# Severe Accident Modeling Under Extended Station Blackout for A Large Scale PWR

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## 1. Introduction

The Fukushima Daiichi accident revealed some vulnerabilities of operational nuclear power plants (NPPs) under an extended Station Blackout (SBO). The extreme conditions of the Tohoku earthquake and the subsequent tsunamis led to an SBO event which lasted for several days and led to the loss of the ultimate heat sink (UHS). Plant' alternating current (AC) power sources were lost and the plant went into a severe accident. As the accident progressed a significant amount of core material melted and relocated to the Lower Head (LH) of the Reactor Pressure Vessel (RPV). RPV integrity was jeopardized and a significant amount of radioactive material was released to the environment.

Accordingly, the NPPs industry extended the Severe Accident Management (SAM) strategies to strengthen the plants' capability to cope with risks associated with extended SBO. The In-Vessel Retention (IVR) Strategy stands as one of the core-melt key SAM strategies and aims to ensure the integrity of the reactor vessel LH under the circumstances of a relocated core materials and the retention within the vessel of the core-melt pool formation by preventing the vessel failure.

To ensure the integrity of the reactor vessel LH, SAM Guidelines (SAMG) utilize a set of high-level candidate actions, specifically: depressurization and external water injection into the primary and secondary systems. The heat removal capacity stands as the main parameter that can be used to qualify the IVR strategy [1], but due to critical heat flux limitation, it may be quite challenging for large scale power reactors, such as APR1400. For these reactors, it's been suggested to combine internal as well as external cooling of the reactor vessel in order to avoid the creep rupture failure and maintain its structural integrity.

## 2. Objective

This paper aims to understand the challenges of successfully implementing the IVR strategy for a large scale pressurized water reactor (PWR). The effectiveness of the IVR strategy is assessed in consideration of both epistemic and aleatory uncertainties. The goal is to identify the success window that guarantees the integrity of RPV is maintained in the event of an extended SBO.

## 3. Methodology

To meet the objective of this work, the model of a typical large scale PWR (APR1400) undergoing a severe accident sequence is developed using RELAP5/SCDAPSIM/MOD3.4 code. A representative severe accident scenario is selected with an SBO as an initiating event given its high contribution to core damage frequency.

The accident progresses to a degraded core due to loss of coolant, resulting in a core melt and corium relocation to the lower plenum which threatens the vessel integrity. However, the vessel integrity can be ensured by proper implementation of the IVR strategy which necessitates assessment of the underlying uncertainties. Accordingly, the Phenomena Identification and Ranking Table (PIRT) is used to identify the key phenomena and hence derive the key uncertain parameters. In this work, we consider both the epistemic uncertainties (i.e. phenomena-related, e.g.: melting and relocation phenomena) as well as aleatory uncertainties (i.e. scenario-related, e.g.: depressurization rate and timing, injection rate and timing).

### *3.1. Model description*

To simulate the response of the plant, RELAP5 module is used to calculate the overall RCS thermal-hydraulics, reactor kinetics, the transport of non-condensable gases and SCDAPSIM module is used for calculation of the heat-up and damage progression in the core. [2]. The model captures the occurrence of creep failure for the RPV based on the Larson-Miller Creep Rupture Model developed in RELAP5/SCDAPSIM/MOD3.4 code [3]. Various severe accident phenomena such as nuclear heat generation, temperature distribution, zircaloy cladding oxidation, fuel deformation, liquefaction, ballooning and rupture of fuel rod cladding, release of fission products and the disintegration of fuel rods into a porous debris of molten materials, all the way to slumping and relocation of the molten corium into the LH of the RPV are considered.

Figure 1 shows the APR1400 nodalization used in this study. The input model includes the Reactor Coolant System (RCS) and two Steam Generators (SGs) on the secondary side. The RCS consists of Reactor Pressure Vessel (RPV), two Hot Legs, four Cold Legs and four Reactor Coolant Pumps (RCPs).

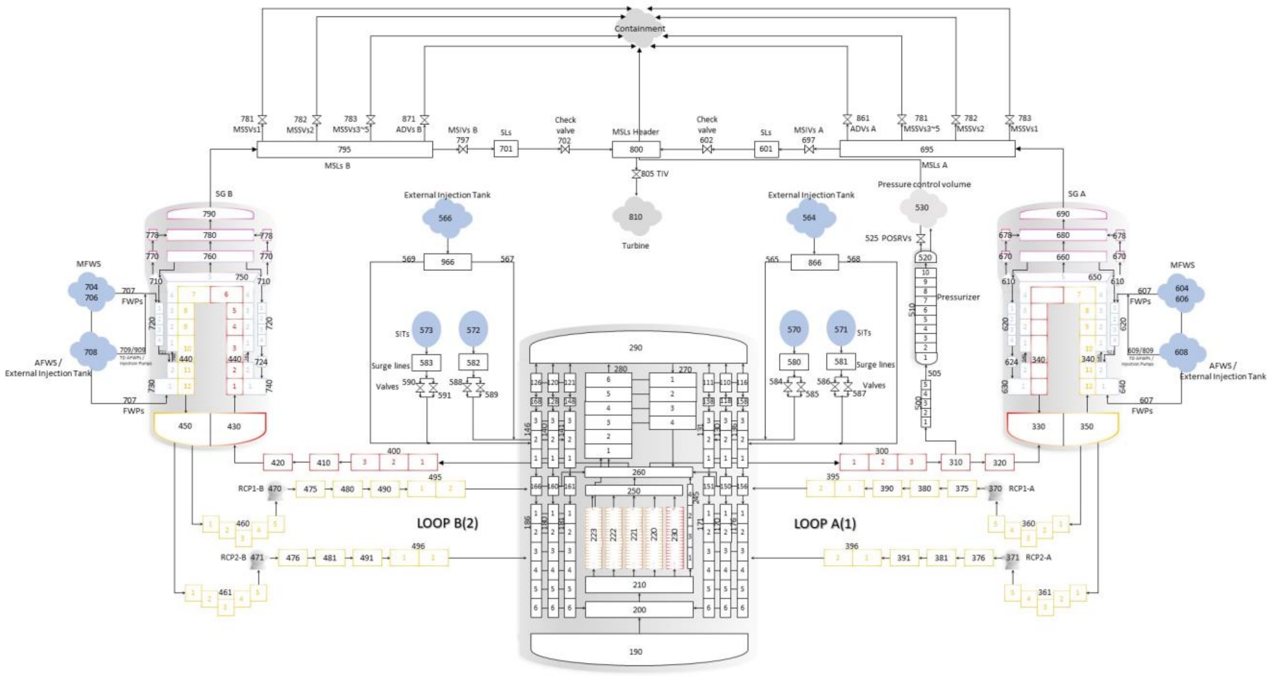


Figure 1 Steady-state validation for APR1400

Pressurizer (PZ) is connected to the Hot Leg and at its top one Pressurizer Safety Relief Valve (PRSV) is modeled to simulate the release of RCS coolant in case of depressurization. The water level in the SGs is controlled automatically over the full operating range by the Main Feedwater System (MFWS). On the secondary side, the main steam system transfers the steam from the SGs to the turbine through the Main Steam Line (MSL), six Secondary Main Steam Safety Valves (MSSVs), two Main Steam Line Atmospheric Dump Valves (MSL-ADVs), two Main Steam Line Isolation Valves (MSLIVs) and Turbine Isolation Valve (TIV) are modeled on the MSL connected to the upper head of the SGs. The MSSVs prevent over-pressurization of the SG automatically, TIV is used to isolate the turbine and the ADVs are used to depressurize the SGs. The turbine is represented as a boundary condition using a time dependent volume. Similarly, the containment is represented by a time dependent volume.

### *3.2. Assumptions and Accident Progression*

The accident scenario and associated sequence were selected based on Probabilistic Risk Assessment (PRA) study reported in APR1400 Design Control Document [4].

The SBO is initiated by a Loss of Offsite Power (LOOP) event along with a concurrent failure of both EDGs and loss of all AAC power sources. All active systems including safety system are inoperable. The only available means of supplying feedwater to the SGs is the TD-AFWPs. A rapid increase in the SGs pressure results in cyclic opening and closing of the MSSVs once the respective setpoints are reached. Consequently, steam is released to maintain the integrity of the secondary pressure boundary. The RCS natural circulation is established, and core cooling is provided. However, since the AC power is not recovered until battery depletion, MSSVs continue to release the pressure until the SGs inventory is depleted. At which point, the natural circulation stops, and heat removal is no longer possible. When the SGs dry out, the RCS pressure will rapidly increase until the POSRVs opening setpoint is reached. At this point, the RCS inventory is continuously discharged, and the core starts to uncover, ultimately leading to fuel damage. Without any provisions for a mitigation strategy, molten corium relocation to LH is inevitable.

However, the consequences of the accident can be mitigated by applying the SAM strategies which involve external water injection using portable equipment into the SGs and RCS. This necessitates depressurization of the secondary and primary sides by opening the ADVs and POSRVs.

### *3.3. Uncertainty Quantification*

The severe accident involves very complex physics which entails a number of modeling uncertainties due to incomplete knowledge and use of simplified models, system representation uncertainties, plant uncertainties and uncertainties induced by the user effect. This situation necessitates quantification of the underlying uncertainties before any conclusion can be drawn regarding the success of the IVR strategy.

Based on a previous study [6], the Phenomena Identification and Ranking Tables (PIRT) developed for a severe accident, key phenomena relevant to the in-vessel phase were identified. Subsequently, a set of uncertainty parameters associated with these phenomena are derived. These key parameters can be divided into two categories: epistemic (phenomena-related) and aleatory (scenario-related). Specifically, parameters related to the melting and relocation for the former, and depressurization rate and timing for the latter. Given the range and distribution, these uncertainties are propagated through the thermal-hydraulic model using a probabilistic methodology based on Wilk's sampling method.

## 4. Results

A steady state simulation was performed to verify the input model of RELAP5/SCDAPSIM. The steady state results compared reasonably well with corresponding values reported in APR1400 Design Control Document, as shown in Table 1.

Table 1. Steady-state validation for APR1400

| Thermal-Hydraulic Parameter | Simulation | DCD |
|---|---|---|
| Total core heat output, MWt | 3983 | 3983 |
| Primary system pressure, kg/cm²A | 155 | 158.2 |
| Reactor inlet coolant temperature, °C | 299.13 | 290.6 |
| Reactor outlet coolant temperature, °C | 330.22 | 323.9 |
| Total coolant flow, $10^6$ kg/h | 75.8 | 75.6 |
| Core-exit average coolant temperature, °C | 331.17 | 325 |
| Pumps speed, rpm | 1190.12 | 1190 |
| Steam generators pressure, kg/cm²A | 76.2 | 84.4 |
| Feedwater temperature, °C | 226.67 | 232.2 |
| Total steam flow per gen, $10^6$ kg/h | 3.92 | 4.07 |

A base case of extended SBO scenario was simulated and the timeline of key events is summarized in Table 2, with time 0 s representing the batteries' depletion time. A sharp increase in the steam generators pressure leads to the cycling of MSSVs until SGs inventory depletion at 50 minutes into the accident. This results in the loss of the natural circulation in the RCS due to loss of ultimate heat sink. After the SGs dry out, the RCS pressure rapidly increases until the POSRVs opening setpoint. At this point, POSRVS start cycling and the RCS inventory is discharged, and the core starts to uncover, ultimately leading to fuel damage. The severe accident entrance point happens after 1h and 48 minutes after losing the TD-AFW. The core starts to uncover at 1h and 22 minutes and is totally uncovered at 2h and 8 minutes.

The first liquified porous debris occurs in the hottest channel after 12 minutes from SAM entrance. The first molten pool formation occurs after 35 minutes from SAM entrance. Without any provisions for a mitigation strategy, molten corium starts to relocate to the LH. By the time 4h and 8 minutes from the SAM entrance, almost all the core is molten and slumped down at the bottom of the core region. The first creep rupture occurs in the surge line at 49 minutes after SAM entrance. This is followed by the rupture of the hot legs sequentially, and then lower head creep rupture occurred.

Table 2. Extended SBO base case accident progression for APR1400

| Time (hh:mm) | Sequence |
|---|---|
| 00:00 | **Initiating event – extended SBO**<br>Reactor TRIP<br>Turbine TRIP<br>RCPs TRIP<br>TD-AFWP TRIP<br>MSIVs TRIP<br>TIV TRIP |
| 00:01 | POSRV OPEN<br>MSSVs START CYCLING |
| 00:50 | MSSVs STOP CYCLING<br>SGs DRYOUT |
| 01:03 | POSRV START CYCLING |
| 01:05 | Boiling START |
| 01:22 | Core UNCOVERY |
| 01:48 | Severe accident ENTRANCE |
| 02:00 | Core DAMAGE |
| 02:08 | Core DRYOUT |
| 02:23 | Molten Pool FORMATION |
| 05:56 | Molten Pool SLUMPED |

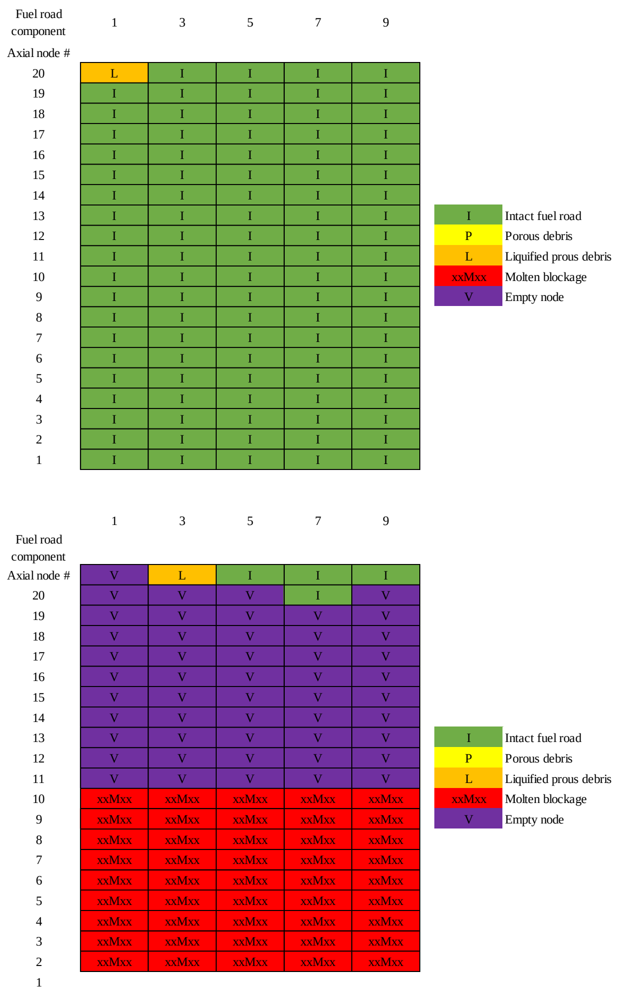


Figure 2 Core node map at the time when first core degradation occurs (upper) and at time when all the core became molten(lower)

Due to time constraints, the uncertainty analysis could not be completed at the time of preparation of this paper. Consequently, the uncertainty quantification will be included in the revised version of the paper. The detailed uncertainty analysis will be presented at the KNS Conference.

## 5. Conclusion

To assess the effectiveness of the IVR strategy for APR1400, this research focuses on analyzing the possible consequences of the severe accident after performing depressurization and external water injections into SGs and RCS.

For accurate and realistic results, the identification of the main uncertain input parameters and their expected range and distribution is imperative.

For the epistemic uncertainty, the models used for core degradation, coolability, oxidation, fission product release and transportation, relocation, debris formation, melt pool formation, and the external cooling of the RPV, the heat fluxes imposed on the RPV by the molten core, are essential to predict the final state of relocated molten material and the thermal loads on the lower head (LH). For the aleatory uncertainty parameters the depressurization timing (i.e. ADVs and POSRVs opening time and action time), depressurization rate (i.e. percentage of valve opening), injection rate and timing, along with the water injection temperature, coolant inventory in SITs, pressure in SITs, SITs coolant temperature, are considered.

Hence it is particularly important to quantify all those underlying uncertainties before identifying the success window of the IVR strategy. The model results would help in identification of the available success window particularly regarding the timing of relevant operator actions during the severe accident.

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