ASSESSMENT OF SBO FUKUSHIMA LIKEWISE SCENARIO FOR AN IPWR DESIGN WITH RELAP5MOD33 AND RELAPSCDAPSIMMOD3.5 CODES

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ABSTRACT

In recent years Small Modular Reactors (SMR) have become very popular within the nuclear industry. These designs allow to reduce costs as well as to enhance the safety due to passive nuclear safety features. Within these systems, the integral Pressurized Water Reactors (iPWR) are very extended because they take advantage of the previous technology developed for Gen II and III PWRs. In this sense, previous Best Estimate system codes like RELAP5 or CATHARE seem to be reliable for Deterministic Safety Assessment (DSA) but need to be assessed for new passive systems in which natural circulation takes a key role. In the present paper, Energy Software Ltd., in collaboration with the UPC, has developed an iPWR input model for both NRC RELAP5 and ISS RELAPSCDAPSIM codes. These models, based on CAREM-25 publicly available data, simulate an SBO Fukushima likewise scenario. Results under Design Basis Accident (DBA) conditions are benchmarked to assess the reliability of the codes to reproduce the plant availability reported in the collected data. Passive systems like Safety Injections and Residual Heat Removal Exchangers have also been included to analyze the code capabilities to reproduce natural circulation under iPWR conditions. Finally, core damage progression is simulated with SCDAP components to analyze the severe accident related phenomena. Results of both simulations seem to confirm the 36 hours grace period for SBO scenario of the CAREM-25 design plus the extended 36 hours grace period associated to the availability of Emergency Injection System (EIS) in Loss of Coolant conditions reported by designer.

KEYWORDS
SMR, RELAP5, SCDAP, iPWR, SA, SBO

1. INTRODUCTION

In the last two decades Small Modular Reactors (SMR) have become very popular within the Nuclear Industry. After the 2015 Paris agreement [1], which pushed to strongly limit the increase of the global average temperatures, worldwide countries have focused their attention on low-carbon sources of energy that can provide steady and reliable response to the grid. In this context, SMRs are of great interest because they guarantee the well-known power availability of the Nuclear Power Plants with lower upfront capital cost and enhanced safety. The main strengths of the SMRs are the modularization, the factory construction, and the wider applicability to non-electrical purposes that improves the thermal efficiency of the utility, like cogeneration, hydrogen production and sea-water desalinization. Currently there are more than seventy SMR designs under development for different applications [2]. Designs can be classified in the following groups: 31 Water Cooled Reactors (6 marine based), 14 High Temperature Gas Cooled Reactors, 11 Fast Neutron Spectrum Reactors, 10 Molten Salt Reactors, and 6 Micro-sized Reactors. As described in C. Zeliang and et. al. in [3], main challenges in the design of SMRs are those related with Modifications to regulatory and licensing and with Passive Safety Systems. In this sense, Water Cooled Reactors take advantage compared with other proposals because they are the most deployed and tested designs in the history of the Nuclear Industry. This reduces the regulatory uncertainties also helping to meet the safety goals and requirements. In addition, some of the Passive Safety Systems (PSS), like the Passive Residual Heat Removal systems (PRHRS), have been previously designed and/or tested in Gen III+ LWRs reactors, such as AP1000 [4] and ESBWR [5]. This also enhances the reliability of the Water-Cooled designs. Between this group, the integral Presurized Water Reactors (iPWR) are the most extended ones. The main feature of the iPWRs is that they include most of the components of PWRs in the Reactor Pressure Vessel (RPV). This eliminates some potential accident initiators also reducing the number of design basis accidents, like Large Break Loss Of Coolant Accidents (LBLOCA). Furthermore, the PSSs embedded to iPWRs minimize the remaining initiators providing an inherent and robust response to mitigate their effects. This allows to include extended accident conditions in the licensing process like those of the Station Blackout occurred in Fukushima. Within the design of Nuclear Power Plants, Deterministic Safety Assessment (DSA) takes a key role to demonstrate the reliability of the safety systems. And for this purpose, it is a mandatory to carry out the verification and the validation (V&V) of the applied codes [6]. For system codes, most of the V&V process carried out for the Gen II and Gen III LWRs [7] and [8] can be extrapolated to iPWRs given that most of the relevant phenomena are quite similar to those that occur in a LWR. Otherwise, the addition of new passive systems that operates at different accident conditions requires further assessment. In this sense international OECD/NEA projects like those performed at PKL experimental facility [9] allow to analyze system codes for PSSs.

In the present paper, well-know system codes like RELAP5mod33 and RELAP/SCDAPSIM/MOD35 are used to simulate an iPWR in an extended SBO scenario like those that happened in Fukushima Daiichi accident [10]. Low pressure core damage sequences have been selected as it is recommended in Severe Accident Management strategies [11] to prevent high-pressure vessel failure. Grace period is assessed for both codes paying attention to the actuation of the PSSs and related phenomena. For core damage progression, main events are described by using SCDAP components with RELAP/SCDAPSIM/MOD35. This code uses the publicly available RELAP5/MOD3.3 [12] and SCDAP/RELAP5/MOD3.2 [13] models, developed by the USNRC in combination with advanced numerics, advanced programming, and SDTP member-developed models and user options. The SCDAP components include representative LWR fuel rods, control rods and general core structures like shroud and grid spacers. SCDAP models simulate the core behavior from core heat-up to core slumping and debris bed formation. SCDAP solves a two-dimensional heat conduction equation to calculate the temperature response for the SCDAP components. The calculations of damage progression include calculations of the fuel rod heat-up, ballooning and rupture, fission product release, rapid oxidation, zircaloy melting, UO_2 dissolution, ZrO_2 breach, flow and freezing of molten fuel and cladding, and debris formation and behavior.

2. IPWR NODALIZATION

The iPWR design used for generating RELAP5mod33 and RELAP/SCDAPSIM/MOD3.5 nodalizations takes as reference some of the public available data of CAREM-25 reactor. Main features and references of the selected information are listed in Table I. CAREM-25 have unique features compared to other iPWR designs as described in [14] and [15]:

- Thermal power is effectively removed by natural circulation for both nominal and shutdown conditions, and Reactor Coolant Pumps (RCP) are not necessary
- RPV is self-pressurized. Heat transfer regime in the core is subcooled boiling and part of the vapour that is generated is accumulated at the top of the vessel to passively pressurize the primary system
- Heat exchangers are placed in 12 cylindrical cartridges at the downcomer region, with embedded coil tubes that heat counter current feedwater up to 30 C of vapour superheating. This design allows avoiding large penetrations in the lower half of the RPV

Table I. Basic CAREM-25 specs

Parameter	Value
Thermal power	100 MW [16]
Pressure (primary/secondary)	12.25 / 4.7 MPa [16]
Core inlet/outlet temperature	557 / 599 K [16]
Mass flow rate	410 kg/s [16]
Feedwater temperature	473 K [16]
Number of Steam Generators (SG)	12 [17]
Number of coil tubes per SG	52 [17]
Length of coil tubes	35 m [18]
RPV height/diameter	11 / 3.2 m [2]
Assembly array	Hexagonal [19]
Number of fuel assemblies	61 [19]
Core active length	1.4 m [19]
Chimney height	4.6 m [16]
Coolant volume	39 m^3 [16]
Second Shutdown System (SSS)	2 m^3 [16]
Emergency Injection System (EIS)	41 m^3 [20]
Passive Residual Heat Removal System (PRHRS) tank	16 m^3

Taking as reference this information, two identical iPWR nodalizations have been generated from scratch for RELAP5mod33 and RELAP/SCDAPSIM/MOD3.5 (see Figure 1). Both input decks have the same hydrodynamic components, control blocks, heat structures and materials with the only difference of the fuel rod components. Main features of the nodalizations are:

- Number of hydrodynamic volumes: 301
- Number of hydrodynamic junctions: 341

- Number of heat structures: 221
- Number of trips: 22
- Number of control blocks: 16

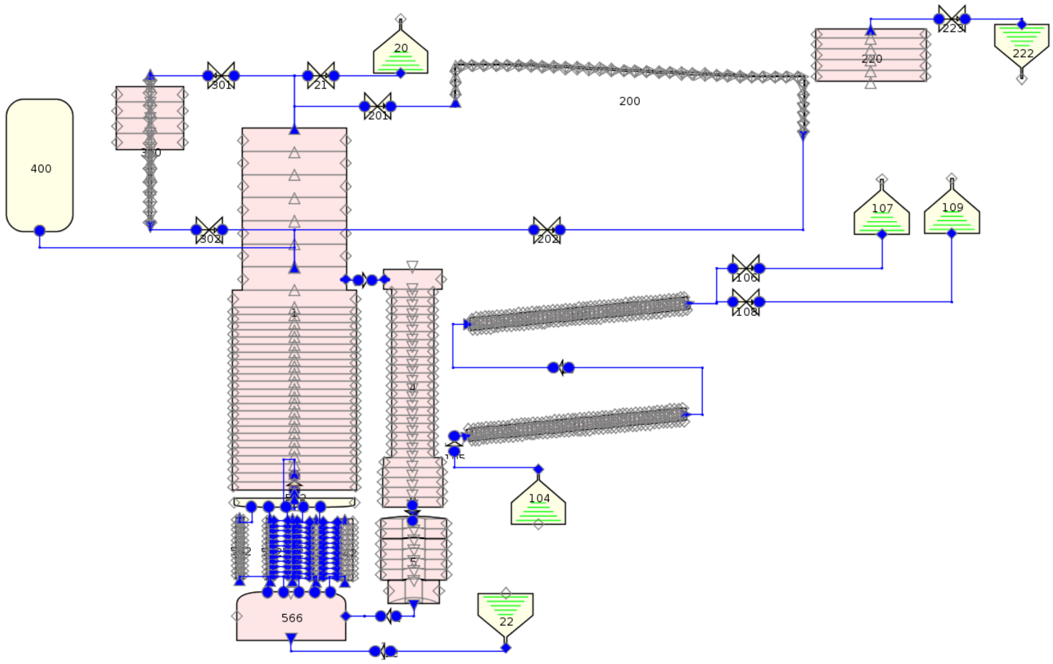


Figure 1. RELAP5mod33 and RELAP/SCDAPSIM/MOD3.5 nodalizations sketch

Three criteria were applied in the definition of the nodalization taking into account the inherent features of the selected iPWR design (natural circulation in the RPV and subcooled boiling in the core):

- to limit the DT between core axial nodes to a maximum of 3 K
- to preserve the finer meshing of the core in the central chimney to correctly reproduce the vapour to liquid heat transfer at subcooled boiling conditions
- to provide great detail in the estimation of primary to secondary heat transfer by limiting the DT in the coil tubes to 1 K

The number of control blocks and trips is very reduced because only one active system is defined for the feedwater. This system adjusts the RPV pressure to the set point value. Otherwise, different passive safety systems are modelled in the nodalizations:

- SSS: high pressure gravity driven injection system with highly borated water to shutdown the reactor if there is a malfunction in the rod insertion

- EIS: emergency accumulator system that injects subcooled water when RPV pressure drops below 1.5 MPa
- PRHRS: passive heat removal system that condensates the vapour of the RPV through horizontal tubes placed in a 16 m^3 tank heat exchanger
- Pressure Relief Valves (PRV): passive control system to avoid pressures higher than 15 MPa in the RPV. It also depressurizes the RPV if core dryout and vapour superheating is detected at the core exit temperature (CET).

In addition, both nodalizations also include the simulation of environmental heat losses at the top and the bottom of the RPV, as well as in the pipes of the passive safety systems. Heat transfer coefficents of the environtment heat structures have been adjusted to obtain a total loss of energy equivalent to 0.1 % of the nominal power.

As regards the fuel components, RELAP5mod33 nodalization uses heat structure components and RELAP5/SCDAPSIM/MOD3.5 uses SCDAP components. For both models core region is divided in 4 fuel and 1 bypass parallel channels of 12 axial levels. Fuel assemblies are not canned within duct walls, hence cross-flow junctions are included. The number of rods and relative powers for each channel follows the distribution shown in Figure 2. For SCDAP components, detailed fuel information is included (He mass and pressure, density, fuel composition, burnup,...) [21] as well as specific components for simulating the guide thimbles and the radiation shielding. SCDAP components also include the simulation of the radiation effects for the different fuel channels and their interaction with the radiation shielding, as well as a COUPLE module to simulate core slumping and debris bed heat exchange with the fluid and the walls of the lower plenum (see Figure 3).

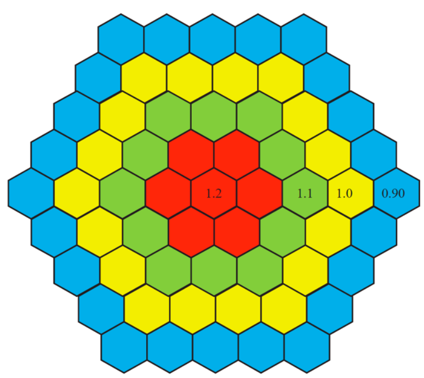


Figure 2. iPWR radial power distribution. Picture from [21]

3. STEADY STATE RESULTS

Table II shows the comparison between the steady state parameters selected in the iPWR design and those obtained for the different nodalizations (Relap5mod33 -R5m33-, RELAP/SCDAPSIM/MOD3.5 with heat structures -RS35 HS-, and RELAP/SCDAPSIM/MOD3.5 with SCDAP components -RS35 SCDAP-). The results show a close agreement in all the simulations with an acceptable maximum deviation of the 1 % for

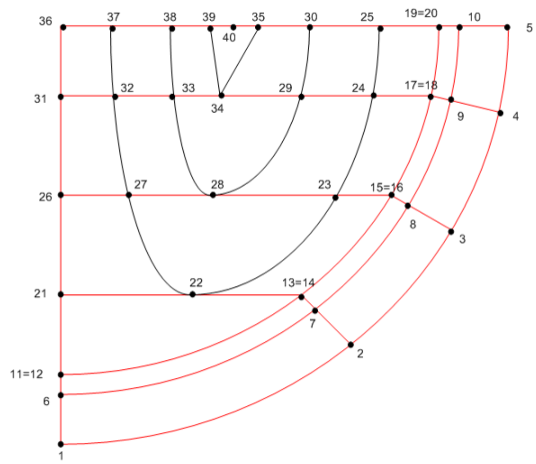


Figure 3. COUPLE meshes of Lower Plenum in RELAP/SCDAPSIM/MOD3.5 nodalization

the RPV mass flow rate of the RS35 simulations. In this sense, it is important to remark that hydrodynamics equations and convection heat transfer modes of RELAP/SCDAPSIM are based on models and correlations of RELAP5mod3.2, hence slight differences can be expected between R5m33 and RS35 simulations. Furthermore, it was observed form losses had to be modified in the core junctions of the RS35 SCDAP nodalization to obtain steady state conditions (from K = 1.8 to K = 1.2). These modifications seem to be related with the SCDAP components that increase the frictional effects in the hydrodynamic components that are coupled. Once the user defined K losses were reduced, the results were quite similar to those obtained for the nodalizations with heat structures.

Table II. iPWR steady state parameters

Parameter	Units	Expected Value	R5m33 K = 1.8	RS35 HS K = 1.8	RS35 SCDAP K = 1.2	Deviation (%)
Thermal power	MW	100.0	100.0	100.0	100.0	0.0
Primary pressure	MPa	12.25	12.25	12.25	12.25	0.0
Secondary pressure	MPa	4.7	4.7	4.7	4.7	0.0
Core inlet temperature	K	557.0	561.0	561.6	561.6	0.8
Core outlet temperature	K	599.0	599.2	599.3	599.3	0.0
RPV mass flow rate	kg/s	410.0	407.9	406.1	406.1	1.0
RPV collapsed liquid level	m	-	6.7	6.7	6.7	-
Secondary inlet temperature	K	473.15	473.15	473.15	473.15	0.0
Secondary outlet temperature	K	563.2	565.0	565.2	565.2	0.3
Secondary mass flow rate	kg/s	-	48.4	48.4	48.3	-

Finally, it is worth mentioning that both RELAP5mod33 and RELAP/SCDAPSIM/MOD3.5 do not include special models to simulate helical tubes. In this paper, default RELAP5 geometry types have been used for both R5mod33 and RS35 nodalizations. This limitation can be affecting the primary to secondary heat transfer as reported by Hoffer et al. in [22]. As it can be observed in Table I, core inlet (DC heat exchanger outlet) temperatures are slightly underpredicted, especially if it is considered that a very fine nodalization was modelled for the coil tubes (102 nodes). Therefore, future code developments could be focussed in the implementation of specific RELAP5 coil tubes models, not only for improving the precision of the results but also the efficiency of the calculations.

4. STATION BLACKOUT: ANALYSES OF THE DESIGN BASIS ACCIDENT PHENOMENA

Table III shows the main events simulated for the selected SBO scenario. Boundary conditions for the transient simulation are:

- Station Blackout takes place at 5000 seconds (SCRAM signal)
- Secondary system is automatically isolated 5 seconds after SCRAM signal
- PRHRS is automatically started with isolation signal
- PRV operating conditions: Open $P_{RPV} > 14.8\ MPa$, Close $P_{RPV} < 14.2\ MPa$
- PRV fully opened if $T_{CET} > 628\ K$
- EIS injection if $P_{RPV} < 1.5\ MPa$

Table III. iPWR Station Blackout events

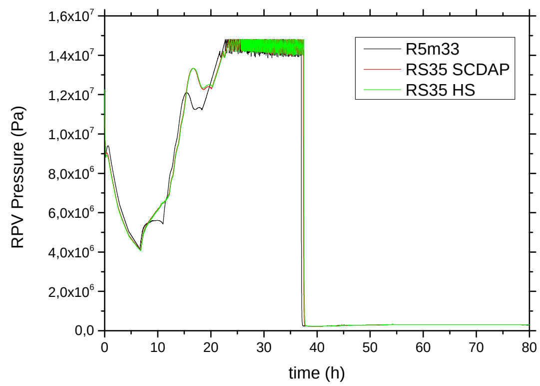
Main events	R5m33	RS35
Start of the Transient (SoT),SCRAM signal	5000 s	5000 s
PRHRS fully opened	5007 s	5007 s
Feewater stopped	5008 s	5008 s
Secondary system isolated	5011 s	5011 s
Loss of natural circulation (PRHRS tank empty)	12 h 25 min	13 h 34 min
PRV opened ($P_{RPV} > 14.8\ MPa$)	24 h 10 min	24 h 49 min
PRV depressurization ($T_{CET} > 628\ K$)	38 h 25 min	38 h 50 min
EIS started ($P_{RPV} < 1.5\ MPa$)	38 h 38 min	39 h 03 min
Core dryout	74 h 20 min	74 h 28 min
Oxidation	-	79 h 10 min
Balloning rupture	-	79 h 20 min
Ceramic formation (U-Zr-O)	-	86 h 5 min
Core Slumping	-	92 h 49 min
Creep rupture	-	95 h

Comparison between R5m33, RS35 with core HS, and RS35 with core SCDAP components (Figure 4 show a quite good agreement between all the simulations during DBA conditions. As it can be seen in Figures 4a, 4b and 4c, the overall behaviour for pressures, peak cladding temperatures and RPV level is equivalent, therefore no relevant discrepancies can be reported. Regardless, it is worth to mention that different pressures are observed between the loss of natural circulation (13 h after SoT) and the opening the PRVs (24h after SoT). In this phase the RPV is totally isolated with the only energy source of the core decay power, hence liquid expansion should be expected as observed in the simulation. Higher pressures are obtained in R5m33 for equivalent and/or lower liquid fractions of water at saturated conditions. Discrepancies seem to be related with code versions and not with SCDAP components (compare black line -R5m33- with red/green lines -RS35- in Figure 4a). As reported in RELAP5 manuals [12], R5m33 (compared to R5m32 and RS35) has different steam tables with new formulations for the light water transport properties (surface tension, viscosity and thermal conductivity), hence different results can be expected during this phase as no big sources/sinks of energy and momentum fade their effect in the numerical solution. Otherwise, such differences do not significantly affect the timing of the CET signal, the EIS injection, and the final core dryout (see Table III), hence simulations can be considered as equivalent. Other aspects to be mentioned are the lower PCT temperatures of RS35 with SCDAP during core dryout (see Figure 4b), and the one hour longer availability of the PRHRS system for RS35 simulations (see Table III). Lower PCTs result from the axial conduction feature of the SCDAP components, that reduces the temperatures of the unwetted regions of the fuel during core dryout (in Figure 4b, compare RS35 with SCDAP components -red line- with R5m33 and RS35 with HSs -black and green lines-). Longer availability of the PRHRS system results from the lower than PRHRS heat removal of the RS35 simulations (see Figure 4e), that reduces the vaporization of the water accumulated in the PRHRS tank also extending the availabily of the system (see Figure 4d). Anyhow, as the final energy balance as well as the mass inventory are the same in all the simulations no significant deviations are reported in the timing the of next events (see Table III).

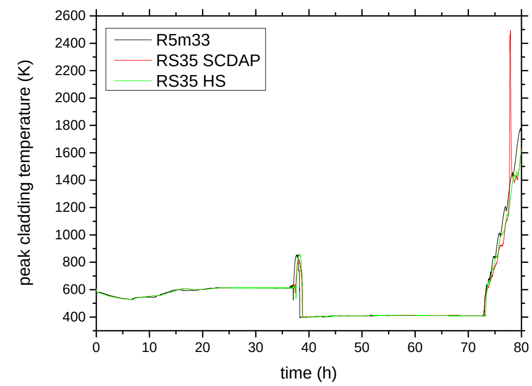
4.1. Assessment of the relavant phenomena

The RS35 SCDAP simulation is used to assess the relevant phenomena associated to DBA conditions (see Figures 5 and 6). Accident sequence starts with reactor shutdown and secondary system isolation. As a result of this, core power is suddenly reduced also affecting to the pressure of the RPV (see Figure 5a). At around 2 MW of core power, saturated conditions are achieved in all the components of the system and pressure starts to increase because of the vapour generated in isolated conditions. PRHRS system does not start to cooldown the RPV until core decay power becomes lower PRHRS heat removal. At this time, two different circulations occur in the RPV (see Figure 5b, the natural circulation induced by the PRHRS, and the circulation through the DC heat exchangers. First circulation results from the vapour accumulated at the top of the vessel that is condensed in the horizontal tubes of the PRHRS system. This circulation cools down the RPV for the first 12 hours (see Figure 5c) until the PRHRS tanks are completely empty. The second circulation is generated by liquid vaporization at the core and density differences with DC water. This circulation is kept until the swell level at the central chimney drops below the inlet of the DC heat exchangers (see Figure 5b).

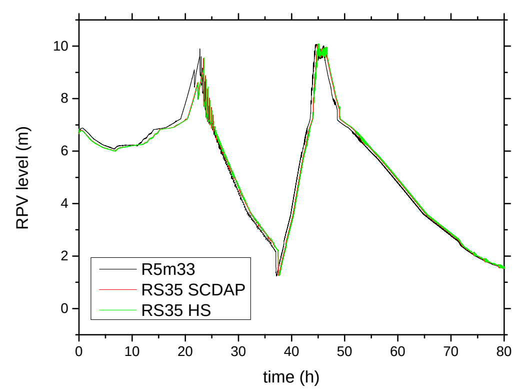
When natural circulation is (partially) lost (at around 7 hours), system pressures starts to increase (see Figure 5d) and liquid is expanded. At around 15 hours, RPV circulation is recovered because swell level achieves again the connection to the DC inlet (see Figure 5b). At this time, water that has been cooled at the bottom of the RPV by the environmental heat losses (see Figure 6a) is moved to the core, also reducing temperatures at chimney and the upper head. This phenomenon slows down the increase of pressure in the



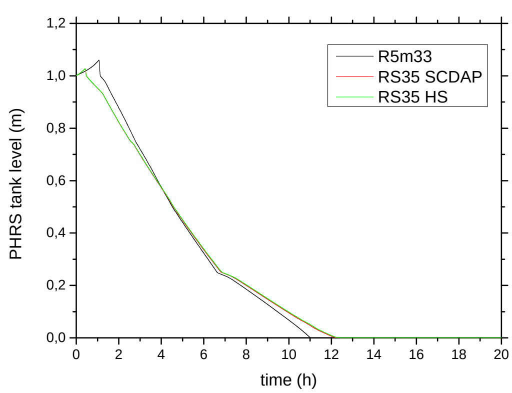
(a) RPV pressure



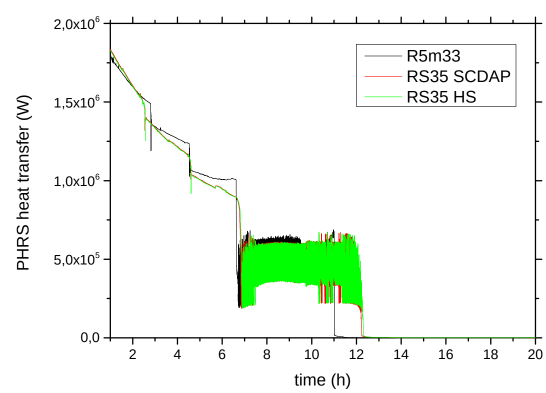
(b) Peak cladding temperature



(c) RPV collapsed liquid level



(d) PRHRS collapsed liquid level

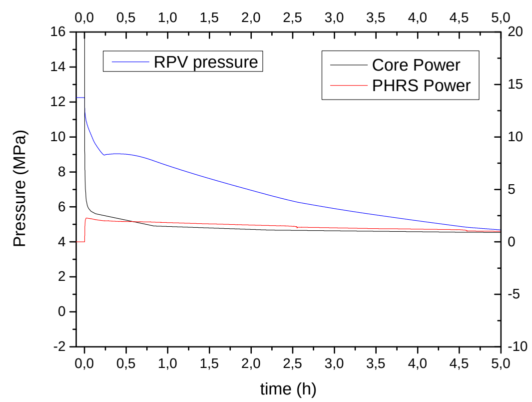


(e) PRHRS heat exchange

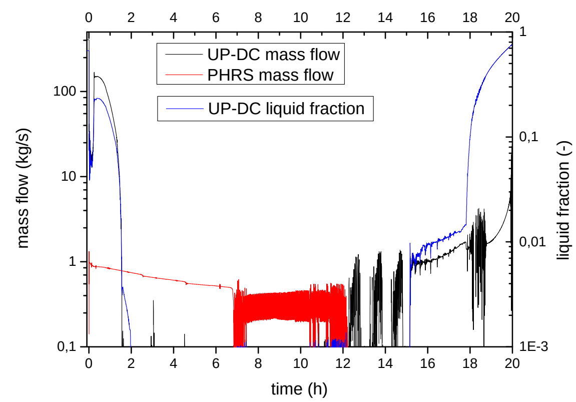
Figure 4. Comparison between R5mod33, RS35 with HS and RS35 with SCDAP

RPV and delays the opening of the PRVs until 24 hours after the SoT (see Figure 6b).

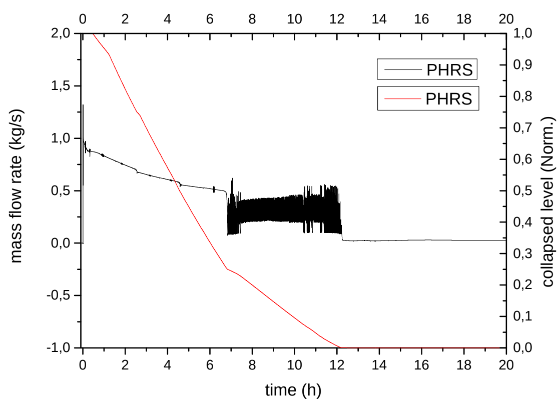
With PRVs action, RPV mass inventory starts to reduce and at around 37 hours after SoT, first core dry out occurs. Because of the CET signal, PRVs are fully opened and system pressure is drastically reduced also enabling the initiation of the EIS (see Figure 6c). The passive injection is extended continuously for more than 8 hours. It is worth mentioning that EIS needs more than one hour to totally quench the core, and that a maximum temperature of 840 K is reported. After EIS injection, liquid levels are recovered in the RPV (see Figure 6d) and grace period is extended up to 74 hours. In this sense, the selected iPWR design fulfills the 36 hours grace period for SBO scenario of CAREM-25 design plus the extended 36 hours grace period associated to the availability of EIS in Loss of Coolant conditions, [16].



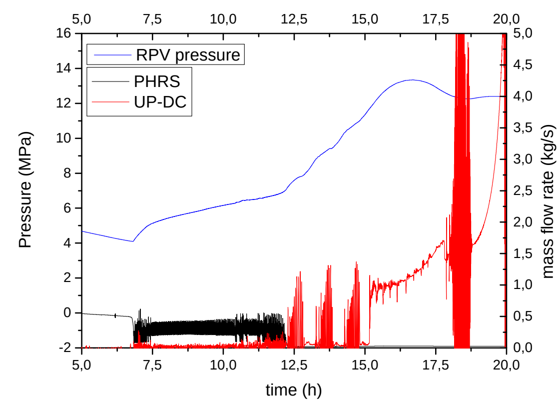
(a) RPV vs. core and RPHRS power



(b) RPV circulation vs. PRHRS circulation

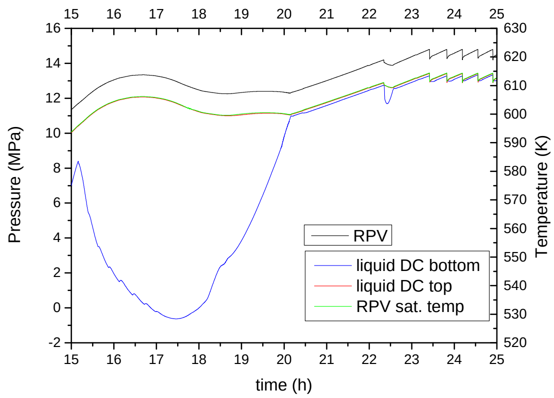


(c) PRHRS circulation vs. PRHRS level

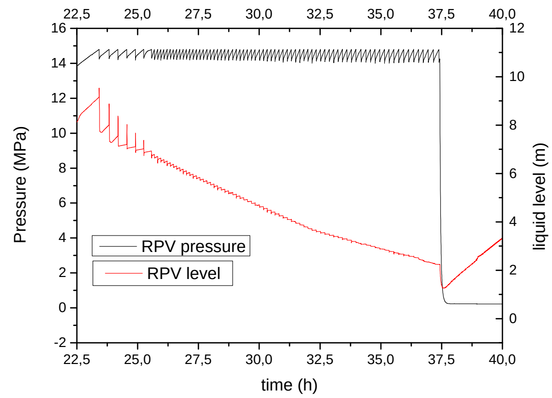


(d) RPV presure vs. PRHRS and RPV circulation

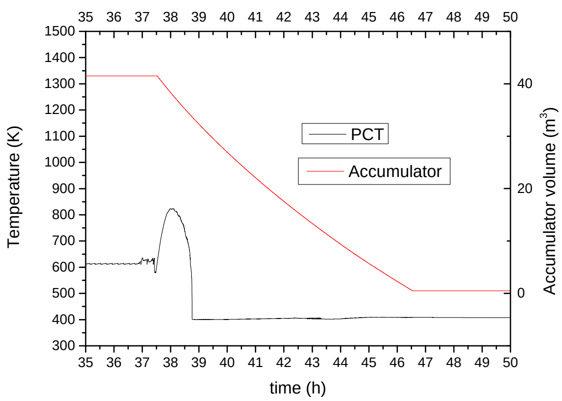
Figure 5. Assessment of the SBO phenomena for DBA conditions I



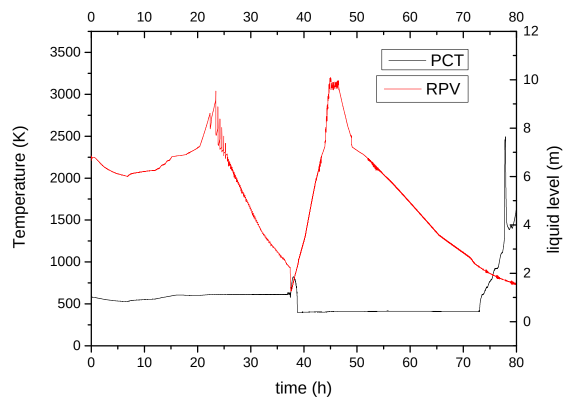
(a) RPV pressure vs. DC temperatures



(b) RPV pressure vs. RPV level



(c) PCT vs. EIS inventory



(d) RPV level vs. PCT

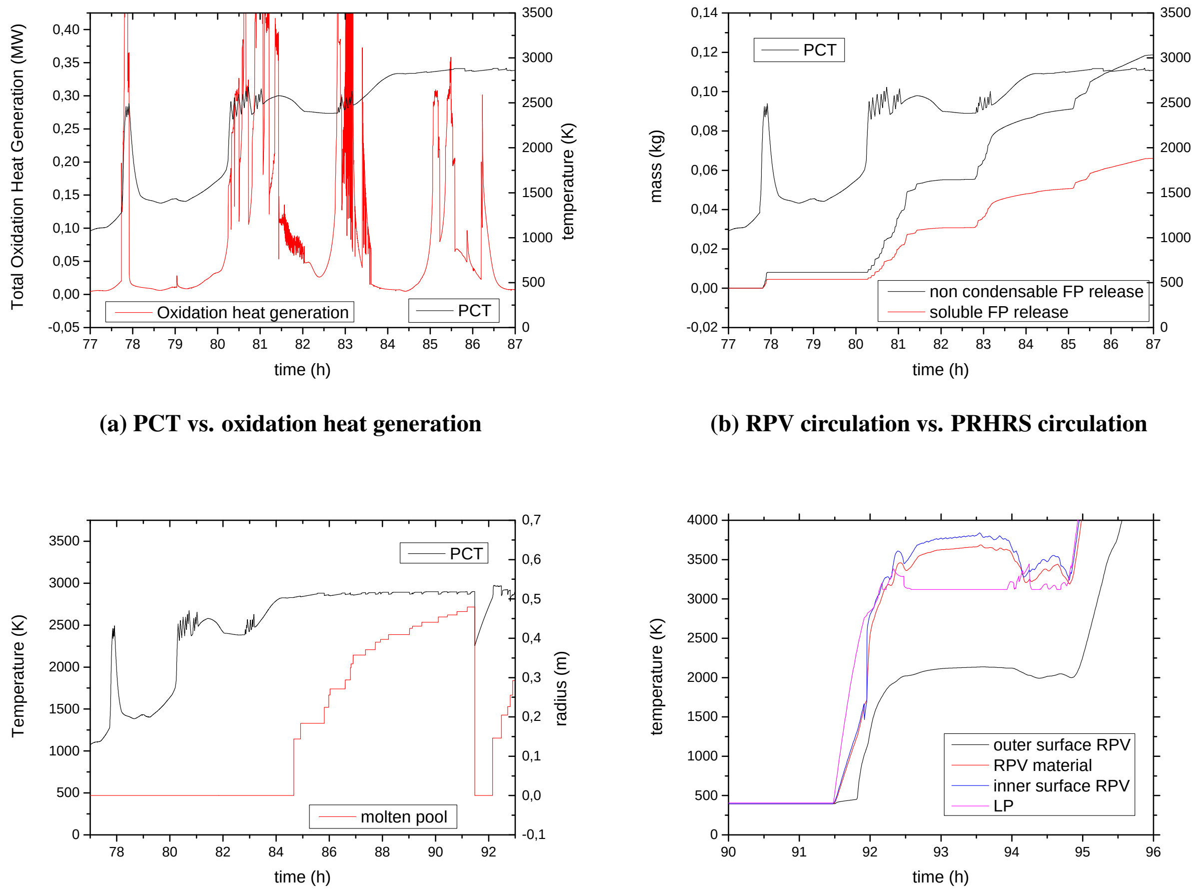
Figure 6. Assessment of the SBO phenomena for DBA conditions II

5. STATION BLACKOUT: ANALYSES OF THE SEVERE ACCIDENT PHENOMENA

RS35 with SCDAP components is used for the assesment of the Severe Accident phenomena (see Figure 7). SCDAP components allow to simulate core damage progression from oxidation to creep rupture. At around 74 hours after the SoT, a second core dryout occurs. When cladding temperatures increase above 1200 K, oxidation starts with spikes in the oxidation heat generation when temperatures achieve the 1477 K. As shown in Figure 7a, the heat generated by oxidation is aproximately 400 kW, that is equivalent to the 70 % of the core decay power (569 kW). As result of this, PCT increases up to 2400 K, causing balloning and rupture and the release of volatile and soluble fission products. This phenomenon will repeat subsequently at different core locations and heights as shown in Figure 7b.

After fuel rupture, oxidation heat generation is interrupted and PCTs are reduced by radiation, convection and axial conduction to other materials and fluid. At around 85 hours after the SoT, when PCTs achieve 2873 K, ceramic formation (U-Zr-O) occurs and molten pool is observed (see Figure 7c). Melted material is accumulated at the bottom of the core for 6 hours, when core slumping occurs.

Figure 7d shows the evolution of the lower plenum temperatures registered by COUPLE module when debris bed is placed at the bottom of the lower plenum. At 92 hours after the SoT, the inner layer of the RPV wall is damaged, penetrating part of the debris bed and increasing the temperatures at the external layer. Finally, at around 95 hours, external wall is damaged either, suddenly increasing the temperatures of the layer and causing the creep rupture. In this part of the simulation, results are not realistic as ex-vessel capabilities have not been included in the input nodalization. Hence, the simulation is considered as finished at 95 hours after the SoT.



(a) PCT vs. oxidation heat generation

(b) RPV circulation vs. PRHRS circulation

(c) PCT vs. molten pool radius

(d) Temperatures at the lower plenum wall of the RPV

Figure 7. Assessment of the SBO phenomena for Severe Accident conditions I

6. CONCLUSIONS

The simulation of an iPWR extended Station Blackout scenario for low pressure core damage sequences has been simulated with success for both RELAP5mod33 and RELAP/SCDAPSIM/MOD3.5 codes. The selected iPWR design was based in some of the public available data of CAREM-25 design. Results showed consistency and quite good agreement between codes, for both the main events and the relevant phenomena. In addition, Results of the simulations seem to confirm the 36 hours grace period for SBO scenario of the CAREM-25 design plus the extended 36 hours grace period associated to the availability of EIS in Loss of Coolant conditions reported by designer.

The analysis of the steady state results showed that SCDAP components seem to introduce additional form losses to the hydrodynamic components of the core. User defined K losses had to be reduced in order to achieve equivalent steady state conditions. In addition, temperatures at the oultet of the DC heat exchangers were slightly overpredicted by the codes. This deviation seems to be related with the lack of an specific RELAP5 model for the coil tubes. Further developments could be focussed in this field to improve not only the precision but also the efficiency of the calculations.

The results of the accident scenario showed some discrepancies in the RPV pressures at isolated conditions (between the loss of natural circulation and the initiation of the PRVs system). Discrepancies seem to be related with the different steam tables and water properties transport formulations of both RELAP5 and RELAP/SCDAPSIM codes. Regardless, such differences do not affect significantly to the timing of the CET signal, the EIS injection, and the final core dryout, hence both simulations can be considered as equivalent.

Finally, core damage progression was studied from oxidation to creep rupture. The use of the SCDAP components provided a friendly and fancy framework to assess and understand the relevant phenomena during severe accident sequence. In addition, RELAP/SCDAPSIM/MOD3.5 code also provided the capability to use Best Estimate tools to assess iPWR design and accident management strategies in one single execution, for all operation, DBA and SA conditions.

ACKNOWLEDGMENTS

This research is supported by the International Atomic Energy Agency under the Coordinated Research Project award I31033 on Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors, launched in 2019.

This work was also partially funded by CSN (Consejo de Seguridad Nacional) through the participation of UPC in CAMP Spain project.

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