

THE CONSEQUENCE OF STATION BLACKOUT IN A NUCLEAR POWER PLANT



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Date

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Abstract

The main purpose of the nuclear power plant is to generate electricity and to supply it to the grid and its auxiliaries systems. Since nuclear safety overrides the production of electricity and therefore nuclear power plants are designed to protect the public and the environment from the radiation release. In order to achieve this design criteria, the nuclear safety regulations requires that the plant must have safety features incorporated in the design to remove the decay heat in a safe manner and to prevent the release of radiation to the environment. Some of the safety systems rely on constant supply of electricity to achieve their function of cooling the reactor core and to safely shutdown the plant. The total loss of Alternating Current power supply to the essential and non-essential equipment of the plant is called Station Black Out. It is critical that the system that removes decay heat in the reactor is effective to prevent the core melt. One of the design basis accidents that the nuclear power plant is design to withstand is loss of coolant accident during Station Black Out. The loss of coolant accident is more likely to occur through the primary pump seals if they do not receive adequate cooling. In this research, the seal leak rate through the primary pump was quantified using a simulator model of a Pressurised Water Reactor type, the quantified leak rates were compared to the recent Station Black Out accident that happened at a Boiling Water Reactor (Fukushima Nuclear plant Unit 1) in Japan. The simulation results of primary pump seals leakage together with the values obtained from Fukushima event were analysed and compared with primary pumps seal leakage flow rates from Westinghouse analysis. Westinghouse assumptions and analysis predicts the primary pump seal leakage during Station Black Out accident to a value of $54\text{m}^3/\text{h}$. The Westinghouse predictions were independently verified by Energy Technology Engineering Centre in response to Nuclear Regulation Council concerns regarding seal leakage rates on the Westinghouse Reactor Coolant Pump designs during the Station Black Out. The Energy Technology Engineering Centre calculations developed finite element structural models for all three seals to predict thermal and pressure distortion. The seal leakage of $45\text{m}^3/\text{h}$ was determined using two-phase correlations based on Dukler constant flow slip flow model [1]. The Pressurised Water Reactor simulation performed in this research predicts a seal leakage of $39\text{m}^3/\text{h}$ which is not far off from the studies.

The radiation release due to Station Black Out was analysed and estimated using a source term code called Rascal. The amount of radiation released by a Pressurised Water Reactor plant determined to be higher than Boiling Water Reactor. The research identified the lessons drawn from the Fukushima accident. The research also highlighted the importance of effectiveness of safeguard systems that are designed to mitigate the accident during SBO.

Discussionadded

Chapter 1: Introduction

1.1 Background

It is a mandatory design requirement that the design of the nuclear power plant makes provision that it supplies its own power from its own dedicated generator to power its essential safety equipment required for the safe shutdown of the reactor; the rest of the electricity produced is supplied to the national grid. In case of a generator trip, the power to the auxiliary equipment will be supplied from the national grid, which powers the equipment essential for the safe shutdown of the plant. In addition to the generator, there are also onsite emergency diesel generators that are installed to serve as back-up essential electrical switchgear to supply the emergency equipment with power when the plant generator and the grid are not available. All these power supplies have a certain frequency of failure associated with them, and should at any stage all of them become unavailable at the same time; it will make removal of decay heat from the reactor core very difficult. Station Blackout (SBO) is a condition when all Alternating Current (AC) to the essential and non-essential switchgear is unavailable for safe shutdown of the plant [2].

The quick restoration of the electrical power is important for nuclear safety to prevent the core melting. Such events can lead major accidents which are classified as level 7 rating on the International Nuclear and Radiological Event Scale (INES). INES classifies events on a scale of Level 1 to Level 7, whereby Level 1-3 are termed “incidents” and Level 4-7 are termed “accidents” [3]. The most recent SBO event took place in Japan Fukushima nuclear power plant when a Tsunami hit the power plant causing the station to lose all AC power supply including the emergency diesel generators, the event was rated Level 7 on the INES scale [4] [5]. Level 7 is defined as “an event resulting in an environmental release corresponding to a quantity of radioactivity radiologically equivalent to a release to the atmosphere of more than several tens of thousands of Terabecquerels of ^{131}I [3].” One of the biggest threats to nuclear safety is a loss of coolant accident (LOCA) which can happen during the station black out. The small break loss of coolant accident (SBLOCA) is highly likely to happen if the reactor coolant primary pumps (RCP) seals (Figure 1) fail due to loss of cooling.

The primary pump seal plays a very important role as part of the boundary of the primary system to prevent hot primary water from leaking out. Under normal operation mode, the boundary is maintained by a buffer of cool water at high pressure which is injected between pump impeller and the seals; the majority of the cool water passes through the seal and exit the pump via a dedicated return line. Another independent cooling water system provides cold water through an integral heat exchanger called thermal barrier which is located between the impeller and the pump bearing [6]. During SBO event the high pressure water system and the independent cooling system are lost which causes primary water to leak out through the seal and ultimately causing LOCA.

A LOCA can lead to radiological release to the environment if the fuel elements are defective and the highly active water is not properly contained. The design base accidents postulated in the design of a nuclear plant considers the release of a variety of fission products. The accident includes amongst others the release of activity normally circulating in the primary coolant. Radioactive materials, which are calculated to be released, include the noble gasses, organic iodine, elementary iodine, and particulates such as cesium, tellurium and ruthenium [7].

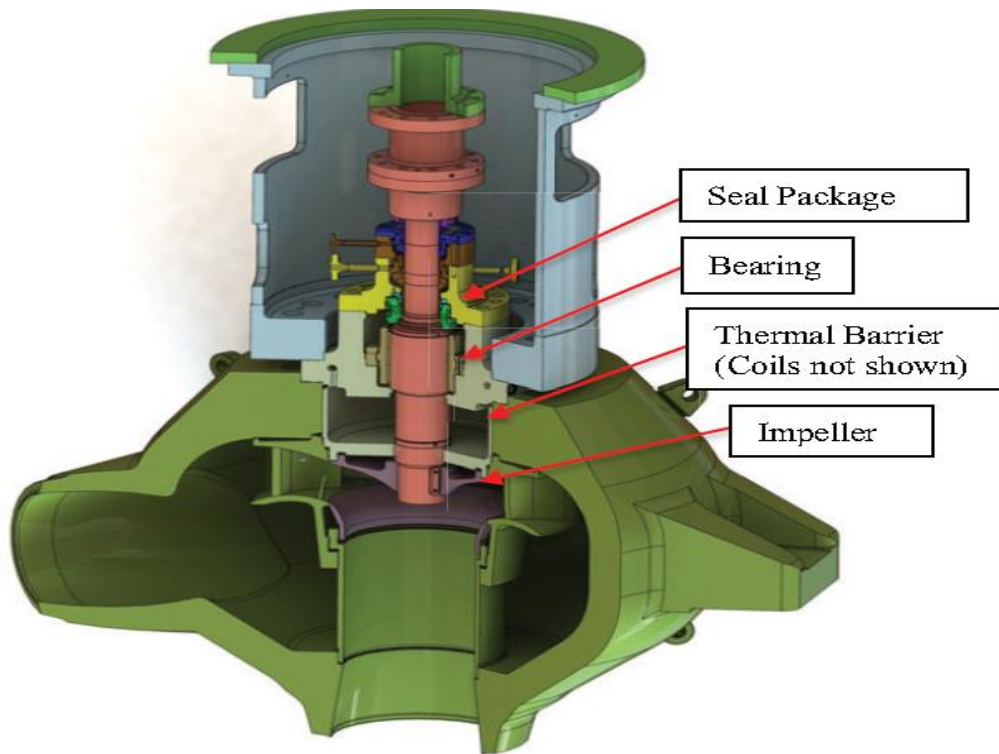


Figure 1: Reactor Coolant Pump [6]

1.2 Research Objective

The research objective is to study the effects of SBO accident and evaluate its potential impact on the Nuclear power plant. The research will focus on the following items,

- Reactor coolant pumps seal failure and subsequent LOCA through the seals
- Release of radioactive material during SBO accident.

This research will study the Fukushima accident and investigate how the leak through the primary pumps seal could have contributed to the initial loss of coolant, which ultimately resulted in the core damage. The leak through the primary pump seal was an inconclusive matter according to TEPCO report [5] and this research seeks to examine the possibility of LOCA due to seal failure.

The research will also evaluate the amount of radioactive release due to leak from the primary pump seals based on the source term.

1.3 Problem Statement

The main research question emanates from one of the common issue #4, which was not adequately explained and was an outstanding matter in the TEPCO Fukushima report.

Common /Issue-4: Pump seal failure

“Meanwhile, mechanical seals were mounted on the primary loop recirculation pumps (PLR pumps) as a shaft seal. During normal operations, sealing water for the shaft seals provided from the control rod drive (CRD) pumps prevents reactor water from leaking. When the external power supply was lost, CRD pumps were shut down and sealing water was lost, then the high-pressure reactor water was discharged to the drywell (D/W) equipment drain sump via the PLR pump shafts and shaft seals. Examinations are needed to determine how much water actually leaked” [5].

1.4 Research method and approach

1.4.1 Reactor coolant pump seal failure and LOCA simulation

The simulation was performed on the Koeberg Nuclear plant simulator second platform. Second platform is a virtual simulator designed for the purpose of training reactor operators and for research purpose by engineers. Koeberg power station is the three loop Westinghouse Pressurised water reactor (PWR) plant design; it was only used as reference since it is different from Fukushima Unit 1 which is boiling water reactor (BWR) plant design. However, the principle of the seal operation is the same for both BWR and PWR reactor coolant pumps irrespective of the differences in designs.

The Simulator model is built using Orchid® software created by L3-MAPPS. The core is built from SCIENCE code using Unit 1 Westinghouse fuel data as input. This is tuned to the Unit 1 Physics Data Book (PDB). The simulator core model uses the Nodal Expansion method (NEM). The scenario uses the cycle 24 core. It starts at full power operations at End of Life (EOL) conditions with 0% fuel cladding damage [8].

A PWR simulation replicates similar events that happened in Fukushima Unit 1 on the day of the accident, which happened on 23 March 2011. The simulation was designed to follow the events of the Fukushima accident as stipulated in the TEPCO report [5]. The events of interest for the research purpose will be from the time of the Earthquake up to LOCA event. These events were selected on the basis that whatever happened beyond event LOCA was the recovery process from the accident, which was operator-induced events. The recovery process is when the operator control the plant based on the accident procedures. The research focus is on what would have been the response of the plant if there were no interventions by operators. The main interest is based on when operators lost control and monitoring of the plant.

The primary pump seal leakage is simulated as a cold leg at the pump break discharge. The break size is selected in such a way that the break flow will equal a pre-selected value of seal leakage rate. The analysis of the break flow is dictated by the critical flow correlations for subcooled and saturated water, this is to accommodate for the changing conditions of the primary systems during the analysis. The approach produces a more realistic representation of the response of a primary pump seal leak in particular the degradation due to overheating of the seal. Variations from the actual leak are expected due to over prediction of the leak rates; the deviation is a conservative approach.

In reality during SBO event the seal degradation, if it occurs, will extend over period lasting from several minutes to hours. The most likely result in this gradual degradation would be a slow increase in pump leakage from only about 1 m³/h per pump at the time of reactor trip to potentially larger leak rates at some point later in time [9].

Due to the several interacting considerations, it is difficult to analytically or experimentally predict the primary pump leakage flow following a total loss of all seal cooling. Furthermore, it is possible that different primary pumps will be affected in different ways with resulting variations in leakage flow. Since it is difficult to accurately predict primary pump seal leakage flow following a loss of all seal cooling, field experience has been reviewed for relevant information and an analysis has been performed to estimate a maximum leakage rate [9].

1.4.2. The source term

The “source term” is the radioactive material composition and its magnitude available within the containment to be leak to the environment. It includes the radioactive materials (chemical and physical properties) within the containment that are available for leakage from the reactor to the environment. Any radioactive material escaping the containment during the accident involving core damage is referred as the radiological release to the environment. The amount of radiological release to atmosphere is determined from the containment leak rate and airborne radioactive inventory in the containment atmosphere. The source term is a function of time and considers the fission products from the reactor core. It also considers the plant accident mitigating features such as containment sprays capability to remove the fission product. The source term is used for judging the acceptability of both the plant site and the effectiveness of engineered safety features [10].

The source term analysis was done using the Radiological Assessment System for Consequence Analysis (RASCAL) code. RASCAL code has been developed for the U.S. Nuclear Regulatory Commission. It is designed to be used in the independent assessment of dose projections during response to radiological emergencies. RASCAL was used by response personnel to conduct an independent evaluation of dose, for consequence projections and for training and drills. It allows consideration of the dominant aspects of source term, transport, dose, and consequences and supplements assessments based on plant conditions and quick estimates based on hand-calculation methods. It was specifically developed to evaluate releases from nuclear power plants, spent fuel pool storage and casks, fuel cycle facilities, and radioactive material handling facilities. RASCAL 4.3 consists of seven tools. Four tools are designed for use in consequence assessments. These tools include: Source Term to Dose (STDose), Field Measurement to Dose (FMDose), Radionuclide Data Viewer, and Decay Calculator. STDose and FMDose are the primary tools used to estimate doses. The Radionuclide Data Viewer provides users with direct access to the RASCAL radionuclide database which contains information on decay and dose coefficients. The Decay Calculator is a legacy code that provides user’s estimates of the activities in the decay chain of a parent radionuclide at some time given the activity of the parent at time zero [11].

In addition to LOCA, which had always been part of RASCAL, it has recently been upgraded to include SBO accident which is based on the State-Of-The-Art Reactor Consequence Analysis (SOARCA) study [12].

Two scenarios were simulated on Rascal code for radiation release results. The first scenario was simulated for a PWR plant type and the second scenario was for BWR of Fukushima unit 1 type plant. In both scenarios RASCAL code was used to determine the following:

- The airborne radioactivity which was available to be released from the containment during the accident;
- The calculation of offsite doses at the exclusion area boundary (EAB) for the first two hours after the accident and at the low population zone (LPZ) boundary for the course of the accident. The need for evacuation based people living downwind from the projected path of plume and bordering sectors. This area, the downwind sector and two adjacent sectors, affords protection from potential wind shifts and plume meander. This is known as a "keyhole" (as shown in Figure 2) because of its appearance.

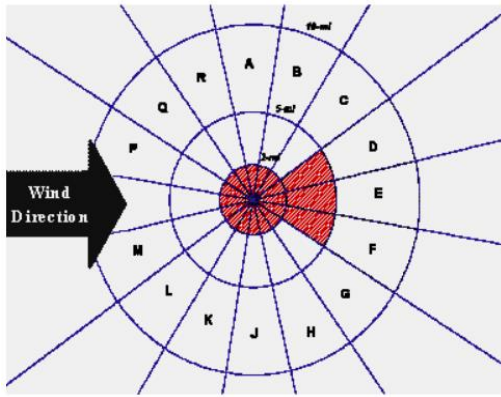


Figure 2: Illustration of the keyhole concept in Emergency Planning

Chapter 2: Literature review

2.1 Station Blackout

NUREG- 1150 [13] provides the summary of the initiating event analysis based on the core damage frequency and containment performance from seismically initiated events. Core damage frequency is defined as the combination of initiating events, hardware failures, and human errors which lead to core uncover with the flooding of the core unexpected. The accidents are estimated to be occurring during normal full power, the events are categorised as internal and external initiating events. The loss of offsite power is historically categorised as an internal event. The internal initiating events are classified as transients which will require reactor shutdown.

The total core damage frequency (see Figure 3) is divided by contribution of plant damage states such as:

- Station blackout (SBO)
- Anticipated trip without scram events (ATWS),
- LOCA (from pipe breaks, pump seal failures and failed relief valves),
- LOCA with containment bypass (steam generator leaks and primary system interconnecting systems),

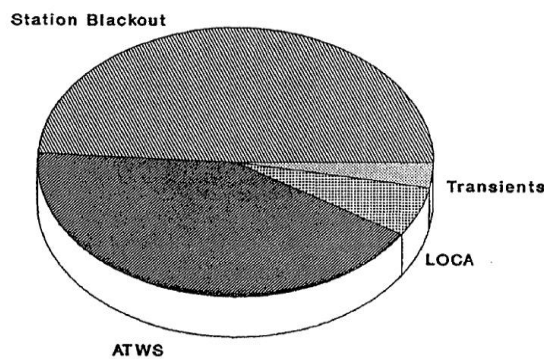


Figure 3: Total Mean Core Damage frequency [13]

The unavailability of power will have a significant adverse effect on the plant to establish and maintain safe shutdown conditions. Risk analysis indicates that loss of all AC power will contribute more than 70% of the overall risk in certain nuclear plants. US Nuclear Regulatory Commission (NRC) issued Station Blackout rule and regulatory guide (RG) 1.155 [2]. This was done to assist the Nuclear Power Plants (NPPs) to have the capability to withstand the SBO. Some NPPs made modification by installing additional emergency AC power source in order to supply the important and vulnerable systems of the plant such as Reactor Coolant Pump (RCP) seals and Direct Current (DC) battery backed boards.

In the SBO event, the unit will fail to supply its own auxiliaries, the onsite emergency diesel generators will also fail to start and supply the essential boards. Most of the essential safety systems required for removing the reactor core decay-heat and containment heat depends on the AC power to operate, their function will be impaired and the only hope for plant will be on the availability of the passive systems that do not require the AC power. The example of passive system used in the PWR is accumulators which operates on a principle of equalising pressures between two volumes and do not rely on any external signal. Passive components are preferably used in variable nuclear plants designs, due to their high level of reliability (low risk to fail) [14].

The regulatory guide expresses three areas of concern when planning for SBO [2];

- Maintaining a reliable AC power system
- Proper procedures and training of personnel to restore AC power timeously.
- Ensuring plant capability to cope with station blackout

2.1.1. Reliability of AC power

Reliability of the ac power system is affected by the vulnerability to common cause failure associated with design, operational and environmental factors. The plant should meet the minimum acceptable SBO capability of 2 to 16 hours; the duration will depend on the plant's characteristics. If the plant actual SBO capability is less than minimum acceptable duration, then a modification is necessary to ensure the plant ability to cope with SBO [15] [2] .

2.1.2. Plant procedures

Plant procedures should include the necessary actions to restore power. The loss of offsite power can be caused the following;

- Grid under-voltage and collapse
- Weather-induced power loss
- Power distribution system faults

The procedure and supporting analysis are primarily structured to address the loss of all ac power as an initiating event that occurs when the plant is in the start-up or power operational mode. However, the guideline should be augmented to provide appropriate guidance should a concurrent loss of reactor coolant, loss of secondary coolant or steam generator tube rupture exists [15].

2.1.3. Plant capability to cope with SBO

Evaluation of plant specific capability is based on the following items [16] [17] [18];

- Evaluation should assume that plant was operating at full power 100% for at least 100 days.
- Evaluation of systems, structures and components (SSC) required for removing decay heat and providing core cooling are availability, examples are back-up batteries, water storage capacity and compressed air.
- Inventory loss evaluation considering shrinkage and leakage from pump seals.
- Effects of environment on safeguard and supporting systems, including fire protection.
- Effects of weather and recovery of on-site power buses including emergency diesel generators
- Evaluation of control room habitability during SBO.
- Evaluation of use of non-safety related equipment to deal with SBO.
- Evaluation of timeous response of the control room and field operators to take necessary actions.
- Evaluation of containment integrity, which includes all valves going in and out of the containment.

2.1.3.1 Reactor power history

The reactor power history is an important factor to consider when evaluating the specific capability of the plant to cope with the SBO [16]. The amount of decay heat that exists after reactor shutdown is a function of time at which the reactor was at power. The residual heat in a sub-critical core consists of fission product decay-energy, decay of ^{238}U neutron capture products and residual fissions due to delayed neutrons [15].

A typical scenario is illustrated in the in Figure 4 [15].

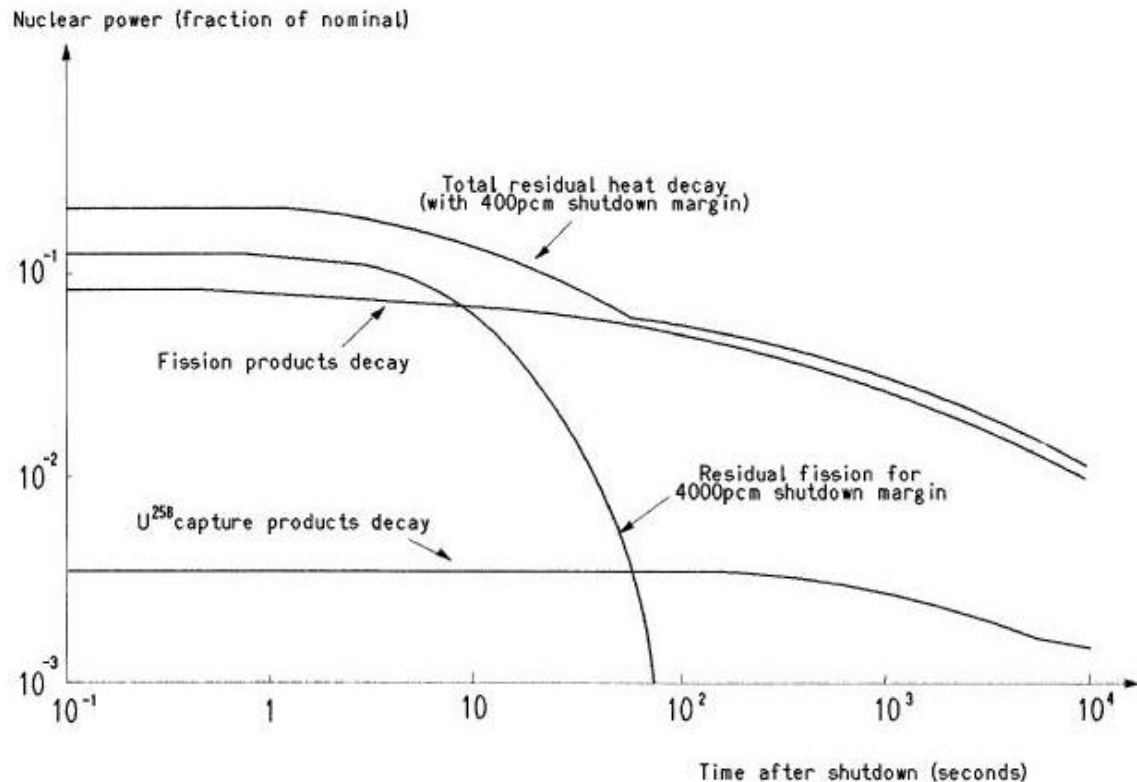


Figure 4: Nuclear power decay after shutdown [15]

Beta and gamma radiation from the decay of ^{239}U (23.5 minutes half-life) and ^{239}Np (2.35 day half-life) contribute significantly to heat generation after shutdown. The residual thermal power following reactor shutdown is split into three components: (1) the residual thermal power due to residual fission, (2) the decay of capture products ^{239}U and ^{239}Np , (3) the decay of fission products and actinides except ^{239}U and ^{239}Np [16]. The thermal power due to residual fission is the highest component of the total residual thermal power during at least the first 50 seconds following a reactor trip. The thermal power due to residual fission after reactor shutdown depends on the number of neutrons generated by the different neutron sources in the core; the potential neutron sources are the following [19]:

- the decay of delayed neutron precursors,
- the spontaneous fission of actinides,
- (α, n) reactions.

The second and the third components are negligible in comparison with the first component under sub-critical conditions. The main parameters that influence the second and third components are [16] [19]:

- Initial fuel enrichment,
- Burn up history of each fuel assembly,
- Outage duration between the consecutive cycles.

2.1.3.2 Evaluate SSC required for removing decay heat and providing core cooling are available, examples are back-up batteries, water storage capacity and compressed air

The prevention and limitation of accidents is based on the concept of “defence in depth”, based on three levels [20]:

- Inherent conservative design of the plant and the quality of the fabrication provides the first level. It provides the maximum ability of the plant to operate safely during normal operations.
- The second level is designed based on the postulated events, which will lead to abnormal transients and incidents. The design is in such a way that it will prevent accidents from happening by safely shutting down the plant.

- The third level is designed based on the hypothetical accidents, which can affect the integrity of the plant and fission product barriers. To mitigate the consequence an accident, at least two redundant safeguard systems are designed with the assumption that one is likely to fail.

All the levels of defence will obviously be complemented by the well-trained licence operators and technical staff equipped with the Emergency Operating Procedures and Emergency Planning [15].

The SSC's that are important for safety must be designed, fabricated, erected and tested to quality standards corresponding to their safety function. Quality assurance programme must be established which used the recognised international standard and quality programmes. The plant should be designed with a system that is capable of removing residual heat. Its safety function shall be to remove the fission product decay-heat and residual heat from the reactor core at the rate that will not challenge the integrity of the reactor vessel and the fuel design limits [20].

In a PWR, systems such as steam generators are used to evacuate heat from the core without exceeding the design limits for the fuel and the reactor coolant boundaries. They are used from the first phase of cool down and they rely on the availability of the condenser and the turbine bypass system [21].

The SG's are backed up by the auxiliary feed-water systems and atmospheric relief systems. At lower temperature and pressures (below 177°C and 2.5 kPa) when the SG's can no longer remove heat, the reactor coolant circulation and cooling is ensured by the residual heat removal system which consist of pumps and heat exchangers. The heat exchangers of the residual heat removal systems are cooled by component cooling system, which is ultimately cooled by the essential services water systems [22].

An emergency core cooling system (ECCS) provides a safety function to remove heat from the reactor core after the loss of reactor coolant at the rate that will reduce the fuel clad damage and clad metal water reaction. The ECCS design function is to maintain the core in coolable geometry and to ensure that the metal-water reaction remains below 1%. The ECCS in Figure 5 consists of the redundant low head and the high head safety injection which are electrically backed from redundant emergency bus [14].

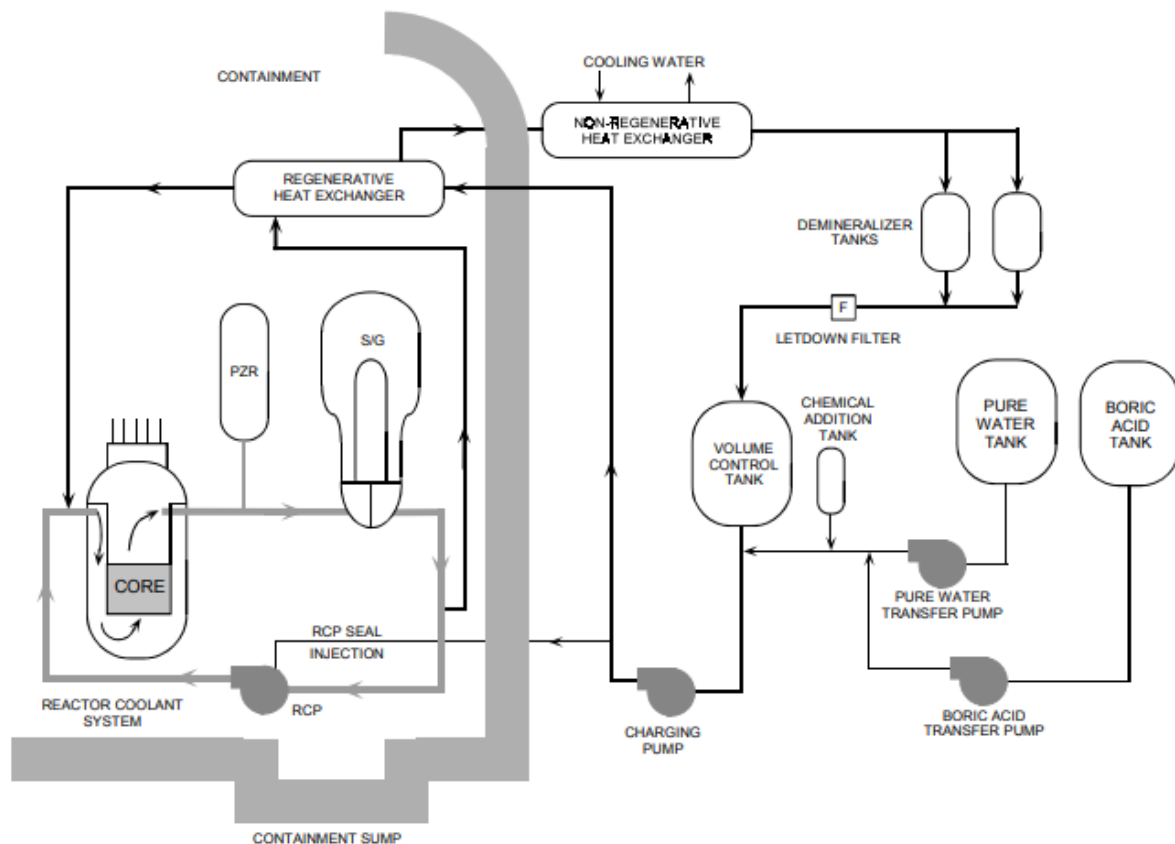


Figure 5: PWR ECCS schematic [23]

Redundant pumps and heat exchangers provide the containment spray system as shown in Figure 6. Its purpose is to cool and depressurise the containment following a large break LOCA. The water storage tank is provided for both ECCS and containment spray system. Both systems are initially connected to the tank and when the tank is used up the connection is automatically changed to sumps to recirculate the water and provide continuous cooling of the reactor coolant systems [8].

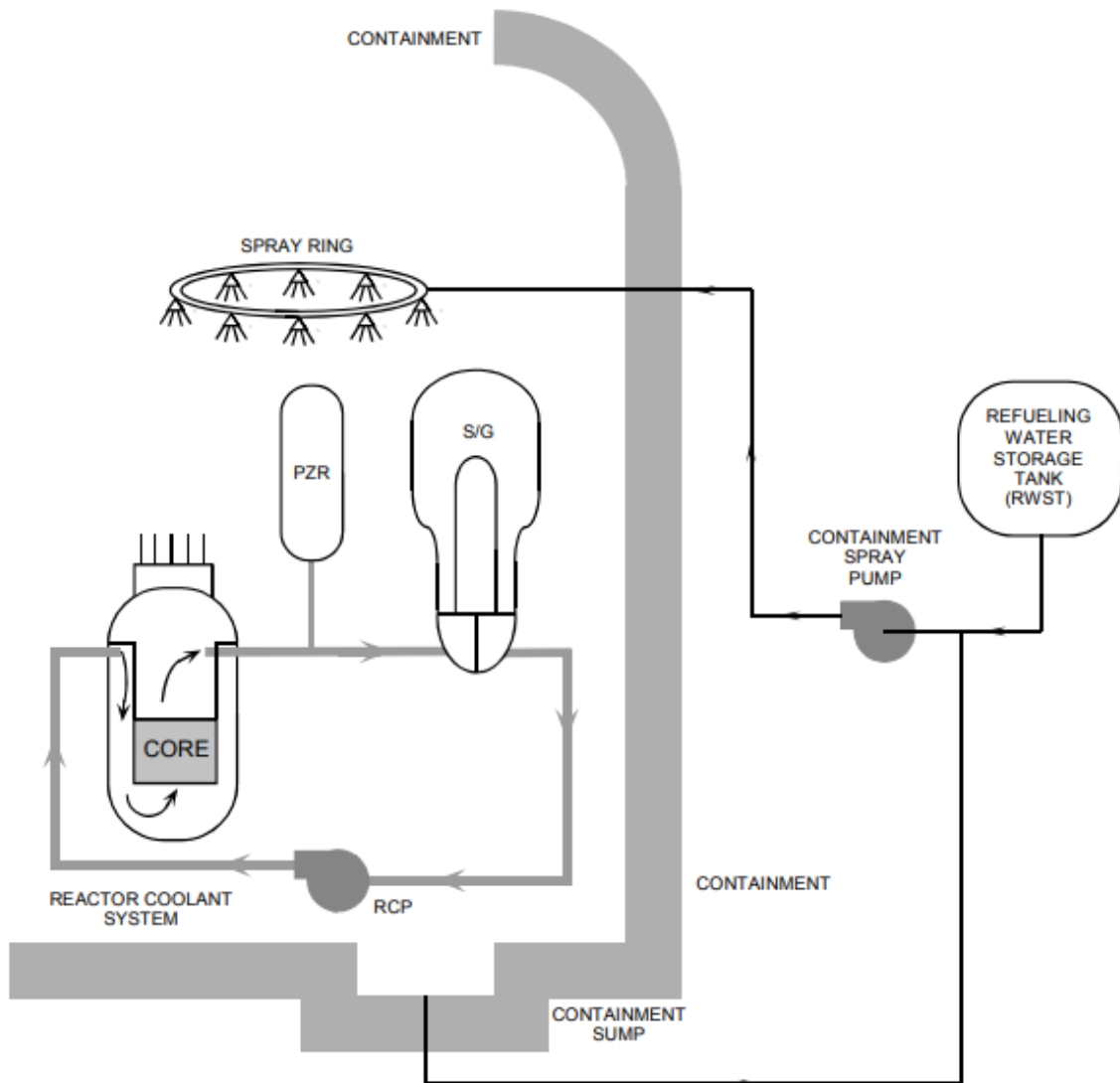


Figure 6: Containment Spray system [23]

The BWRs are also equipped with a safety grade ECCS to mitigate adverse effects of a LOCA by maintaining core cooling to prevent core damage. These systems also play a role in beyond design basis accidents (BDDBA), and anticipated transient without scram (ATWS) and SBO. High pressure coolant injection system (HPCI) is a safety system designed to maintain adequate reactor vessel water inventory and core cooling during small break LOCAs as shown in Figure 7 it also aids in depressurization of the reactor vessel to allow low pressure ECCS to inject during intermediate break LOCAs. The HPCI system is independent of ECCS; it does not require AC power or instrument air to perform its function. The steam from the main steam supply system drives the turbine. The HPCI system consists of a turbine, turbine driven pump and the auxiliary systems required for turbine operation and associated piping and instrumentation [24].

In a PWR a similar the pressuriser uses the same level programme, the sufficient make-up should be evaluated to ensure that core temperatures are maintained at acceptable values on loss of coolant through the pump seals for the duration of the SBO [26].

During the loss of level which beyond the capability of the normal makeup system the ECCS are designed to compensate for the inventory loss [14].

2.1.3.4 Environmental effects on safeguard and supporting systems, including fire protection

The design of the Safeguard and support systems should be such that they are able to withstand the environmental factors such as the vibration, thermal expansion, radiation, corrosion, dynamic effects, mechanical loadings, humidity, high temperature, high pressure, low temperature, vacuum, water, steam or chemicals. The safeguards and supporting systems important to safety shall be designed to withstand the effects of the environmental conditions during normal operations, testing, maintenance and postulated accidents [27]. Firefighting system shall be designed with the adequate capacity and capability to reduce the adverse effects of fire on the SSC's important for safety. The fire system operation should not impair the SSC functionality [17] [18] [2].

2.1.3.5 Effects of weather and recovery of onsite power buses including emergency diesel generators

Evaluation on the severe weather such as storms that can affect and disrupt the power supply to the station should be performed. The power supply failures due to weather can be divided into two groups [18]:

1. Weather induced SBO events which do not affect the restoration time. Examples include lightning and other events that are not severe. These types of weather-related power supply losses are either grid related or plant centred.
2. Weather induced SBO events that cause the adverse conditions over a broad area such that the restoration time is long. The examples of these are severe weather conditions such as hurricane, high winds, snow, ice storms and tornadoes [17] [18] [9].

2.1.3.6 Evaluate control room habitability during SBO

Control shall be designed to allow actions to be taken promptly to operate the plant during normal conditions and during accident conditions. Adequate measures must be put in place to ensure personnel are protected from radiation exposure during accidents. The radiation exposure of personnel should not exceed the regulatory limit of 50 mSv (5Rem) whole body count, or its equivalent to any of the body during accidents [28].

Alternative shutdown panel outside the control room shall be provided to allow capability to maintain prompt hot shutdown (HSD) conditions and the capability to subsequently establish the cold shutdown conditions (CSD). The shutdown panel shall have all necessary instrumentation and the required procedures to place the plant in a safe state [29] [17] [18]. Functions to be carried at the shutdown panel:

- Heat removal,
- control of primary system pressure and volume,
- reactivity control,
- reactor protection,
- control of auxiliary circuits,
- communications

2.1.3.7 Evaluate the use of non-safety related equipment to deal with SBO

Removing the heat from the core remains the highest priority during SBO, in a PWR design, the steam generators remains the ultimate heat sink and the feed water supply needs to be maintained to evacuate the heat. The steam driven turbine pump will remain the single source of feed, the water source to make up the tank will need to be made up from non-safety system such as firefighting water or portable water

systems. For spent fuel pit, other source of make-up such as firewater should be used to maintain the level in the pit and ensure that the fuel is covered and cooled all the time. Mobile emergency diesel engines should be used to connect to the essential emergency buses in the absence of normal emergency diesel generators [30] [18].

2.1.3.8 Evaluate the timeous response of the control room and field operators to take necessary actions

Timeously restoring power can lessen the consequence of loss of power accident; the equipment required to mitigate SBO for the first 8 hours should be available to minimise the consequence. The time it takes for SBO to lead to core damage accident depends on the capability and the reliability of the decay heat removal systems that are not dependent on AC power. The capability and the availability of the decay heat removal system will allow time to restore the AC power to systems normally used to cool the core [9] [17].

2.1.3.9 Evaluate the containment integrity, which includes all valves going in and out of the containment

Reactor containment shall be designed in such a way that it ensures leak tight barrier against the uncontrolled release of radioactivity to the environment. The containment isolation valve should have double isolation with one valve situated outside where it can accessed and closed in case of failure of automatic closure [17].

2.2 LOCA review

LOCA is defined as “A breach of the reactor coolant pressure boundary which results in a leak rate beyond the normal makeup capability of the plant” [31]. Failure of large coolant pipe is one of the design bases of water-cooled reactors. The emergency core cooling needs to be in place to mitigate the consequences of a break. According to the NRC 10 CFR 50.6 the acceptance criteria for emergency core cooling systems for light-water nuclear power reactors is as follows [32]:

- *Peak cladding temperature*: The calculated maximum fuel element cladding temperature shall not exceed 1204°C.
- *Maximum cladding oxidation*: The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- *Maximum hydrogen generation*: The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
- *Coolable geometry*: Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- *Long-term cooling*: After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The fuel must maintain its coolable geometry during the LOCA sequence and the structural integrity of the rods should be maintained. The fuel cladding properties and behaviour is of the particular interest as it involves complex dynamics. The zirconium alloy used in the cladding, tubes for the fuel will oxidize slowly in the reactor water by the growth of the thin oxide layer on the surface (Figure 8); this happens during the normal operation and it forms a hexagonally closed packed α -phase zirconium. During the LOCA accident, when clad temperature increases to about 800°C the α -phase zirconium transforms into β -phase zirconium. The oxidation reaction significantly increases the oxide scale which becomes more brittle during the quenching of the hot fuel to lower temperatures. Ductility of cladding could be severely degraded if the degree of oxidation is high. It was discovered that if the embrittled cladding fragments into small pieces; the coolability of the core could be seriously impaired [32].

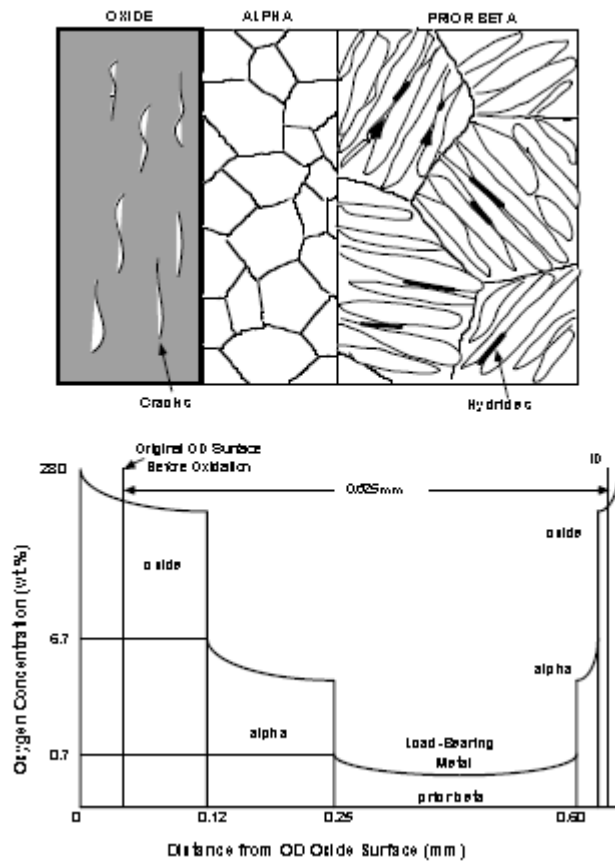


Figure 8: The schematic illustration of microstructure (top) and oxygen distribution (bottom) in oxide [32].

2.2.1 LOCA categories

Table 1 provides summary LOCA categories based on the size of the break, the flowrate or leak rate are depended on the size and the plant operating characteristics. BWR operates at the lower pressure than the PWR and therefore BWR will experience less flowrates.

Table 1: LOCA categories and break sizes [31]

Categories	BWR	PWR
Category 1 Small break LOCA Size (1.2 to 5 cm) Flowrates (27 to 409 m ³ /h)	It is a steam or liquid break in the reactor coolant system (RCS) where the leak-rate is within the capability of the makeup system	It is a steam or liquid break in the reactor coolant system (RCS) where the leak rate is exceeding the charging flow. The secondary system is required to cool the RCS and reaches the equilibrium pressure.
Category 2 Medium break LOCA Size (5 to 15 cm) Flowrates (409 to 1364 m ³ /h)	It is the liquid or steam break in the RCS where the makeup system cannot maintain the inventory. The break is however too small to depressurise the RCS for injection of the low pressure systems	The liquid or steam break is large enough to remove the decay heat without utilising the steam generators. The break is however too small to depressurise the RCS for injection of the low pressure systems
Category 3 Large break LOCA Size (> 15 cm) Flowrates (>1364 m ³ /h)	It is the steam or liquid break, which will rapidly depressurise the RCS. The High pressure system will not have adequate flow rates to match the break and level in the RCS will be lost.	It is the steam or liquid break, which will rapidly depressurise the RCS. The High pressure system will not have adequate flow rates to match the break and level in the RCS will be lost. The pressure will decrease to below the accumulators and low pressure injections.

2.2.2 LOCA through the failed RCP pump seal during SBO

Reactor coolant pump (RCP) seal failure refers to the loss of integrity of the primary coolant system pressure boundary through failure of the RCP seals designed to minimize reactor coolant leakage along the RCP shafts. Since leak rates resulting from the loss of RCP seal integrity may exceed the capability of the reactor coolant makeup system, the failure of RCP seals is a potential initiating event for small-break LOCAs [18].

A few older BWRs rely on maintaining primary system integrity to provide core cooling during SBO. These designs do not have the usual independently powered emergency makeup systems but utilize an isolation or emergency condenser that condenses steam from the core and returns it to the primary system. Pump-seal-induced LOCAs could be important to risk in this type of plant design since there is no means to maintain primary coolant inventory during station blackout [17].

NUREG- 1401 [33] proposed a resolution to reduce the of severe accidents associated with RCP seal failure by reducing the probability of seal failure, thus making RCP seal failure a small contributor to total core damage frequency. The following items were included in the proposed resolution [33]:

1. Treat the RCP seal assembly as an item performing a safety-related function similar to other components of the reactor coolant pressure boundary, applying quality assurance requirements consistent with Appendix to 10 CFR 50 and applicable General Design Criteria of Appendix A,
2. Provide RCP-manufacturer-recommended instrumentation and instructions for monitoring RCP seal performance and detecting incipient RCP seal failures, and

3. Provide RCP seal cooling during off-normal plant conditions involving loss of all seal cooling such as station blackout (SBO).

The third proposal is most important for LOCA accidents with SBO, many utilities has since resolved to modify their plants by building an independent emergency diesel which directly supply an emergency seal injection pump. The diesel generator is positioned at an elevated place at the level higher than postulated Tsunami height. This is to ensure a continuous supply of DC power for instrumentation and control and AC power to the backup pumps and valves required to supply seal water in case of total loss of power. The absence of such modification at Fukushima added to the difficulties of dealing with SBO accident [9].

BWR plants exhibit significantly lower leak rates from seal failures than do PWR plants, primarily because of the lower system pressure in BWRs. The use of reactor core isolation cooling (RCIC) systems, larger high-pressure coolant injection (HPCI) systems and feed-water makeup capabilities, and the presence of isolation valves in the reactor recirculation loops in BWRs reduces the potential for significant leak rates as a result of pump seal failures [18].

The major concerns involve seal failures due to adverse temperature effects on secondary seal elastomer materials as well as performance instabilities at the primary seal faces related to coolant flashing and two-phase flow. Current RCP secondary seal materials are susceptible to accelerated degradation when seal cooling is lost and seal temperatures approach normal reactor coolant system operating temperatures. Seal failures resulting in a small-break LOCA under station blackout conditions could lead to core damage due to the concurrent loss of normal and emergency reactor coolant makeup capabilities.

The estimated time for a core to be uncovered is between 8 to 12 hours if the reactor coolant pump seal leakage is larger than 2.7 to 5 m³/h per pump. If the primary makeup system is lost and the seals leak rate is more than 23 m³/h, the core can be uncovered within minutes [18].

2.3 Fukushima Unit 1 review

2.3.1 Normal operations cooling

BWR use a closed, direct steam cycle loop, as shown schematically on Figure 9. The working fluid is water that is used both as the coolant to remove heat and the moderator for controlling reactivity. Coolant water boils in the reactor core at a pressure of approximately 7MPa, and the steam that is generated is used to drive turbines to generate electricity. After passing through the turbines, the steam is condensed back to water by being cooled by the condenser tubes that are filled with cold water taken from a heat sink. The water resulting from condensation is then pumped back to the reactor as feed water [34].

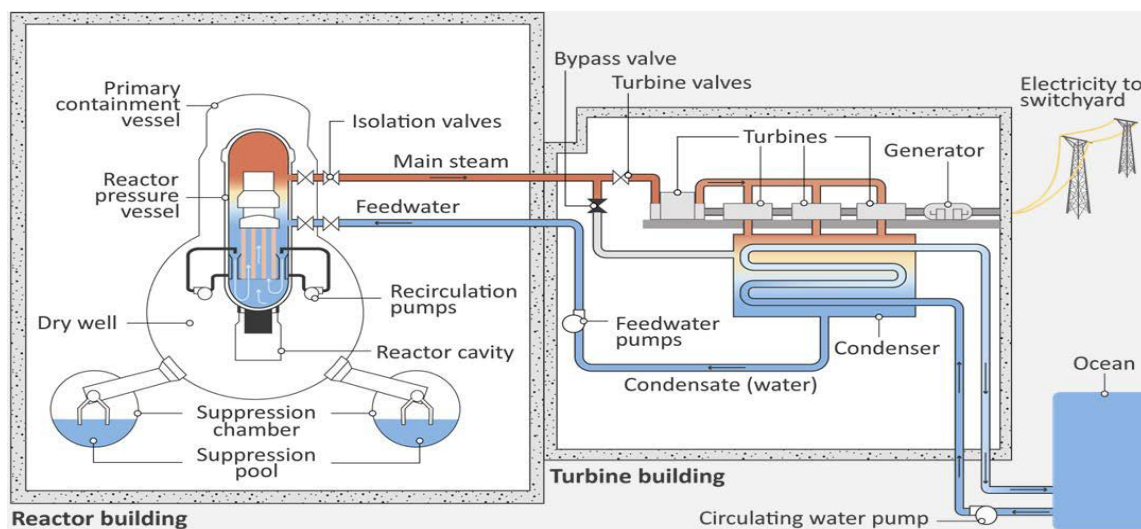


Figure 9: Schematic of a boiling water reactor [34]

A BWR level is normally controlled at 14 m by level control loop as shown in Appendix 2 and Figure 10. The top of the active fuel is at about 10 m which is very important, as the fuel should always be covered by water to prevent it from damaged.

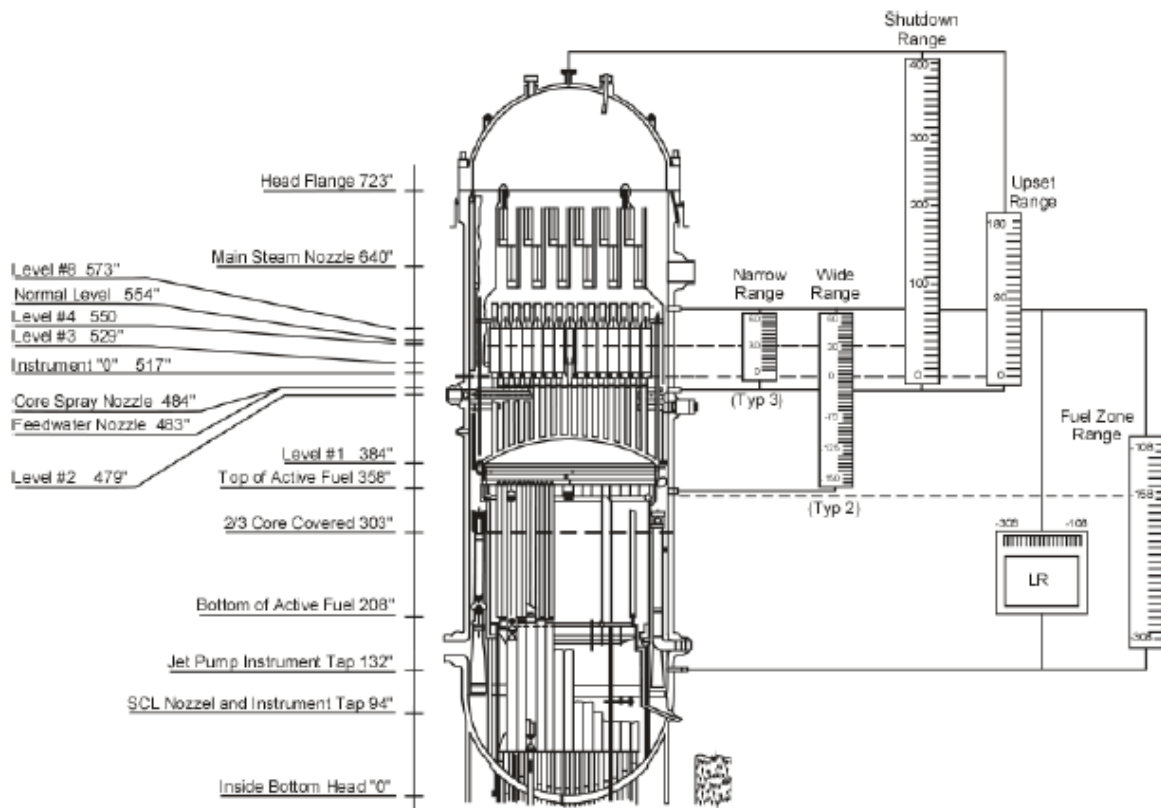


Figure 10: BWR Reactor level [24]

2.3.2 Emergency operations of BWR

Emergency core cooling

BWRs are equipped with a safety grade emergency core cooling system (ECCS) to mitigate adverse effects of a LOCA by maintaining core cooling to prevent core damage. These systems also play a role in beyond design basis accidents (BDBA), and ATWS and SBO [34].

High pressure coolant injection system (HPCI)

HPCI shown in Figure 7 is a safety system designed to maintain adequate reactor vessel water inventory and core cooling during small break LOCAs. It also aids depressurization of the reactor vessel to allow low pressure ECCS to inject water during intermediate break LOCAs. The HPCI system is an independent ECCS; it does not require AC power or instrument air to perform its function. The HPCI system consists of a turbine, turbine driven pump and the auxiliary systems required for turbine operation and associated piping and instrumentation. The HPCI system initially operates on an open cycle, taking water from the Condensate storage tank (CST), and injecting into the reactor vessel via a main feed-water line. When the CST water level is low or the suppression pool water level is high, the HPCI pump suction is aligned to the suppression pool and the suction line from the CST is isolated. Heat from the primary system is transferred to the suppression pool by the continuous or intermittent operations of the safety relieve valves (SRVs). Suppression pool cooling must be initiated to complete the heat transfer path to the

ultimate heat sink. The HPCI system provides primary system make-up at RPV pressures ranging from the operating pressure to injection pressure of low pressure ECCS. By supplying cool water and using the steam generated by the decay heat for driving the HPCI turbine, the reactor depressurizes and allows the injection by the low pressure ECCS. When the primary steam pressures are less than about 1 MPa, the operation should be terminated because of poor steam conditions for the two-stage HPCI turbine drive that would affect the system's reliability. System initiation can be accomplished by automatic signals or remote-manually from the main control room (MCR). Receipt of either a low reactor water level or high drywell pressure indication will automatically start the HPCI system. The 125 V DC power supply provides the energy for the controller, the motor operated valves; hence, if DC power is lost, the function to control the HPCI will cease [34].

Automatic depressurization system (ADS)

Under the scenario of a high pressure core cooling system failure, the ADS is designed to rapidly reduce the RPV pressure to allow the low pressure emergency core cooling systems to work upon a predetermined low reactor water level signal and high primary containment vessel (PCV) pressure signal. The actuation signal of the ADS controls solenoid valves feeding nitrogen gas into the SRV cylinder from the ADS accumulator to open the SRV [34].

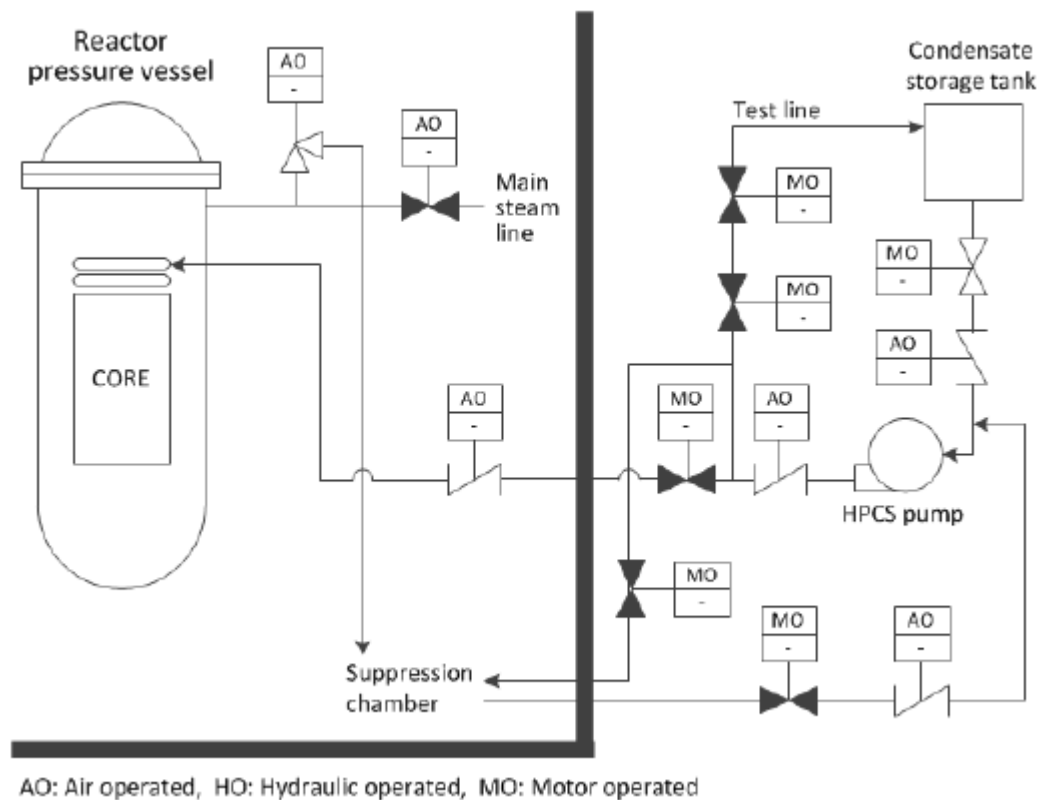


Figure 11: Schematic of high pressure core spray system (HPCS) [25]

High Pressure Core Spray (HPCS) system

The purpose of the core spray system in Figure 11 which was in the design of all Fukushima Daiichi NPP units, is to provide low pressure make-up water to the reactor vessel for core cooling under LOCA conditions. All Fukushima Daiichi NPP units, like all BWRs, the core spray system are designed to inject

water from the suppression pool to the core at low pressures using one or more spray spargers located above the core. The CS system automatically initiates upon receipt of either a predetermined low level reactor vessel water level signal or a high drywell pressure. The water is delivered by motor-driven pumps that required AC power to operate [34].

Isolation Condenser(IC) system

In the Fukushima Unit 1 design, there were two separate and redundant isolation condenser loops. In these closed loops, the primary side of the isolation condenser received steam generated in the reactor and condensed it by cooling inside the heat exchanger tubes that were submerged in colder water tanks (isolation condenser pools) located outside the primary containment vessel. Condensed steam is then sent as cold water back to the reactor by gravity, Figure 12. Without mixing with the radioactive primary side water, the secondary side water in the isolation condenser pools boiled, and the evaporated steam is vented to the atmosphere, which served as the heat sink. The secondary side water volume of the isolation condenser (both trains together) is sufficient to provide for eight hours of cooling before requiring replenishment from a dedicated water source [34].

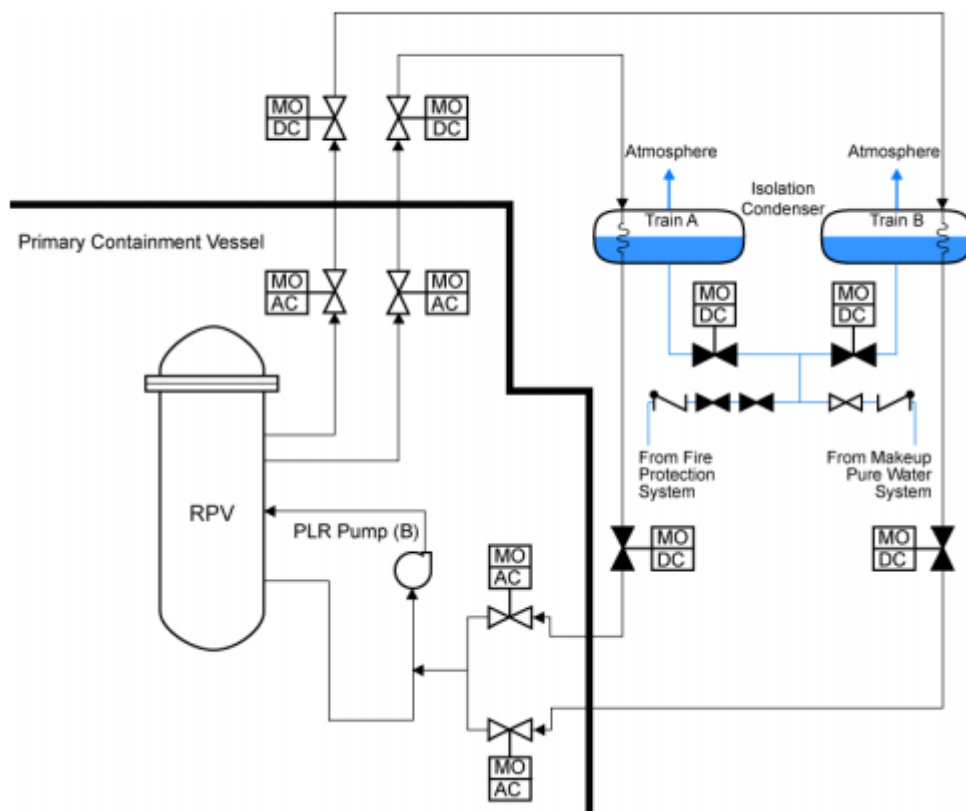


Figure 12: schematic of Isolation Condenser [24]

2.4 Primary pump review

The reactor coolant pump as illustrated in Figure 13 is of a single stage centrifugal type with one impeller/diffuser assembly. The impeller imparts energy to the reactor coolant. The pumps are designed to maintain a continuous flow of primary coolant through the reactor core and thereby ensure the removal of the heat energy generated and its transfer to the steam generators. This cooling function is essential to avoid overheating and possible failure of the fuel cladding with the resulting release of fission products into the primary fluid. The flow-rate of one RCP pump is 21 075 m³/h [35].

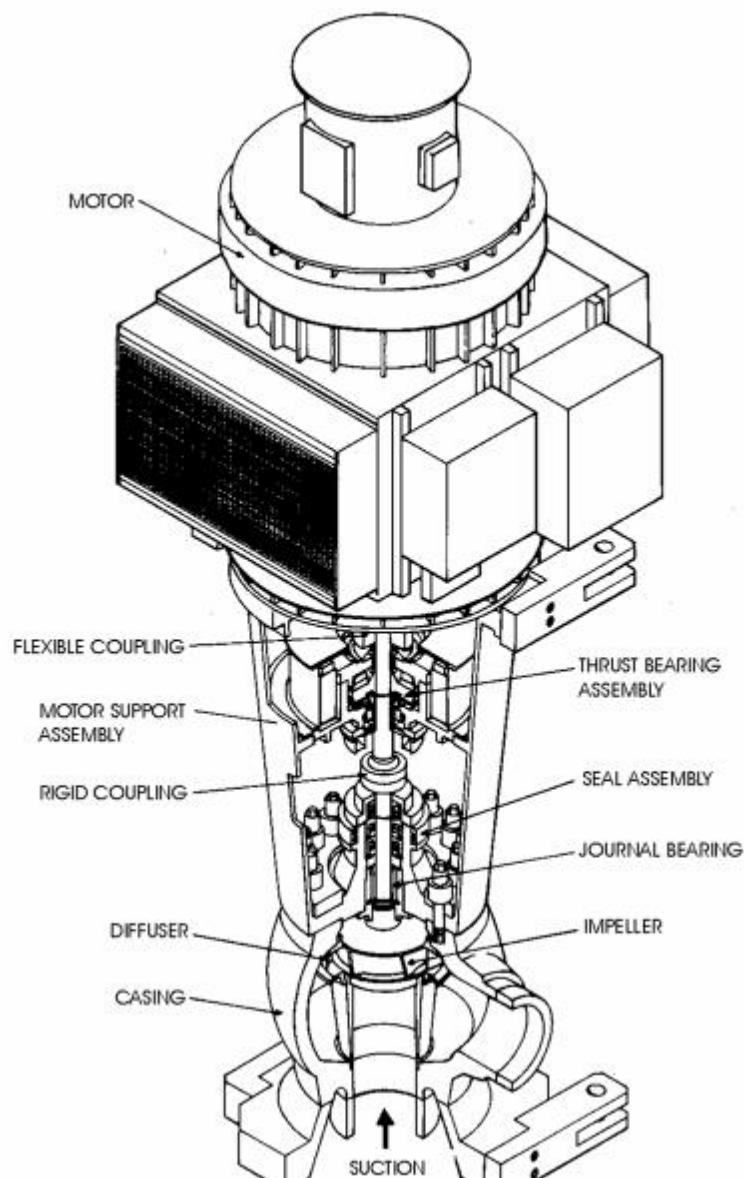


Figure 13: Primary pump schematic [36]

2.4.1 Thermal Barrier

The thermal barrier provides the thermal isolation as shown in Figure 14 of the impeller section of the pump. The barrier is made up of zigzag tubes connected together as a heat exchanger and is located directly above the pump impeller. The temperature drop across the barrier is from 290 °C at the impeller to 90 °C at the lower radial bearing and seals area. Component cooling water which is called RRI system at Koeberg (Figure 14) circulates through the heat exchanger to achieve the above temperature drop. The

thermal isolation of the lower bearing seals must be achieved as they operate at very close clearances and the seals in particular are part of a very delicate and high precision construction, and are very sensitive to high temperatures. For this reason, relative cool fluid supplied by the chemical and volume control system called RCV at Koeberg (Figure 14) is injected through a connection on the thermal barrier. This water supply is known as seal injection, it is supplied at the pressure higher than the RCP system [15]. Seal injection flow acts as a buffer to prevent reactor coolant from entering the pump seal and bearing section of the pump. If seal injection is lost, the hot reactor coolant can flow up the pump shaft. Under these conditions, the primary pump thermal barrier functions as a heat exchanger to cool the hot reactor coolant before it enters the primary pump bearing and seal area [37].

Following restoration of ac power after SBO, it is desirable to restore primary pump seal cooling as soon as possible in order to reduce seal temperatures and mitigate potential continued degradation of the primary pump seals. However, the restoration of the seal cooling must be performed in a controlled manner in order to avoid thermal shock and related damage to the pump parts. Uncontrolled restoration of seal cooling to a hot primary pump can lead to bent shafts and damage seals. The bent shafts are primarily attributable to the rapid introduction of cold seal injection flow into the seal area, which results in abnormal thermal gradients and thermal stresses across the primary pump shaft. Failure of the seals resulting in excessive leakage and loss of RCP system inventory equated to a cold leg break [37].

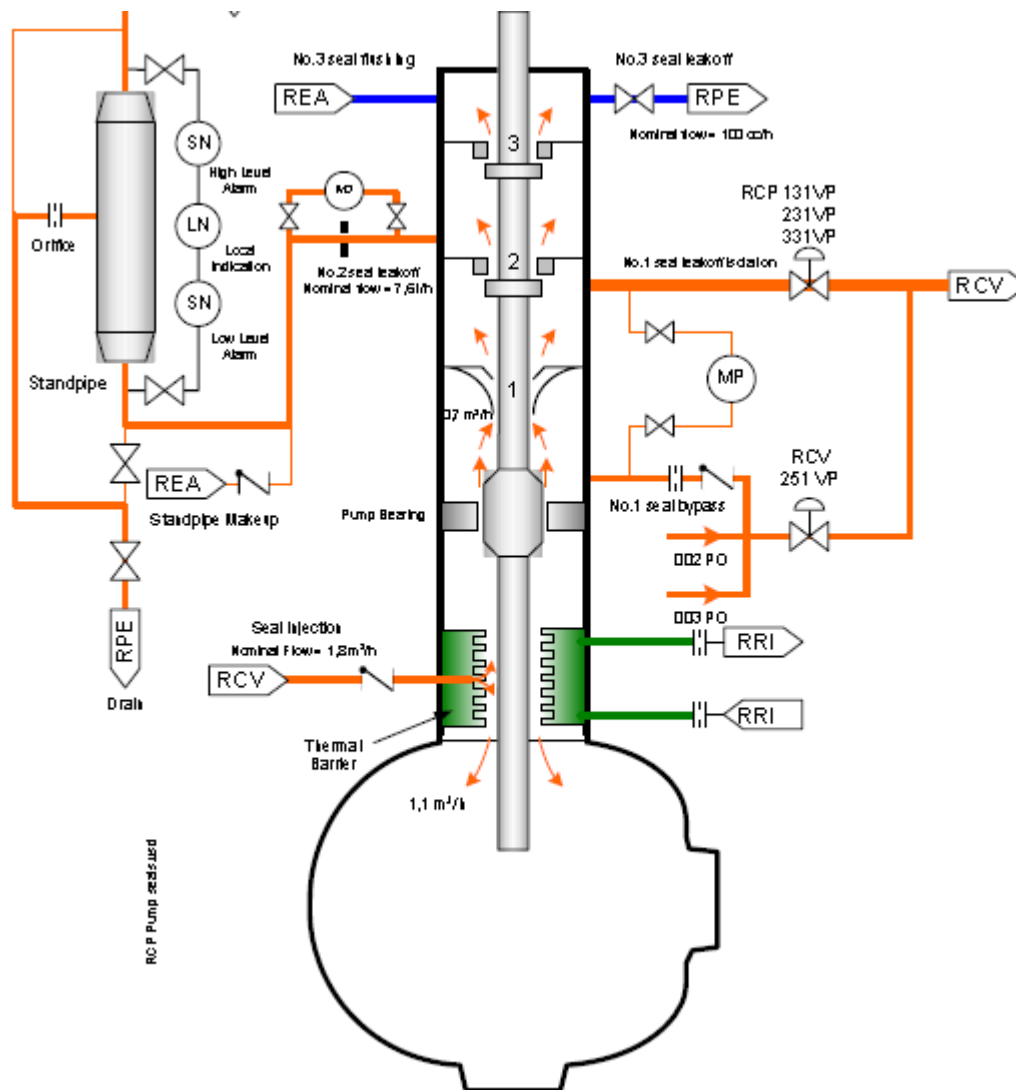


Figure 14: Thermal Barrier cooling [35]

2.4.2 Pump Radial Bearing

The radial bearing is a hydrodynamic swivel bearing. The bearing surface, or journal, is provided with a graphite lining. A stellite ring is hot shrunk on the mating surface of the shaft. The bearing is located around the shaft above the thermal barrier and serves as a means of alignment and radial support for the shaft assembly. The radial bearing is lubricated and cooled by seal injection water [35].

2.4.3 RCP pump motor

The RCP pump motor is an air-cooled 6, 6 kV squirrel-cage induction motor, supplied from the non-essential switchboards. The motor is physically located above the pump using a common shaft and rotates the pump impeller at a speed of 1485 rpm. The motor and pump are connected by a three-part coupling, comprising a spool piece and coupling flanges, keyed (backup security) and shrunk to a tapered shaft on the pump and shrunk (no key) on the parallel motor shaft. The motor windings are cooled by containment air circulating through them. The air is cooled in a separate heat exchanger supplied by the component cooling water system before returning to the containment atmosphere. The RCP motor further consists of the following; an oil-lubricated Kingsbury thrust bearing, two oil-lubricated radial bearings, an oil lift subsystem and a flywheel with anti-reverse mechanism [35].

2.4.4 Primary pump seals

The function of the seal section is to prevent reactor coolant leakage along the pump shaft to the containment during normal operating conditions, and to limit reactor coolant losses through the seals in the event of a loss of seal water injection. Three seals mounted in series on the shaft provide this pressure isolation as shown in Figure 15. The first seal, referred to as the no. 1 seal is of the film-riding-face type, is the most important because it is most vulnerable point of the RCP system. The second and third, referred to as the no. 2 and no. 3 seal respectively, are of conventional design with rubbing faces.

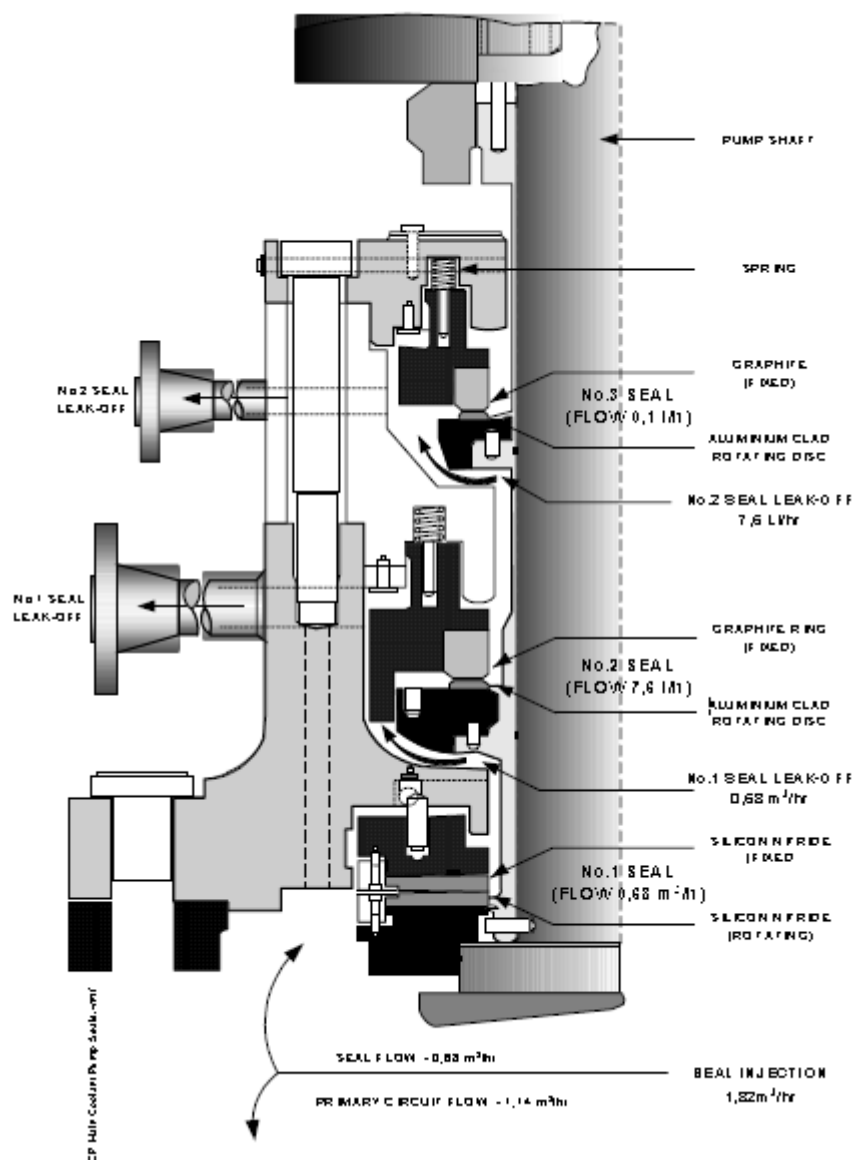


Figure 15: Simplified Primary pump seal configuration [36]

No. 1 Seal

The no. 1 seal is the main seal of the reactor coolant pump shown on Figure 15. The seal is a balanced, control leakage-type seal, with contact prevented between the seal ring and the runner by a thin layer of water.

The primary components of this seal are a runner, which rotates with the shaft and a non-rotating seal ring attached to the lower housing. These sliding faces are made of silicon nitride and are mounted on stainless steel supports with an interface gap of 0.01 mm.

The pressure on the lower side of the seal ring is the same as the RCP pressure; Volume Control Tank (VCT) pressure maintains the pressure of the upper side, which is the collection point for the controlled leakage from this seal. The normal leakage flowrate is around 200 – 400 l/h, at normal operating pressure.

The term "controlled leakage" is used because the leakage from the seal is predetermined and controlled, this done by ensuring that the gap between the seal ring and the seal runner is always held to a constant value. This is achieved by designing for a stable balance of hydrostatic forces on the ring. The opening force of the seal is due to the differential pressure across the seal and is of utmost importance as it maintains the film between the sliding faces [35].

No. 2 Seal

The no. 2 seal is a rubbing face seal comprised of an aluminium ring and a graphite runner that is fixed on the pump shaft, Figure 15. The seal ring rubs against the runner, providing the seal along the shaft. Under normal operating conditions, this seal is designed to accommodate a differential pressure of 350 kPa and limits its leak rate to 7.6 l/h. It is however, designed to support the full RCP system pressure in the event of a no. 1 seal failure. Under these conditions, the no. 2 seal withstands a differential pressure of 15.4 MPa and limit RCP leakage to approximately 3 m³/h. The no. 2 seal provide a back-pressure on the no. 1 seal, thus directing the major portion of no. 1 seal flow to the no. 1 seal leak-off line. No. 2 seal leak-off is directed to a standpipe where it overflows via an orifice to the drains. The standpipe is situated is such a way to give a back-pressure of approximately 20 kPa to this leak-off [35].

No. 3 Seal

This seal is designed to maintain the back-pressure on the no. 2 seal as illustrated in Figure 15. A rubbing face seal comprised of aluminium clad rotating discs on graphite rings. The seal is loaded by way of springs that ensure a tight seal between shaft and runner. The third seal limits normal leakage to 0,1 l/h. This leakage is due to the static pressure placed on the seal by the standpipe and is sufficient to lubricate and cool the rubbing faces. The static pressure maintains a differential pressure of approximately 20 kPa across the seal. The seal has a continuous clean uncontaminated flushing water supply from primary coolant make-up system, which prevents the seal from running dry in case of an insufficient seal leakage. This continuous small flow is fed to the back-face of the seal and serves to stop any crystallisation of boron on the seal. The leak-off from the no. 3 seal is directed to the vents and drain system [35].

2.4.5 Seal Injection

To protect the seals, high-pressure borated water is injected into the pump, entering at the top of the thermal barrier housing (see Figure 14). The borated water is used to prevent dilution of the primary system; any change in water boron concentration will change the reactivity in reactor. This seal injection water, at a temperature of less than 60°C and a pressure slightly higher than the primary pressure, prevents the outward flow of the primary coolant. The water is extremely pure, having been filtered and treated by chemical and volume seal injection subsystem. The seal injection water serves three main purposes [37]:

- It permits highly filtered water (with particulate matter less than 2 microns) to flow through the seals instead of the primary coolant, in which corrosion particles are suspended that could cause severe seal damage.

- It acts as a double safety precaution in the thermal isolation of the seals from the pump hydraulic section. It lubricates and cools the pump radial bearing section.
- It lubricates and cools the pump radial bearing.

2.4.6 Concerns regarding Loss of Seal Injection and Thermal Barrier Cooling

Following a coincident loss of seal injection and thermal barrier, the major concern is hot water rising up the shaft and subsequent flashing downstream of the number 1 seal and O-ring failure. This would result in rapid seal failure with subsequent LOCA. To minimise the consequences of such an event, the RCP pump(s) must be stopped as soon as possible. It is also important to ensure that the pump bearing and seals do not undergo sudden thermal shock that may result in component damage. Thermal shock would occur if the cooling water or seal injection were reinstated to any affected pump in an uncontrolled manner. To prevent this from occurring, the operator must ensure that seal injection flow and cooling flow to the affected pump(s) is isolated. Tests have shown that the number 2 seal can act as a pressure boundary providing the RCP pumps be stopped. To enable this, the RCP pump oil pumps are started and then the affected RCP pump(s) stopped [9] [37]. Any primary pump(s) that have simultaneously lost both seal injection and thermal barrier cooling must not be started until a status evaluation is performed at cold shutdown.

Under SBO condition, the RCP pumps support systems may not be restored timeously before the hot coolant is introduced into the seal. When the temperature exceeds 95°C, the seal will start to deteriorate and lose its sealing capability. The seal deterioration at high temperature is due to several interactions, including [1] [9]:

- The seal O-ring material softens with increasing temperature, affecting the O-ring sealing ability and life.
- The thermal gradients affect the faceplate tapers of the number 1 seal ring and the runner and the shrink fit of the no.2 seal ring insert, affecting sealing surfaces. Non-uniform thermal gradients and extrusion of O-rings may result in non-uniform sealing surfaces.
- The leakage of reactor coolant through the seals could result in crud blockage of the seals.

2.4.7 Responding to a Loss of Seal Injection and Thermal Barrier Cooling

This event is normally associated with a total loss of ac power. When the loss of seal injection flow and thermal barrier cooling is confirmed, the operator must act quickly to avoid damage to the affected RCP pumps. The reactor trip must be verified before the affected RCP pump is stopped. As this is a planned operator induced reactor trip, it is not foreseen that problems would occur on concurrent implementation of time stressed actions [9].

Containment isolation valves on the seal return line are closed to protect the downstream volume control system equipment. Failure to do this may result in the meltdown of the seal return filter with the potential of damage to the charging pumps. Following isolation of the seal return line, the seal leak-off safety valve will lift and discharge into the pressuriser relief tank. By leaving the number 1 seal leak-off open with diverted seal leak-off flow to pressure relief tank, the number 2 seal will not be exposed to full RCP pressure, thus reducing the potential for a small break loss of coolant accident (SBLOCA) [15].

2.4.8 Seal leakages from seal failures analysis and predictions

Due to the several interacting considerations, it is difficult to analytically or experimentally predict the primary pump leakage flow following a total loss of all seal cooling. Further, it is possible that different primary pumps may be affected in different ways with resulting variations in leakage flow. Since it is difficult to accurately predict primary pump seal leakage flow following a loss of all seal cooling, field

experience has been reviewed for relevant information and an analysis has been performed to estimate a maximum leakage rate [9]. The Energy Technology Engineering Centre (ETEC) conducted an independent analysis of the leakage rates from the primary coolant system through the Westinghouse Reactor Coolant (RCP) shaft seals and associated downstream piping for postulated station blackout conditions [1]. The analysis indicates that seal failure is important in determining the time required to uncover the core. The probability of the seal failure should be determined experimentally. The analysis of the boil off downstream of the no.2 seal needs to be determined as it will provide the valuable information on the seal stability and closure. The temperature gradient across the No.2 seal will cause the seal to rotate close; this will be caused by the decreasing temperature gradient when the water flashes into steam downstream of the seal. The labyrinth seal flow resistance flow when all the seals have failed will need to be evaluated. The labyrinth seal is situated between the impeller and the first stage seal as illustrated in Figure 16.

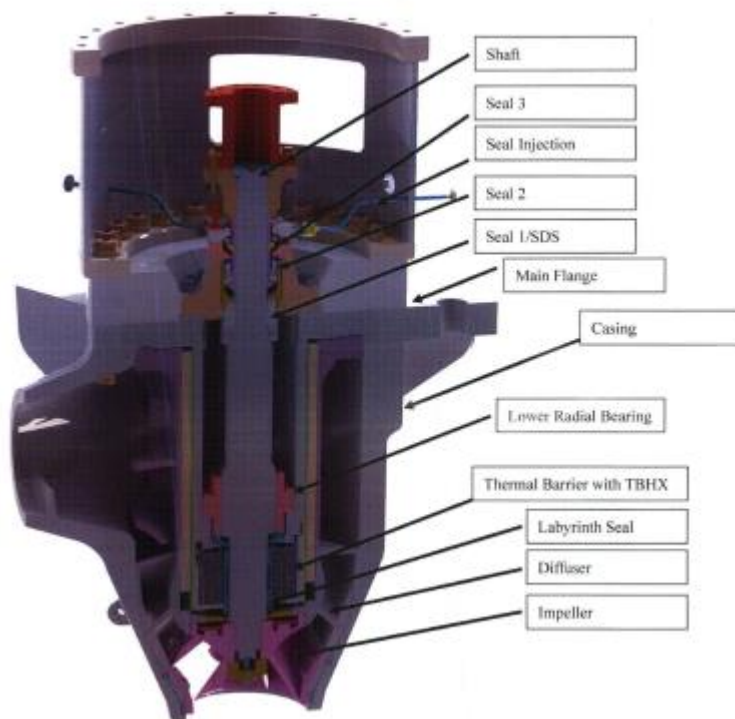


Figure 16: Cutaway of Reactor coolant with Labyrinth seal [38]

The ETEC and Westinghouse [1] studies encompass three different cases in the analysis of seal leakage for station blackout.

Case 1: Initial leakage with the assumption that all seals are functioning.

Case 2: Initial leakage with the assumption that no.1 seal failed while no.2 and no.3 seals remain functional.

Case 3: Initial leakage with the assumption that all seals have failed.

Table 2 list the seal leak rates for the three cases for both ETEC and Westinghouse.

Table 2 : Seal leak rates prediction from ETEC and Westinghouse [1]

	Case 1	Case 2	Case 3
ETEC prediction (m ³ /h)	13	44	287
Westinghouse prediction (m ³ /h)	14	51	327

The seal analysis addresses the three important factors:

- The effect the seal failure has on the RCP leakage
- The effect of RCP cooldown and depressurisation on leakage rates and
- The sensitivity of the leak-rates to the dimensional and thermal hydraulic parameters.

In all above cases, it is assumed that the RCP pump stop to rotate immediately after the SBO.

This research will focus on Case 2 which is assuming that no.1 seal fails open while no.2 and no.3 seal remains intact. Case 2 leak rates correlates with initial leak rates experienced at Fukushima Unit 1 based on TEPCO report [5].

The initial conditions for the modelling the seal failure according to the ETEC and Westinghouse [1] studies was that the plant was at nominal pressure of 15.5 MPa and temperature of 287 °C. A conservative assumption was made that the seal will fail immediately after upon the SBO, this assumption gives the maximum leak rates. The main leak path will be through the no.1 leak-off line with only minimal flow path through the no.2 seal. There will some leakage through the no.2 seal due to machining tolerances, the leakage was estimated to be about 0,06 m³/h.

2.5 Source term review

The source term in this research becomes a particular point of interest in the small break LOCA through the pump seals as results of SBO. In most fuel accident analysis cases the fuel is estimated to have a damage of up to 1% due to flaws that emanate from manufacturing of the fuel or due to fuel being exposed to harsh conditions during the operations [15].

The regulation for nuclear plants prescribes the calculation of offsite doses at the exclusion area boundary (EAB) for the first two hours after the accident and at the low population zone (LPZ) boundary for the course of the accident, which for dose analysis is assumed to continue for 30 days. Control room doses are also calculated for a 30-day period. Although the existing requirements involve the calculation of whole body and thyroid doses, in practice the thyroid dose, resulting from the iodine, has generally been the limiting dose for plants. The EAB and the LPZ size is such that the individual will not receive a total radiation dose to the whole body in excess of 0.25 Sv or a total radiation dose in excess of 0.3 Sv to the thyroid from iodine exposure [10].

The Emergency Planning declaration is based on the following criteria, Alert, Site Emergency and General emergency. An Alert is declared when there is a situation that could develop into a Site or General Emergency and therefore requires notification of all Emergency Personnel in order to obtain a state of readiness to respond. A Site Emergency is declared when there is an emergency condition that involves a serious radiation hazard on-site but involves no serious radiation hazard beyond the public exclusion boundary. A General emergency is declared when there is an emergency condition that involves, or potentially involves, a serious radiation hazard beyond the public exclusion boundary. Table 3 shows a typical guide used in declaration of an emergency plan [15].

Table 3: Emergency planning declaration guide [15]

	Alert	Site	General
Fission product barriers	<u>Barrier 1</u> : Primary activity of greater than $> 3,7 \times 10^{10} \text{Bq/m}^3$ (1 Ci/m^3) equivalent as measured from radiochemical analysis OR Local radiation monitoring indicating greater than 200Gy/hr (20 rad/hr) OR <u>Barrier 2</u> : RCP system leak rate greater than normal RCV make-up system.	<u>Barrier 1</u> : Primary activity of greater than $> 3,7 \times 10^{10} \text{Bq/m}^3$ (1 Ci/m^3) equivalent as measured from radiochemical analysis OR Local radiation monitoring indicating greater than 200Gy/hr (20 rad/hr) AND <u>Barrier 2</u> : RCP system leak rate greater than normal RCV make-up system	<u>Barrier 1</u> : Primary activity of greater than $> 3,7 \times 10^{10} \text{Bq/m}^3$ (1 Ci/m^3) equivalent as measured from radiochemical analysis OR Local radiation monitoring indicating greater than 200Gy/hr (20 rad/hr) AND <u>Barrier 2</u> : RCP system leak rate greater than normal RCV make-up system AND <u>Barrier 3</u> : Containment failure
Spent fuel Damage	Containment radiation greater than 100mGy/hr ($>10 \text{ rad/hr}$)	Fuel building radiation $> 10 \text{mGy/hr}$ (1Rad/hr). Damage to seal of dry spent fuel storage casks	
Radiation dose		On-site dose measured or predicted: 1mSv (100mrem) whole body dose equivalent in a year Or 10mSV (1000mrem) thyroid dose in a year	Off-site dose measured or predicted: 5mSv (500mrem) whole body dose equivalent in a year Or 50mSV (5 rem) thyroid dose in a year

2.5.1 Protective action guide (PAG) and ingestion exposure pathway (IEP) limits [39]

PAG where issued to assist the public health and the government institution in deciding on how much of a radiation hazard constitute the initiation of emergency protective action. The PAGs expressed in the units of the radiation dose (rem/mSv), they represent the trigger points or the initiation points which warrant pre-selected protective actions for the public if the projected (future) dose received by an individual in the absence of a protective action exceeds the PAG. They do not assure that a given level of exposure to individuals in the population is prevented; the exposure of individuals to dose may vary based on the distance from the source of release [39].

The following four principles form the basis for setting up values for PAG's:

- Acute effects of health should be avoided.
- The risk of the delayed effects of health should not be exceeded.
- PAG should not be higher than justified in terms of cost and collective risk of effects health.
- The risk to health from a protective action should not itself exceed the risk to health from the dose that would be avoided.

The PAGs are numerically defined as follows because of the release pathway considered

- *The plume exposure pathway (PEP)*: it relates to the whole body external exposure to gamma radiation from the plume and from deposited material, and for inhalation exposure from the passing radioactive plume. The time of potential exposure could range from hours to days. PEP area is 16 km radius
- *The ingestion exposure pathway (IEP)*: it relates to the contamination due to ingestion of contaminated foods. The time of potential exposure could range from hours to months. The EIP area is 80 km radius [40].

This are some of the examples of protective actions necessary during the an accident with the uncontrolled source of radiation: Evacuating an area; Sheltering-in-place within a building or protective structure; Administering potassium iodide (KI) as a supplemental action; Relocation; Acquiring an alternate source of drinking water; and Interdiction of food/milk.

Radiological Incident Phases and Applicability of Protective Actions

Emergency plan divides the response to radiological incident into three phases;

- Early phase
- Intermediate phase
- Late phase

Table 4 : gives the summary of the phases, the recommended protective actions and guidance. The phases do not have timelines and activities can sometimes overlap, they provide a useful framework response during the emergency response planning.

Table 4 : Summary Table for PAGs, Guidelines, and Planning Guidance for Radiological Incidents [40]

<u>Phase</u>	<u>Protective Action Recommendation</u>	<u>PAG, guide line or planning guidance</u>
Early phase	Sheltering-in-place or evacuation of the public	PAG: 1 to 5 rem (10 to 50 mSv) projected dose over four days
	Supplementary administration of prophylactic drugs – KI as per Table 5	PAG: 5 rem (50 mSv) projected child thyroid dose from exposure to radioactive iodine
	Limit emergency worker exposure (total dose incurred over entire response)	Guideline: 5rem (50mSv)/year (or greater under exceptional circumstances)
Intermediate phase	Relocation	PAG: ≥ 2 rem (20mSv) projected dose in the first year 0.5rem (5mSv)/year projected dose in the second and subsequent years
	Apply simple dose reduction techniques	Guideline: < 2rem (20 mSv) projected dose in the first year
	Food interdiction	PAG: 0.5 rem (5 mSv)/year projected whole body dose, or 5 rem (50 mSv)/year to any individual organ or tissue, whichever is limiting
	Drinking water	PAG: 100 mrem (1 mSv or 0.1 rem) projected dose, for one year, to the most sensitive populations (e.g., infants, children, pregnant women and nursing women); 500mrem (5 mSv or 0.5 rem) projected dose, for one year, to the general population
	Limit emergency worker exposure (total dose incurred over entire response)	Guideline: 5 rem (50 mSv)/year
	Re-entry	Guideline: Operational Guidelines (stay times and concentrations) for specific re-entry activities
Late phase	Clean-up	Planning Guidance: Brief description of planning process
	Waste Disposal	Planning Guidance: Brief description of planning process

The Early-Phase PAGs (TEDE > 1 rem, Thyroid (iodine) CDE > 5 rem)

Total effective dose equivalent (TEDE) includes cloud-shine dose and inhalation committed effective dose equivalent (CEDE) and ground-shine dose. During the early phase, the sequence of events includes evaluation of conditions at the location of the incident, notification of responsible authorities, prediction

or evaluation of potential consequences to the public, recommendations for action and implementation of actions for the protection of the public [39]. The main protective actions for the early phase are evacuation and sheltering-in-place. The decision to evacuate must be based on the risk of the anticipated radiation dose to the individuals versus the feasibility of the evacuation itself and the time determined for evacuation. The evacuation of dense population comes with the statistical risk of injury or death from transportation hazard or increased exposure. Planning in advance and emergency exercises can assist in facilitating the decision making process. Things to consider for evacuation planning:

- Evacuation may be the only effective protective action close to the plume source
- Evacuation will be most effective if it is completed before arrival of the plume
- Evacuation may increase exposure if carried out during the plume passage
- Evacuation is appropriate for protection from ground-shine in areas with high exposure rates and from deposited radioactive materials when suitable shelter is not available.

In areas where the airborne radioactivity is present but the PAGs are not exceeded, consideration must be taken by instructing people to stay indoors and where practically possible instruct individual has to cover the airways (nose and mouth) with available filtering material. Things to consider for sheltering planning:

- Sheltering may provide protection equal to or greater than evacuation for areas not designated for immediate evacuation
- Sheltering may provide a better communication with the public
- Sheltering may be effective if evacuation is impeded by severe environmental conditions, uncertainty about contamination of the routes, health constraints (e.g. nearby hospitals or nursing homes), long mobilisation (farmer workers or prisoners) or physical constraints (e.g. blockage of roads due to debris)

The administration of radio-protectant KI should be considered when a significant quantity of radioiodine was released. KI is most effective if taken prior to exposure and it may require approval of state medical officials. Table 5 gives a summary of recommended doses of KI for different risk groups.

	Predicted Thyroid gland exposure (cGy) (1 cGy = 1 rem)	KI dose (mg)	Number or fraction of 130 mg tablets	Number or fraction of 65 mg tablets
Adults over 40 years	≥ 500	130	1	2
Adults over 18 through 40 years	≥ 10			
Pregnant or lactating women	≥ 5	65	1/2	1
Adolescents, 12 through 18 years ^a		32	Use KI oral solution ^b	1/2
Children over 3 years through 12 years		16	Use KI oral solution ^b	Use KI oral solution ^b
Children over 1 month through 3 years				
Infants birth through 1 month				
^a Adolescents approaching adult size (≥ 150 pounds) should receive the full adult dose (130 mg). ^b Potassium iodide oral solution is supplied in 1 ounce (30 mL) bottles with a dropper marked for 1, 0.5, and 0.25 mL dosing. Each mL contains 65 mg potassium iodide. Source: FDA, "Guidance: Potassium Iodide as a Thyroid Blocking Agent in Radiation Emergencies" (December 2001): http://www.fda.gov/downloads/Drugs/GuidanceComplianceRegulatoryInformation/Guidances/UCM080542.pdf (FDA 2001); and FDA, Frequently Asked Questions on Potassium Iodide (KI): http://www.fda.gov/Drugs/EmergencyPreparedness/BioterrorismDrugPreparedness/ucm072265.htm (Last Updated: 10/27/2014).				

Table 5: Threshold thyroid radioactive exposure and recommended doses of KI for different risk groups [39]

Intermediate-Phase PAGs: 1st year - 2rem, 2nd year - 0.5rem.

Intermediate phase activities are intended to reduce or avoid dose to the public, to control worker exposures, to control the spread of radioactive contamination, and to prepare for late phase clean-up operations. The PAG for relocation of the public is 2rem (20mSv) in the first year and 0,5rem (5mSv) in any subsequent year [39].

Relocation [39]

Relocation is the principal protective action against whole body external exposure from deposited radioactive material and internal exposure from inhalation of radioactive particulates. People may need to be relocated for weeks or months.

Dose reduction techniques [39]

Dose reduction techniques, can be applied in areas where levels of deposited radioactivity are not high enough to warrant relocation. Dose reduction techniques actions are as follows;

- Scrubbing of surfaces
- Flushing of surfaces
- Removal and disposal of small spots of contaminated surfaces
- Spend more time indoors
- Spend more time at low dose areas

Food interdiction [41]

Protective actions are taken on food to:

- avoid or limit, through precautionary measures, the amount of contamination that could become incorporated in human food and animal feeds, or
- delay or limit consumption of human food and animal feeds suspected of being contaminated until the concentration of contamination has been determined, or
- Reduce the amount of contamination in human food and animal feeds. To control radionuclide uptake by crops, animals and feed foliage, the regulator prescribes Authorised Annual Discharge Quantities of the radionuclides that a power operator can release the environment. This cannot be exceeded. Regarding the control of how much is deposited on the soil that depends on the concentration of the material in the atmosphere often referred to as Chi/Q , which is the function of the meteorological condition at the time of the accident.

The limits on food are called Derived Intervention levels (DILs) and are based on the radionuclide activity concentration permitted in human food as listed in Table 6. The protective actions for animal feeds are included as measures to reduce or prevent subsequent contamination of human food. The principal radionuclides for which the DILs are developed in the nuclear reactor accidents are ^{131}I ; ^{134}Cs + ^{137}Cs ; ^{103}Ru + ^{106}Ru .

DERIVED INTERVENTION LEVELS (Bq/kg)						
(individual radionuclides, by age group, most limiting of either PAG)						
Radionuclide	3 months	1 year	5 years	10 years	15 years	Adult
Sr-90	308	362	616	389	160	465
I-131	196	167	722	1200	1690	2420
Cs-134	1600	2190	1940	1530	958	930
Cs-137	2000	2990	2810	2180	1370	1360
Cs group ^(a)	1800	2590	2380	1880	1160	1150
Ru-103	6770	8410	12200	16400	25000	28400
Ru-106	449	621	935	1340	2080	2360
Pu-238	2,5	21	17	14	12	10
Pu-239	2,2	18	14	13	10	9,8
Am-241	2	17	13	11	9,1	8,8
Pu+Am group ^(b)	2,2	19	15	13	9,6	9,3
(a) Computed as: (DIL for Cs-134 + DIL for Cs-137)/2						
(b) Computed as: (DIL for Pu-238 + DIL for Pu-239 + DIL for Am-241) /3						

Table 6: Derived intervention Levels

Drinking Water [39]

The drinking water PAG's are based on the following principles;

- Prevent acute effects
- Balance protection with other factors and derive benefits from actions
- Reduce risk of chronic effects

The two-tier PAG consists of 0,5 rem (5 mSv) for the general population, and a stricter PAG of 0,1 rem (1 mSv) in order to inform protective actions for pregnant women, nursing women, and children under age of 15.

Derived Response Levels (DRL's)

DRL's can be defined as the response based on the information gained the intermediate phase, including identification of specific isotopes, release patterns, and associated decay functions. There are specific radionuclides, including ¹³⁷Cs, ¹³¹I and (⁹⁰Sr/⁹⁰Y) that are of particular interest for major radiological incident scenarios where drinking water sources might be contaminated. The PAG's will need to be converted into DRL's in Bq/L or pCi/L for each of the radionuclides of particular interest. DRL's provided in Table 7 are used to make quick decisions about drinking water provided by public water systems in the event of nuclear emergency.

Isotope	DRLs for pregnant women, nursing women and children age 15 and younger – 1 mSv dose	DRLs for the general population – 5 mSv dose
⁹⁰ Sr/ ⁹⁰ Y	27027 Bq/L	200000 Bq/L
¹³⁷ Cs	167567 Bq/L	459459 Bq/L
¹³¹ I	22162 Bq/L	270270 Bq/L

Table 7: Derived Response levels - Drinking Water Concentrations Corresponding to Specified Doses (mSv) of Select Radionuclides, Assuming One Year of Exposure at Constant Levels [39]

Limit emergency worker exposure [39]

Emergency workers should be adequately informed of, and have an adequate understanding of the risks they may experience during any missions they accept, including the risk of short-term and long-term health effects from exposure to ionizing radiation. Emergency workers should have relevant training, personal protective equipment (PPE), and monitoring instruments so they are able to protect themselves and others. Incident commanders are responsible for balancing the individual risks and public benefits of each worker action.

Late Phase

Key issues that need to be addresses in late phase are the following;

- The stakeholders are involved in setting priorities and determining actions. The process undertaken should be flexible to adapt to a variety of situations.
- Plan for worst case scenarios and address small radiological incidents with existing emergency response and environmental clean-up programs.
- Reoccupation of the households and businesses should be considered based on the results from efforts taken on reducing the radiation risk through the decontamination, radioactivity decay, and managing contaminated waste.
- Following a nuclear accident, the states bear primary responsibility to identify and provide waste management options, including disposal capacity; in the event of a terrorist attack, the federal government can offer a range of assistance to states to identify and implement waste management options
- The release of the fission product time dependent, thus reduction of doses will be strongly influenced by safety features which are time sensitive.

It may be impossible to produce an analysis during a severe accident that will provide additional insight into possible consequences and appropriate offsite protective actions. Therefore, initial protective action decisions for core damage accidents (general emergency) should never be based on source term or dose calculations performed at the time of the accident. These decisions must be based on predetermined plans [7].

2.5.2 Severe accidents

For a severe accident which includes the core damage, the immediate predetermined protective actions are taken based on in-plant indicators of core and containment conditions. This indication includes for example core and containment temperatures and pressures. If core damage is projected or exist, the people near the plant should be evacuated based on the emergency plan. The protective action strategy is a based on the wide range of core damage accidents involving containment failure, bypass, or leakage. Other core damage accidents may require options such as venting; the source term based on the prompt response will need to be assessed during the accident.

Five bases may be used to estimate source term (radionuclide release) from a severe reactor accident. These five bases are:

- Effluent monitor readings
- Accident analysis contained in the safety evaluation report
- Various severe accident consequence studies such as the Reactor Safety Study (WASH-1400)
- Detailed analysis of plant conditions conducted during an accident
- Pre-calculated estimates that relate dominant accident conditions to potential radionuclide releases (source terms).

2.5.3 Plant radiation monitoring

The plant monitoring systems are installed to monitor possible sources of irradiation. The monitoring systems first function is to protect the nuclear personnel against high exposure to radiation; the second function is to protect the public against exposure to radiation; the third function is to perform indirect prevention by monitoring the barriers and the forth function is perform the automatic action to protect the personnel [15]. Under severe accident conditions, the radiation monitors become inadequate to estimate the amount of release by the plant; it is also undesirable to wait for the release to occur before taking necessary additional protective action because effective protective action requires prompt implementation. Therefore, an attempt should be made to project the magnitude of a release before it occurs. Plant monitoring systems important for source term are:

- LOCA radiation monitoring

Threshold 1 is set just above the current activity value, so that normal fluctuations of the reading do not trigger the alarm. The alarm threshold is based on 10% of the actual dose-rate levels listed in the Emergency Plan procedure that requires evacuation of the Reactor Building. Threshold 1 will initiate a local alarm and a grouped alarm in the control room. Threshold 2 is set in accordance with the threshold set points as stated in - Radiation Monitoring Management procedures.

- Primary system radiation monitoring

The let-down line monitor is in place to detect the RCP activity and can be used as a measure of the integrity of the fuel rods. Threshold 1 will initiate a local alarm and a grouped alarm in the control room. Threshold 2 is indicative of an increase in the reactor coolant activity, which can be due to fuel cladding leakage, or a rapid load transient.

- The containment atmosphere radiation monitoring

Alpha Beta Particulate Monitor is used in containment to monitor the presence of Alpha Beta Particulate activity in the atmosphere. This can be directly related to the amount of a water leak from the primary system. Noble Gas Monitor is used in containment to monitor the presence of noble gas activity in the atmosphere. This can be directly related to the amount of a water leak from the primary system. Iodine Monitor is used in containment to monitor the presence of iodine activity in the atmosphere.

2.5.4 Release Phases of a Severe Accident

Despite the differences in designs of the plants and sequence of the accidents, releases can be generally categorised in phases associated with the degree of fuel melting and relocation, reactor pressure vessel integrity, and attack upon concrete below the reactor cavity by molten core materials. The general phases of severe accident for light water reactor are as follows [32]:

- Coolant Activity release

The coolant activity phase period is from the postulated pipe rupture until the first fuel rod has been estimated to fail. The activity released to the containment atmosphere during this phase is small amount

of radioactivity dissolved in the coolant itself. For a large break LOCA, the coolant activity phase period is estimated to last for about 25 seconds in the Westinghouse PWR designs and about 13 seconds B&W PWR designs. This activity phase is expected to last longer for BWR.

- Gap Activity release

The gap activity release start from the onset of fuel cladding failure. The activity released is from the gap between the fuel pellet and cladding. The release to containment includes a few percentage of the total inventory of the more volatile radionuclides, in particular noble gases, iodine, and cesium. The most of the fission products are retained by fuel itself. When the fuel pellet bulk temperature increases significantly some of the fission cannot be retained anymore by the fuel, this implies the end of the gap activity release phase. The release of the fission products from the bulk of the fuel is estimated to commence between 30 minutes and 60 minutes for PWR and BWR respectively from the onset of the accident. The gap activity release is estimated to last for 0.5 hours.

- Early in-vessel release

When the fuel and the other structures in the core reaches high temperatures, the fuel melt and the core geometry is longer maintained relocating to the bottom of the reactor vessel. The early in vessel release phase start at this point, the large amount of the volatile nuclides in the core inventory as well less volatile nuclides are estimated to be released into containment. These release phase when the reactor vessel fails and the molten core falls below. The duration of the release phase is about 1.3 hours and 1.5 hours for both PWRs and BWRs respectively.

- The ex-vessel release

This phase begins when the molten core has cooled enough that the fission products are no longer being released. The phase duration takes 2 and 3 hours for PWR's and BWR's respectively.

- The late in-vessel release

The late in-vessel release phase commences at vessel breach and proceeds simultaneously with the occurrence of the ex-vessel phase. However, the duration is not the same for both phases. During this release phase, some of the volatile nuclides deposited within the reactor coolant system earlier during core degradation and melting may re-volatilize and be released into containment.

For seal leak related accident, the coolant activity release is of particular interest. The leak through the seal resembles the small pipe break or rapture. LOCA is listed as one of the postulated initiating events. Postulated initiating events (PIE's) are as a result of components failure, system malfunction, human errors, external event and or combination of all. PIE's are listed as follows [31];

- Loss of electrical supplies
- Insertion of excessive reactivity
- Loss of flow
- Loss of coolant
- Erroneous of handling of equipment or components
- Special internal events
- External events
- Events involving experimental devices

In a Fukushima accident several of the above mentioned PIE's were involved, however the external events such earth quake followed by the tsunami are to be considered as the initiating event.

Event 4: On-site power loss

Event 3, 4 and 5 are reported to have happened at the same time. The on-site emergency diesel generators were flooded by water from the tsunami. The control room lighting was lost and only emergency lighting remained [5].

Event 5: Direct Current (DC) power loss

Control indicators, displays and monitoring instruments were lost due to loss of DC power; the backup batteries were flooded by Tsunami [5].

Event 6: Loss of core cooling

The control panel indications for HPCI were barely lit but slowly faded to black. The operators determined HPCI was not operable because indicators on the control panel had gone out [5].

Event 7: Off-site and on-site power recovery

Part of the DC power supplies was temporarily recovered at around 16:42 to 17:00 on 11 March 2011; this made it possible to monitor the level in control from wide range level indicators. The level indicated water at the top of the active fuel (TAF) +2130mm. It was also observed that the IC subsystem A (see Figure 12) was operable and indicated closed. The IC was subsequently opened by operator to cool the reactor core. A portable generator temporarily provided lighting in the control room. The alternate current AC distribution had a crosstie connection, which made it possible to connect Unit 1-2. Unit 2 switchgear connection was chosen after assessment by the workers revealed that the standby liquid pumps were available and flooded or damaged. The attempts by workers to connect the mobile generators, which were placed next Unit 2, were successful however, the ability to stabilise the unit was hampered by the explosion on Unit 1 [5].

Event 8: Reactor Pressure Vessel (RPV) depressurisation

Before Tsunami, the operators has cycling the IC valves to control the RPV pressure and maintaining the cooldown below.

Event 9: Water injection

Three fire engines were available for firefighting capability, the fire engines could also be used for low-pressure injection source to the reactors, however one was damaged as the results of floods and second one could not reach unit 1 due to debris and damage on the road. Only one fire engine was ultimately connected after struggling to clear the debris and fuel oil tank, which was swept away by tsunami and was displaced into one road leading to Unit 1 access [5].

3.1.2 Simulation of Fukushima Unit 1 events using PWR simulator

The simulation was performed in a PWR simulator to replicate similar events that happened in Fukushima Unit 1 on the day of the accident, which happened on 23 March 2011. The simulation was designed to follow the sequence events (SE) of the Fukushima accident as stipulated in the TEPCO report [5]. The events of interest for the research purpose will be from Event 1 (Earthquake) to Event 7 (Loss of cooling) only as illustrated in Figure 18. These events were selected on the basis that whatever happened beyond event number 6 was the recovery process from the accident, which was operator-induced events. The research focus is on what would have been the response of the plant if there were no intervention by operators. The main interest of the accident evolution is when the control room staff lost the control and monitoring of the plant due to SBO and the consequence thereof.

The simulator is modelled in such a way that the primary pump seal leakage is simulated as a cold leg at the pump break discharge. The break size is selected in such a way that the break flow will equal a pre-selected value of seal leakage rate. The analysis of the break flow is dictated by the critical flow correlations for subcooled and saturated water, this is to accommodate for the changing conditions of the primary systems during the analysis. The approach produces a more realistic representation of the response of a primary pump seal leak in particular the degradation due to overheating of the seal. Variations from the actual leak are expected due to over prediction of the leak rates; the deviation is a conservative approach. Assumption made in the analyses with regard to pump leaks were that the leak will start at the pre-selected rate as soon as the power was lost. No time delay was included to simulate the effects of gradual seal degradation. In reality during SBO event the seal degradation, if it occurs, it will extend over period lasting from several minutes to hours. The most likely result in this gradual degradation would be a slow increase in pump leakage from only about 1 m³/h per pump at the time of reactor trip to potentially larger leak rates at some point later in time [9].

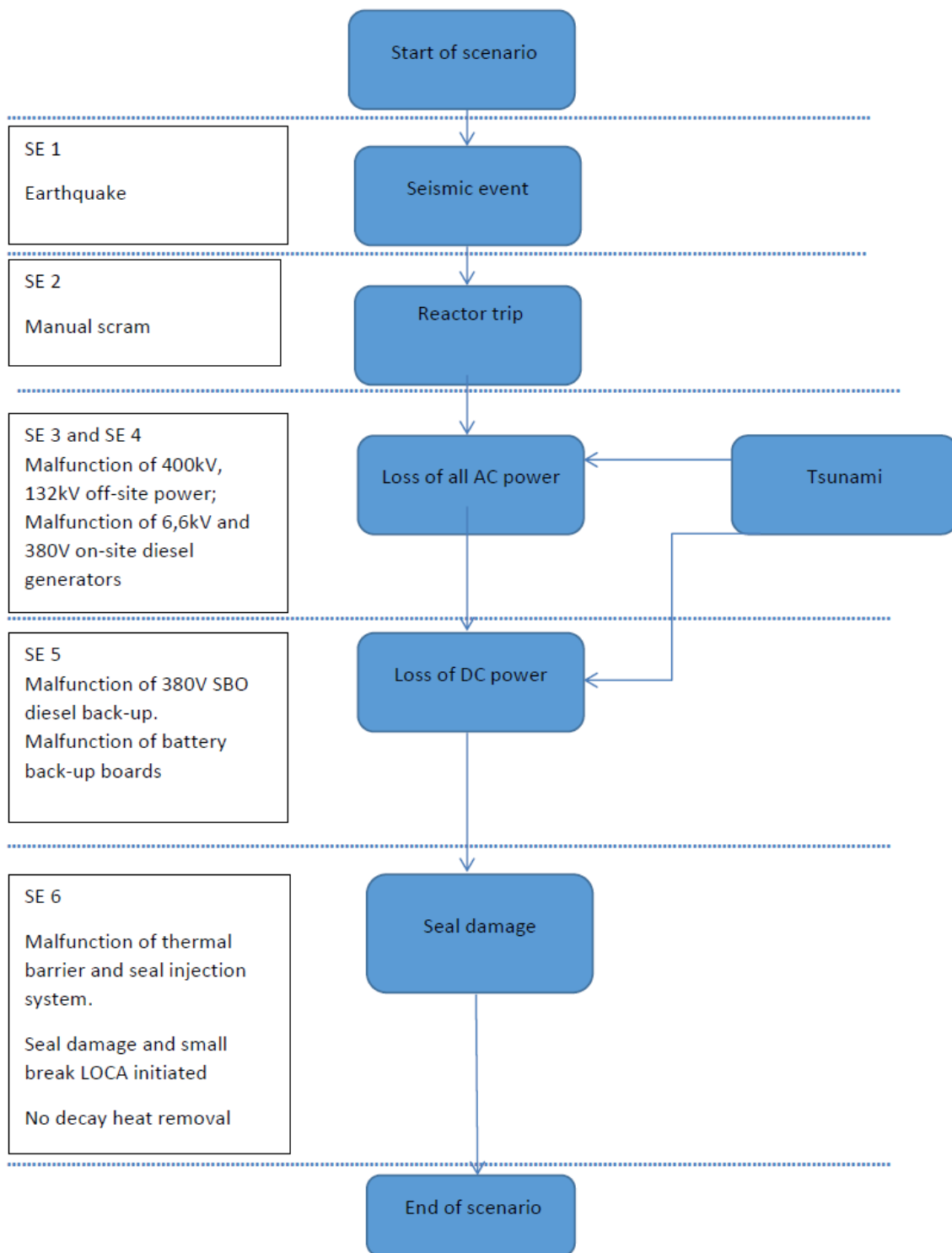


Figure 18: Sequence of events (SE) from Orchid® software simulator [8]

3.1.2.1 Sequence of simulation events (SE):

Sequences of events simulated are shown in Figure 18 and is further explained below.

SE1: Earthquake

Earthquake could not be simulated at Koeberg because the design of the nuclear island and the structures supported by it are seismically designed for the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE) such that the plant equipment and systems are able to:

- Maintain the primary system integrity,
- Shutdown the reactor and remove the residual heat from the reactor,
- Maintain the possible radioactive releases around the site at an acceptable level.

The model however is equipped with a system that can only detect the earth quake. The automatic action of the reactor protection is not available from the seismic event. The detection of a seismic event will actuate an alarm in control room. Subsequent operator actions are determined by the magnitude of the earthquake, which is determined from the seismic event recording equipment.

SE2: Reactor scram

The reactor is scrammed manually in the simulation since the auto trip due to seismic event is not part of the PWR simulator design, unlike the Fukushima Unit 1 BWR design which is designed to auto scram on the detection seismic event. For a PWR of this design, the action to shut down the plant will be based on the magnitude greater than Operating Basis Earthquake which is 0.15g in the horizontal direction or 0.1g in the vertical direction. The reactor trip is achieved by pulsing any of the redundant trip buttons which is train A or train B [15].

SE3: Offsite power loss

The off-site power loss is simulated by opening 400kV breaker and inhibiting the auto change over to alternative offsite power supply 132kV. Figure 19 demonstrate the simplified version of the power distribution in nuclear plant. The Power Evacuation system has three main transformers; the power is transferred in a transformer from the primary to the secondary windings, in terms of Faraday's Law of Induction. They transform the voltage of the electrical energy supplied by the generator (24kV ac); to appropriate voltages for the circuits it is intended to supply.

The main generator transformer steps the generator voltage up to 400kV ac for distribution over the national grid. The unit generators are connected to the 400kV ac grid systems. In the event of a loss of off-site power, the emergency diesel generators are designed to start to ensure a power supply to the essential loads. Under certain conditions it may be desirable to supply these essential loads from 132kV back up supply line so as to conserve the diesel generators for their standby emergency power source function. The off-site backup supply (132kV) is also inhibited in this simulation to prevent the supply of non-essential boards. This standby source is automatically brought into service when an under-voltage and / or under-frequency condition is monitored on the 132 kV bus-bars. A manual start is also possible if required by the operator [9].

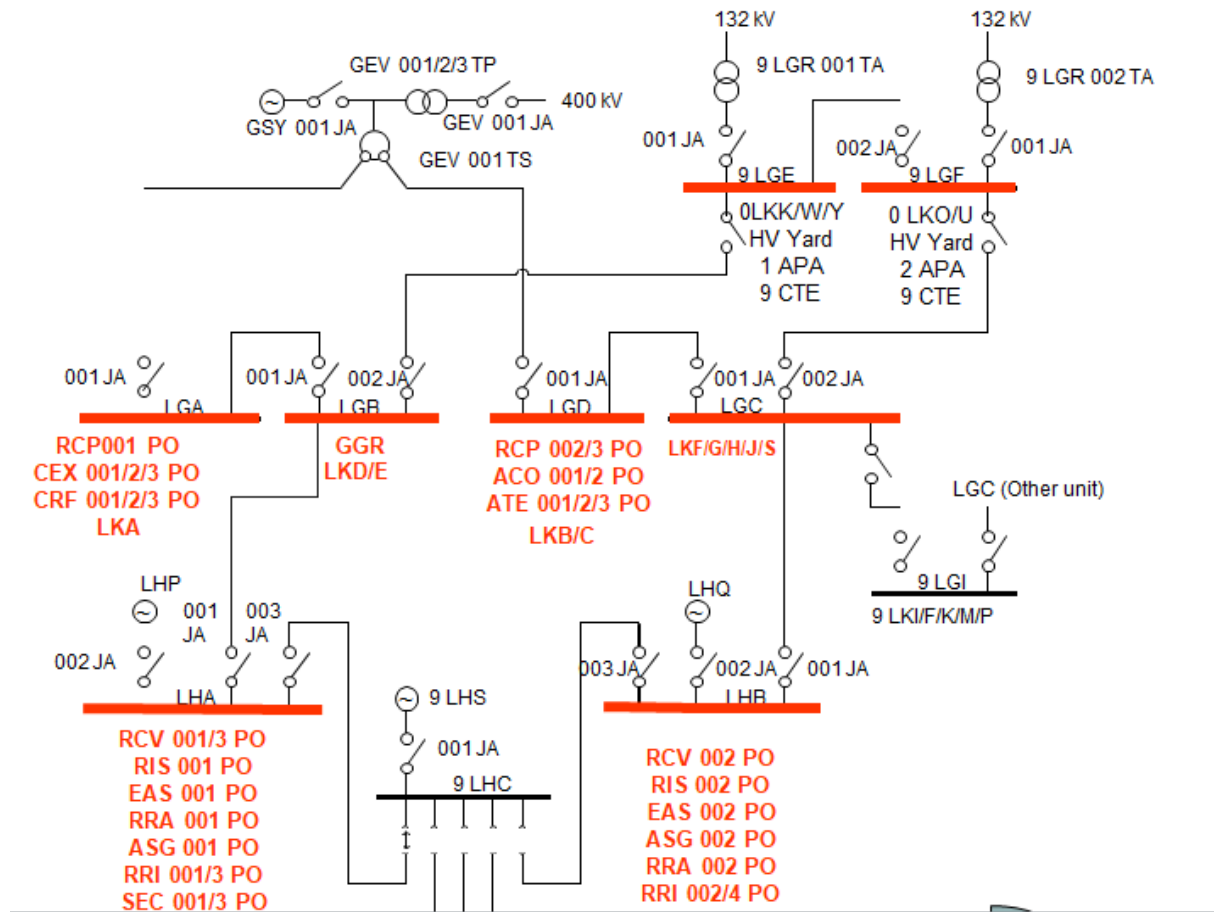


Figure 19: Electrical distribution diagram [8]

SE4: On-site power loss

The on-site diesel generators are inhibited from starting. The standby diesel generators are designed to back-up the essential boards in case of loss of power. The system supplies power at 6,6 kV ac to all the essential auxiliaries of the unit, from the essential Service boards LHA (Train A) and LHB (Train B) as illustrated in Figure 19. The power supplies to essential boards LHA and LHB are from either the LGB or LGC boards supplied from either the unit transformer GEV 001TS or the station transformers LGR 001 & 002TA), or from the diesel generators LHP & LHQ respectively. A fifth diesel generator 9LHS and its board 9LHC may replace one of the diesel generators on either unit, during an outage of one of the latter. The emergency diesel generator are inhibited to disable their design function which is to supply electrical power to the safeguard auxiliaries and enable a safe shutdown of the unit without damage to the equipment (i.e. to ensure the cooling of the reactor). With loss of all onsite AC power the seal injection from the charging pump and the back-up seal supply pump is lost. The thermal barrier cooling is also lost [9].

SE5: Direct Current (DC) loss

The SBO diesel generator that is designed to back up RCP pump seal water supply and DC battery boards is also inhibited. Without the battery, a loss of off-site power (loss of battery charger supplies) results in the immediate loss of digital controls, logics, and indications. This condition is undesirable during the short time required for the emergency diesel generators to run up and re-energise the essential switchboards (and battery chargers) [8].

SE6: Loss of core cooling

In this sequence event the plant safeguards system are not available to cool the reactor. The reactor coolant will be on natural circulation flow, which by design is driven by the steam generators, which are elevated higher than reactor and the residual heat from the reactor. At this point, the major concern is a

LOCA through the primary pump seals, caused by the simultaneous loss of all essential cooling system and seal injection system pumps. The loss of seal injection means that high pressure water from the pump casing will start to flow up into the seals, and as there is no cooling to the thermal barrier the water temperature will be approximately 280°C, instead of the normal seal water temperature of ~ 45°C. Although the seals are designed to withstand RCP system temperature up to 95°C but beyond this temperature, they deteriorate and lead to a leakage of water from the system [35].

3.2 The source term analysis

Due to the irradiation of the fuel assembly in the core there will be a build-up of fission products such as; ^{133}Cs , ^{143}Nd , ^{151}Sm , ^{155}Gd , ^{99}Tc , ^{145}Nd , ^{147}Sm , ^{150}Sm , ^{151}Sm , ^{152}Sm , ^{151}Eu and ^{153}Eu in the core. The amount of nuclear fuel (^{235}U and ^{238}U) will decrease with time and as burnup *increase*. These fission products are particularly important since they have a high neutron absorption cross-section, and as a result will decrease reactivity by absorbing neutrons in the system thus reducing the amount of neutrons that are available to participate in the fission process and subsequently lead to a decrease in k_{eff} . [43]. Activity in the core based on the different burn-up is listed in Appendix 3.

The fission products are radioactive and this will lead to increase in the activity in the core. The activity vs burnup relationship is shown in Figure 20 [44] . The activity increase more evident for the isotopes such as ^{85}Kr , ^{134}Cs and ^{137}Cs , which have relatively long half-lives.

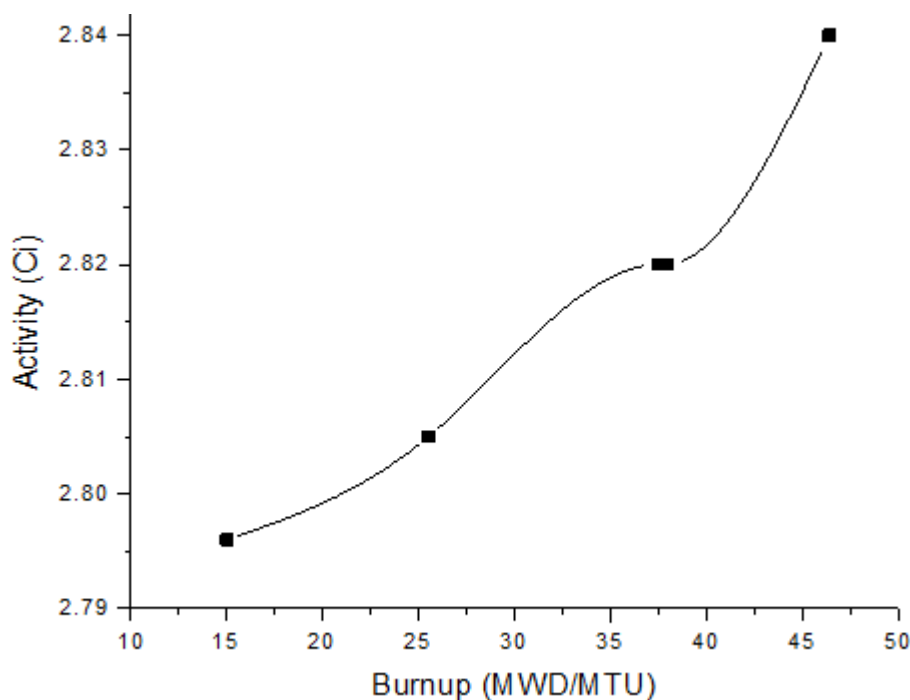


Figure 20: Activity as a function of Burnup (Based only on ^{235}U fuel)

In order to observe the activity in the core, two scenarios were simulated on Rascal code for radiation release results. Scenario 1 was simulated for a PWR plant and Scenario 2 was for BWR of Fukushima unit 1 type plant. In the two scenarios the SBO accident is chosen based on the events of Fukushima, it must be noted that the Fukushima plant is not available on Rascal list of nuclear plants; however Rascal allow list to be populated manually.

3.2.1 Initiating event

SBO in this analysis is initiated by seismic event and is used as basis for the consequential events and condition at the plant. Damage caused by the earthquake is assumed to result in a total loss of offsite power. In addition, onsite AC power is unavailable, due to failures of diesel generators to start or run as needed.

3.2.2 Release pathway

3.2.2.1 Dry containment leakage or failure

Design of the nuclear power plant is such that it fully complies with Defence in depth (DiD) principle which requires that barriers or radiological material with which the reactor is designed are effective enough to prevent escape of radioactive release by bypassing the containment. The containment is assumed to have failed due to the passing of the seal return line, which failed to close on containment isolation signal. This is demonstrated in Figure 21 for PWR and Figure 22 for BWR. The operating experience from a number of nuclear operators shows that the most common bypass through the containment is leakage through a penetration that was not properly sealed. This compromises the effectiveness of the containment to confine the radioactive material within the containment. To minimise the effect of radiation release, the PWR has a number of filters fitted to trap small quantities of airborne fission products from exhaust air from leaving the containment because of leaks in systems.

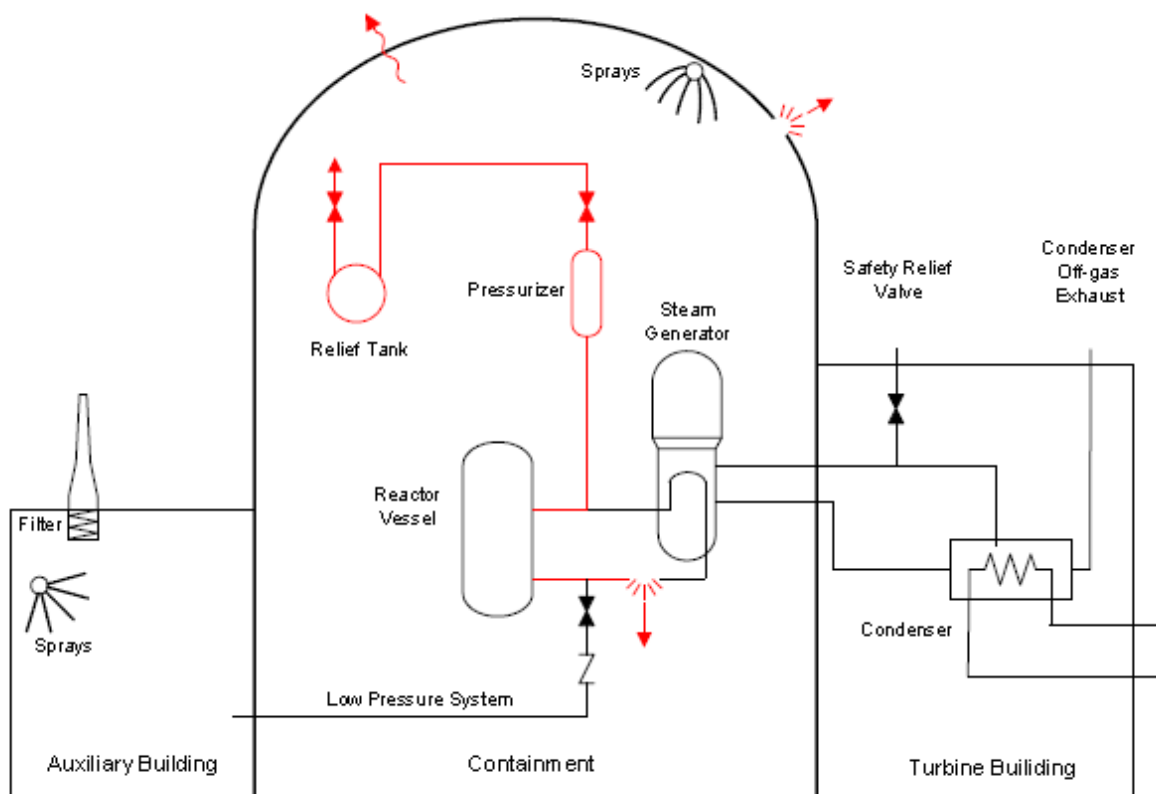


Figure 21 Containment leakage or failure of PWR plant [11]

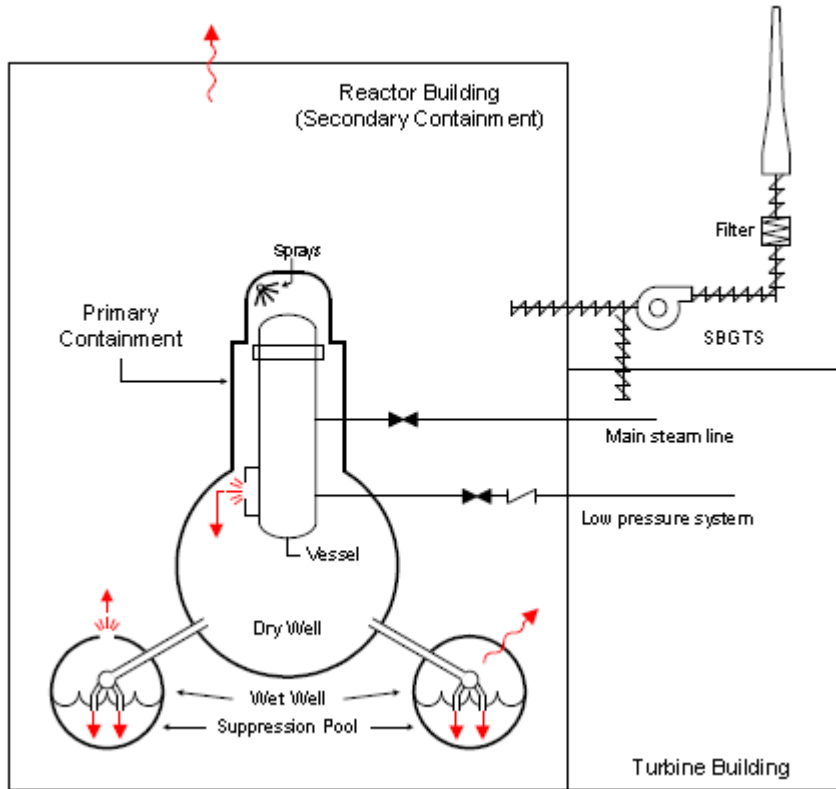


Figure 22: Release through wet well via direct, unfiltered pathway of BWR plant [11]

3.2.3 Time-dependent variation of the activity

As the irradiation of nuclear fuel continues in the reactor core, a continuous build-up of decay products occurs as a result of nuclear transmutation. When that happens, a full spectrum of these isotopes is formed, but only a few are of importance to radiation protection. The importance of nuclides in curie ranking is a function of the burnup, Appendix 3 shows nuclides ranking based on the burn up. The primary actinide contributors at early decay period include ^{241}Pu , ^{238}Pu , and ^{241}Am . In later decay periods the primary ^{239}Pu and ^{240}Pu become the primary contributors. For higher burnups ^{243}Am and ^{239}Np are important contributors to the activity. The dominant fission product contributors to the total curie levels at decay period less than 200 years exist in parent-daughter pair secular equilibrium include among others ($^{137}\text{Cs}/^{137\text{m}}\text{Ba}$); ($^{90}\text{Sr}/^{90}\text{Y}$); ($^{144}\text{Ce}/^{144}\text{Pr}$); ($^{106}\text{Ru}/^{106}\text{Rh}$) ^{85}Kr , ^{147}Pm and ^{134}Cs [45].

3.2.4 Scenario 1: Summary for PWR plant

Scenario 1 was simulated on Rascal code for PWR plant and is summarised in Table 8. Detailed simulation assumptions are provided in Appendix 5.

Table 8: Summary of PWR plant scenario

Title: LOCA with containment leak Run date/time: 2019/07/30 17:37	
<u>Description of Scenario</u> LOCA with all ECC's not functioning; Containment valves are passing releasing radioactive effluent to the atmosphere. The fuel cladding has failed due to high temperatures and ballooning.	
<u>Location</u> Name: Koeberg - Unit 1 City, county, state: Melkbosstrand, Western Cape Lat / Long / Elev: 33,6768° S, 18,4315° E, 18 m Time zone: South African Population: 52 million	<u>Reactor Parameters</u> Reactor power: 1700 MW _t Average burnup: 30000 MWd/MTU Containment type: PWR Dry Ambient Containment volume: 56633.69 m ³ Design pressure: 400kPa Design leak rate: 0, 17 %/d Coolant mass: 2, 00E+05 kg Assemblies in core: 217 Steam generator type: U-Tube SG water mass: 42184 kg
<u>Source Term</u> Type: Long Term Station Blackout (SOARCA) Shutdown: 2011/03/11 14:46 Release from core starts: 2011/03/11 22:46 Core damage estimated by: Core recovered status Core recovered: No Inventory: Default	<u>Release Pathway</u> Type: PWR - Dry Containment Leakage or Failure Description: Containment isolation valves failure Release height: 10 m Release events 2011/03/11 22:46 Sprays Off 2011/03/11 22:46 Leak rate (press/hole) 500 kPa, 5 cm diameter, 200 °C

3.2.5 Scenario 2 Summary for BWR plant

Scenario 2 was simulated on Rascal code for BWR plant and is summarised in Table 9. Detailed simulation assumptions are provided in Appendix 4.

Table 9: Summary of BWR plant scenario

Title: LOCA with containment leak Run date/time: 2019/07/30 22:38	
Description LOCA with all ECC's not functioning; Containment valves are passing releasing radioactive effluent to the atmosphere. The fuel cladding has failed due to high temperatures and ballooning.	
Location Name: Fukushima Unit1 City, county, state: Futaba, Japan, Fukushima Lat / Long / Elev: 37,2330° N, 141,0150° E, 10 m UTC Offset: 9 hours Population: not available	Reactor Parameters Reactor power: 1380MWt Average burnup: 30000MWd / MTU Containment type: BWR Mark I Containment volume: 7079.2 m ³ Design pressure: 413kPa Design leak rate: 0, 54 %/d Coolant mass: 1, 25E+05 kg Assemblies in core: 550
Source Term Type: Long Term Station Blackout (SOARCA) Shutdown: 2011/03/11 14:46 Release from core starts: 2011/03/11 20:46 Core damage estimated by: Core recovered status Core recovered: No Inventory: Default	Release Pathway Type: BWR - Release Through Wet Well via direct, unfiltered pathway Release height: 10 m Release events 2011/03/11 20:46 Leak rate (% vol) Design 2011/03/11 20:46 Wet well Subcooled

Chapter 4 Results and Interpretation

4.1 Results of simulation of Fukushima events using PWR Koeberg simulator

In this section the reactor parameters behaviours from a PWR seal failure are explained and the parallels are drawn to ascertain the same behaviour in a BWR plant like Fukushima Unit 1. The analysis is done to assist with the investigation to determine if LOCA existed at Fukushima at the early stage of the accident. The primary pump seals are known to be the weakest point on the system, which can lead to leaks if cooling is lost to them.

The PWR simulation trended the following reactor parameters,

- Reactor coolant Level
- Reactor coolant Pressure
- Reactor coolant Temperature
- No.1 Seal downstream temperature
- No.1 Seal upstream and downstream pressures
- No.1 seal leak off flow rate

4.1.2 Reactor coolant level

The best indication of the level in a PWR plant is in the pressuriser; level sensors installed in the pressuriser actuate all the protections associated with the level. The operator in the control room does not have any other reactor level monitoring indication except for the Pressuriser level indication. The make-up and the let-down system of the Reactor coolant system during normal operation is actuated by changes in the pressuriser's level. The level in the pressuriser is programmed to change with actual reactor power as illustrated in (Figure 23) [26].

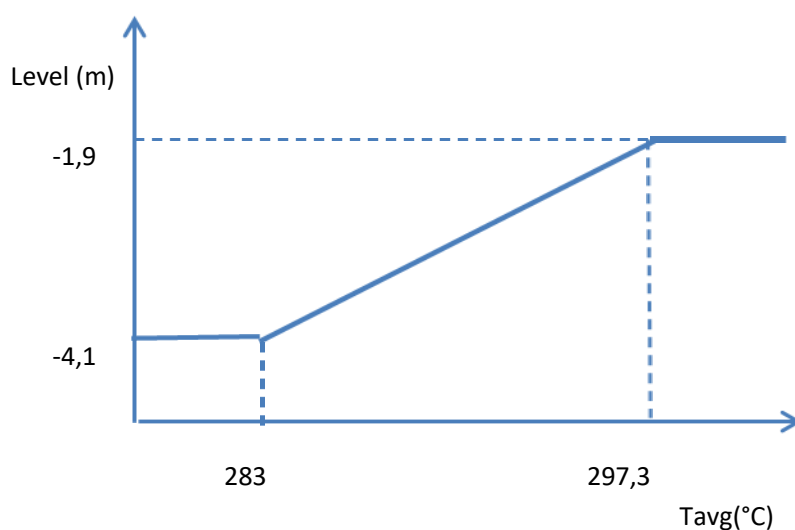


Figure 23: Pressuriser level program [26]

During power operation, there is a minimum pressuriser level which must be maintained in order to ensure correct pressuriser operation. To provide this minimum operating level, an adjustable no-load average temperature (T_{avg}) signal is provided, which is set at a value corresponding to a pressuriser level of -4.1 m. This signal is compared to auctioneered average temperature, which is an indication of density changes in the reactor coolant system. Programming is such that the programmed level will increase as power and T_{avg} rise. In this way, sufficient water volume in the pressuriser is assured, and charging water flowrate remains relatively constant for any given power level. The circuit is designed such that auctioneered T_{avg} is added to the basic set-point and causes programmed level to increase linearly with power to the level of -1.9 m at 100% power.

The normal level operation level at full power is maintained by a mass balance between the let-down flow; charging flow; seal injection flow and seal leak-off flow. When the SBO event starts, the charging and seal injection are lost and the let-down flow is automatically isolated. The level in the pressuriser start to decrease until it reaches the bottom (-6 m) of the pressuriser. The level decreases observed in (Figure 24) happens in two phases, Phase 1 (-2 m to -4 m) level decrease takes place after the reactor trip and this happens due to nuclear flux diminishing and the primary flow coasting down due to primary pumps trip and Phase 2 (-3 m to -4 m) happens as result of loss of coolant. The automatic makeup system fails to replenish the pressuriser to a no-load value due to unavailability of power to the make-up system pumps.

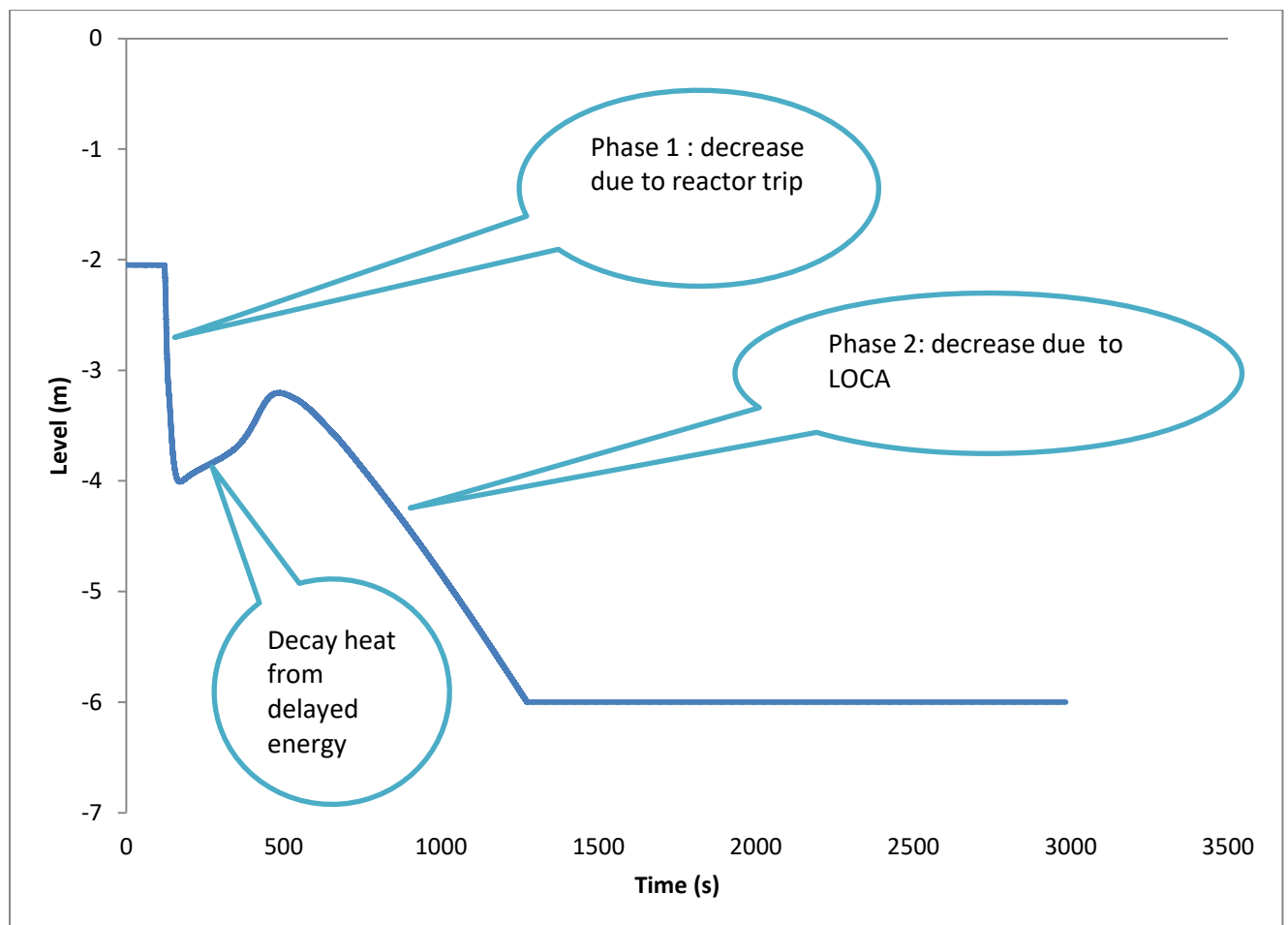


Figure 24: Pressuriser level

When the reactor trips, the prompt energy released from the fission reaction diminishes and only delayed energy released remains. The prompt energy release is mainly from fission fragments (~168

MeV), the remainder of the prompt energy release is from prompt neutrons (~5 MeV), prompt gamma rays (~7 MeV) and gamma rays released by neutrons captured in the core by ^{238}U (~6 MeV). The distribution of such energy with the products of fission, together with the estimated distance at which the energy is released, is given in Table 10 [46].

Prompt energy release	
fission fragments	168 MeV
prompt neutrons	5 MeV
prompt gamma rays	7 MeV
gamma rays from (n,γ) reactions	6 MeV
total prompt energy release	186 MeV
Delayed energy release	
β from fission fragment decay	7 MeV
γ from fission fragments decay	6 MeV
β and γ from nuclei produced by (n, γ)	1 MeV
total delayed energy release	14 MeV
Total energy release	
	200 MeV

Table 10: Distribution of energy distributed by ^{235}U fission [46]

The level in Phase 1 (Figure 24) decreased when the reactor trips and later increased again due to the decay heat produced by delayed energy release. The decay heat was a result of energy released by β^- and γ decays of fission products and radioactive nuclides resulting from (n, γ) reactions. Since the steam dump system to atmosphere was disabled by the loss of ac power, secondary pressure could no longer be limited to the no-load steam pressure but continued to rise to the secondary safety valve set pressure. The decay heat in the reactor is function of the length of reactor power operation prior to shut-down; the longer a reactor operates, and the more long lived nuclides accumulate in the core and therefore affecting the decay heat [46]. The decay heat available one minute after reactor shutdown is about 3%, which is about 83MW_{th} as shown in Figure 25.

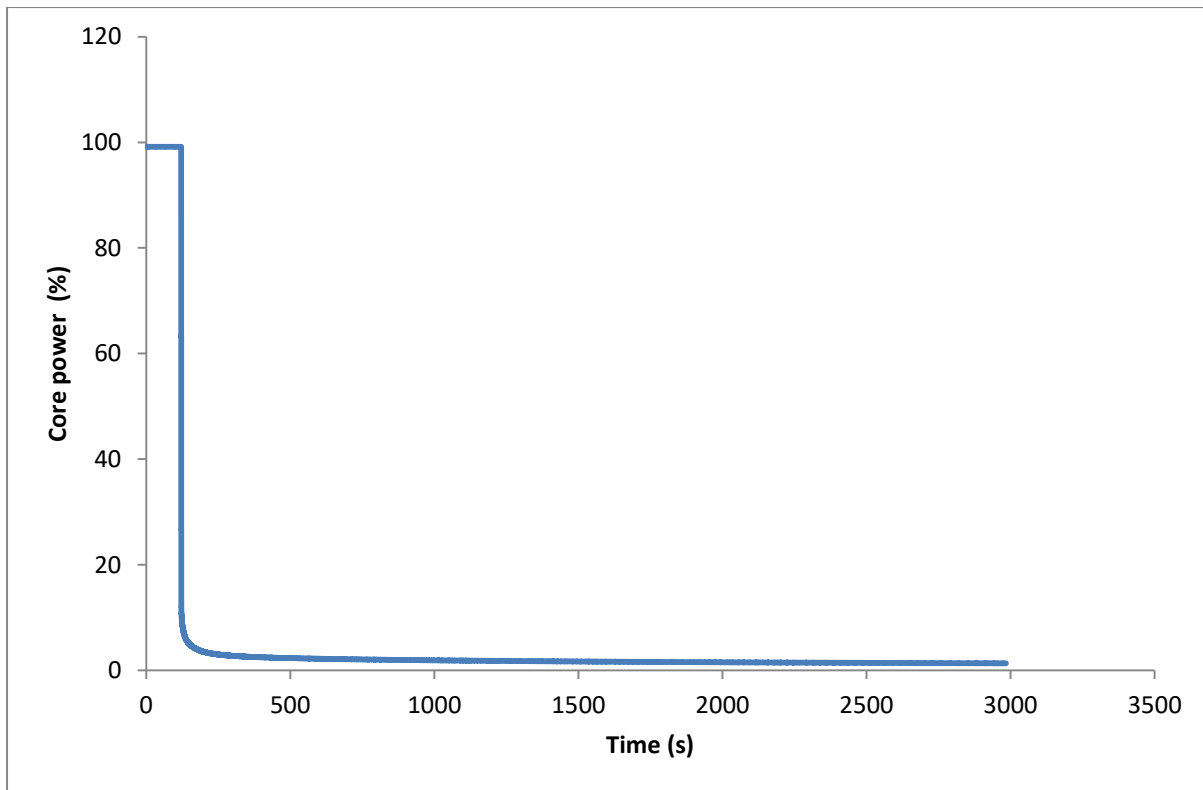


Figure 25: Core power after shutdown

The phase 2 level decrease in Figure 24 can be related to the loss of inventory in the RCP system, which can be attributed to the primary pump seal leakage since pressuriser outflow paths are isolated. The automatic normal and emergency make-up system loses power and therefore the charging pumps and the low heads pumps are not available to replenish the pressuriser to compensate for leaks on the primary system. Phase 2 happens at approximately 400 s after the SBO event, the reactor power at this point is about 2.4 % ($66 \text{ MW}_{\text{th}}$). With the residual power of $66 \text{ MW}_{\text{th}}$ and the continuation of the level decrease will ultimately lead to core to uncover.

The level will continue to decrease until it is off scale low on the pressuriser measuring instrumentation. The pressuriser level is followed by the decrease in pressure; it will continue to decrease until the RCP system is saturated at the set-point pressure of the steam generator safety valves. Seal leakage will continue to deplete the RCP inventory until the draining of the upper head and this will cause voids to form in the steam generators U-tubes. Voiding in the U-tubes will impede natural circulation through the RCP coolant loops and reflux boiling will be the mode of heat transfer between core and steam generators to remove decay heat [46].

The same behaviour was observed in a Fukushima Unit1 accident, the level decreased during the SBO. At 1hour 19 minutes after the SBO event the level was reported to be at +2 m above the top of active fuel (TAF), this was reported after the DC power was recovered and some instrumentation became available to the operator. The volume change of water that was lost at Fukushima during this period amount to 34 m^3 , which equates to $16 \text{ m}^3/\text{h}$ leak-rate [5].

4.1.3 Reactor coolant temperature and pressure

The RCP pressure and temperature are nominally at 15.5 MPa and 330 °C during power plant operation respectively. After a reactor trip, the temperature of the water was instantaneously decreased when the fission process was stopped by addition of negative reactivity from the control rods. The reactor coolant temperature decreased to a no load value of about 286°C about 22 s after the reactor trip; this was shortly followed by temperature increase which peaked at 326°C (344 s after the trip) due to residual heat from the core as it is illustrated in Figure 26. Once the reactor was shutdown, the energy released in radioactive decay and the primary pumps provided the main source of heating. Hence, the coolant needed to be maintained after the termination of the neutron-induced fission process in a reactor. The cooling of the reactor core by forced circulation was lost when the primary pump stopped during SBO conditions and the natural circulation of water began to take place.

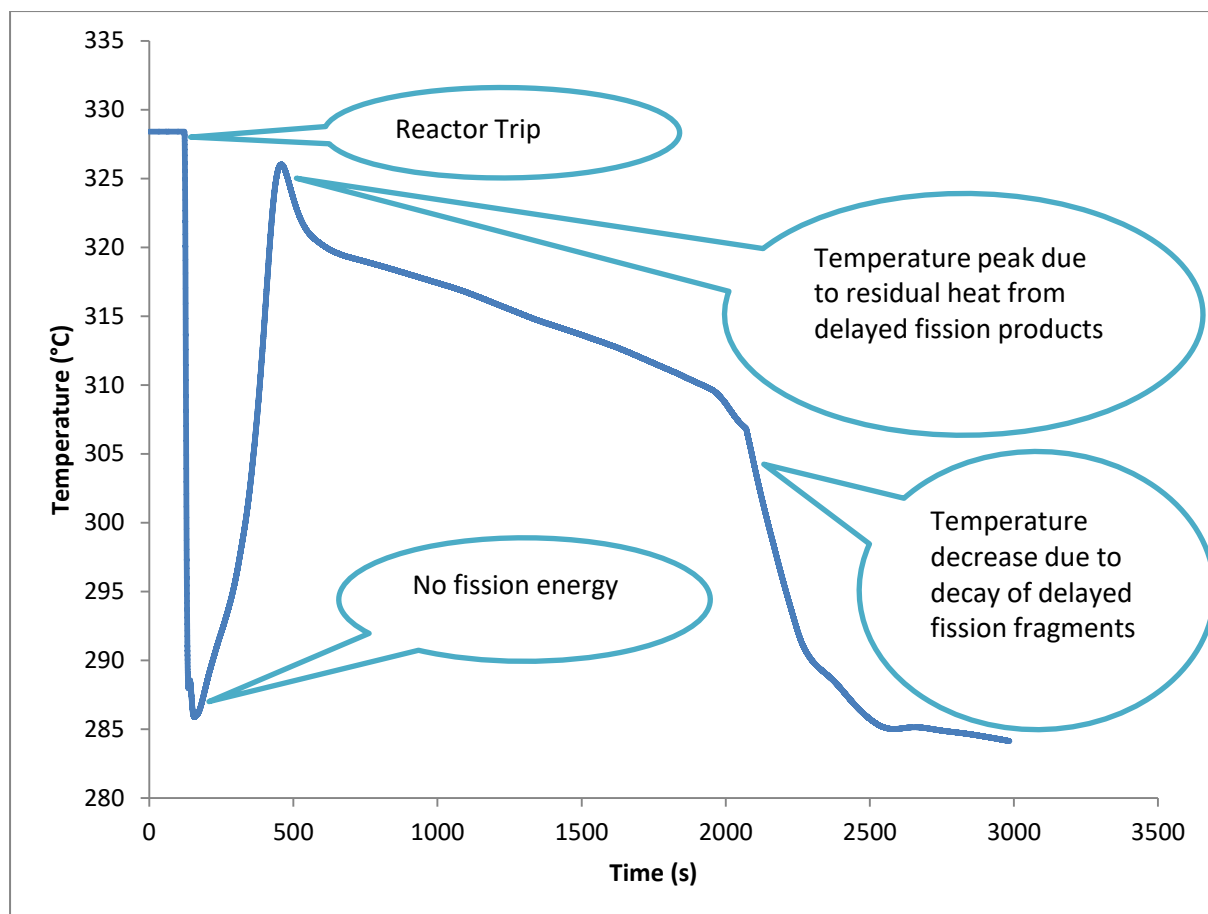


Figure 26: Reactor coolant temperature

4.1.4 Temperature downstream no.1 seal

The seal was designed to withstand up to temperature of 121°C, beyond these temperatures the seal started to fail. This temperature for complete seal failure was reached at 461 s after the reactor trip as illustrated in Figure 27. The high temperature downstream of the seal was an indication of hot water from the RCP system passing through the seal. When the temperature exceeded 95°C (at 344 s after reactor trip), the seal started to deteriorate. The seal O-ring material softened with increasing

temperature, affecting the O-ring sealing ability and life. This condition affected the angle between the faceplates of the RCP seals and the gap between the faceplates of the number 1 film riding seal.

Under normal circumstances the seal water will return to volume control system VCT via seal return line, the line is equipped with a heat exchanger that cool the seal water before is used again as seal injection. The seal return line has the containment isolation valve which isolates during the accident to prevent the primary water from leaving the reactor building. The seal return line was isolated and the no.1 seal failed, the safety valve on the return line opened to pressuriser relief tank and relieved the pressure in the line. The primary water leaking through the seal will ended in the pressuriser relief tank, which ultimately filled up, the rupture disc collapsed and the primary coolant leaked out to the floor sumps [8].

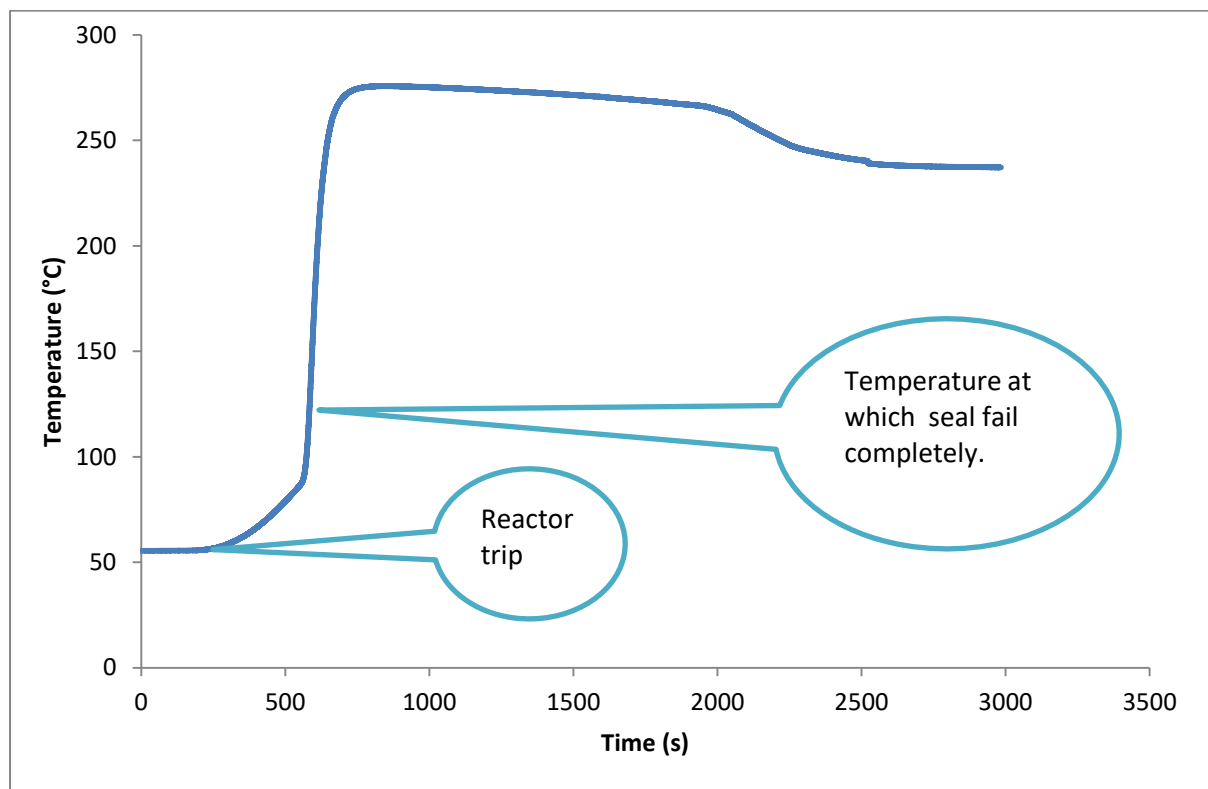


Figure 27: Temperature downstream no: 1 seal

4.1.5 Pressure upstream and downstream the no. 1 seal

Figure 28 shows the equalising of pressure difference across the seal, which was an indication of the seal failure. The decrease on the pressure difference across the seal coincided with the temperature rise downstream of the no.1 seal in Figure 27 which signified the failure of the seal as more hot water passed through. The number one seal gap was normally maintained by balanced opening and closing forces. The closing force on the seal was proportional to the differential pressure across and acted on the upper surface of the seal ring. The pump stator was free to float in an axial direction and an equilibrium position was established by the force generated by the pressure distribution within the fluid film between the faces was balanced by the closing forces due to the pressure acting on the top of the seal stator assembly. This arrangement allowed the seal to accommodate axial travel of the pump shaft while maintaining a nearly constant film thickness and leakage rate [6].

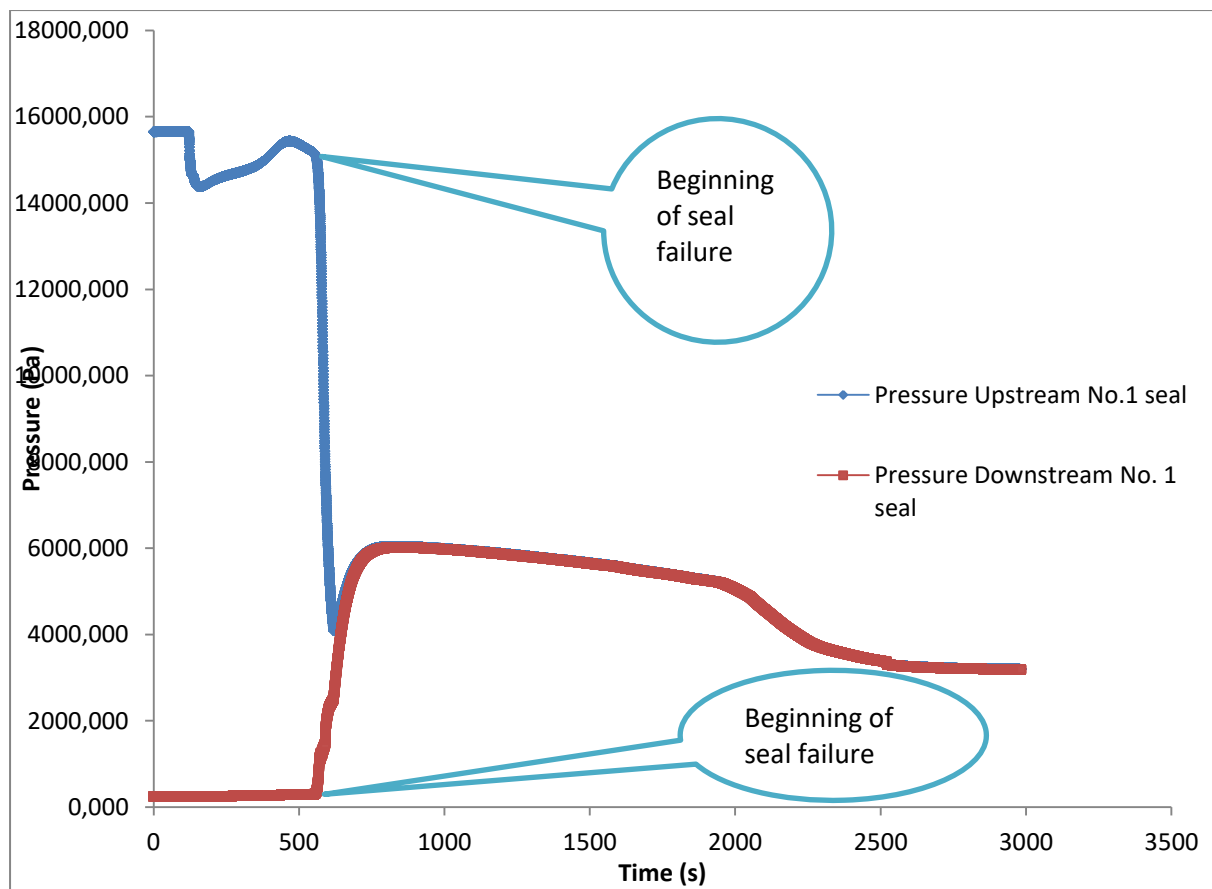


Figure 28: Pressure upstream and downstream the no.1 seal

4.1.6 The no.1 seal leak off flow

The seal leak off flow also increased at 344 s after the reactor trip which also indicated the failure of the no. 1 seal as shown in Figure 29; this was consistent with the increase of temperature and equalising pressure difference across the seal as shown in Figure 27 and Figure 28 respectively. The flow path was along the pump shaft through the labyrinth seals, through the cartridge seal internals, and either through one of the three seal leak-off lines or out the top of the pump casing. The flow rate also depended on the no.1 seal leak off line pressure which has a pressure relief valve that lifts at pressure.

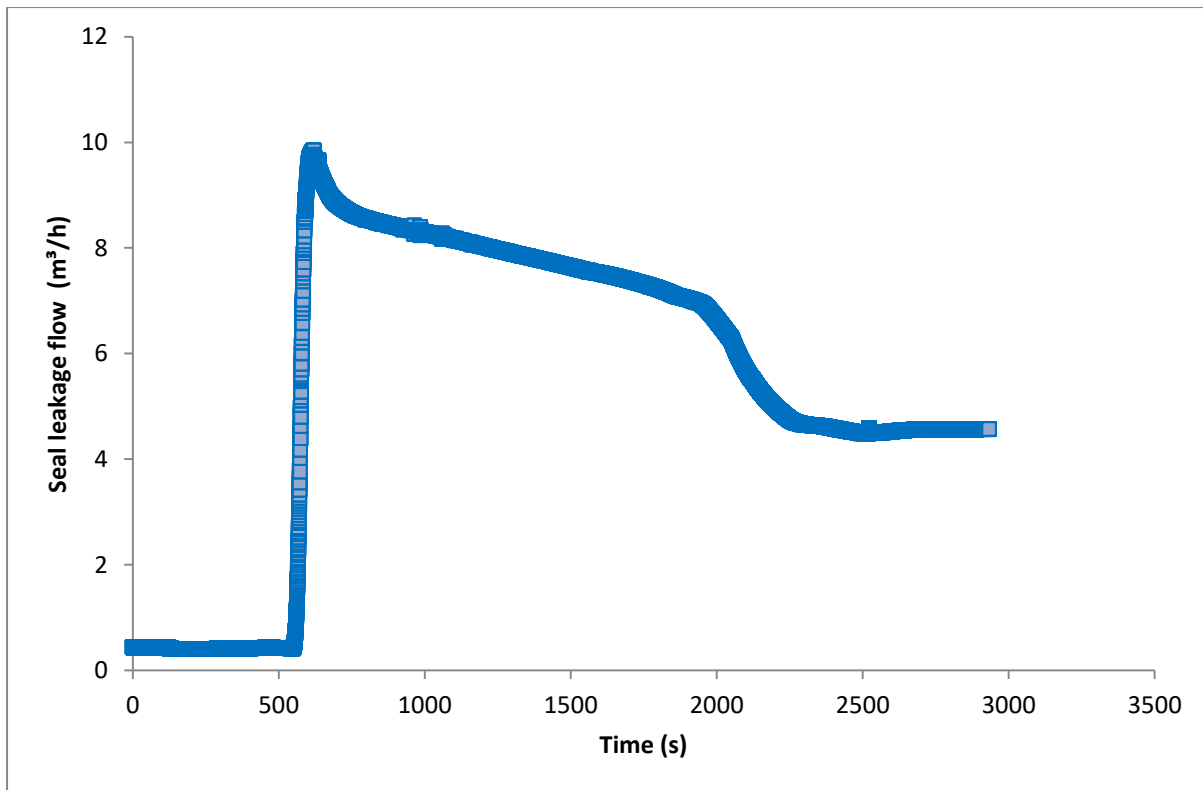


Figure 29: The no: 1 seal leak off flow

The maximum leak rate from each loop was $13.5 \text{ m}^3/\text{h}$ as illustrated in Figure 29. PWR plant used for this simulation was a three loop therefore total of $40 \text{ m}^3/\text{h}$ leak rate was based on three primary pumps that failed at same time. The total leak rate from the simulation data was averaged at $40 \text{ m}^3/\text{h}$, this leak rate was within the range of Westinghouse and ETEC predicted scenario of initial leakage with no.1 with no.2 and no.3 seal functional [2].

The prediction by Westinghouse ($51 \text{ m}^3/\text{h}$) and ETEC ($44 \text{ m}^3/\text{h}$) come from proven studies as discussed in chapter 2 section 2.4.8 [33]. The events of Fukushima that led to loss of coolant accidents are discussed in section 3.1 of this paper and they form part of the basis for this research. The full events timelines are provided in Appendix 1 and were used to determine the seal leak rate of $39 \text{ m}^3/\text{h}$ (see Appendix 2). The Fukushima leak rates are based on the level changes provided in the TEPCO [5] reports which can be attributed to the mass loss through the RCP pump seal. The Koeberg simulation are relatively within the range of the Fukushima accident estimated leak rate, the leak rate was found to be approximately $41 \text{ m}^3/\text{h}$ based on the evolution of events during the accidents. Figure 30 shows a comparison of the seal leakage for ETEC, Westinghouse, Koeberg and Fukushima.

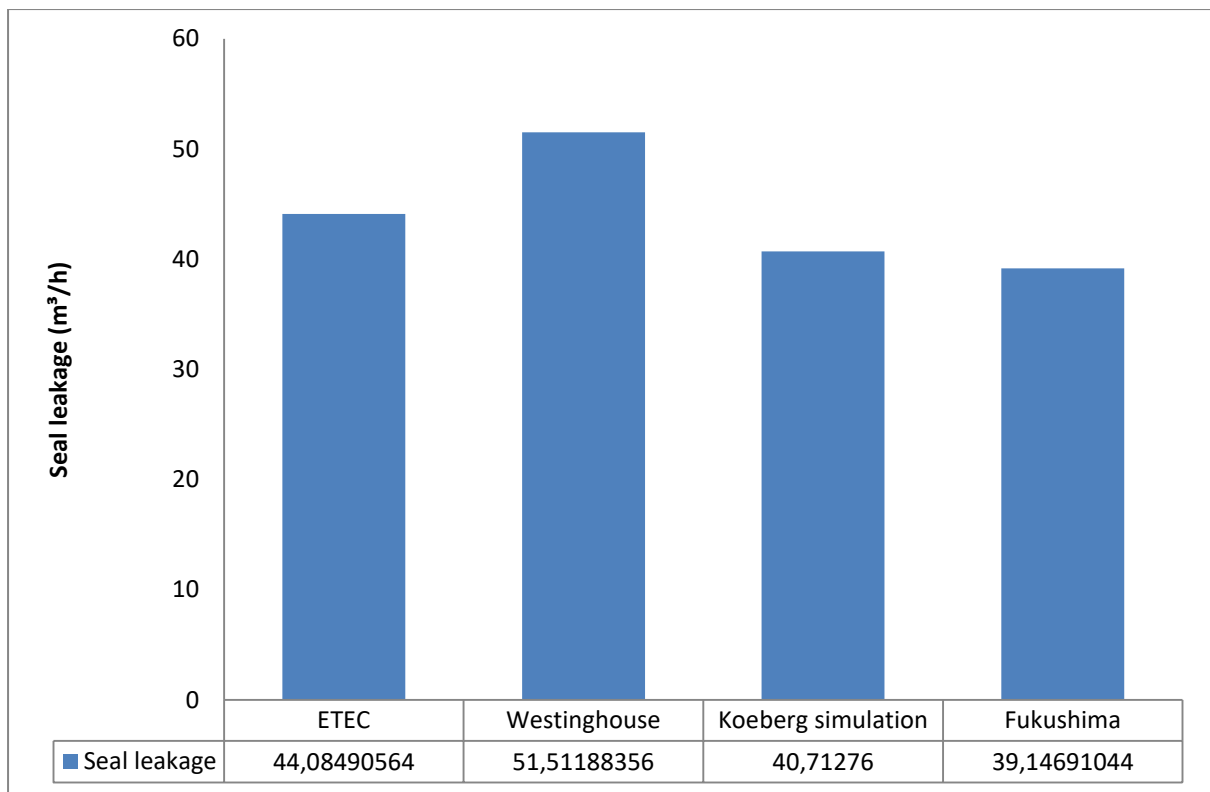


Figure 30: Seal leakage comparison

4.2 Source term results

4.2.1.1 Release pathway results for PWR scenario 1

Table 11 lists the maximum dose results from the scenario 1, which were based on PWR plant. The following to be noted from the results;

- Inhalation dose coefficients used by Rascal are from International Commission on Radiological Protection (ICRP) 60/72 [11].
- Doses exceeding EPA PAGs are underlined.
- The early phase PAG's: TEDE - 10mSv,
- Thyroid CED - adult thyroid dose from exposure to all radionuclides as related to 1992 EPA PAGs (no interim PAG - not underlined)
- Child Thyroid CED - 1y old result (most limiting) from exposure to radio-iodine's only - for KI administration considerations
- Intermediate-Phase PAGs: 1st year - 20mSv, 2nd year - 5mSv

Table 11 : Summary of dose results from PWR plant [11]

Maximum Dose Values (Sv) - close-in								
Distance from the release (kilometers)	0,16	0,32	0,48	0,8	1,13	1,61	2,41	3,22
Total ED	<u>6,50E+00</u>	<u>2,30E+00</u>	<u>1,30E+00</u>	<u>6,20E-01</u>	<u>3,70E-01</u>	<u>2,00E-01</u>	<u>1,20E-01</u>	<u>9,40E-02</u>
Thyroid CED	4,60E+01	1,50E+01	7,80E+00	3,50E+00	2,10E+00	1,20E+00	6,90E-01	5,30E-01
Child Thyroid CED	<u>8,80E+01</u>	<u>2,80E+01</u>	<u>1,50E+01</u>	<u>6,80E+00</u>	<u>4,00E+00</u>	<u>2,30E+00</u>	<u>1,30E+00</u>	<u>1,00E+00</u>
Inhalation CED	3,90E+00	1,20E+00	6,60E-01	3,00E-01	1,80E-01	1,00E-01	5,90E-02	4,60E-02
Cloudshine	1,30E+00	6,50E-01	4,40E-01	2,20E-01	1,30E-01	6,90E-02	4,20E-02	3,40E-02
4-day Groundshine	1,30E+00	4,10E-01	2,20E-01	9,90E-02	5,90E-02	3,30E-02	2,00E-02	1,50E-02
Inter Phase 1st Yr	<u>1,00E+01</u>	<u>3,20E+00</u>	<u>1,70E+00</u>	<u>7,80E-01</u>	<u>4,60E-01</u>	<u>2,60E-01</u>	<u>1,50E-01</u>	<u>1,20E-01</u>
Inter Phase 2nd Yr	<u>5,00E+00</u>	<u>1,60E+00</u>	<u>8,40E-01</u>	<u>3,80E-01</u>	<u>2,30E-01</u>	<u>1,30E-01</u>	<u>7,40E-02</u>	<u>5,80E-02</u>
Maximum dose values (Sv) - to 16,1 km								
Distance from the release (kilometers)	4,8	6,4	8	11,3	16,1			
Total ED	<u>6,60E-02</u>	<u>5,80E-02</u>	<u>4,70E-02</u>	<u>3,10E-02</u>	<u>1,90E-02</u>			
Thyroid CED	3,50E-01	3,10E-01	2,50E-01	1,70E-01	9,90E-02			
Child Thyroid CED	<u>7,40E-01</u>	<u>6,60E-01</u>	<u>5,40E-01</u>	<u>3,60E-01</u>	<u>2,10E-01</u>			
Inhalation CED	2,90E-02	2,60E-02	2,10E-02	1,40E-02	7,70E-03			
Cloudshine	2,80E-02	2,40E-02	2,00E-02	1,40E-02	8,60E-03			
4-day Groundshine	9,30E-03	8,20E-03	6,60E-03	4,30E-03	2,40E-03			
Inter Phase 1st Yr	<u>7,20E-02</u>	<u>6,30E-02</u>	<u>5,10E-02</u>	<u>3,30E-02</u>	1,90E-02			
Inter Phase 2nd Yr	<u>3,50E-02</u>	<u>3,10E-02</u>	<u>2,50E-02</u>	<u>1,60E-02</u>	<u>9,50E-03</u>			

The results in Figure 31 showed that a total of seven of 22 ½° sectors were affected by the release, out of that the regions which had the worst impact were those on the plume path (sectors D to F) which were exposed to a total effective dose equivalent greater than 50mSv at the distance of 0.8 km from the point of release.

Total Effective Dose

Accumulated between 2011/03/11 22:46 and 2011/03/12 06:46

Loca with containment leak

Koeberg - Unit 1

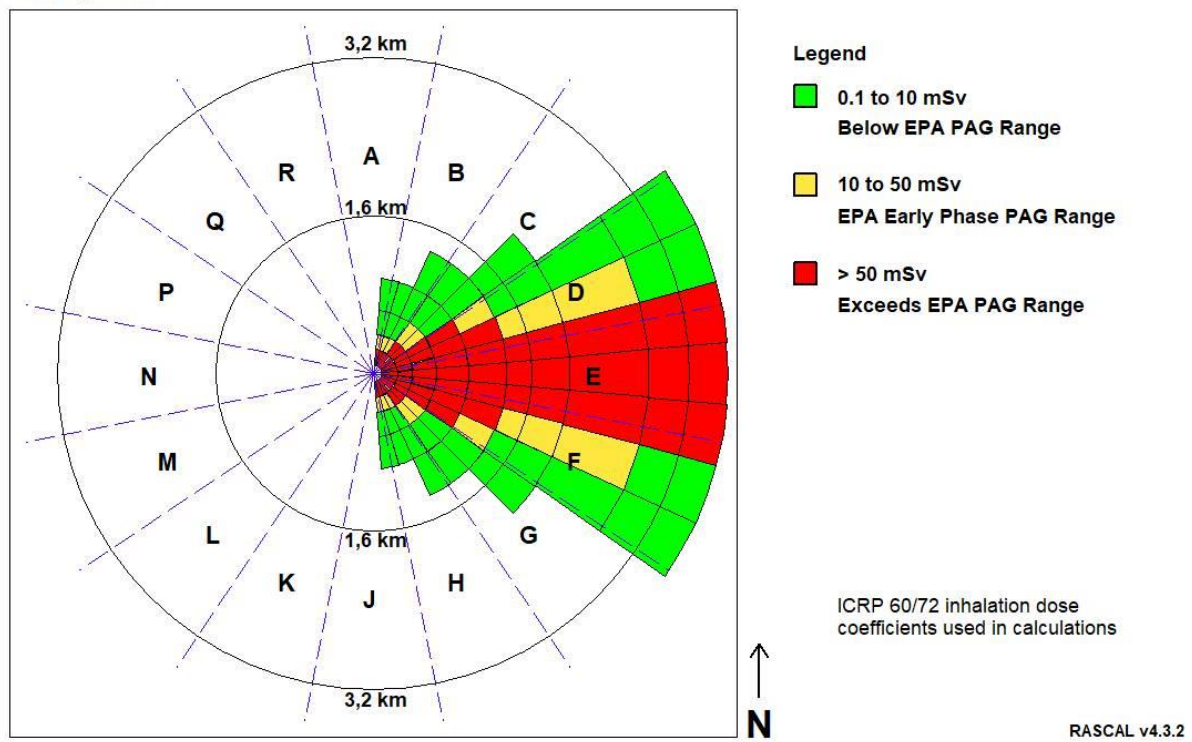


Figure 31: TEDE for PWR for 3,2km radius [11]

The total number of $22\frac{1}{2}^{\circ}$ sectors that were impacted by various levels of radioactive release is five, with one whole sector (E) and two fractions of D and F having been exposed to a dose that is higher than 50mSv. Sectors C and G had the least impact (with the dose ranging from 0.1 to 10 mSv) because they are further away from the centre of the plume. The areas on the periphery of the plume path were exposed to the dose ranging from 10 to 50 mSv and 0.1 to 10 mSv depending on their proximity to the nuclear power station. Figure 32 showed that the overall distance that was impacted extends to 16.1 km from the power station.

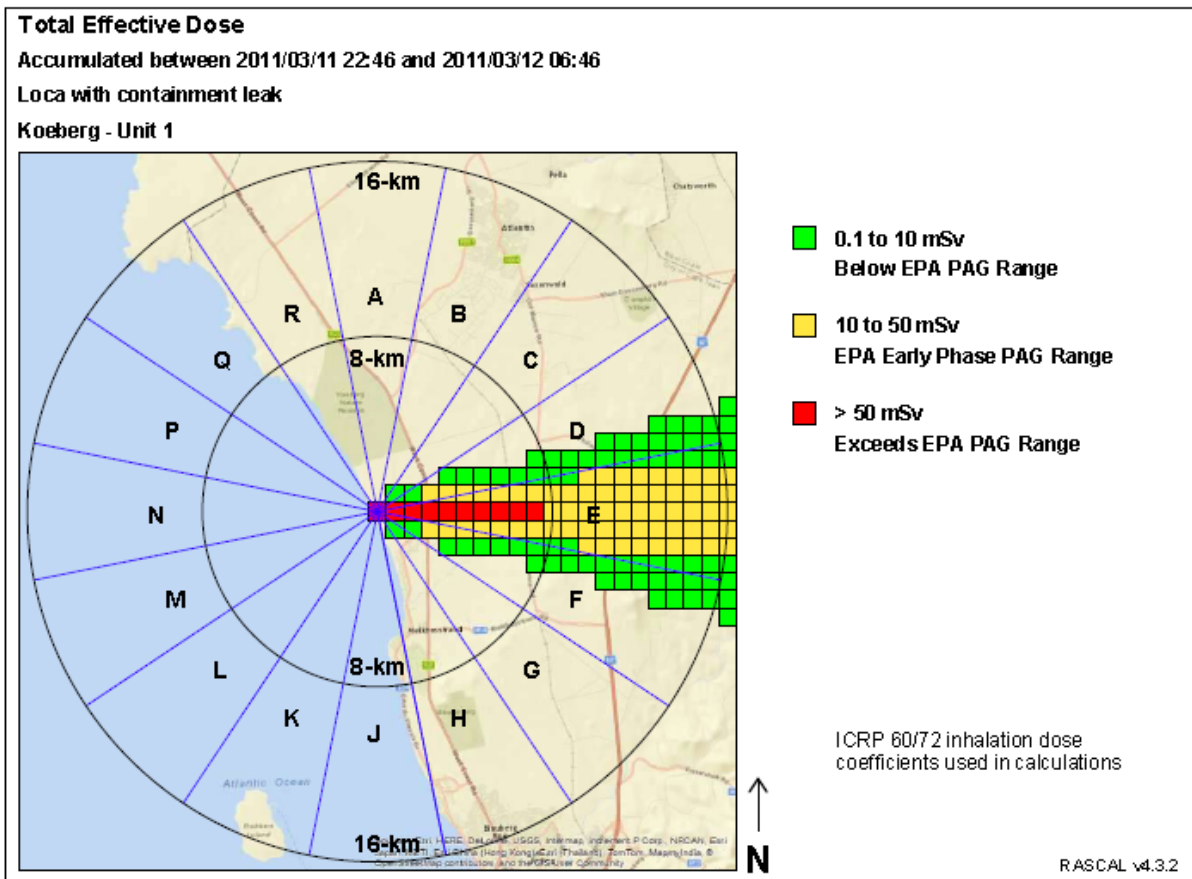


Figure 32: TEDE for PWR for 16.1 km [11]

According to the results, the early phase PAG's recommended [39] were the following:

- Sheltering-in-place or evacuation of the public
- Prioritise personal decontamination
- Simple actions: rinsing roofs and streets, sweeping streets
- Supplementary administration of prophylactic drugs – KI; See Table 5.
- Limit emergency worker exposure (total dose incurred over entire response).
- Health physics support for workers.

For the intermediate phase PAG's recommended [39] were as follows:

- Relocation of the public.
- Early clean up (Vacuuming, Washing and vegetation removal).
- Dose projections are higher than what is required for dose reduction techniques
- Food interdiction [22] - (precautionary measures for contaminating human and animal feed; limit consumption of food suspected to be contaminated and reduce amount of contamination of food)
- Drinking water – Apply similar approach as food interdiction and provide the alternative water supply (e.g. blending water source, import tanker trucks and bottled water)
- Limit emergency worker exposure by ALARA principle and Dose and risk relationship

For late phase PAG's recommended [39] were as follows:

- Clean up and waste removal

4.2.1.2 Time- dependent variation of the activity results for PWR scenario 1

- 1) The results in Figure 33 showed that there is a non-linear relationship between the activity and the decay period of the release, and that the overall trend was that, the activity decreased with time. The observed variation in the activity as the decay period progresses was due to the decay of unstable nuclides and the build-up of other radioactive isotopes. The reason for a sharp drop in the material of ^{95}Kr in PWR versus in BWR is because of plate-out of on the SSC of the PWR which are assisted by the fact that PWR by its nature does not undergo boiling and BWR does. This high temperature from boiling slows down the plate-out process.

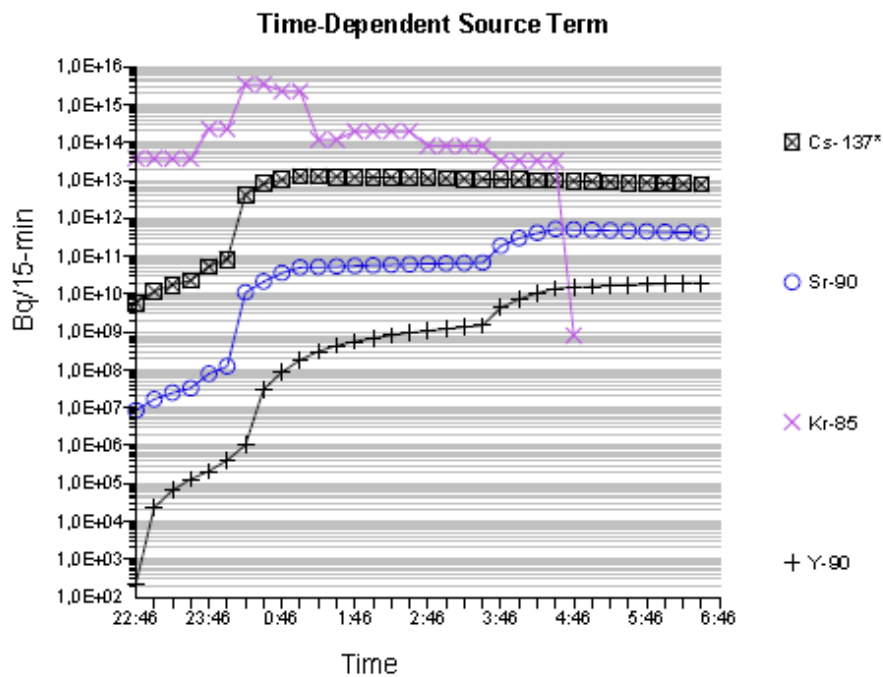


Figure 33: Individual Activity as function of time after release for PWR [11]

The following activities (curies) were predicted in various structures, systems and components (SSC) of the PWR plant: total activity, noble gasses, total I activity, ^{85}Kr , ^{135}Xe , ^{131}I , ^{135}I , ^{137}Cs , ^{89}Sr , ^{132}Te , ^{106}Ru , ^{99}Mo and ^{141}Ce . Figure 34 was an excerpt of the time-dependent distribution of total activity in various SSC. The activity shown in the Rx Core (reactor core), Cont (containment), and Env (environment) columns was the activity at the end of the time period. The activity in the core and containment was adjusted for decay and depletion. The activity in the Env column was the total activity released to the environment, not corrected for decay. The activity shown in the Rx-Cont and Cont-Env columns was the activity transfer during the period [11].

Time (s)	Rx Core	Rx-Cont	Cont	Cont-Env	Env
-21600	2.72E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0	2.25E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
900	2.22E+09	1.80E+07	1.46E+07	3.12E+02	3.12E+02
1800	2.13E+09	8.34E+07	8.03E+07	1.87E+03	2.18E+03
2700	2.01E+09	1.10E+08	1.56E+08	3.59E+03	5.77E+03
3600	1.89E+09	1.09E+08	2.19E+08	5.28E+03	1.11E+04
4500	1.83E+09	5.48E+07	2.28E+08	5.95E+03	1.70E+04
5400	1.79E+09	2.85E+07	2.17E+08	6.22E+03	2.32E+04
6300	1.77E+09	9.99E+06	1.95E+08	6.23E+03	2.95E+04
7200	1.76E+09	9.95E+06	1.79E+08	6.25E+03	3.57E+04
8100	1.75E+09	1.61E+05	1.68E+08	6.17E+03	4.19E+04
9000	1.74E+09	1.60E+05	1.65E+08	6.10E+03	4.80E+04
9900	1.74E+09	1.59E+05	1.61E+08	6.05E+03	5.40E+04
10800	1.73E+09	1.58E+05	1.58E+08	5.99E+03	6.00E+04
11700	1.72E+09	2.78E+04	1.55E+08	5.94E+03	6.60E+04
12600	1.72E+09	2.77E+04	1.52E+08	5.89E+03	7.19E+04
13500	1.71E+09	2.75E+04	1.49E+08	5.85E+03	7.77E+04
14400	1.71E+09	2.74E+04	1.47E+08	5.80E+03	8.35E+04
15300	1.70E+09	2.72E+04	1.44E+08	5.76E+03	8.93E+04
16200	1.69E+09	2.71E+04	1.41E+08	5.72E+03	9.50E+04
17100	1.69E+09	2.69E+04	1.39E+08	5.68E+03	1.01E+05
18000	1.68E+09	2.68E+04	1.37E+08	5.63E+03	1.06E+05
18900	1.68E+09	2.66E+04	1.34E+08	5.60E+03	1.12E+05
19800	1.67E+09	2.65E+04	1.32E+08	5.56E+03	1.17E+05
20700	1.67E+09	2.63E+04	1.30E+08	5.52E+03	1.23E+05
21600	1.66E+09	2.62E+04	1.28E+08	5.48E+03	1.28E+05
22500	1.66E+09	1.60E+05	1.26E+08	5.45E+03	1.34E+05
23400	1.65E+09	1.59E+05	1.25E+08	5.42E+03	1.39E+05
24300	1.65E+09	1.58E+05	1.23E+08	5.38E+03	1.45E+05
25200	1.64E+09	1.58E+05	1.21E+08	5.35E+03	1.50E+05
26100	1.64E+09	1.57E+05	1.20E+08	5.32E+03	1.55E+05
27000	1.63E+09	1.57E+05	1.18E+08	5.29E+03	1.61E+05
27900	1.63E+09	1.56E+05	1.17E+08	5.26E+03	1.66E+05
28800	1.62E+09	1.56E+05	1.15E+08	5.23E+03	1.71E+05

Figure 34: Total activity (Ci) balance among various SCCs of the plant

4.2.2.1 Release pathway results for BWR scenario 2

Table 12 lists the maximum dose results from scenario 2, which was based on BWR plant. The following should be noted from the results [11];

- Inhalation dose coefficients used was ICRP 26/30.
- Doses exceeding EPA PAGs were underlined.
- The early phase PAG's: TEDE - 10 mSv, Thyroid (iodine) CDE - 50 mSv
- *** indicates values less than 10 µSv

Table 12: Summary of dose results from BWR plant

Maximum Dose (Sv) close in								
Distance (kilometers)	0,16	0,32	0,48	0,8	1,13	1,61	2,41	3,22
Total ED	<u>1,90E-01</u>	<u>5,00E-02</u>	<u>2,30E-02</u>	8,20E-03	4,30E-03	2,10E-03	9,40E-04	5,30E-04
Thyroid CDE	<u>1,70E+00</u>	<u>4,40E-01</u>	<u>2,00E-01</u>	7,20E-02	3,70E-02	1,80E-02	8,20E-03	4,60E-03
Inhalation CEDE	8,40E-02	2,20E-02	9,80E-03	3,60E-03	1,80E-03	9,10E-04	4,10E-04	2,30E-04
Cloudshine	1,10E-03	4,90E-04	3,00E-04	1,40E-04	7,70E-05	4,50E-05	1,10E-05	***
4-day groundshine	1,10E-01	2,80E-02	1,20E-02	4,50E-03	2,30E-03	1,20E-03	5,20E-04	2,90E-04
Inter Phase 1st Yr	<u>7,20E-01</u>	<u>1,90E-01</u>	<u>8,40E-02</u>	<u>3,10E-02</u>	1,60E-02	7,80E-03	3,50E-03	2,00E-03
Inter Phase 2nd Yr	<u>3,40E-01</u>	<u>8,70E-02</u>	<u>3,90E-02</u>	<u>1,40E-02</u>	<u>7,40E-03</u>	3,60E-03	1,60E-03	9,20E-04
Maximum Dose (Sv) - up to 16,1 km								
Distance (kilometers)	4,8	6,4	8	11,3	16,1			
Total ED	2,20E-04	1,30E-04	9,40E-05	6,10E-05	4,40E-05			
Thyroid CDE	1,90E-03	1,20E-03	8,20E-04	5,40E-04	3,90E-04			
Inhalation CEDE	9,50E-05	5,70E-05	4,10E-05	2,70E-05	1,90E-05			
Cloudshine	***	***	***	***	***			
4-day groundshine	1,20E-04	7,10E-05	5,10E-05	3,30E-05	2,40E-05			
Inter Phase 1st Yr	8,00E-04	4,90E-04	3,50E-04	2,30E-04	1,60E-04			
Inter Phase 2nd Yr	3,80E-04	2,30E-04	1,60E-04	1,10E-04	7,60E-05			

The results in showed Figure 35 that in total of four of 22 ½° sectors were affected by the release; none of the regions beyond 1,6 km had been seriously impacted by the plume path.

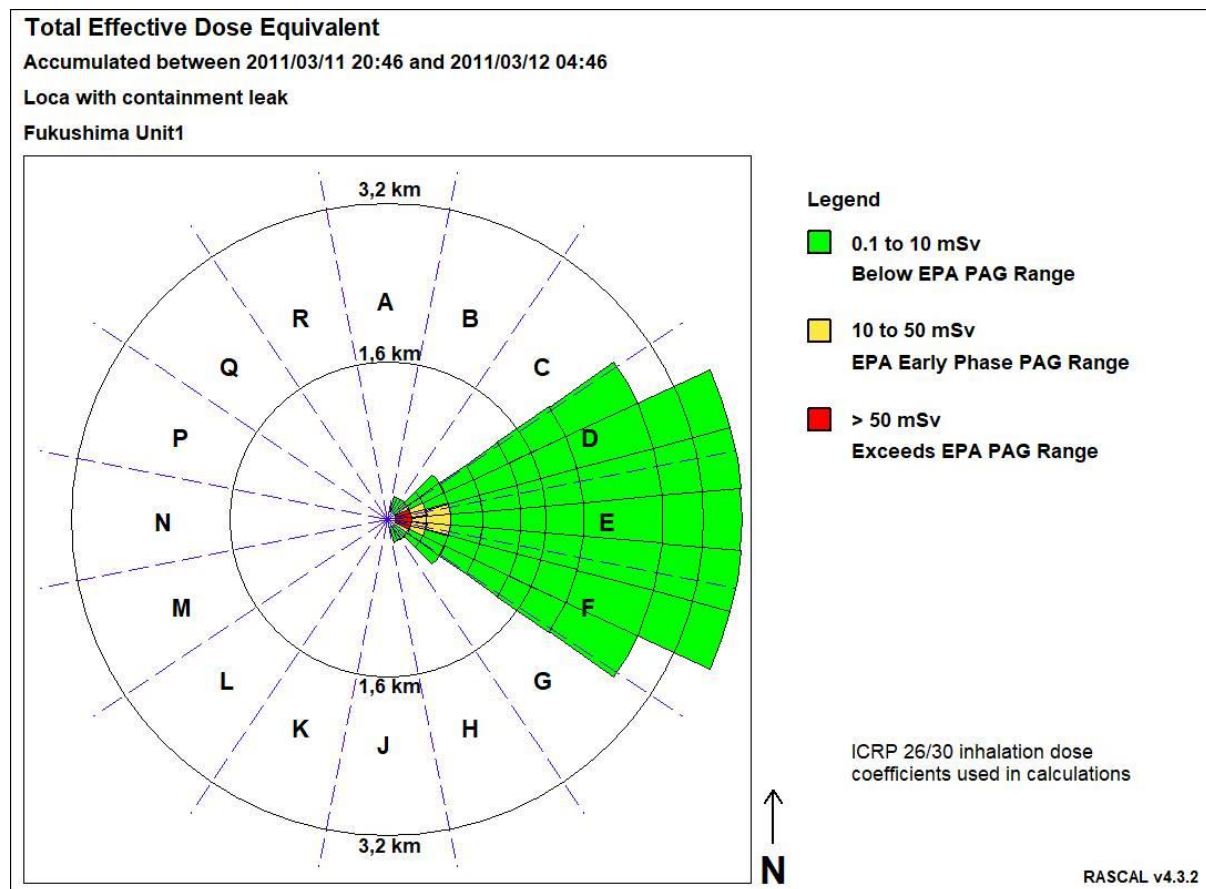


Figure 35: TEDE for BWR at 3.2 km radius

The highest affected sectors were D to F which were exposed to a total effective dose equivalent greater than 50 mSv at the distance of 0.16 km from the point of release (refer to Figure 35). The areas on the periphery of the plume path were exposed to the dose ranging less than 10 mSv depending on their proximity to the nuclear power station. Figure 36 shows that the overall distance that was impacted extends to 8 km from the power station.

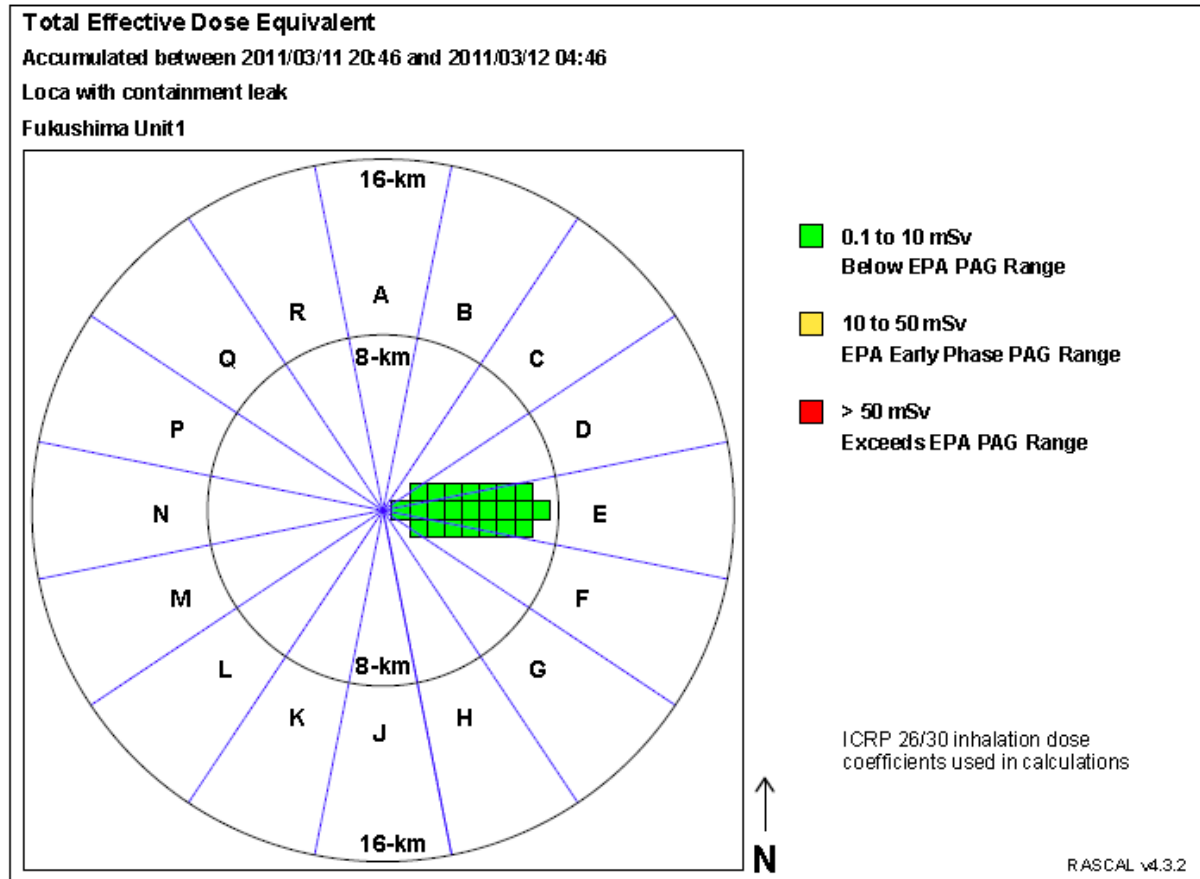


Figure 36: TEDE for BWR for 16.1 km

According to the results above, the dose released during the accident did not require any of the protection action to be implemented, however the protection of worker at the proximity of the power station and good practices were required at all times.

Dose reduction techniques as recommended in the intermediate phase required to be applied as follows;

- scrubbing of surfaces;
- flushing of surfaces;
- removal and disposal of small spots of contaminated surfaces;
- Spending more time indoors and spend more time at low dose areas should be applied as prescribed for intermediate phase.

4.2.2.2 Time-dependent activity results for BWR scenario 2

As discussed in section 4.2.1.2, the importance of nuclides in curie ranking is a function of the burnup as is indicated in Figure 37. The results in Figure 37 show that there was a non-linear relationship between the activity and the decay period of the release, and that the overall trend was that the activity decreases with time. The activity (Ci) of each radionuclide released to the environment is shown for each 15-min period.

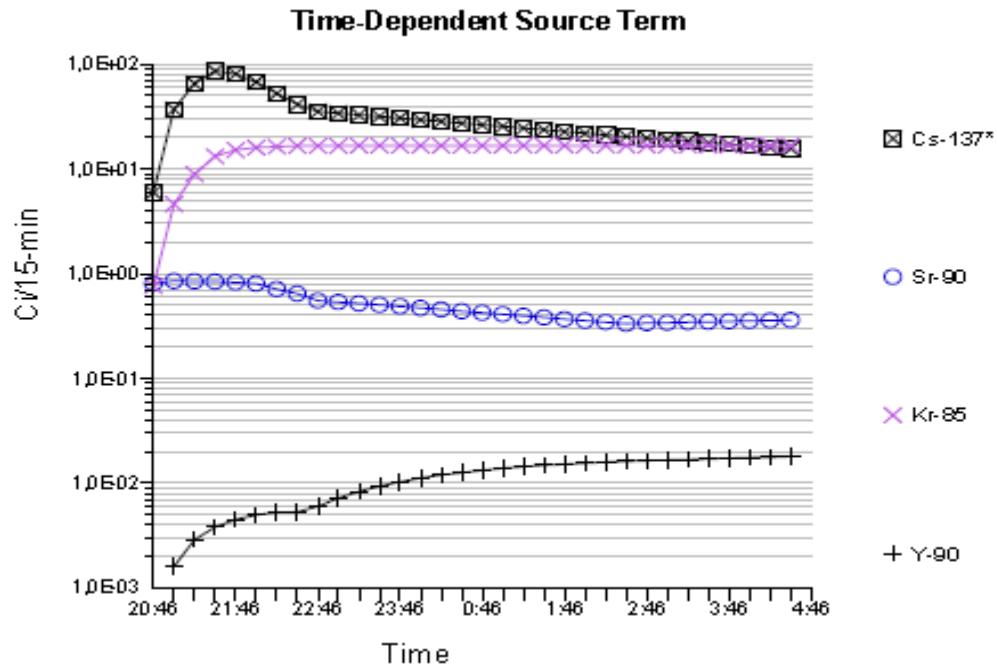


Figure 37: Individual activity as function of time for BWR scenario 2

The following activities (curies) (see Figure 38) were predicted in various structures, systems and components (SSC) of the BWR plant: total activity, noble gasses, Total I activity, ^{85}Kr , ^{135}Xe , ^{131}I , ^{135}I , ^{137}Cs , ^{89}Sr , ^{132}Te , ^{106}Ru , ^{99}Mo and ^{141}Ce . The activity shown in the Rx Core (reactor core), Cont (containment), and Env (environment) columns is the activity at the end of the time period. The activity in the core and containment is adjusted for decay and depletion. The activity in the Env column is the total activity released to the environment, not corrected for decay. The activity shown in the Rx-Cont and Cont-Env columns is the activity transfer during the period [11].

Time (s)	Rx Core	Rx-Cont	Cont	Cont-Env	Env
-21600	2.72E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0	2.25E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
900	2.22E+09	1.80E+07	1.46E+07	3.12E+02	3.12E+02
1800	2.13E+09	8.34E+07	8.03E+07	1.87E+03	2.18E+03
2700	2.01E+09	1.10E+08	1.56E+08	3.59E+03	5.77E+03
3600	1.89E+09	1.09E+08	2.19E+08	5.28E+03	1.11E+04
4500	1.83E+09	5.48E+07	2.28E+08	5.95E+03	1.70E+04
5400	1.79E+09	2.85E+07	2.17E+08	6.22E+03	2.32E+04
6300	1.77E+09	9.99E+06	1.95E+08	6.23E+03	2.95E+04
7200	1.76E+09	9.95E+06	1.79E+08	6.25E+03	3.57E+04
8100	1.75E+09	1.61E+05	1.68E+08	6.17E+03	4.19E+04
9000	1.74E+09	1.60E+05	1.65E+08	6.10E+03	4.80E+04
9900	1.74E+09	1.59E+05	1.61E+08	6.05E+03	5.40E+04
10800	1.73E+09	1.58E+05	1.58E+08	5.99E+03	6.00E+04
11700	1.72E+09	2.78E+04	1.55E+08	5.94E+03	6.60E+04
12600	1.72E+09	2.77E+04	1.52E+08	5.89E+03	7.19E+04
13500	1.71E+09	2.75E+04	1.49E+08	5.85E+03	7.77E+04
14400	1.71E+09	2.74E+04	1.47E+08	5.80E+03	8.35E+04
15300	1.70E+09	2.72E+04	1.44E+08	5.76E+03	8.93E+04
16200	1.69E+09	2.71E+04	1.41E+08	5.72E+03	9.50E+04
17100	1.69E+09	2.69E+04	1.39E+08	5.68E+03	1.01E+05
18000	1.68E+09	2.68E+04	1.37E+08	5.63E+03	1.06E+05
18900	1.68E+09	2.66E+04	1.34E+08	5.60E+03	1.12E+05
19800	1.67E+09	2.65E+04	1.32E+08	5.56E+03	1.17E+05
20700	1.67E+09	2.63E+04	1.30E+08	5.52E+03	1.23E+05
21600	1.66E+09	2.62E+04	1.28E+08	5.48E+03	1.28E+05
22500	1.66E+09	1.60E+05	1.26E+08	5.45E+03	1.34E+05
23400	1.65E+09	1.59E+05	1.25E+08	5.42E+03	1.39E+05
24300	1.65E+09	1.58E+05	1.23E+08	5.38E+03	1.45E+05
25200	1.64E+09	1.58E+05	1.21E+08	5.35E+03	1.50E+05
26100	1.64E+09	1.57E+05	1.20E+08	5.32E+03	1.55E+05
27000	1.63E+09	1.57E+05	1.18E+08	5.29E+03	1.61E+05
27900	1.63E+09	1.56E+05	1.17E+08	5.26E+03	1.66E+05
28800	1.62E+09	1.56E+05	1.15E+08	5.23E+03	1.71E+05

Figure 38: Total activity (Ci) balance among various SCCs of the BWR plant

Chapter 5: Conclusion

The aim of this thesis was to establish that SBO event would lead to LOCA, which will ultimately result in the release of radioactive nuclides to the public and the environment. This question emanated from Fukushima disaster report on the existence of LOCA through the primary pump seals during the accident on Unit 1. The Fukushima BWR unit 1 plant accident was analysed in chapter 3 and the relative SBO scenario in the context of Koeberg PWR plant was examined in chapter 4. The main findings were the following:

- a) PWR simulation showed that it will take 460 seconds after the reactor trips before the pump seals start to fail during SBO event. The comparison of number 1 seal leakage flow rate made between a PWR plant, BWR Fukushima Unit 1, Westinghouse and ETEC case studies as shown Figure 30 indicated that the SBO will result in seal primary leakage flow of about 40 m³/h. The seal leakage flow of this magnitude can result into a category 1 LOCA (small break) as explained in section 2.2 Table 1 of this report. If the leak continues unabated without any make-up to the reactor coolant system, the core will be uncovered as it was the case in Fukushima unit 1. Therefore it can be concluded from the results that seal damage and leakage will be the consequence of the SBO event.
- b) Judging from the results in section 4.2 (Figure 31 and Figure 35), under accident conditions when comparing the PWR and BWR plant design the amount of radiation exhibited by BWR is lower if compared to PWR plant. The significant high radiation activity observed in PWR is attributed to the high pressure at which the plant is normally operated (15.4 MPa) as opposed to activity released by BWR plant, which is normally operated at lower pressures (7 MPa). The results of dose at distance of 0.16 km closest to the reactor in BWR scenario (Table 12) indicates a total EDE of 0.19 mSv, which is lower than the early phase PAGs limit of 10 mSv. Although the dose is lower than what was reported in the Fukushima report [5], which was 1.2 mSv, detected in front of the reactor building by plant radiation monitoring instruments (discussed in section 2.5.3); it is clear the spread of contamination did occur as result of SBO event. At the time when dose rates were detected the reactor core was still intact, the fuel was still covered with water, however the level in the reactor was still decreasing due to the leaking seals.

The results show that the SBO event remains the main contributor to core damage frequency as it was discussed in section 2.1, therefore the following are best industry practices drawn from the results and can inform the utilities when selecting the type of reactor technologies for the new nuclear plant builds

- Safeguard systems should be designed to be independent of electrical supply, modification of plants should include passive systems to make-up the RCS in case of LOCA. The protection and engineered safety systems must be designed in such a way that it performs adequate reactivity control; maintain the water inventory in the reactor vessel and remove the residual heat generated by the fuel to mitigate the consequences of the accident.
- The procedures must be written in such a way that it will guide the operators to perform a safe shutdown of the plant and improve the monitoring and control of the reactor level during the severe accidents such as SBO. New methods should be developed to control leakage at the early stage of the incident to counteract the abnormal leakage rates as observed in simulation results in section 4.1.6. New seals should be explored which are composed of robust type of materials such as stainless steel (410 type) with ceramic and high strength carbon graphite with resilience to high temperatures.

The limitations and assumptions made in this research were the following;

- The difference between in dose rates results obtained from the BWR scenario and dose rates reported in Fukushima report highlights the limitation of the Rascal software to provide dose reliable estimates in real-time events. Rascal is designed to be used as an independent assessment tool for dose projections during response to radiological emergencies. The floods at Fukushima damaged the radiation monitoring which made it difficult to monitor the dose rates as the accident evolved. The first dose rate was recorded eight hours after the Tsunami. All assumptions used in Rascal are detailed in Appendix 4 and 5.
- This research focused on Case 2 of the ETEC and Westinghouse analysis, which is assuming that no.1 seal fails open while no.2 and no.3 seal remains intact. The reason for choosing Case 2 leak-rates is that it correlates with initial leak rates experienced at Fukushima Unit 1.
- A conservative assumption was made that the seal will fail immediately after the SBO; this assumption gives the maximum leak rates. The main leak path will be through the no.1 leak-off line with only minimal flow path through the no.2 seal. There will some leakage through the no.2 seal due to machining tolerances, the leakage was estimated to be about 0,06 m³/h. The simulator is modelled in such a way that the primary pump seal leakage is simulated as a cold leg at the pump break discharge. The break size is selected in such a way that the break flow will equal a pre-selected value of seal leakage rate. The analysis of the break flow is dictated by the critical flow correlations for subcooled and saturated water, this is to accommodate for the changing conditions of the primary systems during the analysis. The approach produces a more realistic representation of the response of a primary pump seal leak in particular the degradation due to overheating of the seal. Variations from the actual leak are expected due to over prediction of the leak rates; the deviation is a conservative approach [9]
- Earthquake could not be simulated at Koeberg because the design of the nuclear island and the structures supported by it are seismically designed for the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE) such that the plant equipment and systems are able to:
 1. Maintain the primary system integrity,
 2. Shutdown the reactor and remove the residual heat from the reactor and
 3. Maintain the possible radioactive releases around the site at an acceptable level [15].

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Acronyms

Mwe: Megawatts Electrical
Pn: Norminal Power
AC: Alternate Current
DC: Direct Current
RCIC Reactor Core Isolation
PWR: Pressurised Water Reactor
BWR: Boiling Water Reactor
NPP: Nuclear Power Plant
SBO: Station Blackout
SRV: Safety Relief Valve
RPV: Reactor Pressure Vessel
NWL: Normal Working Level
ECCS: Emergency Core Cooling System
BDBA: Beyond Design Base Accident
ATWS: Anticipated Trip Without Scram
HPCI: High Pressure Coolant injection
CST: Condensate Storage Tank
HPCS: High Pressure Core Spray
IC: Isolation Condenser
RCP: Reactor Coolant System
VCT: Volume Control Tank
ETEC: Energy Technology Engineering Centre
EAB: Exclusion Area Boundry
LPZ: Low Population Zone
ADS: Automatic Depressurisation System
PCV: Primary Containment Vessel
CRD: Control Rod Drive
TAF: Top of Active Fuel
LOCA: Loss of Coolant Accident

Appendix 1: Fukushima sequence of events [5]

Date of unit at full power	2010/10/05	
Date of the accident	2011/03/11	
No. of DAYS AT POWER	152	
Time of reactor shutdown	14:46	
Time of subcriticality	15:02	
Time of the SBO	15:27	
Event	Date & Time	Discussion
Earthquake	2011/03/11 14:46	9.0 on Richter scale
Loss of off-site power		damaged breakers and power lines
Reactor shutdown		Auto trip on seismic Rx protection
	2011/03/11 14:47	first level decrease to no load value but level remain within control band
Turbine manual trip		
MSIV closed automatically		
Emergency Diesels start		
Isolation Condenser (IC)	2011/03/11 14:52	auto initiated by increasing pressure in the RPV
Rx subcritical	2011/03/11 15:02	Rx confirmed subcritical
	2011/03/11 15:03	feed from isolation condenser isolated due high cooldown >55 °C/h
		RPV controlled between 6 to 7 Mpa by opening and closing MO-3A, operated 3x before Tsunami
Tsunami first wave	2011/03/11 15:27	14 to 15m (design basis = 5,7m)
Tsunami second wave	2011/03/11 15:35	
Loss of all AC power Unit1	2011/03/11 15:37	Loss of AC and DC
	2011/03/11 15:39	
	2011/03/11 16:56	level at +2130mm and decreasing
	2011/03/11 18:10	level at 0mm Top of active fuel
DC board partially energised	2011/03/11 18:18	
heat sink	2011/03/11 18:18	Condenser placed in service, one MSIV(MO-2A) and Feedwater valve (MO-3A) opened
loss of feedwater	2011/03/11 18:18	MO-3A FW valve closed
	2011/03/11 20:07	Reactor pressure 6,9MPa
	2011/03/11 21:19	Water level 200 mm above fuel
	2011/03/11 21:30	MO-3A FW valve opened
	2011/03/11 22:00	Reactor water level indicate 550mm above the top of active fuel
	2011/03/11 23:00	dose rates of 120 mrem/hr (1,200 µSv/hr) in front of the reactor building north door on
	2011/03/11 23:50	containment pressure at 600KPa
	2011/03/12 02:24	30 rem/hr (300 mSv/hr) intorus room
	2011/03/12 02:30	containment pressure at 840KPa
	2011/03/12 02:45	Reactor pressure 800 Kpa
	2011/03/12 04:00	0.0069 mrem/hr (0.069 µSv/hr) at the main gate
	2011/03/12 04:19	CTM pressure unexpectedly decreased and stabilized near (0.78 MPa) without venting.
	2011/03/12 04:23	0.059 mrem/hr (0.59 µSv/hr) at the main gate
	2011/03/12 04:50	0.1 mrem/hr (1 µSv/hr) is measured at the site boundary
	2011/03/12 05:46	freshwater injection initiated via core spray system
	2011/03/12 07:55	3 tons of fresh water added, level indicate low
	2011/03/12 08:15	4 tons of fresh water added
	2011/03/12 08:30	5 tons of fresh water added
	2011/03/12 09:15	Containmemnt venting initiated
	2011/03/12 09:40	21 tons of fresh water added
	2011/03/12 11:15	Ctm pressure at 750KPa
	2011/03/12 12:05	Rx water level 1500mm below active fuel
	2011/03/12 14:50	ctm pressure at 580KPa
	2011/03/12 14:53	total of 80 tons of fresh water injected into the reactor
	2011/03/13 15:36	Hydrogen expollosion occurred.

Appendix 2: BWR reactor level programme

BWR level parameters	meters				
Flange level	18,3642				
Main steam nozzle	16,256				
Level #6	14,5542				
Normal level	14,0716				
Level #4	13,97				
Level #3	13,4366				
Instrument "0"	13,1318				
Core Spray nozzle	12,2936				
feedwater nozzle	12,2682				
Level #2	12,1666		11,2232	1:19:00	
Level #1	9,7536				
Top of Active Fuel	9,0932	0	9,0932	2:51:00	
2/3 core covered	7,6962			20:38:00	
Bottom of Active Fuel	5,2832				
Jet pump Instrument tap	3,3528				
SCL Nozzle and Instrument tap	2,3876				
Inside bottom head	0				
Fukushima leak rate estimations					
Time [h:m:s]	1:19:00	2:51:00	5:52:00	6:33:00	20:38:00
Level control (m)	11,2232	9,0932	9,29	9,50	7,6962
Volume of the reactor (m³)	203,0902426	164,5467	168,1079	171,908	139,2672
Volumetric flow rate (m³/h)	39,14691044	25,13711	-1,18051	-5,56108	-8,33382
	1,32	2,85	5,87	6,55	2,63
		1,53	3,02	0,68	-3,92
	-51,54343208	-90,087	-86,5258	-82,7257	-115,367

Appendix 3: ACTIVITY AVAILABLE (NUCLIDE INVENTORY) IN THE REACTOR CORE BASED ON ²³⁵U FUEL

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
ag109	0	0	0.00E+00	0.00E+00	0.00E+00
ag109m	2.325E-23	2.169E-23	2.00E-23	1.98E-23	2.35E-23
ag110	3.972E-22	3.705E-22	3.42E-22	3.39E-22	4.02E-22
ag111	1.54E-25		2.93E-08	1.34E-25	1.60E-25
ag111m	1.703E-20	1.596E-20	1.48E-20	1.47E-20	1.75E-20
ag112		6.644E-21	6.59E-21		
ag113	2.736E-21	2.715E-21	2.69E-21	2.69E-21	3.35E-21
ag113m	6.179E-18	6.07E-18	5.95E-18	5.94E-18	7.34E-18
ag114	1.406E-14	1.422E-14	1.44E-14	1.44E-14	1.81E-14
ag115	1.465E-17	1.461E-17	1.46E-17	1.46E-17	1.82E-17
ag116	2.131E-15	2.152E-15	2.17E-15	1.48E-15	1.85E-15
ag117	2.046E-15	2.035E-15	2.02E-15	2.04E-15	2.56E-15
ag117m	2.814E-14	2.799E-14	2.79E-14	2.79E-14	3.49E-14
ag118	9.718E-14	9.6E-14	9.48E-14	9.46E-14	1.18E-13
ag118m	1.681E-13	1.657E-13	1.63E-13	1.63E-13	2.02E-13
ag119	4.327E-13	4.242E-13	4.15E-13	4.14E-13	5.13E-13
ag120	2.337E-13	2.205E-13	2.07E-13	2.05E-13	2.47E-13
ag120m	7.185E-13	6.78E-13	6.37E-13	7.89E-13	9.49E-13
ag121	8.258E-13	7.891E-13	7.51E-13	7.56E-13	9.15E-13
ag122	3.862E-13	3.641E-13			3.68E-13
ag123	6.38E-13				
am241	1.383E-12	1.383E-12	1.38E-12	1.38E-12	1.38E-12
am242m	0	4.242E-12	4.24E-12	0.00E+00	0.00E+00
am242			3.27E-07		3.27E-07
am243	8.116E-14		8.12E-14	8.12E-14	8.12E-14
am244m					9.61E-29
as78	1.803E-18	1.823E-18	1.84E-18	1.85E-18	2.33E-18
as79	3.576E-16	3.617E-16	3.66E-16	3.66E-16	4.61E-16
as80	6.567E-14	6.643E-14	6.72E-14	6.73E-14	8.48E-14
as81	1.337E-13	1.352E-13	1.37E-13	1.37E-13	1.72E-13
as82	4.882E-13	4.932E-13	4.98E-13	4.99E-13	6.27E-13
as82m	1.688E-13	1.694E-13	1.70E-13	1.70E-13	2.14E-13
as83	1.673E-12	1.684E-12	1.70E-12	1.70E-12	2.13E-12
as84	5.219E-12	5.218E-12	5.22E-12	4.03E-12	5.04E-12
as85	5.737E-12	5.699E-12	5.66E-12	5.66E-12	7.05E-12
as86	3.422E-12	3.316E-12	3.21E-12	3.20E-12	3.92E-12
as87	6.987E-12	7.019E-12	7.05E-12	7.05E-12	8.82E-12
as88	7.794E-11	7.882E-11	7.96E-11	7.98E-11	1.00E-10
ba136m				6.07E-16	7.62E-16

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
ba137	0		6.43E-17	0.00E+00	0.00E+00
ba137m	6.308E-17	6.371E-17		6.44E-17	8.08E-17
ba139	1.003E-15	1.012E-15	1.02E-15	1.02E-15	1.28E-15
ba140	1.713E-08	1.713E-08	1.71E-08	1.71E-08	1.71E-08
ba141	1.077E-13	1.09E-13	1.10E-13	1.11E-13	1.39E-13
ba142	3.403E-13	3.443E-13	3.48E-13	3.49E-13	4.39E-13
ba143	2.073E-11	2.094E-11	2.11E-11	2.12E-11	2.66E-11
ba144	2.583E-11	2.605E-11	2.63E-11	2.63E-11	3.30E-11
ba145	3.509E-11	3.521E-11	3.53E-11	3.54E-11	4.43E-11
ba146	3.58E-11	3.575E-11	3.57E-11	3.57E-11	4.46E-11
ba147	2.625E-11	2.603E-11	2.58E-11	2.58E-11	3.20E-11
ba148	5.681E-12	5.504E-12	5.33E-12	5.31E-12	6.49E-12
ba149	1.265E-12	1.196E-12	1.13E-12	1.12E-12	1.34E-12
br82m					4.45E-18
br83	1.575E-16	1.594E-16	1.61E-16	1.62E-16	2.03E-16
br84	6.868E-16	6.952E-16	7.03E-16	7.05E-16	8.86E-16
br84m	3.392E-15	3.43E-15	3.47E-15	3.47E-15	4.37E-15
br85	9.663E-14	9.778E-14	9.89E-14	9.91E-14	1.25E-13
br86	6.145E-13	6.206E-13	6.27E-13	6.29E-13	7.91E-13
br87	1.7E-12	1.717E-12	1.73E-12	1.74E-12	2.18E-12
br88	6.427E-12	6.484E-12	6.54E-12	6.55E-12	8.23E-12
br89	1.953E-11	1.957E-11	1.96E-11	1.96E-11	2.46E-11
br90	2.591E-11	2.582E-11	2.57E-11	2.56E-11	3.19E-11
br91	3.985E-11	3.947E-11	3.91E-11	3.90E-11	4.84E-11
br92	1.081E-11	1.051E-11	1.02E-11	1.02E-11	1.25E-11
br93	6.311E-12	6.163E-12	6.05E-12	6.04E-12	7.60E-12
cd115	3.476E-22			3.54E-22	
cd117	9.022E-19	9.13E-19	9.23E-19	9.25E-19	1.17E-18
cd117m				2.33E-18	2.93E-18
cd118	4.421E-17	4.461E-17	4.50E-17	4.51E-17	5.69E-17
cd119	1.214E-15	1.223E-15	1.23E-15	1.23E-15	1.56E-15
cd119m	1.381E-15	1.391E-15	1.40E-15	1.40E-15	1.77E-15
cd120	1.466E-14	1.467E-14	1.47E-14	1.47E-14	1.85E-14
cd121	3.142E-14	3.097E-14	3.05E-14	3.05E-14	3.79E-14
cd121m	5.111E-14	5.037E-14	4.96E-14	4.96E-14	6.17E-14
cd122	2.603E-13	2.562E-13	2.52E-13	2.52E-13	3.12E-13
cd123	6.314E-13	6.166E-13	6.01E-13	5.99E-13	7.38E-13
cd124	1.12E-12	1.099E-12	1.08E-12	1.07E-12	1.33E-12
cd125	1.153E-12	1.123E-12	1.09E-12	1.04E-12	1.27E-12
cd126	1.393E-12	1.386E-12	1.38E-12	1.38E-12	1.71E-12
cd127	1.69E-12	1.698E-12	1.70E-12	1.71E-12	2.13E-12

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
cd128	9.518E-13		9.63E-13	9.63E-13	
cd130	3.749E-11	3.796E-11	3.84E-11	3.85E-11	4.83E-11
cd131	1.401E-11	1.419E-11	1.44E-11	1.44E-11	1.81E-11
ce141	6.72E-09	6.72E-09	6.72E-09	6.72E-09	6.72E-09
ce143	1.587E-07	1.587E-07	1.59E-07	1.59E-07	1.59E-07
ce144	7.668E-10	7.668E-10	7.67E-10	7.67E-10	7.67E-10
ce145	3.398E-14	3.44E-14	3.48E-14	3.49E-14	4.39E-14
ce146	5.111E-14	5.174E-14	5.23E-14	5.24E-14	6.60E-14
ce147	1.276E-12	1.29E-12	1.30E-12	1.31E-12	1.64E-12
ce148	1.651E-12	1.666E-12	1.68E-12	1.68E-12	2.11E-12
ce149	1.052E-11	1.056E-11	1.06E-11	1.06E-11	1.33E-11
ce150	8.568E-12	8.55E-12	8.53E-12	8.53E-12	1.07E-11
ce151	1.034E-11	1.02E-11	1.01E-11	5.82E-12	7.21E-12
ce152	2.061E-12	2.006E-12	1.95E-12	1.95E-12	2.39E-12
ce153	5.62E-13	5.337E-13	5.06E-13	5.02E-13	6.04E-13
cm242	1.342E-09	1.342E-09	1.34E-09	1.34E-09	1.34E-09
cm243	2.055E-11	2.055E-11	2.06E-11	2.06E-11	2.06E-11
cm244	3.305E-11	3.305E-11	3.31E-11	3.30E-11	3.30E-11
cm245	0	7.037E-14	7.04E-14	0.00E+00	0.00E+00
cs133	0	0	0.00E+00	0.00E+00	0.00E+00
cs134	2.896E-10	2.896E-10	2.90E-10	2.90E-10	2.90E-10
cs134m	3.327E-20	3.311E-20	3.29E-20	3.29E-20	4.09E-20
cs135m		5.777E-18	5.82E-18		7.31E-18
cs136	1.827E-19		1.66E-08	1.87E-19	2.37E-19
cs136m	1.093E-14	1.106E-14	1.12E-14	1.12E-14	1.42E-14
cs137	1.989E-11	1.989E-11	1.99E-11	1.99E-11	1.99E-11
cs138	8.776E-15	8.907E-15	9.04E-15	9.07E-15	1.15E-14
cs138m	9.102E-14	9.208E-14	9.31E-14	9.32E-14	1.17E-13
cs139	1.717E-13	1.738E-13	1.76E-13	1.76E-13	2.22E-13
cs140	2.407E-12	2.433E-12	2.46E-12	2.46E-12	3.11E-12
cs141	8.871E-12	8.947E-12	9.02E-12	9.04E-12	1.14E-11
cs142	1.057E-10	1.063E-10	1.07E-10	1.07E-10	1.34E-10
cs143	6.536E-11	6.539E-11	6.54E-11	6.54E-11	8.17E-11
cs144	4.307E-11	4.254E-11	4.20E-11	4.19E-11	5.19E-11
cs145	1.628E-11	1.592E-11	1.56E-11	1.57E-11	1.93E-11
cs146	4.533E-12	4.362E-12	4.20E-12	4.17E-12	5.08E-12
cs147	1.155E-12	1.131E-12	1.11E-12	1.13E-12	1.39E-12
eu153	0	0	0.00E+00	0.00E+00	0.00E+00
eu154		6.961E-11			6.95E-11
eu155		1.258E-10	1.26E-10	1.26E-10	1.26E-10
eu156	1.438E-08	1.438E-08	1.44E-08	1.44E-08	1.44E-08

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
eu157		3.454E-07	3.45E-07		
eu158	3.841E-18	3.848E-18	3.85E-18	3.86E-18	4.83E-18
eu159	1.356E-17	1.349E-17	1.34E-17	1.34E-17	1.67E-17
eu160		3.052E-16			3.67E-16
ga76		2.589E-15	2.62E-15	2.63E-15	
ga77	2.412E-14	2.444E-14	2.48E-14	2.49E-14	3.16E-14
ga78	1.577E-13	1.592E-13	1.61E-13	1.61E-13	2.03E-13
ga79	4.88E-13	4.903E-13	4.93E-13	4.93E-13	6.20E-13
ga80	5.815E-13	5.831E-13	5.85E-13	5.85E-13	7.34E-13
ga81	5.821E-13	5.814E-13	5.81E-13	5.81E-13	7.25E-13
ga82	8.816E-13	8.805E-13	8.79E-13	8.79E-13	1.10E-12
ga84	9.085E-12	9.193E-12	9.29E-12	9.31E-12	1.17E-11
gd155	0	0	0.00E+00	0.00E+00	0.00E+00
gd159		4.605E-21			
ge77	1.15E-18	1.168E-18	1.19E-18	1.19E-18	1.52E-18
ge78	9.835E-17	9.977E-17	1.01E-16	1.02E-16	1.29E-16
ge79	4.669E-14	4.719E-14	4.77E-14	4.78E-14	6.04E-14
ge79m	2.272E-14	2.297E-14	2.32E-14	2.33E-14	2.94E-14
ge80	2.512E-13	2.542E-13	2.57E-13	2.58E-13	3.25E-13
ge81	1.237E-12	1.249E-12	1.26E-12	1.26E-12	1.59E-12
ge81m				0.00E+00	0.00E+00
ge82	2.177E-12	2.19E-12	2.20E-12	2.20E-12	2.76E-12
ge83	2.333E-12	2.326E-12	2.32E-12	2.32E-12	2.90E-12
ge84	2.123E-12	2.096E-12	2.07E-12	2.05E-12	2.54E-12
ge85	6.982E-13	6.741E-13	6.51E-13	6.47E-13	7.90E-13
ge86	5.12E-10	5.184E-10	5.24E-10	4.70E-10	5.90E-10
i128		5.295E-20	5.32E-20	5.32E-20	6.65E-20
i130	2.544E-19	4.242E-07	4.24E-07	2.59E-19	3.24E-19
i130m	8.936E-18	9.029E-18	9.11E-18	9.13E-18	1.15E-17
i131	4.047E-19		2.72E-08	4.13E-19	5.19E-19
i132	9.431E-17	9.416E-17	9.39E-17	9.39E-17	1.17E-16
i132m	1.56E-16	1.558E-16	1.55E-16	1.55E-16	1.94E-16
i133	8.992E-17	9.041E-17	9.09E-17	9.11E-17	
i133m	7.481E-13	7.522E-13	7.57E-13	7.58E-13	9.54E-13
i134	1.234E-14	1.243E-14	1.25E-14	1.25E-14	1.58E-14
i134m	1.35E-13	1.357E-13	1.36E-13	1.37E-13	1.71E-13
i135	7.98E-07	7.98E-07	7.98E-07	7.98E-07	7.98E-07
i136	1.244E-12	1.251E-12	1.26E-12	1.26E-12	1.58E-12
i136m	2.107E-12	2.118E-12	2.13E-12	2.13E-12	2.67E-12
i137	8.628E-12	8.655E-12	8.68E-12	8.68E-12	1.09E-11
i138	1.996E-11	1.991E-11	1.99E-11	1.99E-11	2.48E-11

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
i139	3.055E-11	3.039E-11	3.02E-11	3.02E-11	3.76E-11
i140	1.892E-11	1.854E-11	1.82E-11	1.81E-11	2.23E-11
i141	1.106E-11	1.084E-11	1.06E-11	1.06E-11	1.30E-11
i142	3.638E-12	3.554E-12	3.47E-12	3.12E-12	3.82E-12
in115m	2.211E-25		2.18E-25	2.18E-25	
in116m		1.444E-22			1.79E-22
in117	1.102E-20	1.11E-20	1.12E-20	1.12E-20	1.41E-20
in117m	9.173E-22	9.267E-22	9.36E-22	9.37E-22	1.18E-21
in118	7.923E-17	7.988E-17	8.05E-17	8.06E-17	1.02E-16
in119	2.301E-16	2.329E-16	2.36E-16	2.36E-16	2.97E-16
in119m	1.089E-18	1.103E-18	1.12E-18	1.12E-18	1.42E-18
in120	5.15E-15	5.201E-15	5.25E-15	5.26E-15	6.63E-15
in121	8.242E-15	8.311E-15	8.38E-15	8.39E-15	1.06E-14
in121m	1.096E-16	1.106E-16	1.12E-16	1.12E-16	1.41E-16
in122	5.689E-14	5.658E-14	5.63E-14	5.63E-14	7.07E-14
in122m	1.657E-14	1.648E-14	1.64E-14	1.64E-14	2.06E-14
in123	7.708E-14	7.595E-14	7.49E-14	7.48E-14	9.33E-14
in123m	1.052E-15	1.025E-15	1.00E-15	9.99E-16	1.25E-15
in124	1.141E-13	1.103E-13	1.06E-13	1.06E-13	1.30E-13
in124m	9.594E-14	9.269E-14	8.95E-14	8.91E-14	1.10E-13
in125	2.936E-13	2.862E-13	2.79E-13	2.78E-13	3.43E-13
in125m	5.697E-14	5.554E-14	5.41E-14	5.39E-14	6.66E-14
in126	3.603E-13	3.436E-13	3.27E-13	3.25E-13	3.95E-13
in126m	3.362E-13	3.206E-13	3.05E-13	3.03E-13	3.68E-13
in127	3.105E-12	3.109E-12	3.11E-12	3.11E-12	3.89E-12
in127m	2.643E-13	2.574E-13	2.51E-13	2.50E-13	3.09E-13
in128	1.688E-12	1.665E-12	1.64E-12	1.64E-12	2.03E-12
in128m	1.969E-12	1.942E-12	1.91E-12	1.91E-12	2.37E-12
in129	4.143E-12	4.117E-12	4.09E-12	4.09E-12	5.08E-12
in129m	1.931E-12	1.916E-12	1.90E-12	1.90E-12	2.36E-12
in130	1.634E-12	1.601E-12	1.57E-12	1.57E-12	1.96E-12
in130m	1.755E-12	1.719E-12	1.69E-12	1.69E-12	2.10E-12
in131	1.859E-12	1.855E-12	1.85E-12	1.85E-12	2.33E-12
in131m	1.487E-12	1.484E-12	1.48E-12	1.48E-12	1.86E-12
in132	4.445E-12	4.309E-12	4.17E-12	4.15E-12	5.07E-12
kr83m	1.675E-20	1.694E-20	1.71E-20	1.72E-20	2.16E-20
kr85	5.222E-21	5.561E-11	5.56E-11	5.36E-21	6.73E-21
kr85m	2.533E-17	2.565E-17	2.59E-17	2.60E-17	3.26E-17
kr87	7.129E-15	7.223E-15	7.31E-15	7.33E-15	9.22E-15
kr88	1.212E-14	1.226E-14	1.24E-14	1.24E-14	1.56E-14
kr89	1.299E-12	1.314E-12	1.33E-12	1.33E-12	1.68E-12

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
kr90	9.973E-12	1.008E-11	1.02E-11	1.02E-11	1.28E-11
kr91	2.818E-11	2.838E-11	2.86E-11	2.86E-11	3.59E-11
kr92	7.365E-11	7.378E-11	7.39E-11	7.39E-11	9.24E-11
kr93	3.501E-11	3.477E-11	3.45E-11	3.45E-11	4.29E-11
kr94	5.55E-11	5.403E-11	5.26E-11	5.24E-11	6.42E-11
kr95	1.324E-11	1.272E-11	1.22E-11	1.21E-11	1.48E-11
kr96	3.602E-11	3.621E-11	3.64E-11	3.64E-11	4.55E-11
la140	2.512E-18	1.302E-07	1.30E-07	2.57E-18	3.23E-18
la141	9.157E-17	9.27E-17	9.38E-17	9.39E-17	1.18E-16
la142	1.239E-15	1.254E-15	1.27E-15	1.27E-15	1.60E-15
la143	3.213E-14	3.262E-14	3.31E-14	3.32E-14	4.21E-14
la144	1.897E-12	1.923E-12	1.95E-12	1.95E-12	2.47E-12
la145	5.623E-12	5.68E-12	5.73E-12	5.74E-12	7.21E-12
la146	9.13E-12	9.19E-12	9.25E-12	9.26E-12	1.16E-11
la146m	5.725E-12	5.762E-12	5.80E-12	5.80E-12	7.28E-12
la147	1.426E-11	1.424E-11	1.42E-11	1.41E-11	1.76E-11
la148	2.56E-11	2.54E-11	2.52E-11	2.52E-11	3.13E-11
la149	1.03E-11	1.004E-11	9.78E-12	9.75E-12	1.20E-11
la150	4.703E-12	4.499E-12	4.30E-12	2.53E-12	3.07E-12
la151	5.3E-13	5.013E-13	4.73E-13	4.69E-13	5.63E-13
mo101	1.486E-14	1.505E-14	1.52E-14	1.53E-14	1.92E-14
mo102	6.791E-14	6.874E-14	6.95E-14	6.96E-14	8.76E-14
mo103	1.119E-12	1.131E-12	1.14E-12	1.14E-12	1.44E-12
mo104	1.464E-12	1.473E-12	1.48E-12	1.48E-12	1.86E-12
mo105	1.771E-12	1.759E-12	1.75E-12	1.75E-12	2.17E-12
mo106	4.79E-12	4.711E-12	4.63E-12	4.45E-12	5.50E-12
mo107	5.287E-12	5.132E-12	4.98E-12	4.96E-12	6.08E-12
mo108	5.798E-12	5.572E-12	5.35E-12	5.32E-12	6.48E-12
mo109	4.799E-12	4.648E-12	4.50E-12	4.48E-12	5.49E-12
mo110	2.985E-12	2.867E-12	2.75E-12	2.46E-12	3.00E-12
mo95	0	0	0.00E+00	0.00E+00	0.00E+00
mo99	7.951E-08	7.951E-08	7.95E-08	7.95E-08	7.95E-08
n16	2.948E-14	2.571E-14			
nb100	1.543E-11	1.56E-11	1.58E-11	1.58E-11	1.98E-11
nb100m	7.742E-12	7.825E-12	7.90E-12	7.92E-12	9.95E-12
nb101	1.968E-11	1.988E-11	2.01E-11	2.01E-11	2.53E-11
nb102	4.808E-11	4.831E-11	4.85E-11	1.47E-11	1.84E-11
nb102m	1.454E-11	1.461E-11	1.47E-11	4.86E-11	6.09E-11
nb103	8.274E-11	8.256E-11	8.24E-11	8.24E-11	1.03E-10
nb104	6.43E-12	6.328E-12	6.23E-12	6.21E-12	7.68E-12
nb104m	3.352E-11	3.299E-11	3.25E-11	3.24E-11	4.01E-11

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
nb105	8.182E-12	7.899E-12	7.62E-12	7.58E-12	9.25E-12
nb106	5.897E-12	5.584E-12	5.28E-12	5.74E-12	6.89E-12
nb107	4.21E-12	3.964E-12	3.72E-12	4.06E-12	4.85E-12
nb108	8.756E-13	6.243E-09			8.96E-13
nb95	6.243E-09	5.577E-21	6.24E-09	6.24E-09	6.24E-09
nb95m	5.508E-21	4.567E-19	5.64E-21	5.65E-21	7.09E-21
nb97	1.733E-16	1.755E-16	1.78E-16	1.78E-16	
nb97m	3.23E-15	3.271E-15	3.31E-15	2.98E-15	
nb98	2.814E-12	2.849E-12	2.88E-12	2.89E-12	3.63E-12
nb98m	8.702E-16	8.811E-16	8.91E-16	8.93E-16	1.12E-15
nb99	1.407E-13	1.424E-13	1.44E-13	1.44E-13	1.81E-13
nb99m	1.831E-13	1.853E-13	1.87E-13	1.95E-13	2.45E-13
nd143	0	0	0.00E+00	0.00E+00	0.00E+00
nd145	0	0	0.00E+00	0.00E+00	0.00E+00
nd147	1.99E-08	1.99E-08	1.99E-08	1.99E-08	1.99E-08
nd149	7.755E-17	7.848E-17	7.93E-17	7.95E-17	9.99E-17
nd151	7.677E-15	7.762E-15	7.84E-15	7.86E-15	9.87E-15
nd152	1.513E-14	1.528E-14	1.54E-14	1.54E-14	1.94E-14
nd153	2.756E-13	2.77E-13	2.78E-13	2.79E-13	3.49E-13
nd154	1.932E-13	1.929E-13	1.93E-13	1.93E-13	2.40E-13
nd155	2.227E-13	2.19E-13	2.15E-13	2.15E-13	2.66E-13
nd156	1.3E-13	1.261E-13	1.22E-13	1.22E-13	1.49E-13
nd157	8.721E-14	8.277E-14			9.37E-14
np238	0	1.032E-07	1.03E-07	0.00E+00	0.00E+00
np239	0	9.273E-08	9.27E-08	0.00E+00	0.00E+00
np240	0				
np240m	0	0	0.00E+00		
o16	0	0	0.00E+00	0.00E+00	0.00E+00
pd109	0	4.049E-23	3.74E-23		
pd111	2.714E-19	2.638E-19	2.56E-19	2.55E-19	3.13E-19
pd112	0	1.97E-19	1.93E-19		
pd113	1.152E-15	1.147E-15	1.14E-15	1.14E-15	1.43E-15
pd114	2.524E-15	2.524E-15	2.52E-15	2.53E-15	3.17E-15
pd115	2.852E-14	2.831E-14	2.81E-14	2.81E-14	3.50E-14
pd116	6.998E-14	6.886E-14	6.77E-14	6.76E-14	8.41E-14
pd117	2.457E-13	2.414E-13	2.37E-13	2.37E-13	2.93E-13
pd118	4.196E-13	4.018E-13	3.84E-13	3.81E-13	4.64E-13
pd119	4.37E-13	4.108E-13	3.84E-13	3.81E-13	4.56E-13
pd120	7.859E-13	7.631E-13	7.39E-13	7.36E-13	9.00E-13
pm147	2.28E-10	2.28E-10	2.28E-10	2.28E-10	2.28E-10
pm148	4.07E-08	4.07E-08	4.07E-08	4.07E-08	4.07E-08

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
pm148m	3.103E-25	5.291E-09	5.29E-09	2.96E-25	3.67E-25
pm149	9.878E-08	9.878E-08	9.88E-08	9.88E-08	9.88E-08
pm150	2.23E-19	2.256E-19	2.28E-19	2.29E-19	
pm151	4.5E-19	1.846E-07		4.60E-19	
pm152	2.722E-16	2.75E-16	2.78E-16	2.78E-16	3.49E-16
pm152m	2.983E-16	3.014E-16	3.04E-16	3.05E-16	3.83E-16
pm153	2.063E-15	2.082E-15	2.10E-15	2.10E-15	2.64E-15
pm154	2.609E-15	2.623E-15	2.64E-15	4.09E-15	5.12E-15
pm154m	4.041E-15	4.063E-15	4.08E-15	2.64E-15	3.31E-15
pm155	2.547E-14	2.548E-14	2.55E-14	2.55E-14	3.19E-14
pm156	2.65E-14	2.622E-14	2.60E-14	2.59E-14	3.22E-14
pm157	3.821E-14	3.725E-14	3.63E-14	3.62E-14	4.45E-14
pm158	3.051E-14	2.931E-14	2.81E-14	2.80E-14	3.40E-14
pr142m	2.066E-22	2.075E-22	2.08E-22	2.08E-22	2.60E-22
pr142		2.742E-07	2.74E-07		
pr143	1.61E-08	1.61E-08	1.61E-08	1.61E-08	1.61E-08
pr144	9.793E-20	9.904E-20	1.00E-19	1.00E-19	1.26E-19
pr144m	2.13E-18	2.152E-18	2.17E-18	2.18E-18	2.73E-18
pr145	1.097E-18	1.11E-18	1.12E-18	1.12E-18	1.41E-18
pr146	1.772E-16	1.793E-16	1.81E-16	1.82E-16	2.28E-16
pr147	3.098E-14	3.137E-14	3.17E-14	3.18E-14	3.99E-14
pr148	2.044E-14	2.066E-14	2.09E-14	2.09E-14	2.63E-14
pr148m	2.329E-14	2.354E-14	2.38E-14	2.38E-14	3.00E-14
pr149	1.578E-13	1.595E-13	1.61E-13	1.61E-13	2.03E-13
pr150	2.739E-12	2.755E-12	2.77E-12	2.77E-12	3.48E-12
pr151	1.009E-12	1.013E-12	1.02E-12	1.02E-12	1.27E-12
pr152	2.634E-12	2.618E-12	2.60E-12	2.60E-12	3.24E-12
pr153	1.059E-12	1.037E-12	1.02E-12	1.01E-12	1.25E-12
pr154	4.595E-13	4.406E-13	4.22E-13	4.20E-13	5.09E-13
pr155	3.293E-13	3.113E-13	2.94E-13	2.91E-13	3.49E-13
pu238	6.82E-12	6.82E-12	6.82E-12	6.82E-12	6.82E-12
pu239	2.481E-14	2.481E-14	2.48E-14	2.48E-14	2.48E-14
pu240	9.116E-14	9.117E-14	9.12E-14	9.12E-14	9.12E-14
pu241	4.186E-11	4.186E-11	4.19E-11	4.19E-11	4.19E-11
pu242	1.595E-15	1.595E-15	1.60E-15	1.60E-15	1.60E-15
pu243	0	0.000001058	1.06E-06	0.00E+00	0.00E+00
rb100	4.732E-11	4.79E-11	4.84E-11	4.85E-11	6.09E-11
rb88	1.511E-15	1.534E-15	1.56E-15	1.56E-15	1.98E-15
rb89	1.671E-14	1.691E-14	1.71E-14	1.71E-14	2.16E-14
rb90	6.285E-14	6.362E-14	6.44E-14	6.45E-14	8.12E-14
rb90m	1.966E-13	1.99E-13	2.01E-13	2.02E-13	2.54E-13

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
rb91	2.764E-12	2.794E-12	2.82E-12	2.83E-12	3.56E-12
rb92	5.145E-11	5.195E-11	5.24E-11	5.25E-11	6.60E-11
rb93	4.043E-11	4.071E-11	4.10E-11	4.10E-11	5.15E-11
rb94	4.828E-11	4.835E-11	4.84E-11	4.84E-11	6.06E-11
rb95	1.76E-10	1.757E-10	1.75E-10	1.75E-10	2.19E-10
rb96	1.038E-10	1.015E-10	9.93E-11	9.89E-11	1.22E-10
rb97	2.041E-11	2.032E-11	2.02E-11	2.03E-11	2.53E-11
rb98	7.247E-12	6.871E-12	6.50E-12	6.45E-12	7.75E-12
rh103	0	0	0.00E+00	0.00E+00	0.00E+00
rh103m	9.685E-24	9.742E-24	9.79E-24	9.80E-24	1.23E-23
rh104	8.336E-21	8.421E-21	8.50E-21	8.51E-21	1.07E-20
rh104m	4.199E-21	4.232E-21	4.26E-21	4.27E-21	5.35E-21
rh105	1.483E-07	1.483E-07	1.48E-07	1.48E-07	1.48E-07
rh105m	3.131E-21	2.959E-21	2.79E-21	2.96E-21	3.61E-21
rh106	2.166E-19	2.021E-19	1.87E-19	1.83E-19	2.17E-19
rh106m	8.213E-22	7.661E-22	7.08E-22	7.00E-22	
rh107	7.729E-20	7.209E-20	6.66E-20	6.59E-20	7.82E-20
rh108	2.103E-17	1.962E-17	1.81E-17	1.79E-17	2.13E-17
rh108m		9.154E-19	8.46E-19		9.93E-19
rh109	5.678E-17	5.327E-17	4.96E-17	4.91E-17	5.86E-17
rh110m		4.581E-16	4.37E-16	4.34E-16	5.27E-16
rh110	4.262E-15	4.08E-15	3.89E-15		4.69E-15
rh111	1.157E-14	1.127E-14	1.10E-14	1.09E-14	1.34E-14
rh112	9.844E-14	9.582E-14	9.30E-14	1.52E-13	1.88E-13
rh113	2.925E-13	2.873E-13	2.82E-13	2.81E-13	3.48E-13
rh114	4.207E-13	4.089E-13	3.97E-13	3.95E-13	4.86E-13
rh115	8.073E-13	7.755E-13	7.43E-13	7.39E-13	9.00E-13
rh116	6.574E-13	6.211E-13	5.84E-13	5.79E-13	6.96E-13
rh117	5.453E-13	5.148E-13	4.84E-13	4.80E-13	5.75E-13
ru101	0	0	0.00E+00	0.00E+00	0.00E+00
ru103	5.565E-09	5.565E-09	5.57E-09	5.57E-09	5.57E-09
ru105	2.996E-19	0.000001181	1.18E-06	2.56E-19	3.03E-19
ru106	5.848E-10	5.848E-10	5.85E-10	5.88E-10	5.88E-10
ru107	9.631E-17	8.997E-17	8.33E-17	8.24E-17	9.79E-17
ru108	2.208E-16	2.063E-16	1.91E-16	1.89E-16	2.25E-16
ru109	7.801E-15	7.573E-15	7.33E-15	7.30E-15	8.96E-15
ru110	8.459E-14	8.398E-14	8.33E-14	8.33E-14	1.04E-13
ru111	6.171E-13	6.088E-13	6.00E-13	5.99E-13	7.44E-13
ru112	7.864E-13	7.677E-13	7.48E-13	7.46E-13	9.20E-13
ru113	1.241E-12	1.203E-12	1.16E-12	1.16E-12	1.42E-12
ru114	8.339E-13	7.977E-13	7.61E-13	7.71E-13	9.36E-13

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
ru115	2.455E-13	2.311E-13	2.16E-13		2.57E-13
sb125	5.348E-23	2.168E-10	2.17E-10	4.96E-23	6.05E-23
sb127	1.634E-18	1.643E-18	1.65E-18	1.65E-18	2.07E-18
sb128	2.617E-17	2.625E-17	2.63E-17	2.63E-17	3.29E-17
sb128m	8.505E-16	8.514E-16	8.52E-16	8.52E-16	1.07E-15
sb129	3.244E-16	3.256E-16	3.27E-16	3.27E-16	4.09E-16
sb129m				0.00E+00	0.00E+00
sb130	6.797E-15	6.851E-15	6.90E-15	6.91E-15	8.67E-15
sb130m	6.9E-14	6.967E-14	7.03E-14	7.04E-14	8.84E-14
sb131	8.983E-14	9.058E-14	9.13E-14	9.14E-14	1.15E-13
sb132	6.147E-13	6.181E-13	6.22E-13	6.22E-13	7.81E-13
sb132m	2.907E-13	2.916E-13	2.93E-13	2.93E-13	3.67E-13
sb133	1.213E-12	1.216E-12	1.22E-12	1.22E-12	1.52E-12
sb134	4.473E-11	4.429E-11	4.38E-11	4.38E-11	5.43E-11
sb134m	3.465E-12	3.43E-12	3.40E-12	3.39E-12	4.21E-12
sb135	1.074E-11	1.051E-11	1.03E-11	1.03E-11	1.27E-11
sb136	2.847E-12	2.723E-12	2.60E-12	2.58E-12	3.13E-12
sb137	2.73E-11	2.752E-11	2.77E-11	1.23E-11	1.54E-11
se79m	5.101E-19	5.16E-19	5.22E-19	5.23E-19	6.58E-19
se81	6.586E-17	6.67E-17	6.75E-17	6.76E-17	8.50E-17
se81m	1.423E-16	1.441E-16	1.46E-16	1.46E-16	1.84E-16
se83	7.711E-15	7.803E-15	7.89E-15	7.91E-15	9.96E-15
se83m	3.422E-14	3.465E-14	3.51E-14	3.51E-14	4.42E-14
se84	2.455E-13	2.483E-13	2.51E-13	2.39E-13	3.01E-13
se85	2.104E-12	2.124E-12	2.14E-12	2.15E-12	2.70E-12
se86	4.287E-12	4.314E-12	4.34E-12	4.65E-12	5.84E-12
se87	1.052E-11	1.056E-11	1.06E-11	1.06E-11	1.33E-11
se88	1.516E-11	1.516E-11	1.52E-11	1.52E-11	1.90E-11
se89	1.481E-11	1.451E-11	1.42E-11	1.42E-11	1.75E-11
se90	1.072E-11	1.045E-11	1.02E-11	1.02E-11	1.25E-11
sm147	5.643E-21	5.643E-21	5.64E-21	5.64E-21	5.64E-21
sm149	0	0	0.00E+00	0.00E+00	0.00E+00
sm150	0	0	0.00E+00	0.00E+00	0.00E+00
sm151	6.646E-12	6.646E-12	6.65E-12	6.65E-12	6.65E-12
sm152	0	0	0.00E+00	0.00E+00	0.00E+00
sm153	1.133E-07	1.133E-07	1.13E-07	1.13E-07	1.13E-07
sm155	7.168E-17	7.237E-17	7.30E-17	7.31E-17	9.18E-17
sm156	6.705E-18	6.76E-18	6.81E-18	6.82E-18	8.56E-18
sm157	4.586E-16	4.594E-16	4.60E-16	4.60E-16	5.76E-16
sm158	6.702E-16	6.689E-16	6.68E-16	6.68E-16	8.35E-16
sm159	8.133E-15	7.976E-15	7.82E-15	7.81E-15	9.66E-15

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
sn121	2.548E-19	2.58E-19	2.61E-19	2.61E-19	3.28E-19
sn123	5.815E-21	1.691E-09	1.69E-09		
sn123m	1.009E-17	1.023E-17	1.04E-17	1.04E-17	1.33E-17
sn125	8.818E-19		2.27E-08	8.87E-19	1.12E-18
sn125m	1.337E-15	1.351E-15	1.36E-15	1.37E-15	1.72E-15
sn127	9.136E-16	9.197E-16	9.26E-16	9.27E-16	1.17E-15
sn127m	3.28E-15	3.281E-15	3.29E-15	3.30E-15	4.18E-15
sn128	3.318E-15	3.341E-15	3.36E-15	3.37E-15	4.23E-15
sn128m	1.809E-12	1.822E-12	1.83E-12	1.84E-12	2.31E-12
sn129	1.625E-13	1.618E-13	1.61E-13	1.61E-13	2.01E-13
sn129m	3.742E-14	3.759E-14	3.78E-14	3.78E-14	4.74E-14
sn130	2.003E-13	2.007E-13	2.01E-13	2.01E-13	2.52E-13
sn130m	4.383E-13	4.392E-13	4.40E-13	4.40E-13	5.51E-13
sn131	7.132E-13	7.106E-13	7.08E-13	7.08E-13	8.84E-13
sn131m	6.839E-13	6.814E-13	6.79E-13	6.79E-13	8.48E-13
sn132	1.401E-12	1.389E-12	1.38E-12	1.38E-12	1.71E-12
sn133	1.018E-11	1.003E-11	9.89E-12	9.80E-12	1.21E-11
sn134	2.489E-12	2.415E-12	2.34E-12	2.33E-12	2.85E-12
sr100	1.451E-11	1.377E-11	1.30E-11	1.29E-11	1.56E-11
sr101	5.811E-12	5.627E-12	5.45E-12	5.42E-12	6.61E-12
sr89	2.779E-19	4.323E-09	4.32E-09	2.85E-19	3.58E-19
sr90	5.658E-21	2.078E-11	2.08E-11	5.81E-21	7.31E-21
sr91	5.187E-16	5.274E-16	5.36E-16	5.38E-16	6.83E-16
sr92	7.995E-15	8.108E-15	8.22E-15	8.09E-15	1.02E-14
sr93	4.133E-13	4.182E-13	4.23E-13	4.24E-13	5.33E-13
sr94	4.351E-12	4.398E-12	4.44E-12	4.45E-12	5.60E-12
sr95	1.435E-11	1.446E-11	1.46E-11	1.46E-11	1.83E-11
sr96	2.647E-10	2.66E-10	2.67E-10	2.68E-10	3.36E-10
sr97	3.349E-10	3.355E-10	3.36E-10	3.36E-10	4.21E-10
sr98	1.14E-10	1.134E-10	1.13E-10	1.13E-10	1.41E-10
sr99	6.308E-11	6.166E-11	6.03E-11	5.99E-11	7.37E-11
tc100	2.528E-17	2.556E-17	2.58E-17	2.64E-17	3.32E-17
tc101	1.338E-17	1.354E-17	1.37E-17	1.37E-17	1.73E-17
tc102	1.254E-13	1.27E-13	1.28E-13	1.29E-13	1.62E-13
tc102m	2.537E-15	2.569E-15	2.60E-15	2.60E-15	3.27E-15
tc103	1.055E-13	1.068E-13	1.08E-13	1.08E-13	1.36E-13
tc104	6.008E-15	6.075E-15	6.14E-15	6.15E-15	7.72E-15
tc105	9.003E-15	8.984E-15	8.96E-15	8.96E-15	1.12E-14
tc106	6.508E-14	6.49E-14	6.47E-14	6.46E-14	8.05E-14
tc107	1.156E-13	1.138E-13	1.12E-13	1.12E-13	1.38E-13
tc108	5.178E-13	5.089E-13	5.00E-13	4.99E-13	6.16E-13

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
tc109	2.393E-12	2.324E-12	2.26E-12	2.25E-12	2.76E-12
tc110	1.982E-12	1.923E-12	1.86E-12	1.86E-12	2.28E-12
tc111	3.124E-12	3.004E-12	2.88E-12	2.87E-12	3.49E-12
tc112	1.258E-12	1.186E-12	1.11E-12	1.14E-12	1.37E-12
tc99	2.833E-15	2.833E-15	2.83E-15	2.83E-15	2.83E-15
tc99m	9.562E-23	9.664E-23	9.76E-23	9.79E-23	1.23E-22
te127	2.479E-21	2.314E-21	2.14E-21	2.12E-21	2.51E-21
te127m		2.004E-09	2.00E-09	3.43E-23	4.07E-23
te129	8.455E-19	7.965E-19	7.44E-19	7.38E-19	8.83E-19
te129m				4.53E-21	5.41E-21
te131	4.562E-15	4.621E-15	4.68E-15	4.69E-15	5.90E-15
te131m	1.547E-16	1.566E-16	6.82E-08	1.43E-16	
te132	4.021E-16	6.819E-08	1.20E-13	4.11E-16	5.16E-16
te133	1.17E-13	1.182E-13	6.86E-14	1.20E-13	1.51E-13
te133m	6.756E-14	6.808E-14	1.89E-13	6.87E-14	8.61E-14
te134	1.86E-13	1.876E-13	1.38E-11	1.89E-13	2.38E-13
te135	1.372E-11	1.375E-11	6.69E-12	1.38E-11	1.72E-11
te136	6.759E-12	6.725E-12	1.54E-11	6.74E-12	8.38E-12
te137	1.579E-11	1.56E-11	6.19E-12	1.54E-11	1.90E-11
te138	6.523E-12	6.353E-12	3.68E-12	6.16E-12	7.57E-12
te139	4.007E-12	3.843E-12	4.35E-12	3.66E-12	4.44E-12
te140	4.316E-12	4.332E-12		4.35E-12	5.43E-12
u234	2.423	2.423	2.42E+00	2.42E+00	2.42E+00
u235	0.03996	0.05047	6.52E-02	6.84E-02	9.26E-02
u235m				1.67E-14	1.83E-14
u236	0.01259	0.01259	1.26E-02	1.26E-02	1.26E-02
u237	8.614E-17	3.237E-08	3.24E-08	7.23E-17	8.36E-17
u238	0.3208	0.3191	3.17E-01	3.16E-01	3.13E-01
u239	4.63E-12	0.00001341	1.34E-05	3.55E-12	3.93E-12
xe131m	2.475E-23		2.53E-23	2.53E-23	3.18E-23
xe133	4.167E-08		4.17E-08	4.17E-08	4.17E-08
xe133m	8.117E-19	8.162E-19		8.22E-19	1.03E-18
xe134m	6.089E-12	6.156E-12	6.22E-12	6.23E-12	7.83E-12
xe135	5.736E-07	5.736E-07	5.74E-07	5.74E-07	5.74E-07
xe135m	1.388E-14	1.405E-14	1.42E-14	1.43E-14	1.80E-14
xe137	1.016E-12	1.028E-12	1.04E-12	1.04E-12	1.31E-12
xe138	4.179E-13	4.223E-13	4.26E-13	4.27E-13	5.37E-13
xe139	8.271E-12	8.333E-12	8.39E-12	8.40E-12	1.05E-11
xe140	2.031E-11	2.04E-11	2.05E-11	2.05E-11	2.57E-11
xe141	6.252E-11	6.227E-11	6.20E-11	6.20E-11	7.72E-11
xe142	3.375E-11	3.346E-11	3.32E-11	3.37E-11	4.19E-11

	15GWD/MTU	25.5GWD/MTU	37.5GWD/MTU	38 GWD/MTU	46.408 MW/MTU
nuclide	Curies				
xe143	1.383E-11	1.348E-11	1.31E-11	2.23E-11	2.74E-11
xe144	3.501E-12	3.347E-12	3.20E-12	1.07E-12	1.29E-12
y100	8.561E-11	8.427E-11	8.30E-11	8.28E-11	1.02E-10
y101	6.88E-11	6.765E-11	6.65E-11	6.64E-11	8.20E-11
y102	7.544E-11	7.53E-11	7.51E-11	6.26E-11	7.79E-11
y103	5.281E-12	4.984E-12	4.69E-12	4.65E-12	5.56E-12
y90	1.389E-21	8.192E-08	8.19E-08	1.42E-21	1.79E-21
y90m	2.786E-20	3.734E-09	0	2.85E-20	3.59E-20
y91	2.306E-21	3.956E-18	3.73E-09	2.37E-21	2.97E-21
y91m	3.908E-18		4.00E-18	4.01E-18	5.04E-18
y92	3.884E-16		3.98E-16	3.98E-16	5.00E-16
y93	1.054E-16	1.07E-16	1.09E-16	1.09E-16	1.38E-16
y93m	4.71E-12	4.783E-12	4.86E-12	4.87E-12	6.16E-12
y94	2.482E-14	2.518E-14	2.55E-14	2.56E-14	3.24E-14
y95	1.286E-13	1.301E-13	1.32E-13	1.32E-13	1.66E-13
y96	3.04E-12	3.072E-12	3.10E-12	3.11E-12	3.90E-12
y96m	1.534E-11	1.55E-11	1.56E-11	1.57E-11	1.97E-11
y97	3.203E-11	3.225E-11	3.25E-11	3.25E-11	4.08E-11
y97m	1.026E-10	1.034E-10	1.04E-10	1.04E-10	1.31E-10
y98	1.643E-10	1.648E-10	1.65E-10	1.65E-10	2.07E-10
y98m	4.501E-11	4.515E-11	4.53E-11	4.53E-11	5.68E-11
y99	1.133E-10	1.133E-10	1.13E-10	1.13E-10	1.42E-10
zn77	1.184E-13		1.21E-13	1.22E-13	
zn78	1.931E-13		1.96E-13	1.96E-13	2.48E-13
zr100	5.274E-11	5.318E-11	5.36E-11	5.37E-11	6.74E-11
zr101	9.898E-11	9.926E-11	9.95E-11	9.96E-11	1.25E-10
zr102	5.411E-11	5.398E-11	5.39E-11	5.39E-11	6.72E-11
zr103	4.276E-11	4.206E-11	4.14E-11	4.13E-11	5.10E-11
zr104	1.142E-11	1.103E-11	1.07E-11	1.06E-11	1.29E-11
zr105	1.608E-11	1.606E-11	1.60E-11	1.60E-11	2.00E-11
zr95	3.412E-09	3.412E-09	3.41E-09	3.41E-09	3.41E-09
zr97	1.277E-15	1.293E-15	1.31E-15	1.31E-15	
zr98	5.99E-12	6.062E-12	6.13E-12	6.14E-12	7.73E-12
zr99	1.252E-10	1.265E-10	1.28E-10	1.28E-10	1.61E-10
total	2.796	2.805	2.82E+00	2.82E+00	2.84E+00

Appendix 4: Summary Report for a BWR plant

Case title: Loca with containment leak
Run date/time: 2 019/07/30 22:38

Case Summary

Event Type Nuclear Power Plant

Case description

LOCA with all ECC's not functioning; Containment valves are passing releasing radioactive effluent to the atmosphere. The fuel pellet temperature has failed due to high temperatures and ballooning.

Location

Name: Fukushima Unit1
City, county, state: Futaba, Japan, Fukushima
Lat / Long / Elev: 37,2330° N, 141,0150° E, 10 m
UTC Offset: 9 hours
Population: not available

Reactor Parameters

Reactor power: 1380 MW_t
Average burnup: 30000 MWd / MTU
Containment type: BWR Mark I
Containment volume: 7079.2 m³
Design pressure: 413 kPa
Design leak rate: 0.54 %/d
Coolant mass: 1.25E+05 kg
Assemblies in core: 550

Source Term

Type: Long Term Station Blackout (SOARCA)
Shutdown: 2011/03/11 14:46
Release from core starts: 2011/03/11 20:46
Core damage estimated by: Core recovered status
Core recovered: No
Inventory: Default

Release Pathway

Type: BWR - Release Through Wet Well via direct, unfiltered pathway
Release height: 10, m
Release events
2011/03/11 20:46 Leak rate (% vol) Design
2011/03/11 20:46 Wet well Subcooled

Meteorology

Type: Predefined - Not site specific
Dataset name: Summer - Afternoon - Windy
Dataset desc: B stab 15 mph No precip 85 F 40% rh
Summary of data
at release point:
00:00
Dir Speed Stab Temp
Type deg mph class Precip °F
Obs 270 15,0 B None 85
Dataset options: Est. missing stability using: Persistence
Modify winds for topography: No

Calculations

Case title:	Loca with containment leak
End of calculations:	2011/03/12 04:46
	Start of release to atmosphere + 8 h
Distance of calculation:	Close-in + to 10 miles
Close-in distances:	0,1, 0,2, 0,3, 0,5, 0,7, 1,0, 1,5, 2,0 miles
Analyst name:	Operator
Inhal. dose coefficients:	ICRP 26/30

Appendix 5: Summary Report for a PWR plant

Case title: Loca with containment leak

Run date/time: 2019/07/30 17:37

Case Summary

Event Type Nuclear Power Plant

Case description

LOCA with all ECC's not functioning; Containment valves are passing releasing radioactive effluent to the atmosphere. The fuel pellet temperature has failed due to high temperatures and ballooning.

Location

Name: Koeberg - Unit 1
City, county, state: Melkbosstrand, <not specified>, Western Cape
Lat / Long / Elev: 33,6768° S, 18,4315° E, 18 m
Time zone: South African
Population: not available

Reactor Parameters

Reactor power: 1700 MW_t
Average burnup: 30000 MWd / MTU
Containment type: PWR Dry Ambient
Containment volume: 56633.69 m³
Design pressure: 400 kPa
Design leak rate: 0.17 %/d
Coolant mass: 2.00E+05 kg
Assemblies in core: 217
Steam generator type: U-Tube
SG water mass: 42184 kg

Source Term

Type: Long Term Station Blackout (SOARCA)
Shutdown: 2011/03/11 14:46
Release from core starts: 2011/03/11 22:46
Core damage estimated by: Core recovered status
Core recovered: No
Inventory: Default

Release Pathway

Type: PWR - Dry Containment Leakage or Failure
Description: Containment isolation valves failure
Release height: 10, m
Release events
2011/03/11 22:46 Sprays Off
2011/03/11 22:46 Leak rate (press/hole) 500 kPa, 5 cm diameter, 200,0 °C

Meteorology

Type: Predefined - Not site specific
Dataset name: Summer - Morning - Calm
Dataset desc: D stab 4 mph No precip 65 F 60% rh

Summary of data	Dir	Speed	Stab	Temp
at release point:	Type deg	mph	class	Precip °F
00:00	Obs 270	4,0	D	None 65

Dataset options: Est. missing stability using: Persistence
Modify winds for topography: No

Calculations

Case title:	Loca with containment leak
End of calculations:	2011/03/12 06:46
	Start of release to atmosphere + 8 h
Distance of calculation:	Close-in + to 10 miles
Close-in distances:	0,1, 0,2, 0,3, 0,5, 0,7, 1,0, 1,5, 2,0 miles
Analyst name:	Operator
Inhal. dose coefficients:	ICRP 26/30