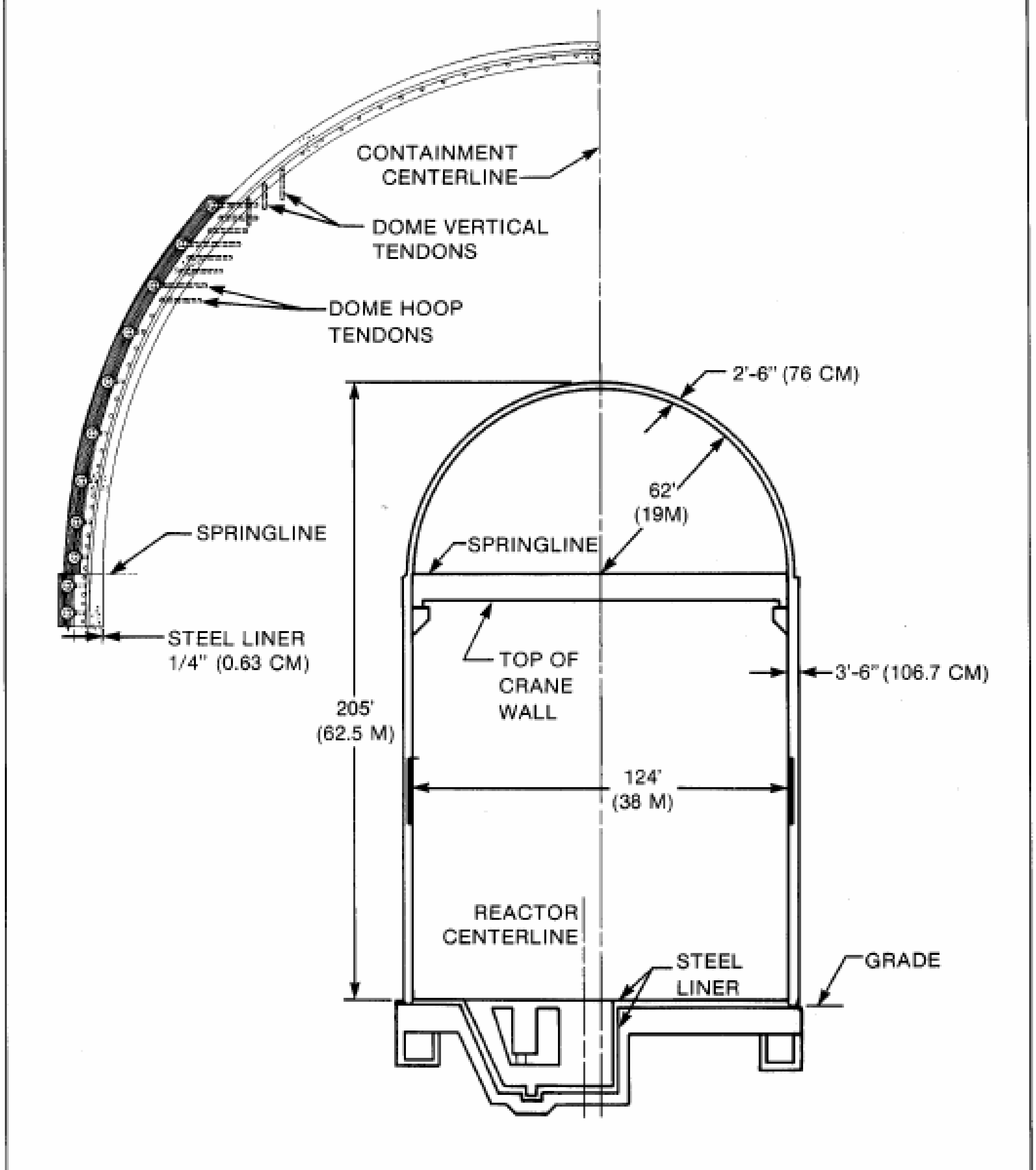

Figure 6-3. Typical Prestressed Concrete Containment, Construction Details

Figure 5: Typical pre-stressed concrete containment *[6]*

The BWR containment structure includes a dry well and a wet well, where a PWR does not contain any wells. A simplistic diagram can be seen in Figure 6 below.

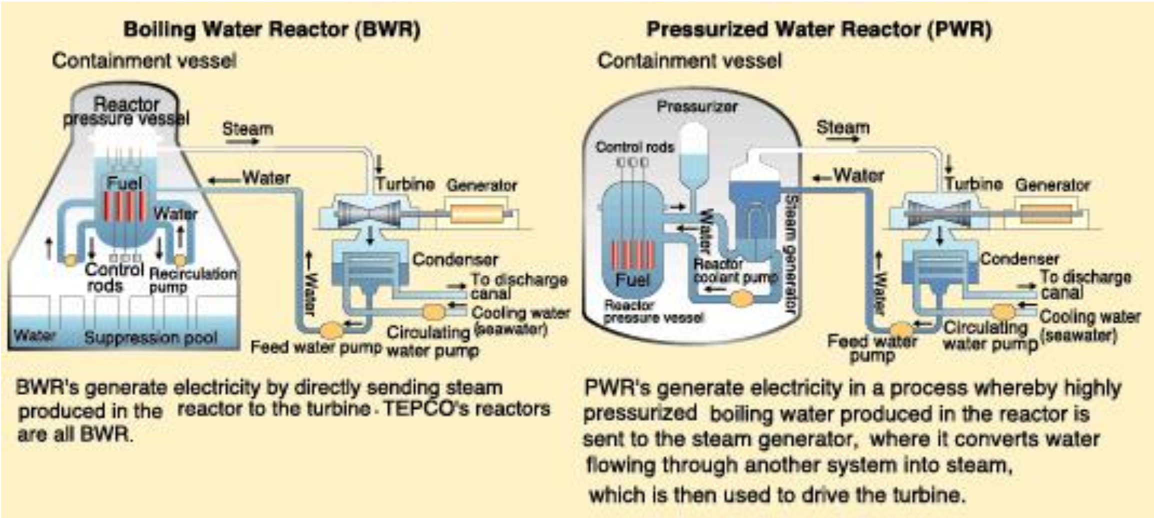

Figure 6: BWR and PWR containment vessels [7]

These containment structures have many functions which include:

- Radiation shielding;
- Provisions for personnel accessibility;
- Anchorage, support and protection of the reactor tank system and other equipment;
- Resistance to jet, pipe whip, and loads produced by emergency conditions;
- A boundary of wet-wells and pool structures, allowing communication between drywell and wet-well (Mark II/second generation designs),
- Lateral stability for the containment;
- Transfer the containment loads to the foundation.

The loading on the containment structure includes:

- Temperature transients and gradients,
- Safe shutdown earthquake loads,

- Internal (and external) missiles,
- Mechanical loads from pipe rupture,
- External pressures,
- Winds and tornadoes.

Particular severe accidents considered for the containment buildings include:

- Over-pressurisation
- Dynamic pressure (shock waves)
- Internal missiles
- External missiles
- Melt-through
- Bypass (equipment failure leading to the reactor coolant being released into the peripheral buildings or directly into the environment)

Some containment structures have special features like vents for containment venting, if required, for controlled venting of gases or vapour to the atmosphere. This is to ensure that the containment structure integrity is maintained in case of a hydrogen release or buildup of steam pressure inside the containment structure. This is predominantly utilized in BWR systems. When hydrogen is released in the containment building, it can create an explosive mixture with air and cause an explosion which may destroy or damage the containment building structures.

The scenario of the accidents that occurred at Chernobyl and Fukushima Daiichi will briefly be discussed below with the main emphasis on the role of the containment building vessel.

1.3.2 Chernobyl Accident

The Chernobyl Power Station situated in Pripyat, approximately 16 km northwest of the city of Chernobyl in Ukraine. The station consisted of four light water boiling type reactors, each capable of producing 1000 MW of electrical power. The Chernobyl nuclear accident happened on April 26th, 1986 as a consequence of two subsequent explosions in one of the four reactors of the Chernobyl

Nuclear Power Plant (NPP). Explosions were caused by a few factors, but mostly by human error and technical deficiencies. Chernobyl utilized the RBMK-1000 boiling water type reactors and below are some of the design features that influenced the sequence of the accident [8]:

- Positive void coefficient of reactivity
- The design of the control and safety rods
- Speed of insertion of the emergency protection rods
- Control of power
- Instrumentation indicating the reactivity margin
- Size of the reactor core
- Capability to alter or bypass safety systems, plant trips and alarms
- Sub-cooling of the inlet water
- Primary coolant system
- No Containment building

Investigation and analysis indicate the onset of the accident was initiated from an after-hours safety test which simulated a station blackout power failure scenario. During the test, the safety systems were intentionally turned off. The engineers at Chernobyl wanted to determine the duration and amount of power a spinning turbine would provide at lower reactor power to certain systems in the plant before the backup diesel generators start up [9].

This poorly designed experiment coincided with the design imperfections and the lack of experienced trained personnel led to the chain reaction in the core to run out of control. Several explosions occurred and caused severe damage to the steel and concrete lid/roof of the reactor. With the reactor containment damaged, a large amount of radioactive material was released into the atmosphere, where it was carried further by the wind and air currents in a very short time period. In addition, high temperatures in the reactor caused further melting of the remaining fuel, which caused prolonged radioactive emissions for weeks after the accident. It was estimated that the radioactive cloud released at the time of explosion was expelled kilometers high in the atmosphere. This was later carried by the winds over many European countries, first over Scandinavian countries, then changing direction of the air currents contaminated Poland, Czechoslovakia, southern parts of Germany and Austria. Finally, South and Southeast winds accompanied by rain showers led to contamination of Balkan countries [10]. It is approximated that the total amount of radioactive release was 5300 PBq (10^{15} Bq), 200 hundred times higher than radioactivity released in Hiroshima with most of the radioactive fallout dispersed over Belarus, Ukraine, and Russia.

1.3.3 Fukushima Daiichi Accident

Brief progression of the Fukushima-Daiichi accident that led to the containment failure:

On March 11, 2011, a severe earthquake occurred off the coast of Japan with the epicentre about 180 km away from the Fukushima Daiichi NPP. This generated a tsunami which caused devastating damage over the whole nuclear site, composed of six BWRs. Three of the six reactors were in operation at rated power output before the event, while the other reactors had been in an outage.

It was noted that the automatic shutdown protocol (SCRAM) worked as it was designed. The earthquake and tsunami caused a loss of all offsite power. Due to the loss of all external electric power sources, the emergency diesel generators automatically started, but with the tsunami impact, all the operational diesel generators were lost including the back-up batteries due to flooding. The lack of AC or DC power supply to the vital operation of decay heat removal systems caused the fuel of the operational nuclear reactors to suffer total and partial melting damage.

A large amount of hydrogen was released as a consequence of the zirconium-steam reaction, resulting in strong explosions damaging the containment buildings of some of the units. This depiction is illustrated in Figure 7.

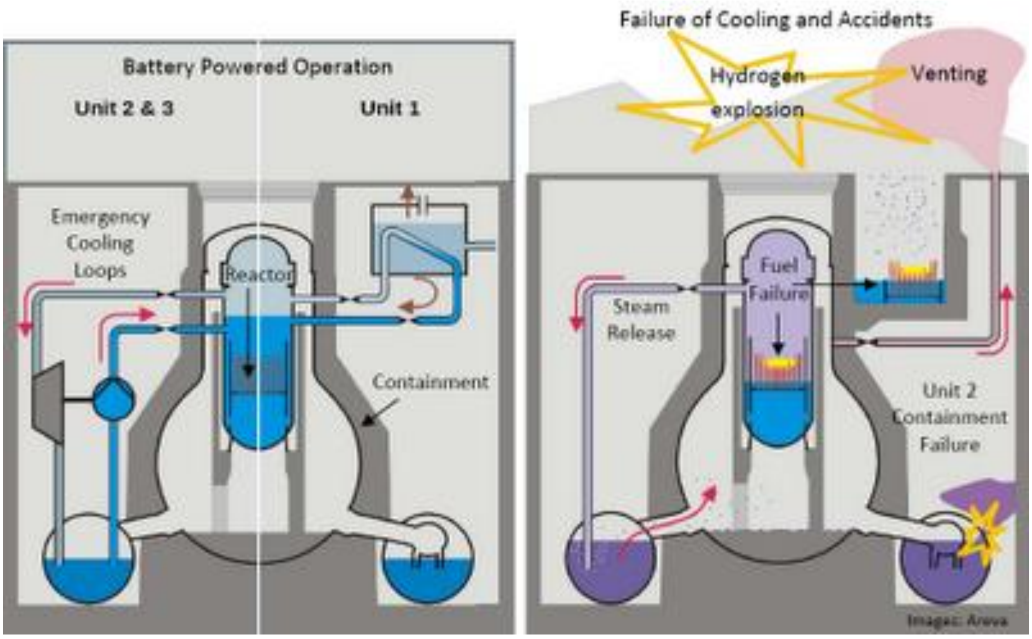

Figure 7: Fukushima Daiichi containment building before and after a hydrogen explosion *[11]*

The Fukushima accident occurred due to the failure of the cooling system at the nuclear power station. This, as a resultant from the 4th largest earthquake recorded in the world since 1900 [12] and the subsequent formation of high tsunami waves that hit the Fukushima NPP which damaged the electrical network and emergency diesel generators on site. This lack of electrical power supply implied the loss of all forced circulation or cooling of the nuclear reactors and spent fuel pools.

In contrast to Chernobyl, the Fukushima reactors were equipped with a concrete containment building. The explosions at Fukushima were solely of a chemical nature (hydrogen explosions) and affected the reactor containment buildings but, based on the best available information, not the reactor pressure vessels or the reactors themselves. The release characteristics were different from the Chernobyl accident. Releases of only gas phase radionuclides occurred in the course of venting operations to relieve the over-pressure inside the vessel which took place approximately one day after the shutdown. In comparison to the uncontrolled, continuous releases of Chernobyl with peak releases at the beginning of the accident, the venting operations at Fukushima NPP happened in pulses over a time period of more than a week, and were often conducted under advantageous weather conditions.

From the beginning, the Chernobyl accident was proclaimed as a major accident, according to the International Nuclear Events Scale (INES) level 7 event, which is the largest scale accident, which endangered the environment and population. The Fukushima Daiichi accident did not arise suddenly, but developed over several days. This accident resulted in the highest number of human casualties and destruction in Japan since World War II. Problems on the NPP began after the devastating earthquake of magnitude 9 and tsunami which followed. TEPCO, the operator of the Fukushima NPP, in June 2011 reported to the IAEA estimating the quantity of radiation released into the atmosphere by the accident to be approximately 15 percent of the radiation released from the Chernobyl incident [12].

1.3.4 Westinghouse 3-loop PWR type power plant design

Nuclear technology and the use of nuclear reactions to produce electricity in NPP were established in the early 1950's. The most common types comprise the use of enriched uranium isotope (U-235) with light water as moderator and coolant. Two designs exist for the method used with enriched uranium, namely the PWR and the BWR type. The other concept utilises natural uranium with heavy-water (water enriched in the deuterium isotope) as coolant and moderator. Fundamentally, all designs utilise the release of nuclear energy to generate heat, which is used to produce steam which is then used in steam turbines to produce electricity.

Some NPP like KNPS utilises the Westinghouse 3-loop PWR plant design. The Westinghouse PWR power producing operation is comprised of three independent closed cycles or loops which includes the primary, secondary, and tertiary loops as can be seen in Figure 8 below.

The primary loop contains the nuclear fuel core, primary pumps and pressuriser. The heat source consisting of the nuclear fuel core positioned within a reactor vessel. The energy generated from the controlled fission reaction is converted into sensible heat in the moderator. The moderator/coolant is pumped to the steam generator where the heat is transferred to a secondary loop through a number of U-tubes, depending on the steam generator design. The reactor coolant then returns to the reactor vessel. The pressuriser connected to the primary loop maintains a pressure above the saturation pressure by means of electrical heaters and cooling water sprays to prevent bulk boiling.

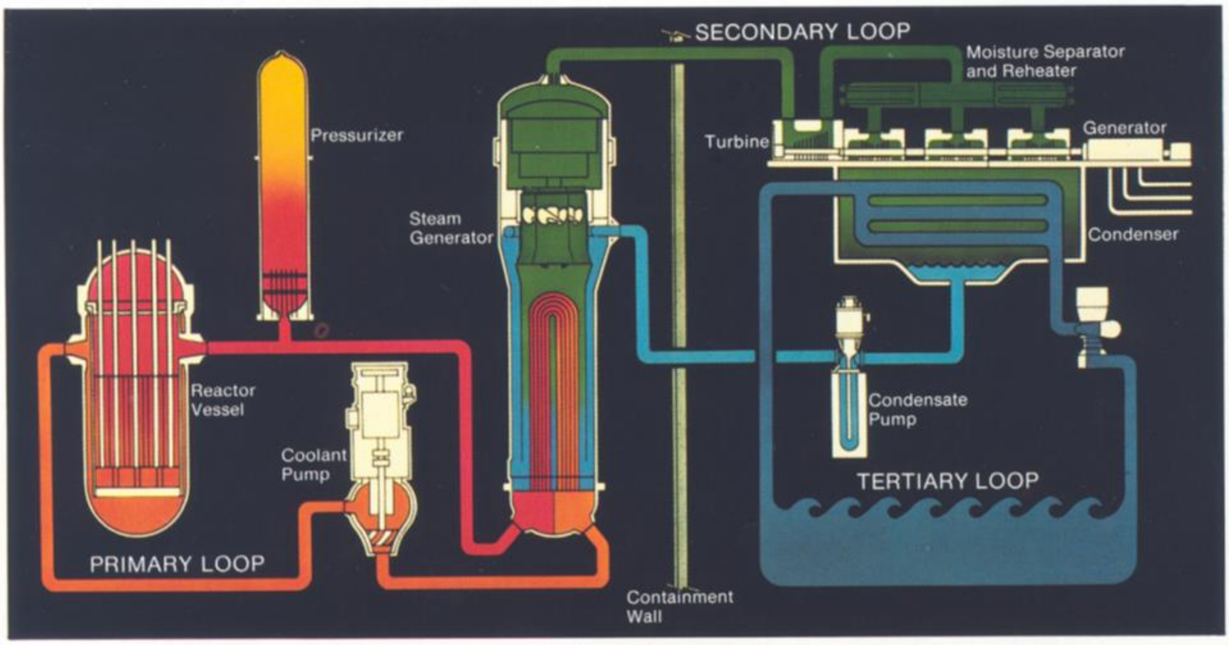

Figure 1-1. Westinghouse PWR Nuclear Power Plant, Simplified Schematic Diagram

Figure 8: Simplified Schematic Diagram of Westinghouse PWR Nuclear Power Plant *[6]*

The secondary loop is the heat utilization cycle where dry steam produced in the steam generator flows to a turbine-generator where it is expanded to convert thermal energy into mechanical energy and produce electrical energy. The expanded steam exhausts to a condenser where the latent heat of condensation is transferred to the tertiary cycle cooling system and is condensed. The condensate from the condenser is pumped back to the steam generators. The tertiary loop is the heat rejection loop where the latent heat of condensation is rejected to the environment through the condenser cooling water [6].

As stipulated by the International Atomic Energy Agency (IAEA) the safe operation of all nuclear installations is fundamental to protect and promote the peaceful use of nuclear energy by the principle of highest nuclear safety. This is enforced and implemented by most nuclear installations with the incorporation of “Defence in Depth”. The concept of Defence in Depth (DID) limits the likelihood and the consequences of potential hazards by the implementation of prevention, control and mitigation measures to ultimately protect both the workers and the public [13].

Defence in Depth consists of an ordered deployment of different levels and redundancy of equipment and procedures in order to maintain the effectiveness of physical barriers placed between radioactive materials, workers, and the public or the environment. This must be ensured in normal operation, anticipated transients and accident conditions at the NPP.

2. Literature review

Since the inception of utilising nuclear power as a commercial energy source, safety and the safe design, construction, operation and decommissioning of these installations was highlighted as of primary importance. In light of this overriding concern for safety, dismissing the possibility of an accident resulting in the release of radioactivity to the environment has been thoroughly investigated and analysed. Probabilistic risk assessments indicate that accidents of this nature are highly improbable but not negligible with severe consequences. The accident at Three Mile Island Unit 2 demonstrated that significant core melting can occur in a commercial nuclear power reactor after losing the first of the three physical barriers of radioactivity. The reactivity-driven explosion at Chernobyl showed that all three barriers can be breached and the public exposed to high levels of radioactivity [13]. A severe accident as with Chernobyl requires the failure of multiple safety systems and the breach of all three physical barriers to the release of radioactivity: fuel cladding, reactor cooling system, and the containment.

To maintain the integrity of the containment building, it is important to mitigate a severe accident involving significant core degradation, where the melted core could react with coolant and structures, thereby preventing the continuous generation of steam, hydrogen and other radioactive gases. Eventually, the containment building would be compromised by overpressure due to the release of these gases if the designed safety systems do not operate as intended [14]. One of these severe accidents is a LOCA. A LOCA is a Design Based Accident (DBA) where the primary system inventory is lost as a result of a break in the primary piping. The consequences of a LOCA are potentially catastrophic and the mitigation thereof necessitates the employment and incorporation of multiple emergency safeguard systems. During accidents, the last barrier to the radioactive products before release into the atmosphere is the containment building. The IAEA for this reason requires from all NPP operators to design their containment system to ensure or contribute to achieving the following safety functions as a minimum requirement [15]:

- Confinement of radioactive substances in operational states and in accident conditions,
- Protection of the plant against external natural and human induced events,
- Radiation shielding in operational states and in accident conditions.

Thus, protecting the last barrier becomes fundamentally important in all operating conditions. Two of the most critical installations and enhancements to nuclear containment building vessels for incident management are containment water spray systems and CFVS.

2.1 Containment spray system

The main function of a containment water spray system is to reduce the temperature, pressure and airborne contamination inside the containment building, following an accident which may challenge the containment integrity. The containment spray system provides a high-pressure, finely divided water spray to the containment atmosphere located in the top parts of the containment building. Heat transfer to the droplets and subsequent condensation of atmospheric steam can produce rapid reductions in temperature, pressure, aerosol and fission product concentrations [13]. The spray droplets, as well as much of the condensate, will collect in a sump at the bottom of the containment. Generally, the initial spray water is taken from a water storage tank. When this source is exhausted, water is pumped from the sump, through a heat exchanger, back to the spray nozzles.
Westinghouse designed a containment spray system with 2 independent trains with redundancy in mind to limit offsite and site boundary doses following the unlikely event of a LOCA [16]. Redundancy refers to the use of more than one or the minimum requirements to fulfil a given safety function, for example the minimum number of pump sets or equipment to fulfil a given safety function. This is an important design principle for achieving high reliability in systems important to safety, and eliminating the single failure criterion requirements for safety systems. Redundancy permits the failure or unavailability of at least one set of equipment to be tolerated and allowed without loss of the system function [17]. In addition, the containment spray system can also be used for containment fire protection for widely spread fires inside the reactor building. This use is however limited to certain reactor power levels.

Borated water supplied to the containment spray system is pumped via independent trains to the spray headers inside the containment building, where it is atomized through spray nozzles. This atomizing spray increases the water surface area available for heat transfer. The fine water mist/droplets also provide nucleating sites to condense any steam generated. As the spray droplets absorb heat, the temperature of the saturated steam decreases and the containment pressure decreases.

The water supplied to this system can be chemically treated or dosed for multiple reasons. At KNPS tri-sodium phosphate (TSP) is added to the containment spray system to maintain the pH above 7. The Material Safety Data Sheet (MSDS) for TSP can be found in Appendix B. The benefit of maintaining the pH above 7 includes:

- Ensuring the released iodine remains in solution, and prevents any significant release of iodine from the sump during recirculation. (Note: Iodine is produced in the fission reaction from the parent product Tellurium (^{132}Te). The iodine remains in the fuel element during normal operation due to containment of the fuel matrix and fuel cladding.)
- The long-term corrosion rate of galvanized steel, aluminium and zinc-based coatings is minimised.
- The minimum pH of 7 is recommended to protect stainless steel from chloride stress corrosion cracking.

The containment spray system can be placed in service either manually or automatically when containment pressure increases to a predetermined setpoint of 240 kPa gauge, in the KNPS case. It is ultimately responsible to protect the integrity of the containment structure from the threat of failure, due to over pressurisation from the energy released by the primary system in the event of a pipe break (LOCA).

2.2 Containment Filtered Venting Systems (CVFS)

CFVS is an accident management system designed to minimise and control the release of fission products when relieving the pressure of the containment building in the case of a severe nuclear accident. The basic idea of a filtered venting system is to open a controlled flow path via a radioactive nuclide filtration unit to the external environment in order to relieve the build-up of pressure caused by steam and non-condensable gases generated inside the containment building structure. By doing this, it is possible to delay or prevent structural failure of the containment building. It also provides additional time to mitigate the accident and reduces the offsite consequences compared to those produced by containment failure [18]. A filtered venting system is designed to remove the exhaust produced in the containment building during a postulated core melt accident. In this case, the exhaust could be composed of gases of H_2, O_2, N_2, steam, CO, CO_2 and radioactive aerosols. There are two main types of CFVS:

- Dry, which utilizes a gravel or sand-bed filter,
- Wet, which uses a scrubber in a liquid solution.

Both allow the removal of most of the aerosols (including cesium, >99.9%) and iodine (>99% in elemental form) contained in the vented atmosphere [19].

Different kinds of filtered venting designs were investigated in the 1980s with different filtered venting strategies following the Chernobyl accident. Several European countries installed containment filtered venting systems with no consistent set of technical specifications. This was all based on the nuclear operator requirements. Similarly, after the Fukushima Daiichi accident the needs analysis highlighted the importance of the containment building function, and containment filtered venting to be considered for severe accident conditions. This was based on the results of the national stress tests conducted and submitted by the end of 2012 with binding action plans to increase the robustness of their nuclear plants against beyond design events. [20]

Young *et al.* [14] modelled a Station Blackout (SBO) accident scenario on a South Korean designed two-loop 1000 MWe PWR Generation II nuclear reactor (OPR-1000), developed by Korea Hydro and Nuclear Power (KHNP) and Korea Electric Power Corporation (KEPCO). An accident analysis for the OPR-1000 under a SBO was conducted using the MELCOR computer code. An SBO accident is where onsite and offsite electrical systems do not work or are lost. It is assumed that the emergency core cooling systems such as High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) are not operational during the accident. At the start of the SBO, the reactor and turbine are tripped, the pumps for the Main Feed Water (MFWs) and Auxiliary Feed Water (AFWs) are stopped, and the Main Steam Isolation Valve (MSIV) is closed.

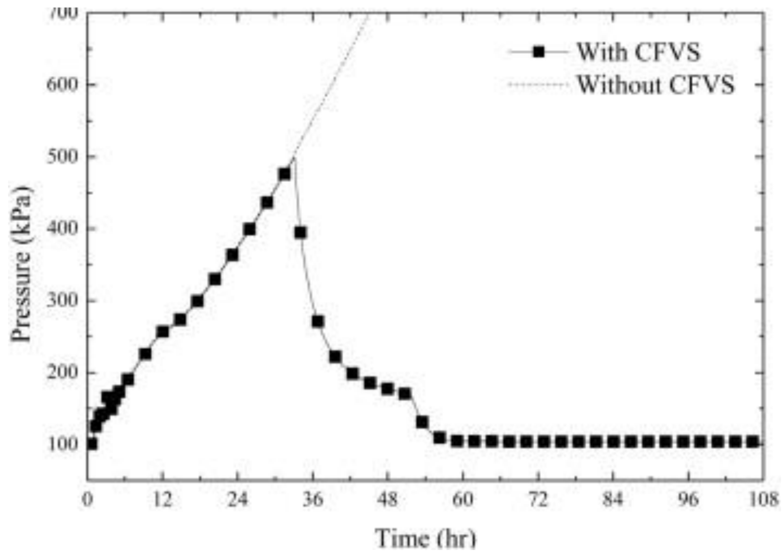

Figure 9: Pressure variations in containment under an SBO accident scenario. *[14]*

The model also incorporated the inclusion of a proposed containment filtered venting system vessel taking suction directly from the containment building. The results of the model can be seen in Figure 9 above, where a solid line indicates the pressure over time transformation in the containment building with a CFVS, while the broken line represents the atmosphere thermohydraulics in the control volume of the CFVS vessel and the water saturation temperature at the pressure of the control volume of the CFVS, respectively. The containment pressure rapidly increases over time without the CFVS or other intervention leading to design and failure limits. It can be observed that pressure increases well above that of the specified 4 bar gauge design pressure of most current containment building designs.

In the scenario where the safety systems are operable at approximately 500 kPa the CFVS system activates passively. It is observed that the pressure and temperature are rapidly decreased and brought under control with the activation of the CFVS. Consideration to factors that will influence the behaviour and the integrity of any nuclear containment building, include:

- Corrosion or any other Aging Mechanisms
- Chemical reactions with CO_2 (carbonation of the concrete)
- Fragility curve of the steel (ASME code for the temperature effect)
- Damage due to impact with internal or external missiles/objects, etc.

In PWR's a postulated LOCA will incite the reactor vessel to depressurise and may cause the core to be uncovered or trigger boiling of the primary system inventory. For this reason, accurate knowledge and understanding of the flow rate of the coolant through a pipe break during a LOCA accident are very important. Similarly, this will be indicative of the rate of containment pressure build up or increase and the selection of reactor safety components.

In a similar study, Sang-won Lee *et al.* [18] analysed and modelled containment over-pressurisation scenarios (both SBO and large break LOCA) using the MAAP4 computer code on an OPR1000 large dry containment PWR in Korea (1000 MWe). Parameters and values for this study are seen in Table 1. In this OPR1000 design, the ultimate containment failure pressure with a 5% probability 95% confidence level was determined to be 1.01 MPa.

Table 1: Initial and boundary conditions for containment over-pressurisation scenarios in Korean simulation *[18]*.

Parameter	Values
Reactor power	2815 MW_{th}
CTMT net free volume	7.73×10^4 m^3
CTMT design pressure	0.5 MPa
CTMT ultimate pressure	1.01 MPa
CFVS opening/closing set-point	0.9 / 0.6 MPa 0.5 / 0.2 MPa
CFVS flow rate	17 kg/s @ 0.9 MPa 10 kg/s @ 0.5 MPa

The CFVS is modelled as a junction/penetration connecting the containment upper compartment to the environment with an assumed flow diameter to be 230 mm (9 inch) and flow resistance through the CFVS is considered. To activate the CFVS, the nuclear facility can opt for either containment isolation valves which will be opened either by the operator (active means) or by rupture disc (passive means) when the set point is reached. A decontamination factor of 1000 for aerosol was assumed in this analysis.

The three cases investigated in this study were:

- Case 1: without venting,
- Case 2: with cyclic venting at severe accident pressure of 0.9 MPa / 0.6 MPa (CFVS-9 case),
- Case 3: with cyclic venting at a containment design pressure of 0.5 MPa / 0.2 MPa (CFVS-5 case).

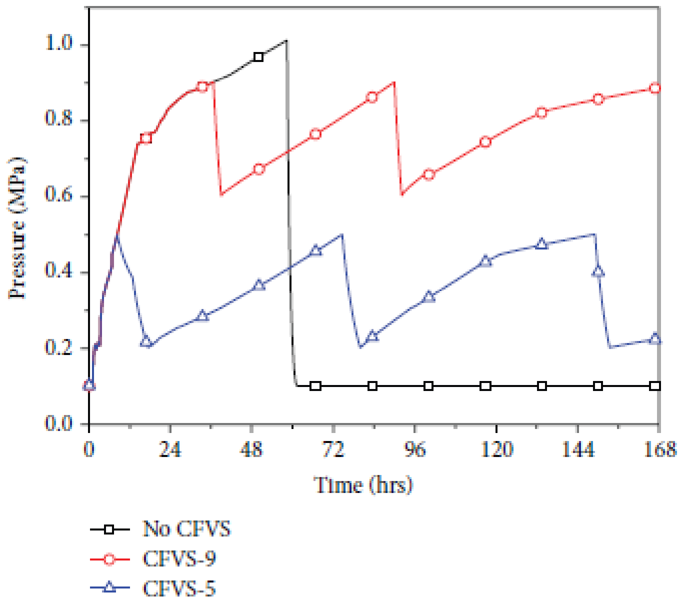

Figure 10: Containment pressure history in LBLOCA and extended SBO scenarios *[18]*

From Figure 10, the results indicate that during the initiating time period, the containment pressure continuously increases due to the break steam flow rate to the containment. Large amounts of steam generated and primary system water flow into the containment vessel. Similarly, hydrogen and non-condensable gases are steadily generated by molten core concrete interaction in the base-mat. The containment pressure reaches its ultimate pressure at approximately 53.8 hrs in Case 1 without venting.

In the venting cases (CFVS-9 and CFVS-5) the initiating scenarios proceed and the containment pressure increases to the point where the CFVS starts to operate. Then, the containment pressure rapidly decreases to the closing set-point of 0.6 MPa and 0.2 MPa respectively.

Radiological release to the environment is reduced to 10^{-3} considering the decontamination factor. In addition, the cyclic venting strategy can reduce the noble gas release by 50% for 7 days [18]. This is important as noble gases, such as radioisotopes of krypton and xenon that are produced by fission that are highly radioactive and is relatively difficult to confine as they occur in gaseous form, are chemically inert, and are relatively insoluble in water. It can be established from the above studies and models that the type of CFVS will affect the thermal load and operating conditions of the particular containment building during and after the postulated accident scenarios. Consequently, the different types and designs of CFVS methods will be evaluated.

2.2.1 CFVS types and designs

As discussed previously there are two main types of CFVS namely a Dry CFVS, which utilises a gravel or sand-bed filter and a Wet CFVS system, which uses a scrubber in a liquid solution.

Wet CFVS systems like the Westinghouse wet-metal design used by all Swedish NPPs take suction from the containment building and are guided by piping system to an opening/inlet into the scrubber tank full of water. The Swedish nuclear fleet utilises a Multi Venturi Scrubber System (MVSS) [21]. The major component is a scrubber system comprised of a large number of small venturi scrubbers submerged in a water pool. Radioactive aerosols and gaseous iodine separate from the containment gas mixture and are contained in the water pool. Venturis are ideally located on different levels and are sealed off by the internal water level. The number of engaged venturis is proportional to the pressure in the containment and so the total relief flow, thus achieving high decontamination factors over a wide flow ranges. The cleaned and non-condensable gases are purged from the scrubber and released via a filter room to the atmosphere. This design is illustrated in Figure 11.

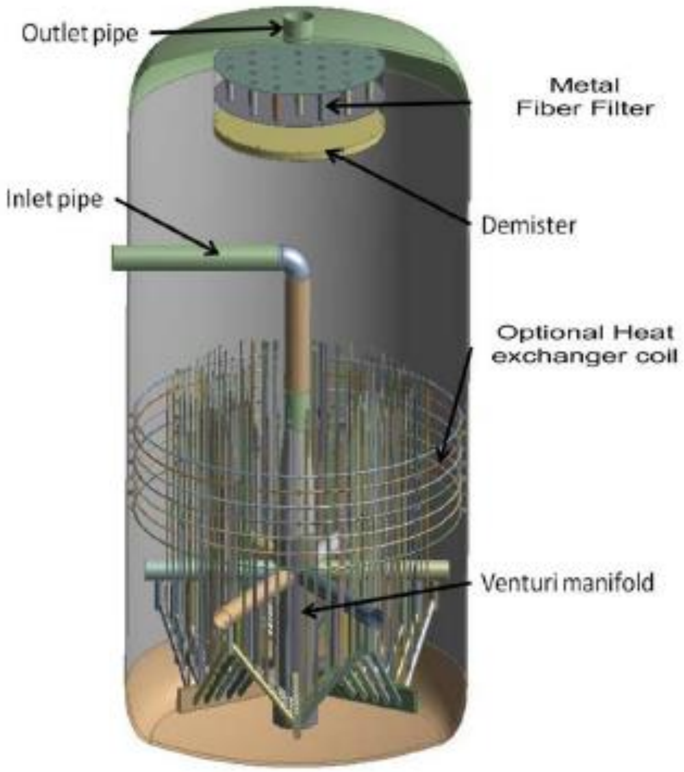

Figure 11: Wet CFVS system design principle *[22]*

The Dry CFVS method is a venting system that can consist of either one type of filter or a combination of two of the following:

- A metal fibre filter that retains airborne radioactive aerosols
- A sand bed filter
- Industrial cartridge filter with pleated sintered fibre
- A molecular sieve with doped zeolite for chemisorption of gaseous radioactive elemental iodine and its organic compounds.

Similarly, the Dry CFVS takes suction from the containment by means of an isolation valve and/or rupture discs and piping system to the inlet of the filter unit. This corresponds to the Westinghouse dry CFVS system with the addition of a separator/pre-filter, preventing water droplets from entering the CFVS. This can be situated inside the containment building. The venting system can be actuated either remotely by opening containment isolation valves or by a rupture disc, depending on regulatory and/or customer requirements. The Dry CFVS method can be seen in Figure 12.

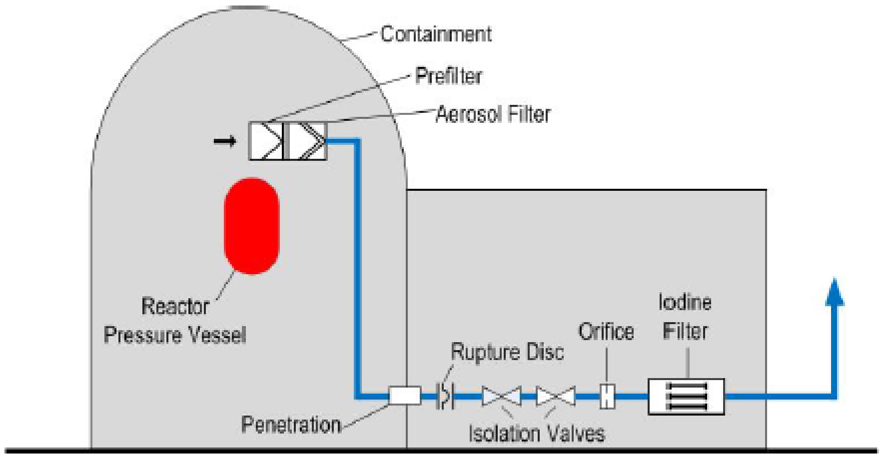

Figure 12: Schematic view of a Dry filter method for CFVS of aerosol filter inside containment *[22]*

Due to the increasing demand for revised nuclear safety culture and OE in the nuclear field over the past 30 years, CFVS developing companies like Areva (the multinational organization specializing in nuclear power) are incorporating designs that utilise both wet and dry CFVS methods. [23]

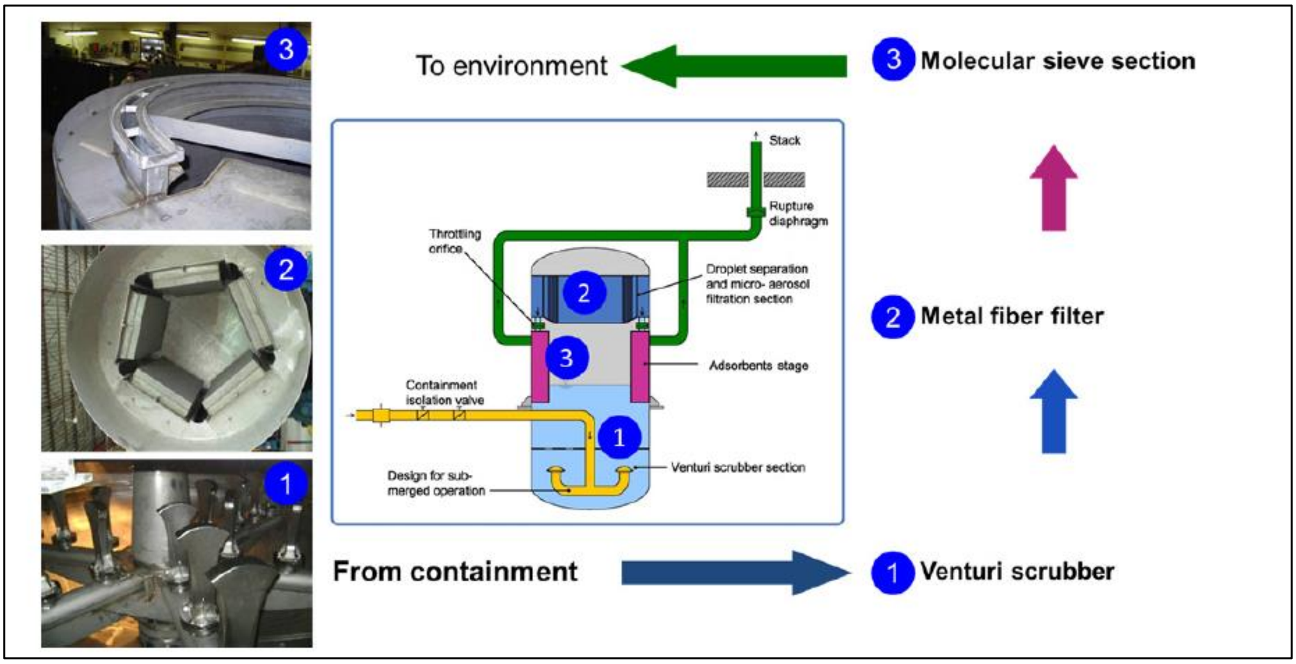

Figure 13: Areva FCVS design *[23]*

In Figure 13, the three stages of the combination type CFVS produced by Areva are distinctly noted. Stage 1, the venturi scrubber stage (wet stage), is comprised of a set of venturi nozzles submerged in a scrubbing liquid. The vented gas from the containment atmosphere enters the first stage where increased dynamic pressure of the vented gas causes the static pressure to decrease. This way it provides adequate absorption time and the bulk mixing of the scrubbing liquid. This stage will retain most of the elemental iodine and large quantities of iodine compounds [23]. Stage 2, metal fibre filter stage (dry stage), is comprised of a droplet separation and micro-aerosol filtration section. This stage will retain exhaust gases downstream of the venturi nozzles, suspended droplets and re-suspended aerosol from the scrubbing liquid. From this section, gases are moved to Stage 3, the molecular sieve section (dry stage) via throttling orifices. In this section, the remaining gaseous organic iodine is removed via adsorption on a molecular sieve adsorbent, which reacts under high temperatures to yield high iodine retention percentages. Subsequent to this stage the treated gaseous waste is expelled via a filter room or stack system.

2.3 Passive Autocatalytic recombination

Following a LOCA, hydrogen gas will accumulate in the containment building, due to the temperature generated within the reactor coolant system. Nuclear stations like Koeberg Nuclear Power Station (KNPS) need to measure the hydrogen concentration in the containment atmosphere, adequately mix the containment atmosphere to avoid accumulation of hydrogen at the top of containment structure and to keep the hydrogen concentration low enough (less than 4,1%) to prevent any risk of ignition of the hydrogen-oxygen mixture. Hydrogen is highly flammable and when mixed with air at certain concentrations could cause an explosion. These flammable regions for hydrogen are between 4% and 75% (burning limits) and between 13% and 60% (explosive limits).

During and following a LOCA, hydrogen accumulation can be the result of production from the following sources:

- Zirconium-water reaction,
- The corrosion of metals caused by the containment spraying solution,
- Radiolytic decomposition of the cooling water (which includes core solution radiolysis and sump solution radiolysis). The radiolysis of water occurs during normal plant operations as follows:

$$2H_2O \rightleftharpoons 2H_2 + O_2$$

After a LOCA, however, the dominant direction of the reaction is towards the formation of H_2 and O_2 gas, prompted by gamma radiation from decaying fission products and other radioactive sources.

Hydrogen is produced when zirconium in the core cladding reacts with steam according to the following equation:

$$Zr + 2H_2O \Rightarrow ZrO_2 + 2H_2$$

The hydrogen produced in this manner is released to the containment atmosphere in the first few minutes after the accident occurs.

To prevent this hydrogen build-up during a LOCA, KNPS and Tricastin NPP have Passive Autocatalytic Recombiners (PAR) installed inside their containment buildings. Detailed hydrogen analysis for a beyond design based accident (BDBA) was performed, together with the analysis of the best suited location for these PAR units at the French station of Tricastin. The basic concept of the PAR recombines hydrogen and oxygen to form water vapour (H_2O) or steam. The catalyst enables the recombination to occur at low temperatures and low hydrogen concentrations and it is self-sustaining until the H_2/O_2 concentration is insufficient to sustain the reaction. The recombination process is an exothermic reaction and this heating causes convection driven flow due to reduced gas density above the catalytic plates relative to the gas density below the catalytic plates.

The PAR units are constructed of stainless steel with a drawer containing the Platinum-Palladium catalyst coated plates at the lower end. This can be seen in Figure 14.

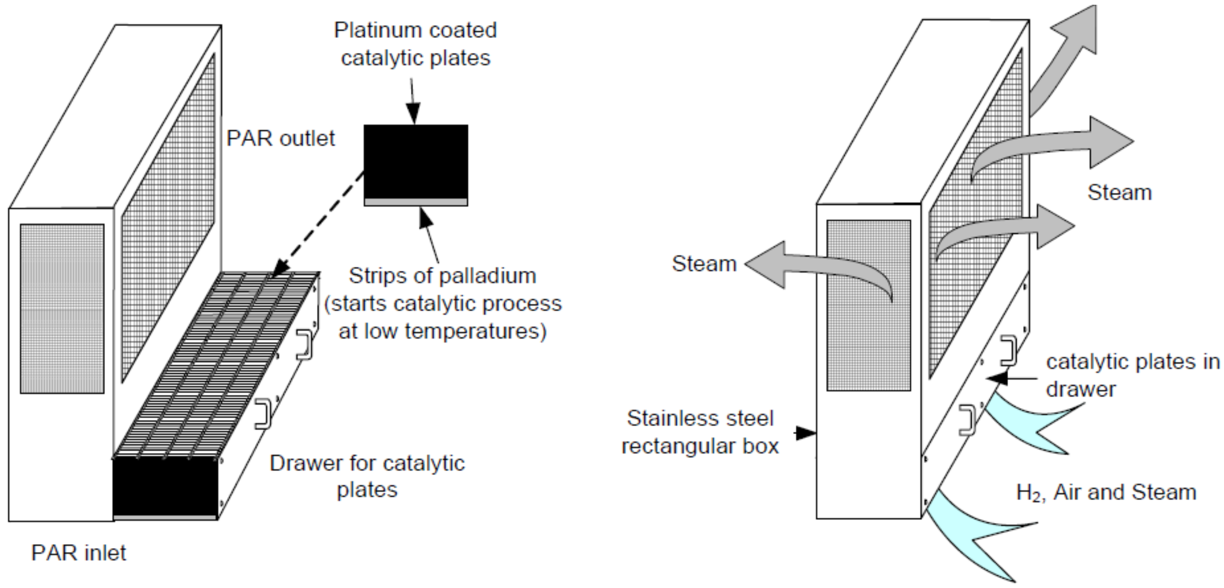

Figure 14: Passive Autocatalytic Recombiner design *[24]*

The recombination process starts when gas mixtures containing between 1% to 2% hydrogen come into contact with the catalyst coated plates. The hydrogen and oxygen reacts chemically on the surface of these plates and is exothermic, generating steam in the process. Thus keeping the hydrogen concentration lower then the flammable concetration range. Testing shows that the hydrogen concentration of the gas mixture will be reduced to approximately 0,5% (volume) at the outlet of the PAR unit. The depletion rate of hydrogen is dependent on the containment pressure and the amount of air, H_2, steam, carbon monoxide and other gases in the mixture.

3. Model formulation and data collection

3.1 Research Methodology

The overall methodology for this research was structured around the case study of the KNPS containment integrity following a double-ended guillotine break LOCA. This was based on literature reviews and previous studies done on various containment buildings such as Young *et al.* [14] and Sang-Won Lee *et al.* [18]. The basic understanding of a PWR containment building design and requirements was evaluated. This was closely followed by considering emergency safeguard systems that will support the containment building integrity following an accident event. These technical evaluations include:

- Design features and requirements for containment buildings
- Containment spray system
- Passive auto-catalytic recombiners
- Design features of CFVS
- Different CFVS type comparison
- Calculations:
    - Model heat up rate in a LOCA
    - Containment pressure and temperature absorption capability

Data collection was performed using the KNPS engineering simulator. This was done due to the complex nature in a rapidly changing atmosphere in the event of a LOCA to ensure accurate data compilation. For this to occur the following were evaluated:

- Understand computer code/simulator used at KNPS
- Simulate system response based on the same input data with different systems and electrical power availabilities
- CFVS incorporation to current KNPS PWR system to cater for “black swan events” like Fukushima Daiichi scale event

Results were presented in basic graph format due to the amount of data gathered and can be requested from the author for further scrutiny.

3.2 Simulator (Koeberg engineering simulator)

With the rapid change and complex nature of the containment atmosphere during transients and accidents the KNPS engineering virtual simulator was utilised to obtain the data for the scenarios.

The simulator is a predictive computer code mainly utilised at the NPP as a training tool for Reactor Operator licencing and requalification purposes. This is a licence binding requirement from the National Nuclear Regulator (NNR). The NNR is the South African nuclear regulatory body responsible for all frameworks to protect people, the environment and property from any harm and damaging effects of ionizing radiation and the risks associated with a specific facility or activity of handling radioactive material. To comply with this licencing requirement guide set out by the NNR, the minimum functional requirements of the simulator are set out but not limited to this. The simulator program yields little to no difference in response compared to the actual unit response under normal and abnormal operations. The software has the capability of mathematical modelling, operating system and utilities design report generation, detail algorithm, general description and all source coding [25].

The simulation boundaries should clearly be identified as the simulator goes into a state where it will “freeze” (where the dynamic simulation is interrupted and remains static until taken out of this mode) when the simulation parameters go beyond that of actual system failure. This occurs as the simulator is not a BDBA code and is not designed to give an output when the analysed system goes beyond failure. For this reason, the containment pressure limit used was the design pressure of 500 kPa gauge. Similarly, the hydrogen generated during the simulation of the LBLOCA was accepted to remain below 4% (below risk of ignition concentrations) for all the simulations as the PARs are passive devices that are incorporated in the simulator code and cannot be disabled or removed.

3.3 Onset conditions for the simulation

This dissertation assumes the following scenario, a severe earthquake occurs off the coast of Cape Town close to the KNPS which generates a tsunami, with waves in excess of 15 m in height. The Earthquake also causes a double ended guillotine break (LBLOCA) on the cold leg of the primary system. All off-site power is lost and the diesel generators trip due to flooding caused by the tsunami.

Simulation scenarios:

Consider a PWR such as KNPS with a primary circuit pressure of approximately 15.4 MPa gauge and the containment pressure comprised of non-condensable gases and air at 96 kPa (abs). To determine the effects of a postulated accident that will increase the containment pressure, temperature and atmosphere rapidly, the following scenarios/ cases were looked at:

- Scenario/case 1: LOCA with one emergency diesel generator

In the initial stage, the containment pressure will increase rapidly due to the coolant (water) being released from a high pressure primary pump discharge pipe break and flashing to steam. In the second stage (normal plant response), offsite power supply is lost with the exception that one emergency diesel generator is operable and available. The passive and safeguard systems will be actuated as per normal plant response.

- Scenario/case 2: LOCA with station blackout conditions

The initiating stage will occur (LOCA), but then no engineered safety system is activated. We assume that the electrical and battery room levels are flooded, rendering equipment inoperable. Thus, allowing the containment building pressure and temperature to increase to design parameters and can be assumed to reach failure or uncontrolled release conditions.

- Scenario/case 3: LOCA with station blackout conditions, with CFVS

The initiating stage and pressure increase occur (same as in scenario 2) in the containment building with the activation of an included containment filtered venting system.

Assuming a CFVS passive actuation at 300 kPa (g) and closes at 200 kPa (g) with an extraction flowrate of approximately 15 kg/s.

The containment pressure increase stage of the case study can be mathematically calculated per unit time incorporating the initial containment pressure. The fluid leaking from the postulated LOCA break is a two-phase mixture due to the primary system operating parameters. Using the Brunell critical flow rate for two-phase flow in short channels, the resulting pressure, P_t, [26] :

$$P_t = P_S + P_a \quad (1)$$

and

$$W = A_n \sqrt{[2\, g\, \rho_P\, P_{sat}\, (1 - c)]} \quad (2)$$

Where:

P_t = Resulting total pressure

P_s = steam pressure

P_a = air pressure

W = critical mass flow rate

P_{sat} (1-c) = pressure at exit from the pipe break/ nozzle

ρ_P = Water density at specific pressure

A_n = break/nozzle area

Assuming the mixture from the break will convert to wet steam in the containment free space and can be written by the energy balance:

$$W\,[H + Cp_S\,(T_h - T_m)] + MCp_a(T_a - T_m) = (W + M) * \frac{(Cp_s + Cp_a)}{2} * (T_m - T_a) \quad (3)\ [26]$$

Model Assumptions:

- The initial velocity of the fluid in the pipe is zero (0).
- At time t =0 (sec) the flashed steam is considered dry saturated condition.

- The coolant flow rate through the break/nozzle is unchanged for scenario 1 where the safety systems activate and the re-flooding of the core and primary system occur for approximately 30 seconds.
- Pressure in the pipe is unchanged for the containment pressure increase stage.
- The water enthalpy in the pipe is unchanged during the blowdown period into containment.
- Two main dimensional/directional components of water velocity at the break nozzle exist. One in the direction of the fluid flow (u_1) and the other is perpendicular to the first (u_2). Thus giving:

$$u_1 = \frac{\dot{m}}{A.\rho_P} \qquad u_2 = \sqrt{\left[\frac{2g\,(P_n - P_a)}{\rho_P}\right]}$$

This gives the kinetic energy of the water:

$$u = \sqrt{((u_1)^2 + (u_2)^2)} \tag{4}$$

The kinetic energy of the steam is determined from the steam mass flow rate (W.X). The steam velocity becomes:

$$v = \sqrt{\left[\frac{(2g\,(P_n - P_c))}{P_s}\right]} \tag{5}$$

As both water and steam possess kinetic energy and enthalpy and the break will increase these in the containment and the energy balance in containment becomes:

$$W\,(1 - X)[(KE)_W + H_W + Cp_W\,(T_h - T_c)] + W.X\,[(KE)_S + H_S + Cp_S\,(T_h - T_c)] + MCp_a(T_a - T_c) = (W + M) * \frac{(Cp_s + Cp_a)}{2} * (T_c - T_a) \qquad (6)\ [26]$$

The above equation gives the new containment temperature after the pipe break occurs and will provide the new pressure assuming the air and steam properties stay unchanged during the one second time step.

Input data used for a typical Westinghouse PWR like KNPS can be seen in Appendix A, Table 4.

3.4 Comparison between different CFVS designs

The combination type CFVS system proves how industry requirements differ from operating unit to unit, therefore the selected design will have to meet a number of unit-specific criteria. Criteria can range from CFVS performance, geographical location to economic viability. A comparison of different CFVS designs and operational functionality was drawn up in Tables 2 and 3 below from various literature sources.

Table 2: Comparison of advantages and disadvantages of wet and dry Containment Filtered Venting systems *[22] [23] [27] [28] [29]*

Type	**Wet CFVS system design**	**Dry CFVS system design**
Supplier/ Examples	• Westinghouse Wet-Metal fibre • Areva • CCI	• Westinghouse Dry design • Sand bed filter (utilised by French PWR's) • Gravel bed
Advantages	• Scrubbing of steam/gas through solution (retaining particles) • Chemical addition for high iodine retention • Predictable behaviour with respect to residual heat capacity • High duration of operation without intervention	• Very high retention rates • Modular design and compactness • Passive operation • Little to no maintenance required after installation • Uses sand beds or stainless steel fibre filters
Disadvantages	• Droplet separation necessary • Re-suspension risk • Limited fine aerosol retention • Need control and operator intervention	• High filter blockage probability • No decay heat transfer ability • No mixed aerosol removal capability • Ability compromised at altered pressures

Apart from Table 2 and the listed disadvantages, it is important to note the possible overall disadvantages of including any CFVS to the design and safety margins of the plant. Some of these include early actuation and actuation method failure.

The disadvantage of the early venting actuation in an accident is that depressurisation occurs at a time when the fission product level in the containment is still high. On the other hand, when the isolation path, i.e. rupture disk or isolation valve fail, would compromise the containment boundary, if not detected, it will lead to an uncontrolled release to the environment.

Another factor to consider is the CFVS extracting effluent flow from the containment building may result in a possible hydrogen concentration/ build-up in the vent line and possibly increasing the risk of hydrogen combustion in the venting lines [22]. This can be limited by nuclear operators with the inclusion of hydrogen recombining processes and equipment for a severe accident.

The advantages and disadvantages do not make the wet or dry type of CFVS suitable but would be dependent on the need and desired functionality of the CFVS.

Table 3: Comparison of the important characteristics of 4 designs of Containment Filtered Venting systems [22] [23] [27] [28] [29]

Types of CFVS Designs	**Westinghouse Dry**	**Westinghouse Wet-Metal fibre (MVSS)**	**Areva (high speed sliding pressure venting)**	**CCI**
Characteristics: Filtration process	Modular metal fibre filter and molecular sieves	Submerged venturi and metal fibre filter	Submerged venturi, metal fibre filter and molecular sieve	Sparger assemblies, co-current scrubber, recirculation zone, droplet separator
Chemicals in scrubbing liquid	N/A	Sodium Hydroxide, Sodiumthiosulphate	Sodium Hydroxide, Sodiumthiosulphate	Sodium Hydroxide, Sodiumthiosulphate, Aliquat
Decay heat capacity	Natural convection of filter only	High, depending on size of vessel	High, depending on size of vessel	High, depending on size of vessel
Venting duration	Limited by metal fibre filter size and capacity	Limited by design and metal fibre filter size and capacity	Limited by design and metal fibre filter size and capacity (the need to recirculate to containment)	> 1 year (depending on customer specification and ability to make-up to vessel)
Aerosol decontamination factor (DF)	> 10 000 at limited aerosol load capacity	> 10 000 at limited aerosol load capacity	> 10 000 at low aerosol concentration and load capacity	> 10 000 at ideal aerosol concentration
Elemental Iodine DF	> 3300 (if molecular sieves are pre-heated) otherwise >1000	> 1000	>100-200	>1000

From Table 3 the comparison of the 4 different CFVS designs were tabulated. The characteristics and values between the 4 CFVS designs were comparable but were dependant on the type and design required by the PWR and BWR. They also varied due to the actual design of the CFVS and what their main purpose for installation was.

The main characteristics of interest were, for eg. the filtration process and the decay heat capacity. From Table 3 we observed that the filtration process differed between the 4, but 2 were similar using a submerged venturi design. The scrubbing chemicals were mostly the same and the decay heat capacity of the dry CFVS used natural convection of the filter only where the other 3 had a high decay heat capacity. It could also be observed that the venting durations differed between the 4 designs, the aerosol decontamination factors were all the same at >10000, the elemental iodine decontamination factors were different between the 4 designs, showing that the dry CFVS worked better at reducing the iodine contamination at >3300, with the lowest value observed from the high speed sliding pressure venting.

4. Results and Discussion

In this study, one of the most dangerous accidents in a NPP was investigated, namely a LBLOCA on the cold leg of the primary system. This accident was selected to model a Fukushima Daiichi scale event and investigate the response to a BDBA. The determined values of the peak containment pressure, temperature and consequently the capability of the containment to maintain its integrity following the LBLOCA were obtained. In the simulation on the Koeberg simulator, it was assumed that no operator actions were performed throughout the accident scenarios.

The initial conditions for the scenarios can be seen in Appendix A, Table 4 with the pipe break initiated at time t = 0 seconds. The data was gathered and resultant data and graphs can be observed in Appendix B.

4.1 Scenario 1 (LBLOCA with one train emergency diesel generator set available)

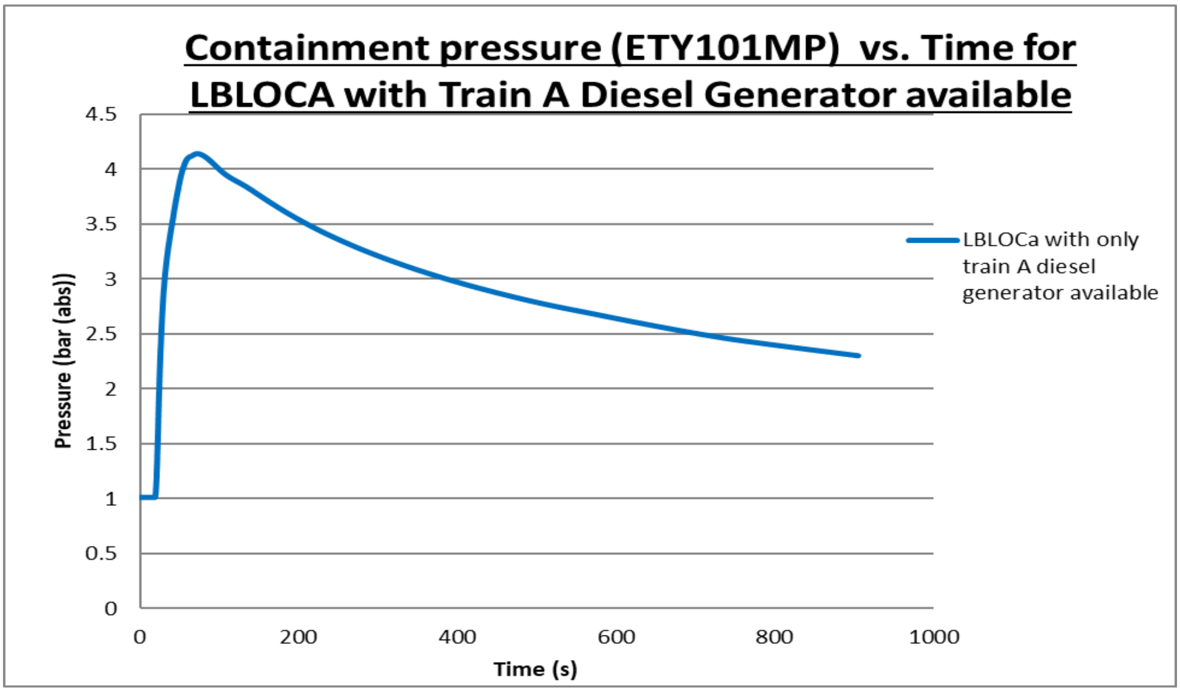

Figure 15: Containment pressure vs. Time for LBLOCA with one Diesel Generator available

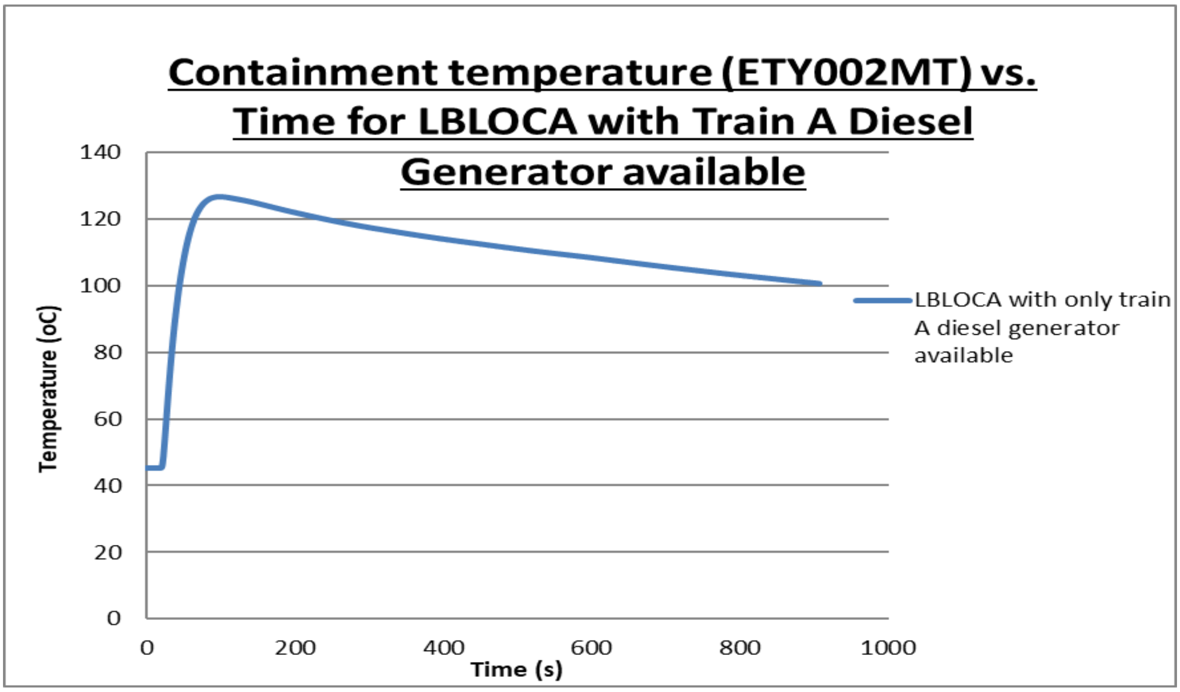

Figure 16: Containment temperature vs Time for LBLOCA with one Diesel Generator available

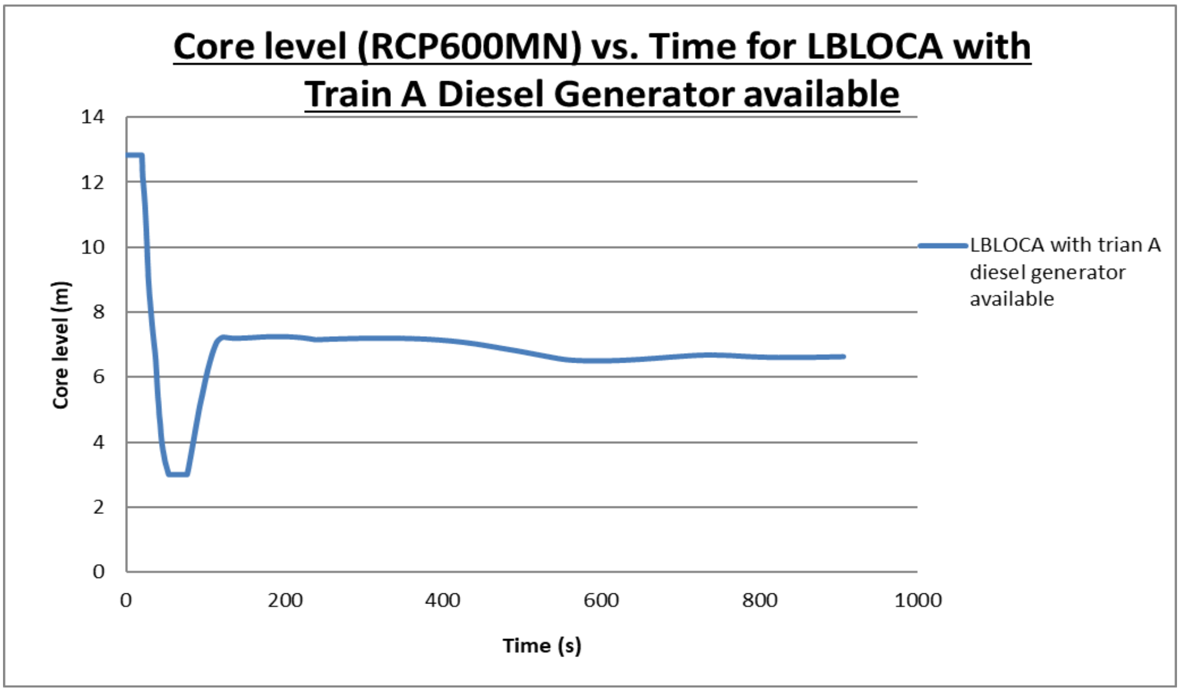

Figure 17: Core level vs Time for LBLOCA with one Diesel Generator available

In scenario 1, the pressure in the containment building rapidly increased and reached a maximum pressure of 4.135 bar (abs) within 70 seconds. This also occurs with the other 2 simulated scenarios reaching 4.011 bar (abs) for scenario 2 (LBLOCA with SBO conditions) and 4.135 bar (abs) for scenario 3 (LBLOCA with SBO conditions with a simulated CFVS) at 54 seconds.

The availability of at least one emergency diesel generator signifies the possibility and capability of a safe shutdown by means of supplying electrical power to the safeguard equipment like containment spray system, reactor safety injection system and auxiliary shutdown systems. With these systems in operation, it can be seen that the containment pressure is being controlled and reduced to less than 2.5 bar(abs) after 15 minutes.

It is also reinforced in Figure 17, where the drop in reactor level from approximately 12,8 m (on level instrument RCP600MN) and the stabilisation to approximately 7.2 m in 120 seconds after the postulated incident is observed. RCP600MN is the instrumentation for measuring the reactor core level in meters during normal and accident operating conditions. Borated water is pumped from the refuelling water storage tank to keep the fuel subcritical and remove residual heat. The containment temperature peaked at approximately 126.8 ^{o}C from the containment temperature of 44.62 ^{o}C. It later decreased steadily to below 105 ^{o}C after 12 minutes.

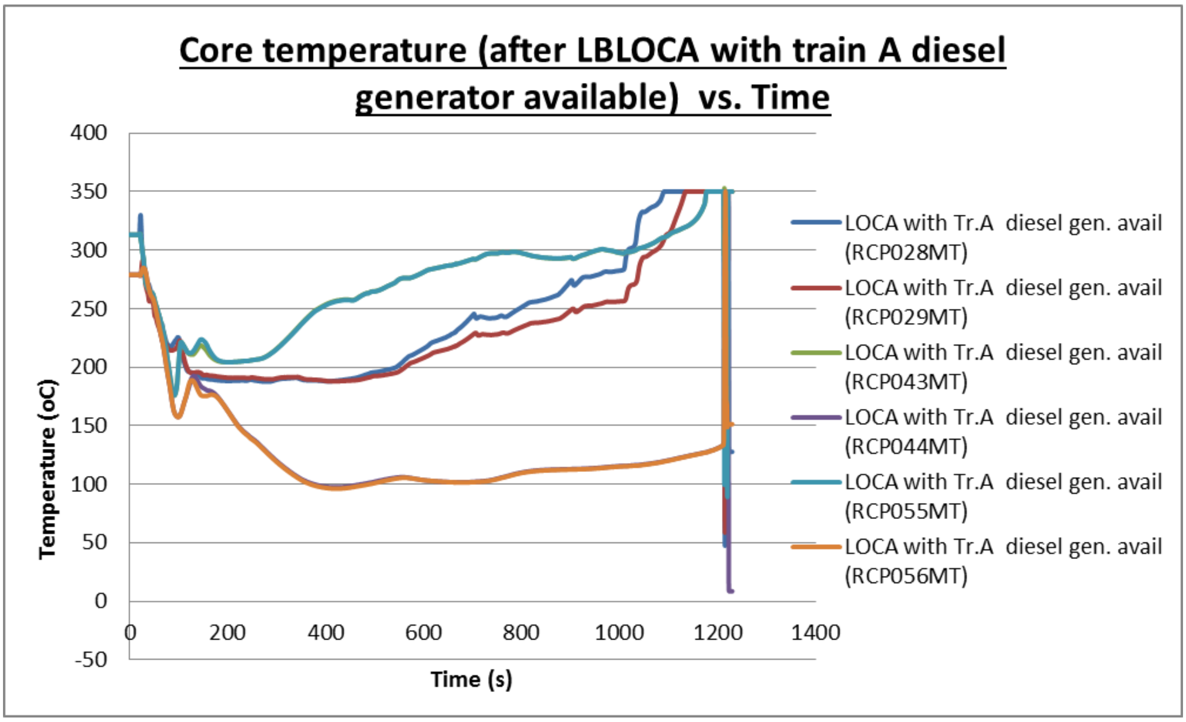

Figure 18: Core temperature profile for LBLOCA with one Diesel Generator available

The simulation for scenario 1 was stopped after 15 minutes, as the results continued to decrease/improve thereafter. This as the plant shutdown process was initiated and the cooling and safety systems were on recirculation mode removing residual heat from the core, protecting the containment integrity by means of containment sprays and emergency coolers, etc.

4.2 Scenario 2 (LBLOCA with SBO conditions)

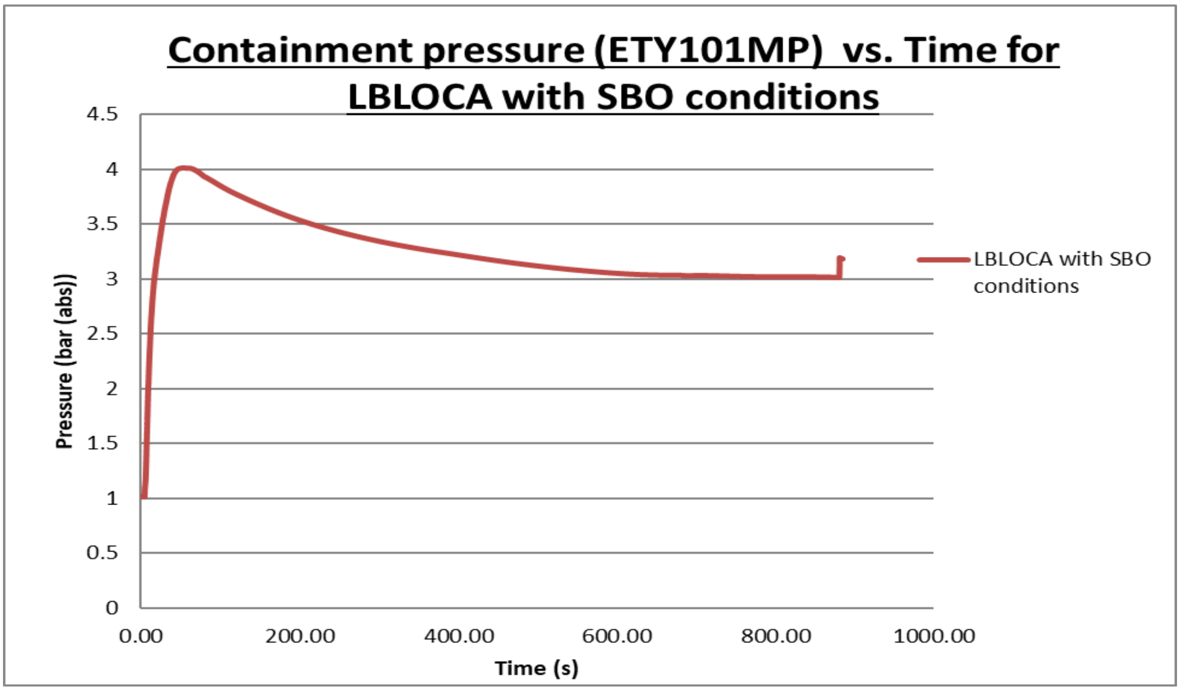

Figure 19: Containment pressure vs. Time for LBLOCA with SBO conditions

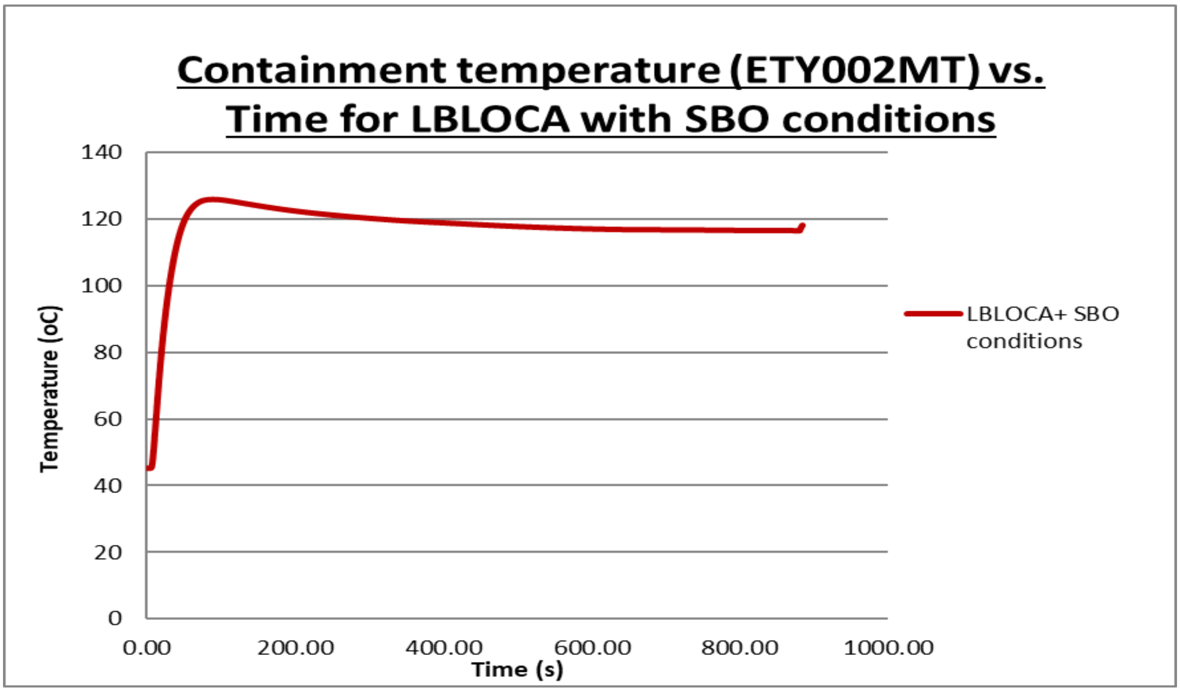

Figure 20: Containment temperature vs. Time for LBLOCA with SBO conditions

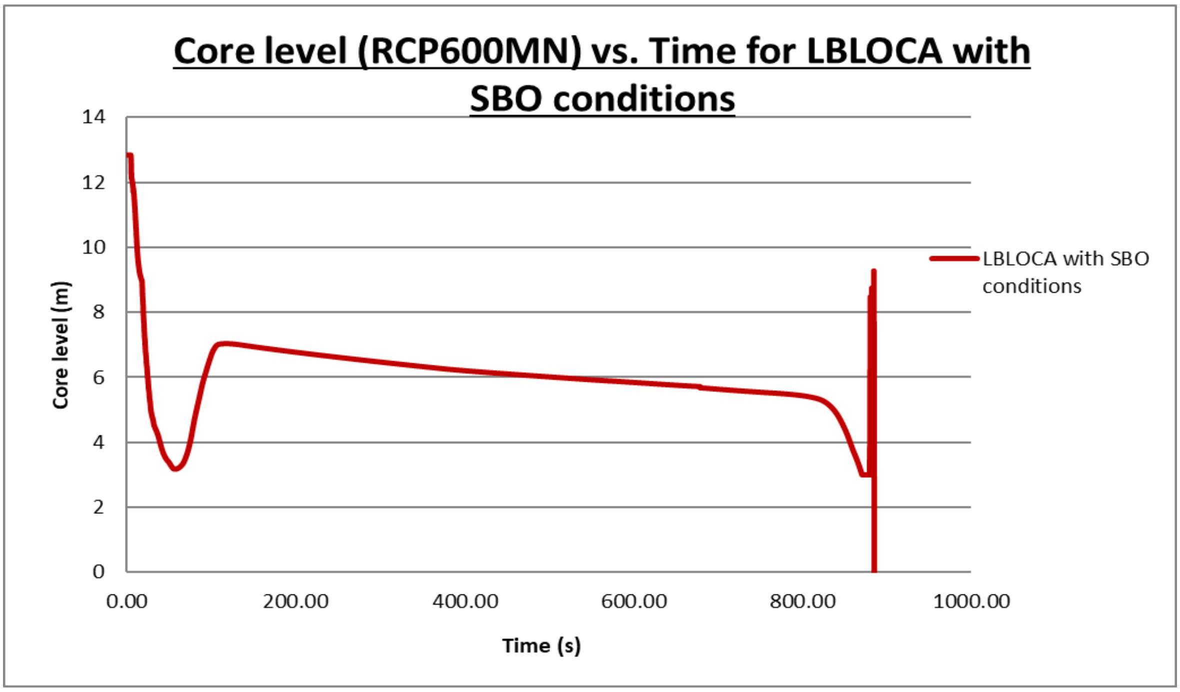

Figure 21: Core level vs. Time for LBLOCA with SBO conditions

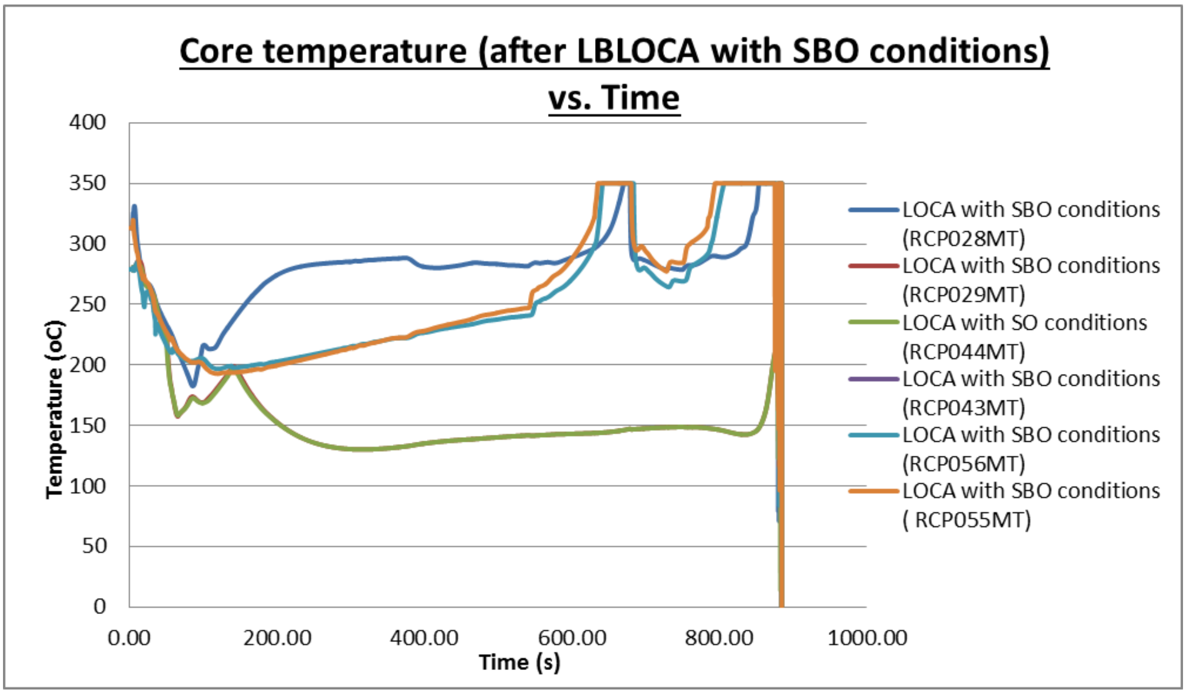

Figure 22: Core temperature profile for LBLOCA with SBO conditions

In simulation scenario 2 the containment pressure peaks at 4.011 bar (abs) after 54 seconds. Consequently, the pressure starts to decrease steadily to approximately 3 bar (abs) after 660 seconds (11 minutes) where it stabilised just before another rapid increase in pressure occurred. This stabilisation is solely due to the pressure absorption capability of the containment structure / building. This can be explained with the calculated heat and pressure capacity of the containment below.

Containment pressure load handling capability:

Mass (m): $$m_{RCP} = m_{cont} = \frac{v_{RCP}}{v_{cont}} = 137228.261 \text{ kg} \quad (7)$$

v_{cont} @ 315°C = 0.001472 $\frac{m^3}{kg}$ [ASME tables]

Energy: $$m(h_{cont} - h_{RCP}) = {}_{cont}^{RCP}Q - {}_{cont}^{RCP}W_{ork} \quad (8)$$

$$h_{RCP} = h_{cont} = 1431.6 \frac{kJ}{kg} \quad (9)$$

Assuming no heat/energy gained or lost, no work done for state 1 to state 2, i.e. the moment before and after the rupture of the primary pipe.

At 400 kPa (from ASME table): $x = \frac{u_{cont} - u_f}{u_{fg}}$ = 0.302 (10)

Where: x = wetness fraction

$$v_{cont} = v_l + x.v_g = 0.1407\,\frac{m^3}{kg} \tag{11}$$

$$V_{cont_{min}} = m_{cont}.v_{cont} = 19308.016\,m^3 \tag{12}$$

Therefore, the minimum containment volume for the pressure not to increase above 400 kPa (abs) (4 bar (abs)) is 19308.016 m^3. The KNPS containment free space volume is 49500 m^3, which is well above the minimum required volume to maintain the pressure at 400 kPa.

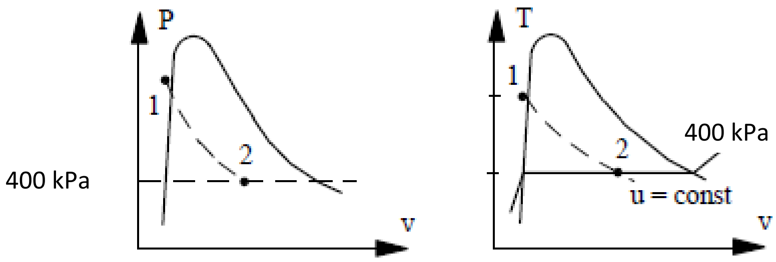

Figure 23: Vaporisation and thermohydraulic effects inside containment vessel following a LOCA

At the integer/input of 870 seconds, the core level was observed, in Figure 21, to deteriorate below 3 meters, shortly after which the simulation concluded due to the simulated system reaching failure limits. This is a consequence of no residual heat removal (water adding capability) from the reactor core. At this point it was evident that the reactor core was uncovered and the fuel started melting with the probability of catastrophic failure.

4.3 Scenario 3 (LBLOCA with SBO conditions and assimilated CFVS)

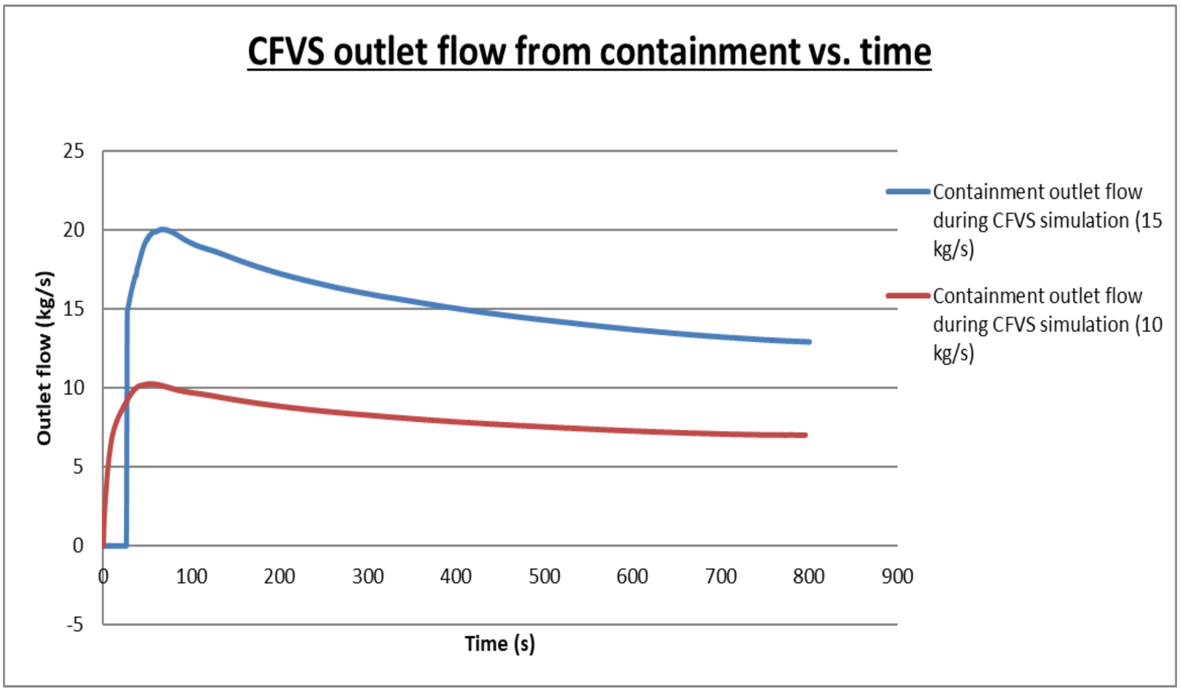

Figure 24: Differences in CFVS exhaust for 10 kg/s and 15 kg/s penetration

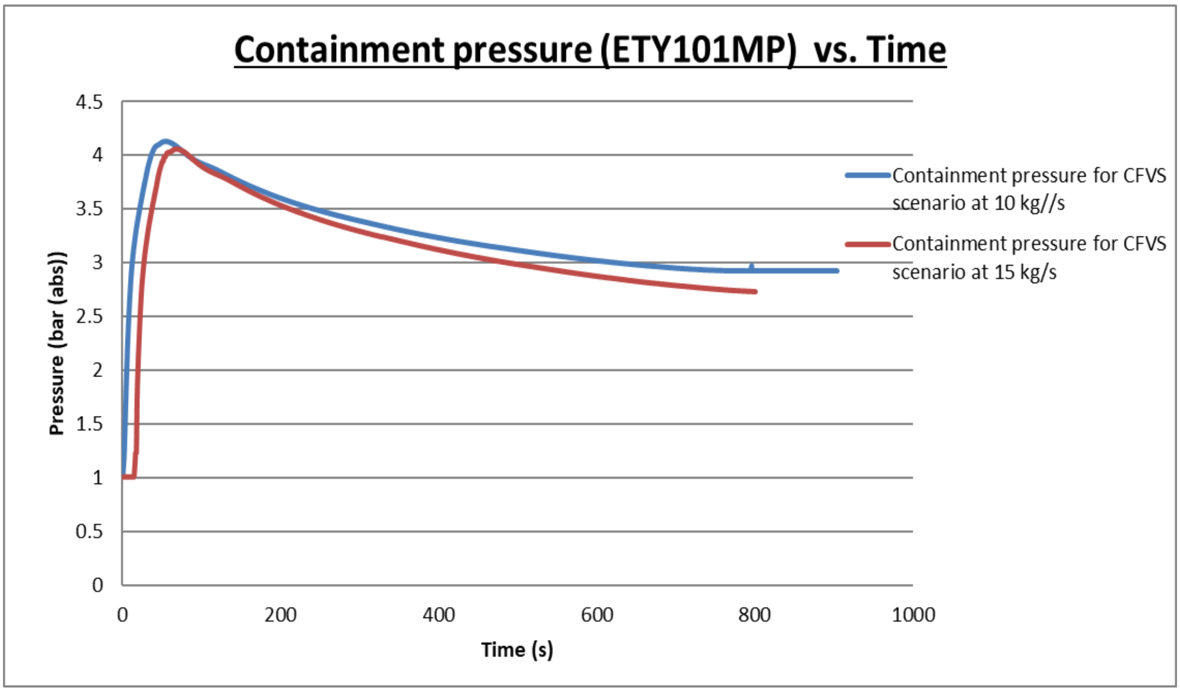

Figure 25: Containment pressure profiles for different CFVS extraction flowrates

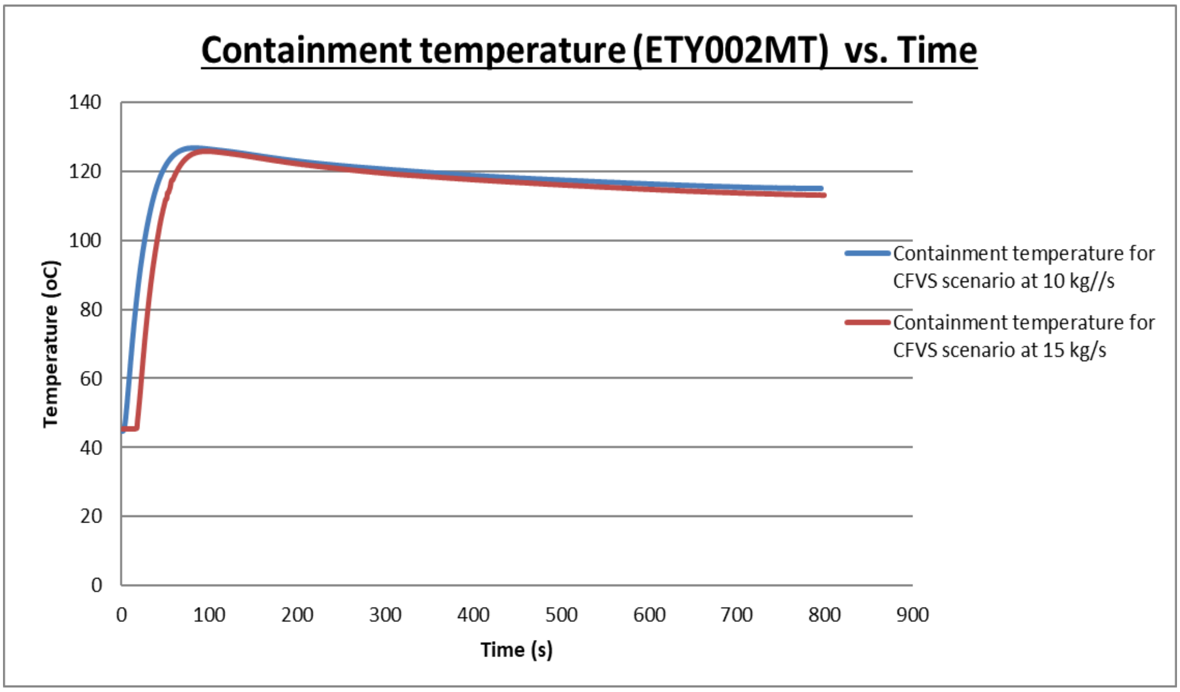

Figure 26: Containment temperature profiles for different CFVS extraction flowrates

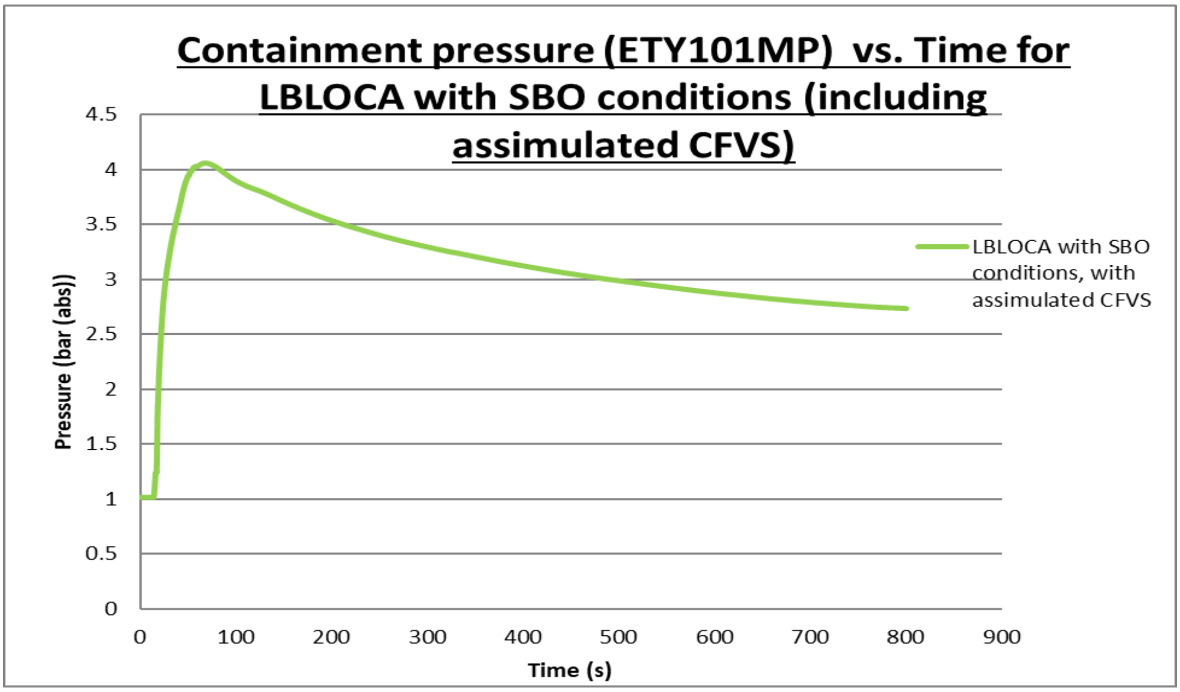

Figure 27: Containment pressure vs. Time for LBLOCA with SBO conditions including assimulated CFVS (extraction rate of 15 kg/s)

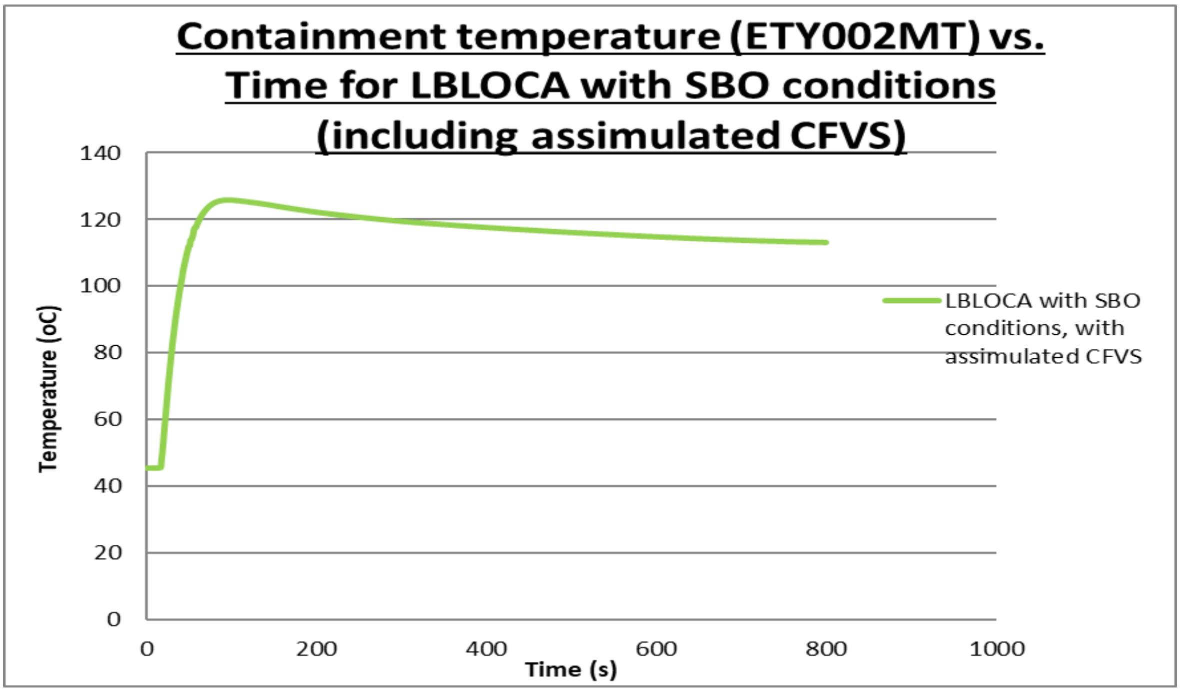

Figure 28: Containment temperature vs. Time for LBLOCA with SBO conditions including assimulated CFVS (extraction rate of 15 kg/s)

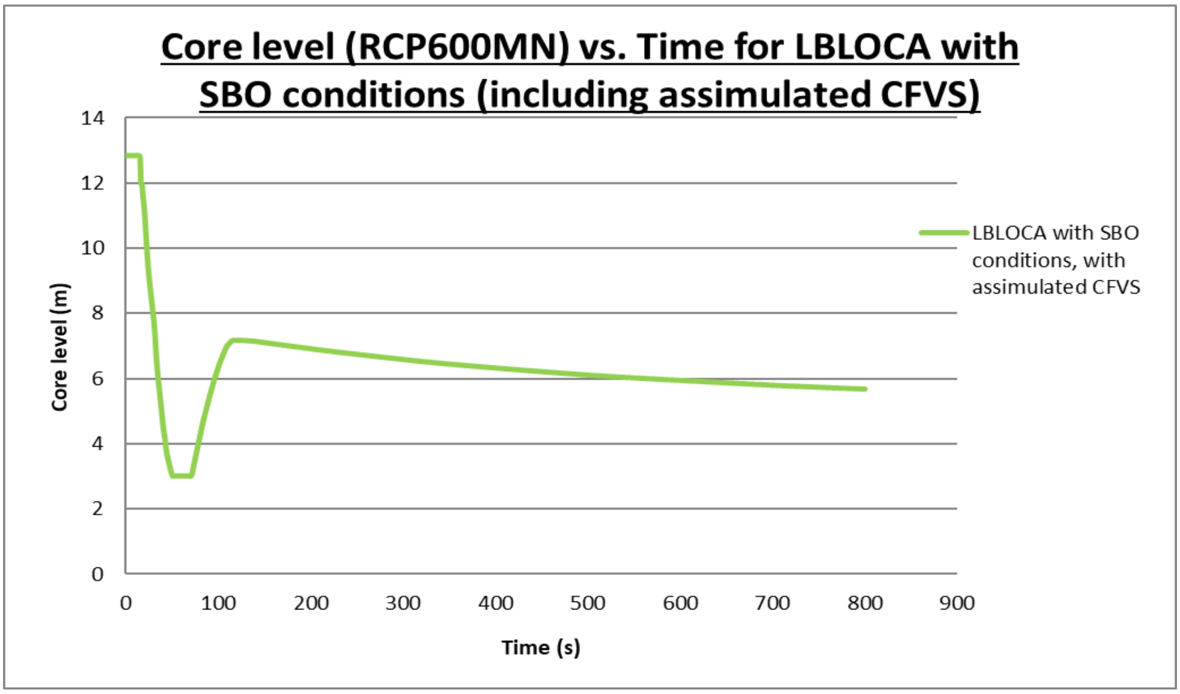

Figure 29: Core level vs. Time for LBLOCA with SBO conditions including assimulated CFVS (extraction rate of 15 kg/s)

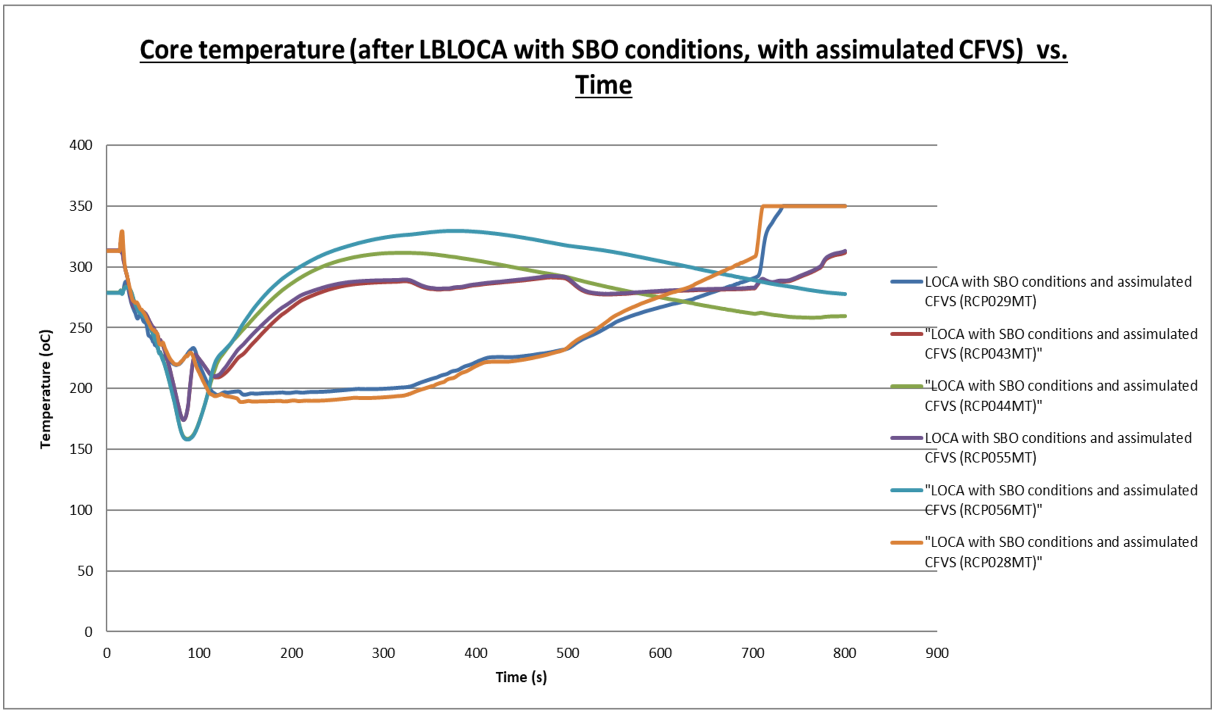

Figure 30: Core temperature profile for LBLOCA with SBO conditions including assimulated CFVS (extraction raet of 15 kg/s)

In scenario 3 a containment penetration point, located in the upper part of the containment vessel, was used as an outlet point with a flow rate of approximately 15 kg/s to simulate a CFVS. Altering the extraction flow rate from the containment building was also investigated and the results can be seen in Figures 24 to 26 using the flowrates of 10 kg/s and 15 kg/s. The temperature and pressure profiles generated from this resulted in similar outputs with the more significant difference being noticeable in the final containment pressure. Here it is noticeable that the containment temperature profile is identical with very little end result by changing the extraction flowrate, thus changing the size of the CFVS had no effect on the temperature profile of containment. The containment pressure profile, however, was clearly influenced by the sizing of the CFVS. The peak containment pressure for a 15 kg/s extraction CFVS system was 4.06 bar (abs), whereas the peak containment pressure for a 10 kg/s extraction CFVS was 4.135 bar (abs). The pressure to which the simulation descended before failing also differed for the two venting extraction rates. The higher venting extraction rate yielded a lower peak containment pressure. This would be desired to ensure that the containment building is not over-pressurised past its integrity. The smaller extraction rate yielded a higher peak containment pressure but only by 3%, where the extraction rate was changed by 50%. The analysis showed the containment exhaust flow is directly related to the containment pressure. It can be seen in Figures 31 and 32 where very little changed in the containment pressure and temperature can be compared to that of scenario 2. This could be due to the consequences of the rapid rate of change of the containment atmosphere conditions compared to the small extraction rate of the simulated CFVS extraction flow rate.

The difference between scenario 2 and scenario 3 can be seen in Figures 19, 27 and 31, where the maximum pressure produced after the break at time integer of 54 seconds was recorded at 4.011 bar (abs) and 4.135 bar (abs) respectively. Even though with a higher peak pressure sustained with scenario 3 (with an assimilated CFVS) the containment pressure reduces faster than that of scenario 2 and decreases to 2.93 bar (abs). This occurs after 13 minutes at which point the simulation reaches failure. To understand this occurrence, observe Figure 30 detailing the reactor core temperature profile for scenario 3. It was observed in seconds after the break and the loss of primary inventory, the core temperature rapidly increased. This also occurred in scenarios 1 and 2. This is a result of the reactor core voiding as it goes through a phase of departure from nucleate boiling. The negative void coefficient of reactivity of the PWR forces the reactor to go subcritical and shutdown causing the temperature to decrease. This can be seen in Figures 18 and 22 for scenarios 1 and 2 respectively.

With little cooling to the core and stored energy being released, the nuclear fuel and the cladding start heating up and the core temperature start increasing. The principle of this phenomenon following a LBLOCA can be seen from literature in Figure 34, Appendix C. This occurred with scenarios 2 and 3, with the only differences being the time at which the effects occur. Stored energy, known as decay heat is generated by:

- Energy released as the products of the fission process continue to decay,
- Energy released from fission caused by delayed neutrons, and
- Energy released from the decay of neutron capture products such as Uranium-239 and Neptunium-239. (Uranium-238, through the absorption of a neutron and the emission of a quantum of energy known as a gamma ray, becomes the isotope Uranium-239. Over a period of time (approximetaly 23.5 minutes), this radioactive isotope loses a negatively charged electron, or beta particle; this loss of a negative charge raises the positive charge of the atom by one proton, transforming it into the element Neptunium-239. Neptunium-239 in turn undergoes beta decay, thus gets converted into Plutonium-239.)

The core level profile for scenario 3 is observed to have the initial drop in level from 12.84 m to 3 m after 52 seconds. Here it stabilises and starts increasing to 7.16 m after 120 seconds. This is a resultant from the reflooding of the primary system from the safety injection system accumulators. This is followed by the gradual decrease in level caused by the loss of inventory via the pipe break. This occurs until the core level decreases below 5.8 m and the simulation ends.

4.4 Summary of results

The different scenario results was superimposed to visibly analyse the differences per unit time following the LBLOCA. This is represented in Figures 31, 32 and 33. When the pressure curve for scenario 1 (LBLOCA with one train diesel generator available) was compared with the pressure curve for scenario 2 (LBLOCA with SBO conditions), together with temperature response profile in scenario 2, the significant effect from the containment spray system on the containment atmospheric temperature is clearly noticeable. This is visible in Figure 31. This coupled with the alleviating effect of the large dry containment design reduces the temperature and pressure of containment adequately.

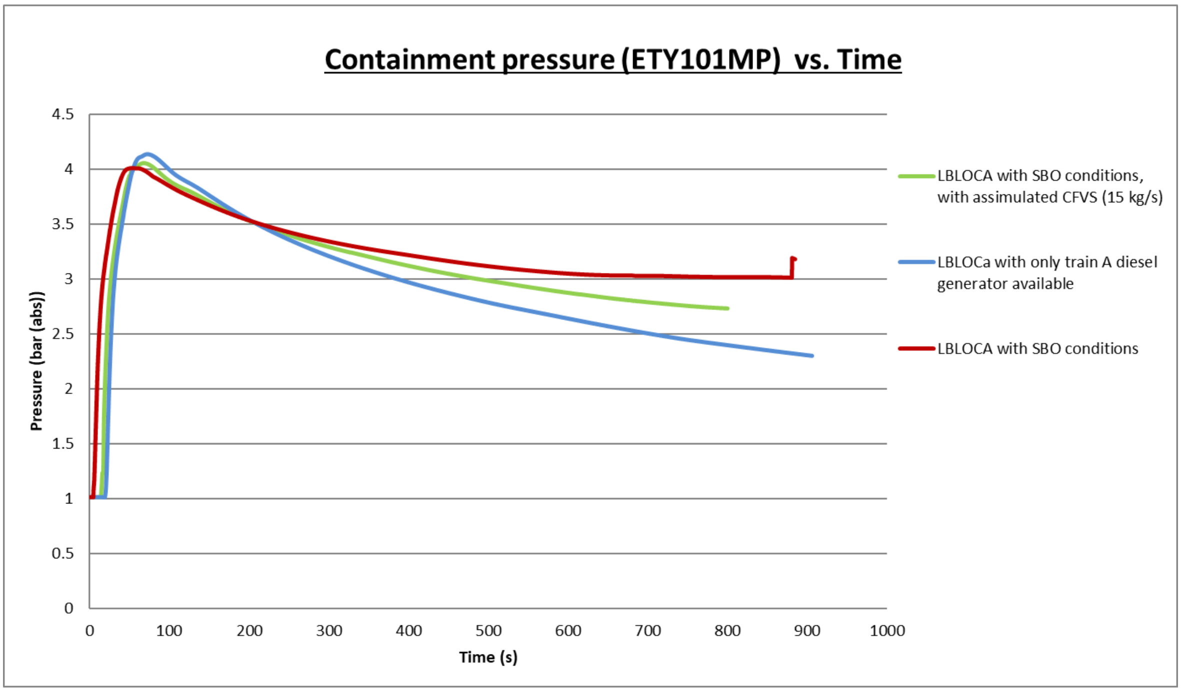

Figure 31: Containment pressure comparison of all simulation results

The other noticeable difference between scenario 2 and 3 is the point at which the simulation or the system failed. Scenario 3 with the assimilated CFVS system continued until 905 seconds, whereas scenario 2 ended only after 884 seconds before system failure.

It can be seen in Figure 31 indicated by the containment pressure measuring instrument, ETY101MP, that the pressure increases rapidly after the onset of the postulated incident scenarios for all the simulation scenarios. This is identical for the temperature profile of containment measured with temperature transmitting instrument, ETY002MT. This is due to the release of large volumes of water and water flashing to steam as a result of the double-ended pipe break in the primary system.

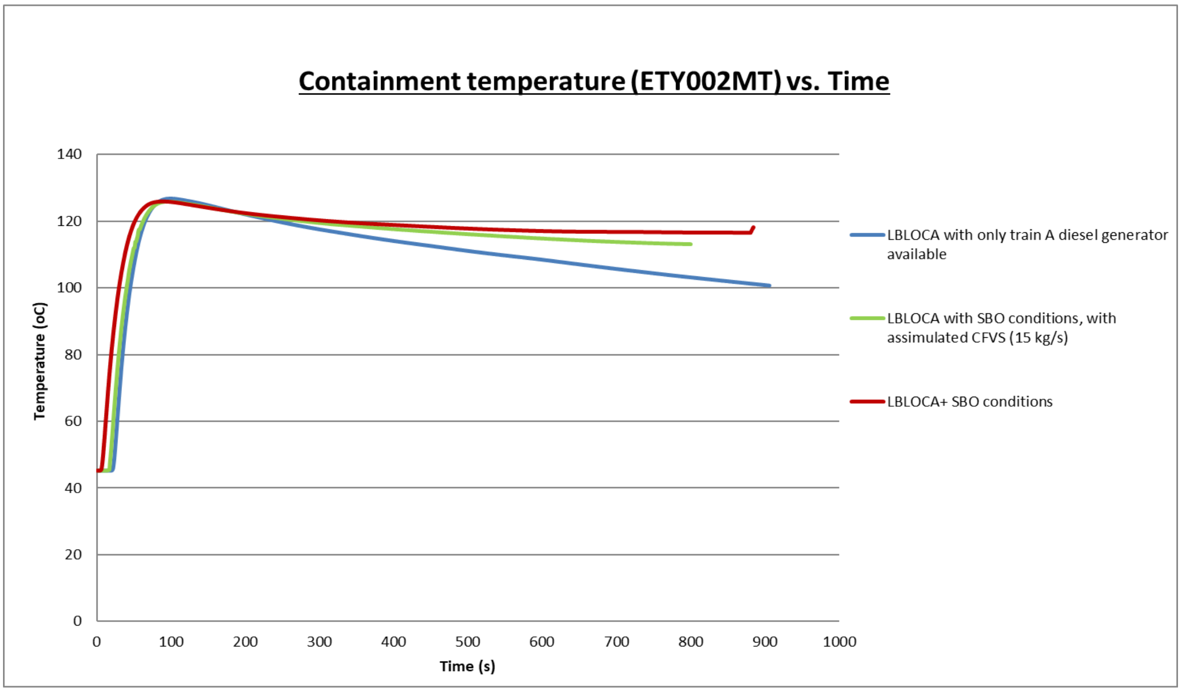

Figure 32: Containment temperature comparison of all simulation results

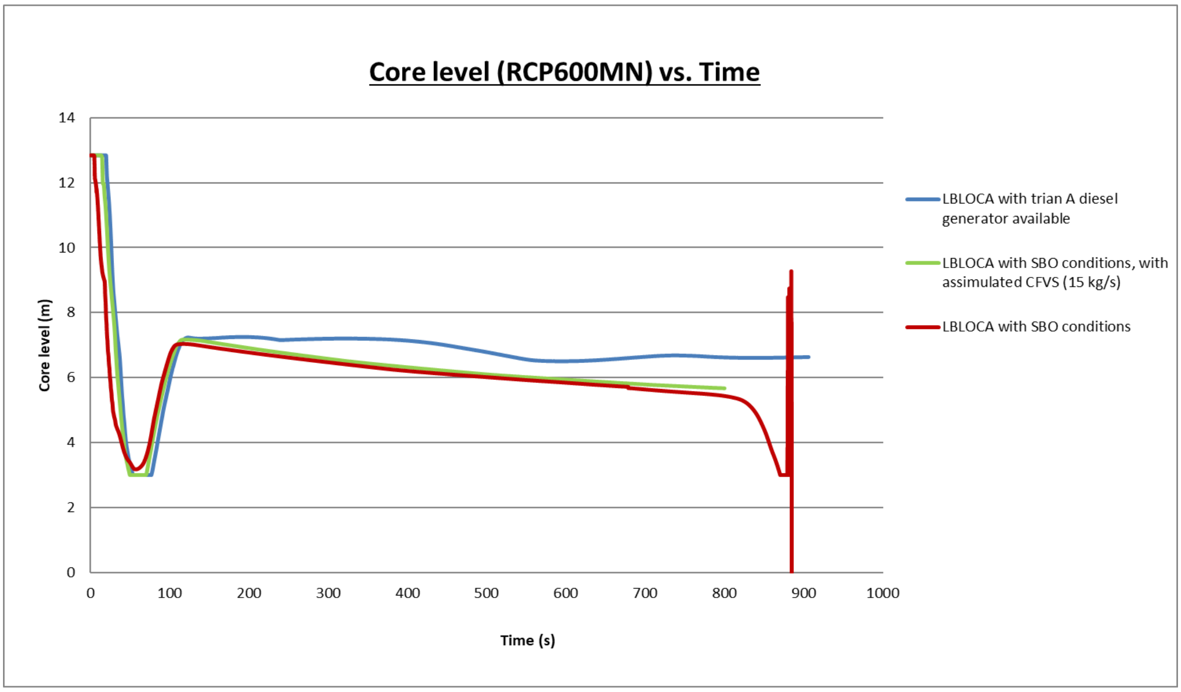

Figure 33: Core level comparison of all simulation results

Correspondingly, the reactor core level can be seen to decrease rapidly after the initial break in Figure 33. This is followed by a sudden increase in core level. The continuous flooding of the primary system from the passive safety system like the safety injection system accumulators contributes to the above occurrence when the pressure decreases below a predetermined value.

Literature suggests two of the most critical installations/enhancements to nuclear containment buildings/vessels for accident management are reliable water injection via sprays and containment filtered venting. The results obtained from the KNPS engineering simulator for the three selected LOCA scenarios supports this theory. With at least one operable emergency diesel generator functioning on the affected unit, the reactor unit can be safely be controlled and shutdown. This would be with the postulated LBLOCA releasing large amount of water and steam into the containment building.

It was observed following the accident input signals to the simulator, that the containment pressure and temperature increased to a maximum value of 4.137 bar (abs) and 126.8 ^{o}C respectively. The pressure and temperature of containment rapidly increased within 2 minutes to their respective thermodynamic peaks due to the high energy released from the pipe break in the primary system.

This was followed by a prompt decrease in containment pressure and temperature, mathematically determined by the containment load handling capability.

The results indicated that the large dry containment design at KNPS (free space volume of 49 500 m3) is well above the minimum required volume (19 308.016 m3) to maintain the containment pressure below 400 kPa in any operating or incident scenario. This was followed by a flattened pressure profile gradient.

Since the construction of the plant, there have been multiple modifications to the systems inside the containment building. These will influence the given free space volume and in turn the consequential pressure during transient and accident conditions.

Scenarios 2 (LBLOCA with SBO conditions) and 3 (LBLOCA with SBO conditions, with an assimilated CFVS) resulted in identical pressure and temperature trend profiles with scenario 1 (LBLOCA with one train emergency diesel generator set available) in contrast which yielded a faster and better alleviating outcome due to the effect of the safeguard systems actuation, i.e. the containment spray system.

The core level data (Figure 17 and 33) also reinforced the effect of the safeguard system actuation/ operational with scenario 1, where it was evident how the reactor primary level decreased (from 12.8 m) with the initial loss of primary inventory but stabilises between 7 m and 6 m after only 140 seconds.

With scenario 2 the initial drop in core level occurred as a result of the pipe break in the primary system, followed by the reflooding of the core plenum by the safety injection accumulators which are passive safety equipment (same as scenario 1). However, without any electrical power supplying the safeguard system in scenario 2, no borated water make-up is possible and the core level starts deteriorating gradually with time. This was followed by a prompt drop in level after 14 minutes to below 3 m, shortly after which the simulation concluded due to the simulated system reaching failure limits.

The reactor core temperatures for the scenarios with the explanatory descriptions of the instruments and their respective functional location in the core can be seen in Appendix B. Figure 22, the core temperature profile for scenario 2, also indicated the sudden increase in core temperature after approximately 10 minutes. Suggesting the reactor core was uncovered and the fuel started melting with the likelihood of catastrophic failure.

In scenario 3 a containment penetration point was used as an outlet/extraction point with a flow rate of approximately 15 kg/s to simulate a CFVS. The containment pressure and temperature profile observed for the simulation incorporating the CVFS mimics that of scenario 2 as the onset conditions are identical, with the difference of the extraction flow from containment. Figures 31 and 32 iterated the little change in the containment thermohydraulic properties compared to that of scenario 2. This indicating the rapid rate of change of the atmosphere conditions compared to the small extraction rate of the simulated CFVS extraction flow rate. This indicated the requirement to investigate the size and effectiveness of the selected CFVS system based on specific BDBA, i.e. a performance based probabilistic risk and safety analysis.

The results suggested the use of a CFVS will be more effective in a gradual over-pressurisation incident compared to a rapid over-pressurisation like a LBLOCA.

The core temperature for scenario 3 is also seen to reach the maximum output for the in-core instrumentation quicker than the other scenarios resulting in system failure.

5. Conclusions and recommendations

This dissertation evaluated the integrity of the PWR containment building as the last barrier following a severe accident, i.e. LBLOCA. This was stimulated after the accident at the Fukushima Daiichi NPP in 2011, which prompted the consideration and requirement to manage severe accidents especially in older generation NPP, like KNPS. At Fukushima, the inability to provide consistent reactor core cooling and reliable containment venting resulted in the final radiological barrier failure and direct release of radioactive nuclides to the surrounding environment.

During the simulated LBLOCA, large volumes of water flashing to steam escaped from the pipe break in the primary system. The pressure and temperature of the last barrier increased rapidly in the first 2 minutes following the break. This was followed by a steady decrease in these parameters based on the containment load bearing capability. This corresponds with studies done on other nuclear containment buildings, i.e. Young *et al.* [14] and Sang-won Lee *et al.* [18]. It was also observed in Figures 15 and 16 the effects of the containment spray system as a safeguard engineered feature reducing the containment building pressure and temperature in a LBLOCA. In addition, the inclusion of a CFVS and its effects was noticed and is evident in Figures 31 and 32. Here it was seen that the CFVS decreased the containment pressure at a faster rate, but is dependent on the size and effectiveness of the CFVS unit. It was also determined that a CFVS would be more effective in a gradual over-pressurisation incident compared to a rapid over-pressurisation like a LBLOCA. This was due to the rapid rate in change of the containment atmosphere conditions associated with a LBLOCA.

In conclusion, the objectives of this dissertation study were met by evaluating literature and obtaining an understanding of the basic construction of the generation III PWR nuclear containment building and its purpose, as well as developing a thermodynamic model based on the LOCA simulation results for the containment temperature and pressure changes. The dissertation also examined the two important severe accident management installations to a nuclear containment building, i.e. containment spray system and CFVS. From the simulation data, it was seen that the containment spray system will effectively aid in reducing the temperature and pressure inside the containment building following a LBLOCA, whereas the CFVS only affected the overall pressure inside containment. The wet and dry type CFVS were also evaluated in terms of advantages and disadvantages as well as the design characteristics of four different CFVS designs.

The results and graphs obtained in literature were similar to those obtained in the 3 simulated scenarios but may have differed due to plant-specific data and design of KNPS which was used as a case study.

With the inclusion or installation of CFVS, it is important to be mindful that it comes with its own consequences. This includes the risks of inadvertent actuation and damage from external hazards since the CFVS is situated outside containment. Likewise, more gaps identified, that must not be overlooked include:

- Vent Initiation and flow rate capacity
- Aerosol, iodine and radioactive nuclide loads and characteristics during BDBA
- Hydrogen accumulation inside CFVS
- Duration of operating CFVS unmanned
- Heat removal capability of the CFVS

Recommendations for future study include acquiring and using a simulation tool that is capable of evaluating beyond design based accident scenarios. Further studies looking at the radioactive nuclides released during and after a severe accident like a LBLOCA, as well as the efficacy of the incorporated CFVS to a PWR. The selection criteria and performance based matrix for this selection of a CFVS.

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Appendix A.

Input data used for simulation of accident scenario (typical Westinghouse PWR like Koeberg NPP)

Table 4: Initial conditions for LOCA model in case study

Initial conditions of the high pressure pipe	**Value**	**Units**
Inner diameter	0.74	m
Initial pressure of water	154	bar
Initial temperature of water	280	°C
Volumetric flow rate of water	6.1	m^3/s
Initial volume of water	202	m^3
Containment initial conditions		
Pressure	0.96	bar
Temperature	30	°C
Steam dryness fraction	1	
Spray system water temperature	20	°C
Boundary conditions		
Spray system injection pressure	15	bar
Volume of containment building	49500	m^3
Rate of water injection	785	$\frac{m^3}{hr}$

Appendix B.

Koeberg Simulator data and results for the scenarios

Table 5: Scenario specified data for engineering simulator

	Scenario 1	**Scenario 2**	**Scenario 3**
Simulated CFVS	No	No	Yes (15 kg/s and 10 kg/s)
Passive Accumulator Safety Injection	Yes	Yes	Yes
Passive Hydrogen absorption (PARs)	Yes	Yes	Yes
Diesel reload signal allowed (1 diesel)	Yes	No	No
Safety systems activated (containment spray and safety injection systems etc)	Yes	No	No

Table 6: Scenario data and results

	Scenario 1	Scenario 2	Scenario 3
Peak Containment Pressure (bar)	4.135 (at 70 s)	4.011 (at 54 s)	4.135 (at 54 s)
Peak Containment Temperature (°C)	126.8	inconclusive	Inconclusive
Containment Integrity maintained	Yes	Inconclusive	Inconclusive
Time to failure (maximum)	15 minutes	11 minutes	13 minutes
Operator actions	none	none	Optional
Minimum reactor core level (m)	7.2 m at 120 s	3 m at 870 s	3 m at 52 s
Maximum core temperature	Inconclusive	Inconclusive	Inconclusive

Note: Pipe break or LBLOCA initiated at time, t=0.

The Core temperature instrumentation information:

- Unit : °C
- Range: Minimum = 0
  Maximum = 350
- Instrument functional location: - RCP028MT/ RCP044MT = Reactor in-core outlet temperature
  - RCP029MT/ RCP43MT = Reactor in-core inlet temperature
  - RCP055MT = Reactor hot leg (SG inlet temperature)
  - RCP056MT = Reactor cold leg (Reactor inlet temperature)

Scenario 1:

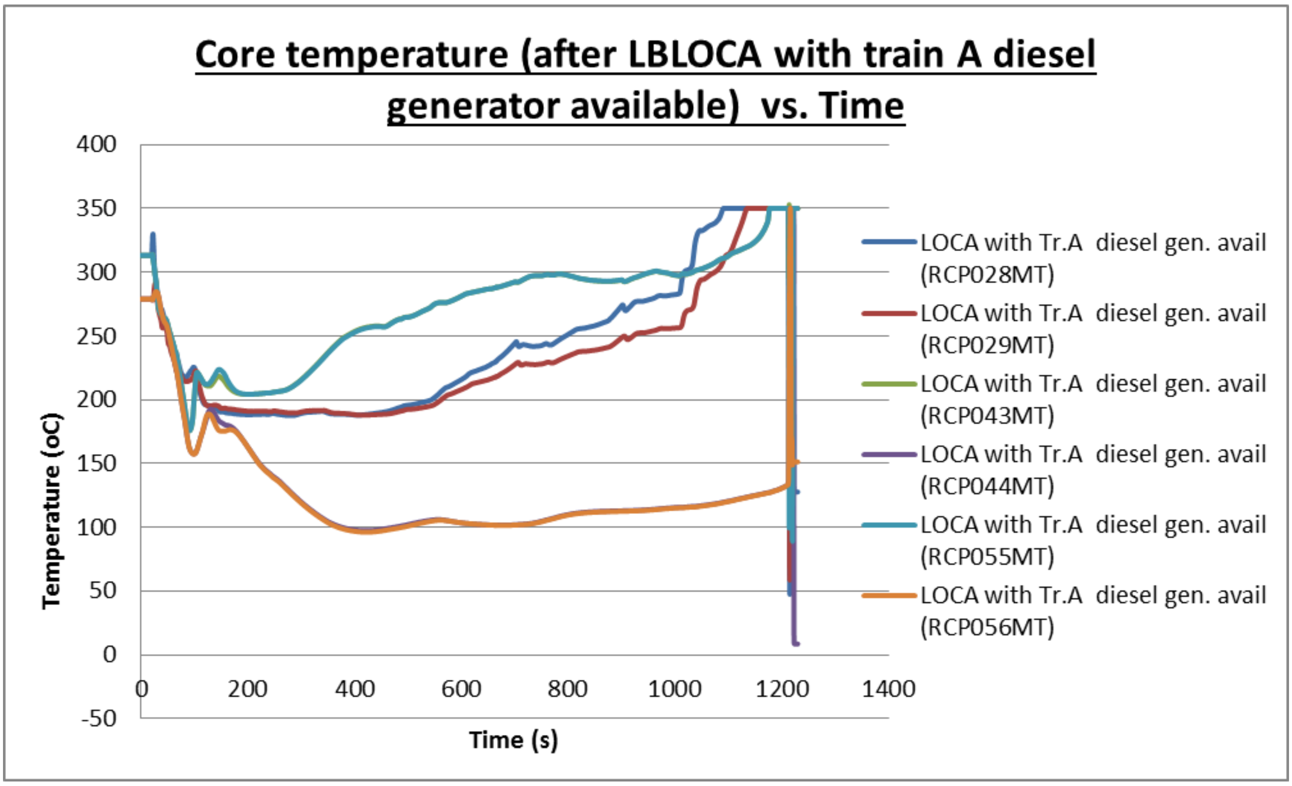

Figure 34~~18~~: Core temperature profile for LBLOCA with one Diesel Generator available

Scenario 2:

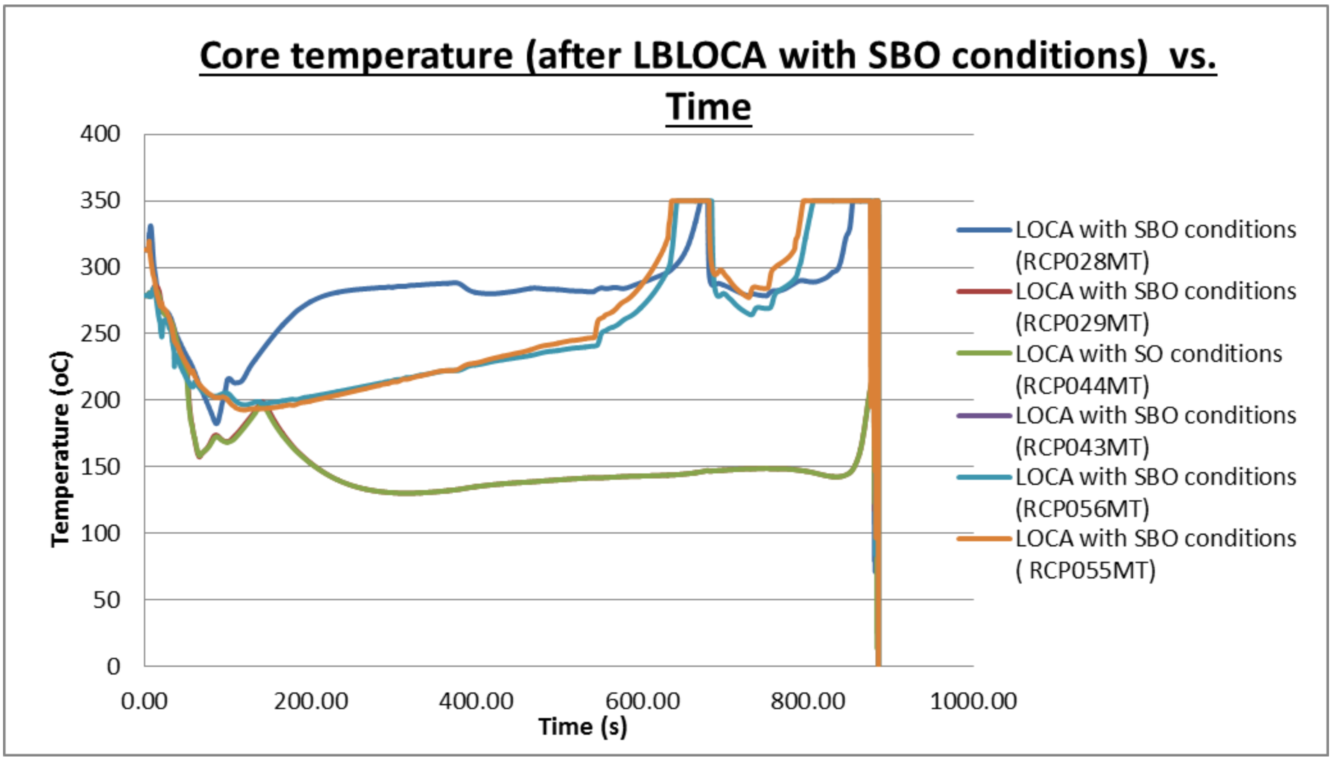

Figure 35~~22~~: Core temperature profile for LBLOCA with SBO conditions

Scenario 3:

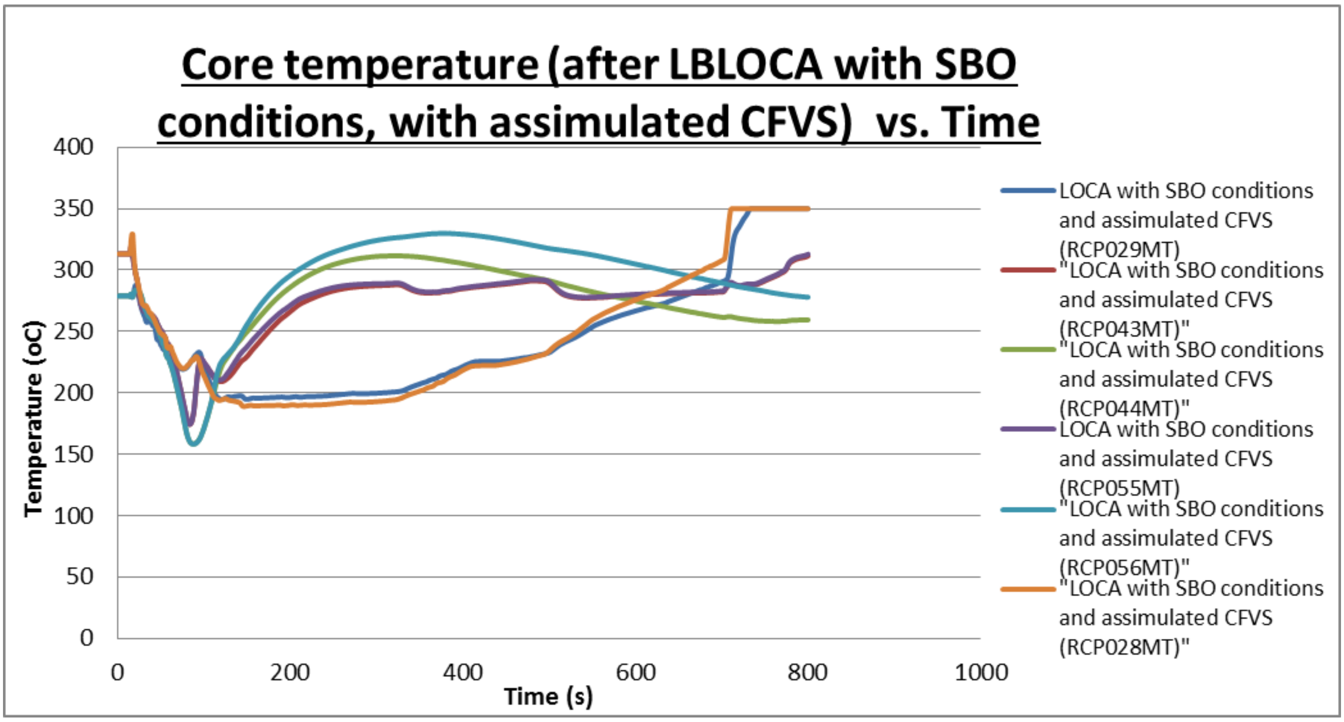

Figure 36~~30~~: Core temperature profile for LBLOCA with SBO conditions including assimulated CFVS (extraction raet of 15 kg/s)

Appendix C.

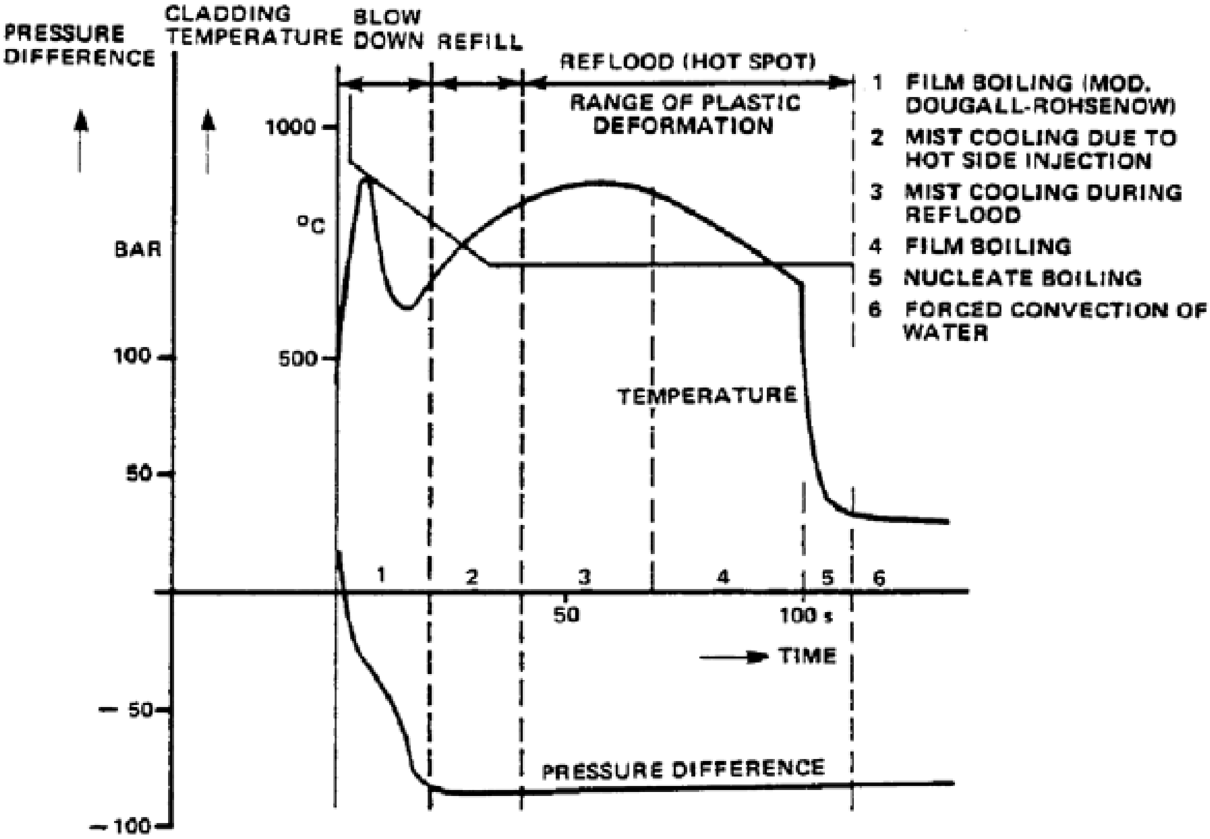

Figure 34: Pressure and temperature change history of a Westinghouse PWR fuel rod following a LBLOCA [30]