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A SE Approach to Assess The Success Window of In-Vessel Retention Strategy

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Abstract : The Fukushima Daiichi accident in 2011 revealed some vulnerabilities of existing Nuclear Power Plants (NPPs) under extended Station Blackout (SBO) accident conditions. One of the key Severe Accident Management (SAM) strategies developed post Fukushima accident is the In-Vessel Retention (IVR) Strategy which aims to retain the structural integrity of the Reactor Pressure Vessel (RPV). RELAP/SCDAPSIM/MOD3.4 is selected to predict the thermal-hydraulic response of APR1400 undergoing an extended SBO. To assess the effectiveness of the IVR strategy, it is essential to quantify the underlying uncertainties. In this work, both the epistemic and aleatory uncertainties are considered to identify the success window of the IVR strategy. A set of in-vessel relevant phenomena were identified based on Phenomena Identification and Ranking Tables (PIRT) developed for severe accidents and propagated through the thermal-hydraulic model using Wilk's sampling method. For this work, a Systems Engineering (SE) approach is applied to facilitate the development process of assessing the reliability and robustness of the APR1400 IVR strategy. Specifically, the Kossiakoff SE method is used to identify the requirements, functions and physical architecture, and to develop a design verification and validation plan. Using the SE approach provides a systematic tool to successfully achieve the research goal by linking each requirement to a verification or validation test with predefined success criteria at each stage of the model development. The developed model identified the conditions necessary for successful implementation of the IVR strategy which maintains the vessel integrity and prevents a melt-through.

Key Words : Systems Engineering, APR1400, Severe Accident, In-Vessel Retention Strategy, Modeling and Simulation.

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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). This led the nuclear industry to develop appropriate Severe Accident Management (SAM) strategies to strengthen the plants' capability to cope with an SBO event that may last for several days. One of these key strategies is the In-Vessel Retention (IVR) strategy which aims to prevent the vessel failure in order to ensure that corium and fission products are contained.

During an extended SBO, all alternating current (AC) power sources are lost and with the depletion of direct current (DC) battery power, secondary heat removal via the Turbine Driven Auxiliary Feedwater Pumps (TD-AFWPs) is also lost and the plant undergoes a severe accident. In case of failure to recover the AC power, the accident progresses and a significant amount of core material would melt and relocate to the Lower Head (LH) of the Reactor Pressure Vessel (RPV). To ensure the integrity of the reactor vessel LH, SAM Guidelines (SAMG) propose to utilize a set of high-level candidate actions to mitigate the accident and minimize the consequences of an extended SBO.

The heat removal capacity stands as the main parameter that can be used to qualify the IVR mitigation strategy (Ma et al., 2016). While the IVR strategy is feasible for smaller power reactors, it may be quite challenging for large scale power reactors due to critical heat flux limitation. For these latter reactors, it's been suggested to combine cooling the molten corium from both inside and outside in order to avoid the creep rupture failure and maintain the vessel structural integrity.

2. Literature review

A number of studies investigated the IVR strategy under severe accident conditions. For example Cho et al. [2], and Ma et al. [4] showed that proper implementation of the SAMG high-level candidate actions related to the in-vessel phase, the consequences can be minimized by providing preserving the vessel integrity.

Cho et al. [2] evaluated the in-vessel phase of the SAM strategy by identifying and assessing the epistemic and aleatory uncertainties. The analysis considers a sensitivity study of the in-vessel phase of SAM for the Korean OPR1000 MWe NPP. The impact of depressurization timing and the flow rate and timing, along with the uncertainties associated with the core melting and relocation process had been analyzed.

According to Ma et al. [4] the success of the IVR strategy is highly dependent on the evolution of the molten corium pool. This is attributed to the fact that different configuration of the molten pool impose different thermal loads on the reactor vessel during the accident.

Park et al. analyzed five different accident scenarios initiated by different events (TLFW, SBO, SBLOCA without SI, MBLOCA without SI, and LBLOCA without SI). The goal is to evaluate the robustness of the APR1400 vessel under different accident conditions. They concluded that different initiating events lead to different melt pool configurations (two-layer formation versus inverted layer configuration), which may impact the thermal load and the time of vessel failure.

Other studies focused on the later stage in SAM for Nordic BWRs such as the vessel failure mode by Goronovski et al. [20],[17], the ex-vessel debris coolability by Yakush et al. [18], and the steam explosion by Grishchenko et al..[19],[21] These studies develop surrogate models to represent the various phenomena underlying the severe accident and hence assess the effectiveness of SAM strategies.

3. Engineering Approach

3.1 Objective

This paper aims to understand the challenges of implementing the IVR strategy for a representative large-scale pressurized water reactor (PWR) in order to identify the success window that guarantees the integrity of the RPV in the event of an extended SBO.

RELAP5/SCDAPSIM/MOD/3.4 is used to conduct the thermal-hydraulic analysis of a severe accident initiated by an extended SBO. This is followed by an uncertainty analysis to assess the effectiveness of the intended IVR strategy.

For this work, a Systems Engineering approach is applied to facilitate the development process of assessing the reliability and robustness of APR1400 IVR strategy in the event of a severe accident initiated by an extended SBO. Specifically, Kossiakoff's four basic activities are applied, as illustrared in Figure 1:

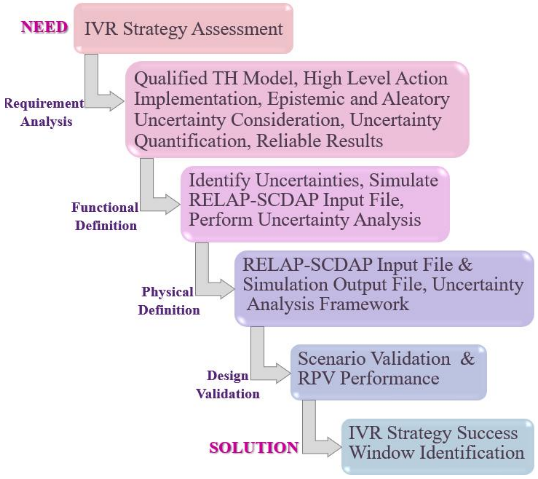
Requirement analysis (problem definition);

Functional definition (functional analysis and allocation);

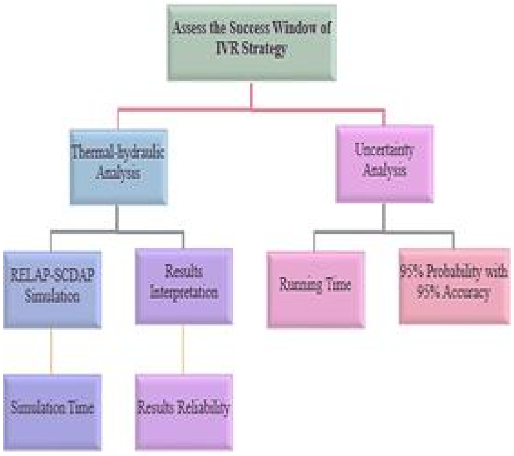
Physical definition (synthesis, physical analysis, and allocation);

Design validation (verification and evaluation).[1]

For this work, an objective hierarchy, shown in Figure 2, was developed in order to identify the most important characteristics relevant to the overall process of completing the mission.



[Figure 1] Engineering Method



[Figure 2] Objective Hierarchy

3.2 Work Breakdown Structure

After a Systems Engineering approach is established, the work breakdown structure should be defined. The work breakdown structure involves seven main stages:

1) Develop, verify and validate an APR1400 plant model to reflect the main components and systems under an extended SBO.

2) Conduct a base case simulation of the severe accident to investigate the corium behavior using RELAP5/SCDAPSIM/MOD/3.4 code under nominal operating conditions.

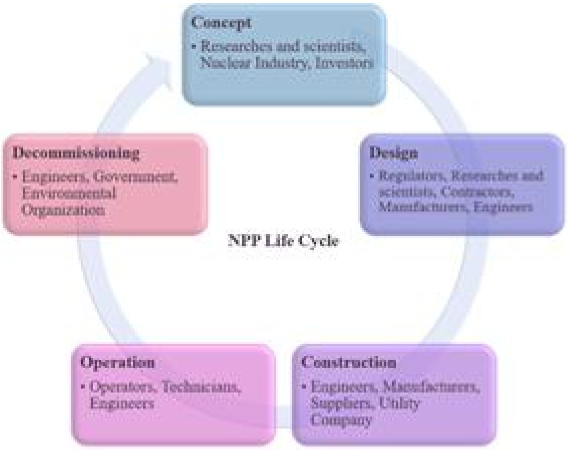
3) Test the implementation of the IVR strategy using the developed model.

4) Identify the key uncertain parameters based on the PIRT developed for severe accidents. In relevance to the extended SBO, key uncertain parameters are identified with ranges and distributions.

5) Quantify the uncertainties for more reliable and realistic results.

6) Identify the time frame and conditions where the strategy is performed successfully.

7) Validate the case study by comparing the results with other published results.



[Figure 3] NPP Stakeholders throughout its life cycle

3.3 Stakeholders Identification

A NPP is a large-scale complex plant that involves a number of stakeholders with vested interest in the project throughout its life cycle as shown in Figure 3, from the initial conceptual phase to the decommissioning phase.

For this work, the stakeholders may be categorized into four main groups with economic, social, environmental and technical interests.

All the involved stakeholders have been deeply affected by the Fukushima Daiichi accident, particularly impacting people's perception of the nuclear industry. It is therefore imperative to enhance the NPP safety, which led the industry to derive additional requirements to be enforced by the regulators to regain the stakeholders' trust in nuclear power plants.

4. Concept Development

4.1 Requirements Analysis

The requirements considered for this work fall into three categories: mission requirement, originating requirements, and system and component requirements, as summarized in Table 1.

The mission requirements are traceable to stakeholders' needs and reflect the needs and goals of the stakeholders to evaluate the success of the new mitigation strategy and the system response.

<Table 1> Severe Accident initiated by Extended SBO Requirements for APR1400

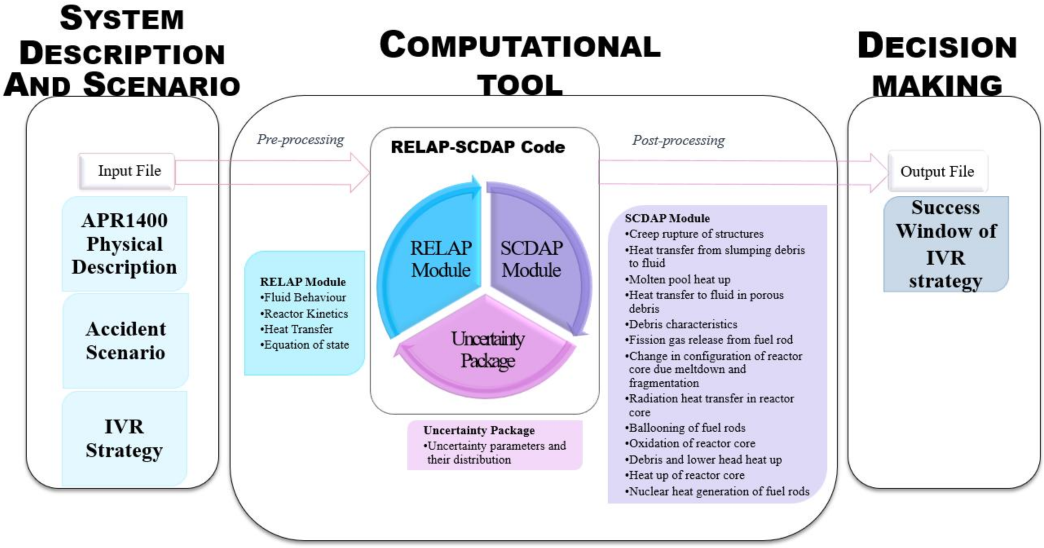
Requirements	Description
Mission Requirements	• APR1400 IVR strategy shall ensure the NPP safety under a severe accident condition initiated by extended SBO.
Originating Requirements	• The vessel integrity shall satisfy the safety criteria under extended SBO condition. • The vessel integrity shall be confirmed with 95% probability and 95% confidence under extended SBO condition.
System and Components Requirements	• All AC power and all equipment powered by AC power shall not be available. • All AAC and emergency diesel generators shall not be available. • The FLEX portable equipment should be aligned at 2 hours. • The plant should provide feed and bleed to cope with severe accident conditions. • Primary injection and secondary injection should be provided to cope with severe accident conditions. • The operator action is expected within 30 minutes from SAM entrance.

The originating requirements are derived from the stakeholders' inputs and mission requirements. The objective of all stakeholders is to avoid or at least minimize the radioactive releases.

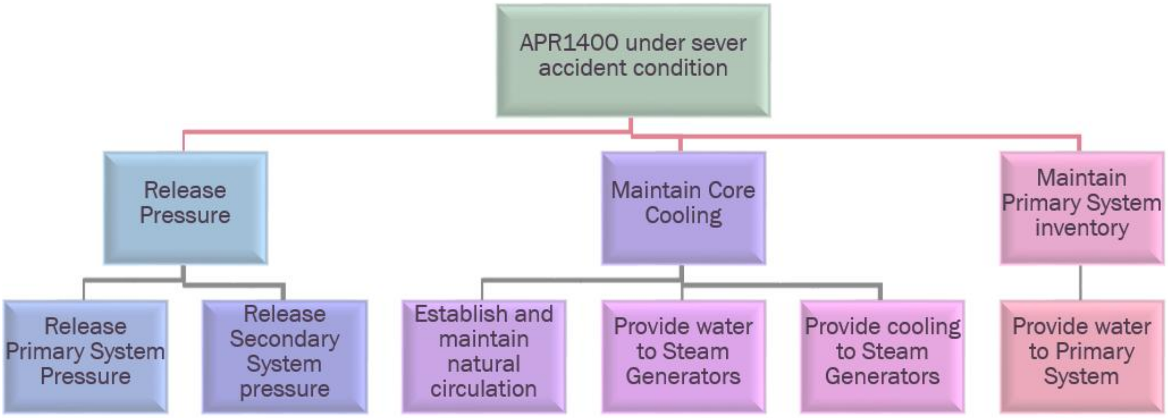
The system and component requirements are about functions and performance constraints to systems, components, or both for the given scenario.

4.2 System and Functional Architecture

The system architecture, shown in Figure 4, illustrates how the different code modules interact with each other during the simulation of the accident scenario. RELAP5/SCDAPSIM /MOD/3.4 can be divided into three main modules: RELAP, SCDAPSIM and UQ modules. The RELAP module and SCDAPSIM module interact with each other exchanging parameters and information to perform the thermal-hydraulic analysis with detailed models that reflect the progression of damage within the core. The UQ (Uncertainty Quantification) module is integrated within the



[Figure 4] RELAP5/SCDAPSIM/MOD3.4 System Interaction and Intergradation Structure



[Figure 5] Functional architecture for IVR Strategy

software and used to conduct the uncertainty analysis.

During the accident progression, three functions are essential to avoid vessel failure, as shown in Figure 5. First, the core shall be cooled to remove the decay heat. Additionally, core coverage and pressure boundary integrity shall be maintained. Cooling may be conducted on the secondary side, the primary side, or both, via a feed and bleed operation. To avoid system over−pressurization the operator should depressurize the system and provide a flow path for the feed−and−bleed operation. To maintain the coverage, it is necessary to provide water to the primary system. The low−pressure head of the external portable pumps may make water injection possible after performing the depressurization process. The IVR strategy, therefore, relies on two high−level actions, namely: system depressurization and injections into the Steam Generators (SGs) and the Reactor Coolant System (RCS).

5. Engineering Development

5.1 Thermal−Hydraulic Model

As mentioned earlier, the progression of the severe accident phenomena is modeled using RELAP5/SCDAPSIM/MOD3.4 to simulate the response of the plant. RELAP5 module is used to calculate the overall RCS thermal−hydraulics, reactor kinetics and the transport of non−condensable gases and SCDAPSIM module is used to calculate the heatup and damage progression in the core.[9] 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.[13]

To simulate the accident, the system description (nodalization) should be first provided. The nodalization used in this study 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). A Pressurizer (PZ) is connected to the Hot Leg and at its top, one Pressurizer Safety Relief Valve (PSRV) 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-ressurization of the SG automatically, TBV is used to isolate the Turbine and the ADVs are used to depressurize the SGs by the operator. The turbine is represented as a boundary condition using a time dependent volume. Similarly, the containment is represented as a boundary condition by a time dependent volume.

5.2 Accident Scenario

The accident scenario and associated sequence were selected based on Probabilistic Risk Assessment (PRA) study reported in APR1400 Design Control Document.[14] The SBO is initiated by a LOOP event along with a concurrent failure of both EDGs. Due to the loss of all AAC power sources, all active systems including the Emergency Core Cooling System (ECCS), Shutdown Cooling System (SCS) and the MD-AFW 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 maintaining the secondary pressure boundary integrity. The RCS natural circulation is established, and core cooling is provided. However, if the AC power is not recovered until battery depletion or until SGs inventory depletion, RCS 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. 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 expected.

Due to the failure of offsite power recovery within 72 h, the AC power is still unavailable. The Diverse and Flexible Coping Strategies (FLEX) can be implemented and portable equipment aligned at 2 hours from the severe accident entry condition and therefore an external power for POSRVs opening and external pumps are employed to provide the SGs and RCS injections. Additionally, the operator action is necessary for opening the ADVs.

5.3 Uncertainty Quantification

The severe accidents involves very complex physics which entails a number of modeling uncertainties due to incomplete knowledge and use of simplified models. Additional sources of uncertainties include 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. The uncertainty quantification starts with Phenomena Identification and Ranking Tables (PIRT). Based on previous studies[6],[7], 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), as shown in Table 2. Specifically, parameters related to the melting and relocation for the former, and depressurization rate and timing for the latter.

<Table 2> Main Uncertainty for Severe Accident initiated by Extended SBO

Category	Description
Scenario-related (Aleatory)	• Depressurization timing (i.e. ADVs and POSRVs opening time and action time); • Depressurization rate (i.e. percentage of valve opening); • Alignment time for the FLEX equipment; • Injection rate and timing, along with the water injection temperature; • Coolant inventory in SITs, along with the pressure in SITs and SITs coolant temperature.
Phenomenological (Epistemic)	• Models used for core degradation; • Coolability; • Oxidation; • Fission product release and transportation; • Relocation; • Debris formation; • Melt pool formation; • External cooling of the RPV; • Heat fluxes imposed on the RPV by the molten core.

Given the range and distribution, these uncertainties are propagated through the thermal-hydraulic model using a probabilistic methodology based on Wilk's sampling method.

5.4 Verification and Validation

The verification and validation process was accomplished following the V-Model Diagram shown in Figure 6. The V-Model consists of four major stages and the test at each stage is conducted to ensure that all the requirements are met.

Once the thermal-hydraulic model is developed, it is validated by comparing the predicted steady-state response to the corresponding values reported in the DCD document as shown in Table 3. The steady-state simulation results of the thermal-hydraulic model are in reasonable agreement with the APR1400 operating conditions with a deviation of less than 10 %. Next, an extended SBO scenario is simulated

<Table 3> Steady-state validation for APR1400

Thermal-Hydraulic Parameter	Simulation	DCD
Total core heat output, MWt	3983	3983
Primary system pressure, kg/cm^2A	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/cm^2A	76.2	84.4
Feedwater temperature, °C	226.67	232.2
Total steam flow per gen, 10^6 kg/h	3.92	4.07

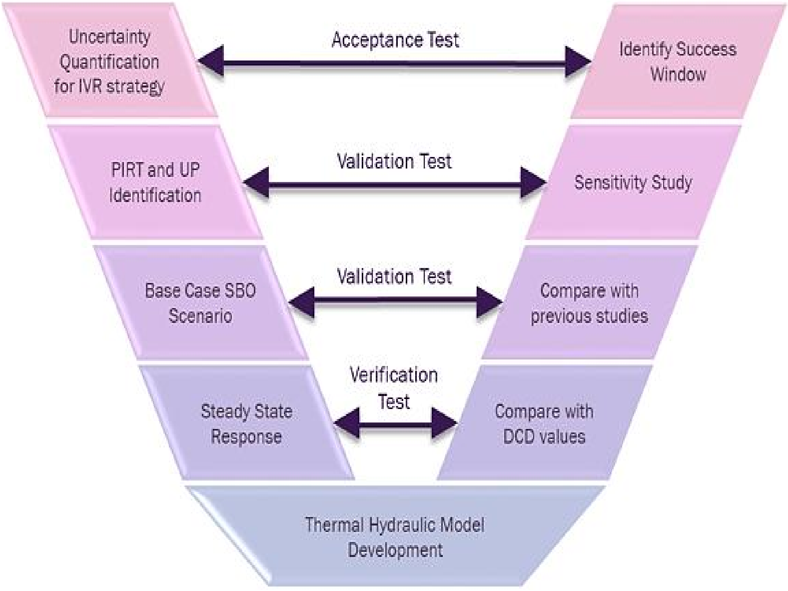
and the results of the base case are compared with previously published data.

The acceptance test shall satisfy the mission requirements. The results of the simulation satisfy the stockholders' requirements and show that IVR strategy can be safely adopted in APR1400 and is able to withstand a severe accident without any radioactive release.

Table 4 illustrate the link between requirements and validation and verification matching process.

<Table 4> Requirements and V/V matching process

Requirements	V/V
APR1400 IVR strategy shall ensure the NPP safety under a severe accident condition initiated by extended SBO.	RPV withstand the severe accident condition and no radioactive material is released.
The vessel integrity shall satisfy the safety criteria under extended SBO condition.	Creep rupture of the RPV is prevented by IVR strategy.
The vessel integrity shall be confirmed with 95% probability and 95% confidence under extended SBO condition.	Uncertainty quantification.
All AC power and all equipment powered by AC power shall not be available.	All safety systems, like RCP, SIP, MD-AFWP, are unavailable
All AAC and emergency diesel generators shall not be available.	All safety systems powered by AC power are unavailable. Additionally, TD-AFWP are unavailable.
The FLEX portable equipment should be aligned at 2 hours.	A time dependent volume had been modeled to reflect in the thermal-hydraulic model FLEX equipment.
The plant should provide feed and bleed to cope with severe accident conditions.	Logic trips had been implemented to activate the depressurization and injection operations.
Primary injection and secondary injection should be provided to cope with severe accident conditions.	Time dependent volumes had been modeled to provide primary and secondary injections.
The operator action is expected within 30 minutes from SAM entrance.	A time delay of 30 minutes is programmed after SAMG entrance condition is achieved.



[Figure 6] V-Diagram for evaluation of IVR strategy under extended SBO

6. Conclusion

A systems engineering approach is adopted to provide a systematic approach to successfully achieve the research mission. A set of verification

and validation activities are followed to ensure that for each stage the requirements are met with predefined success criteria.

The work involves developing a thermal−hydraulic model of APR1400 undergoing an extended station blackout using RELAP5/SCDAPSIM/MOD3.4 tool. The model assesses the effectiveness of the In−Vessel Retention Strategy to retain the structural integrity of the reactor vessel. The underlying uncertainties are quantified to identify the successful window of the IVR strategy.

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