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Review of Design Extension Conditions Experiments and Analyses for Non-degraded Core

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ABSTRACT

The second generation nuclear power plants were designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety during design basis accidents (DBAs). In the transient and accident analysis the effects of single active failures and operator errors were considered. There are also accident sequences that are possible but were judged to be too unlikely and therefore were not fully considered in the design process of second generation reactors. In that sense, they were considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand. Such accident sequences have been analysed in the past to fully understand the capability of a design.

The requirements to analyse such sequences for existing reactors have been introduced after Fukushima Dai-ichi accident. In 2012 the design extension conditions (DECs) were introduced in the International Atomic Energy Agency (IAEA) requirements for the design of nuclear power plants (NPPs). Western European Nuclear Regulators Association (WENRA) requirements of existing reactors for DEC were introduced in 2014. The purpose of considering DEC is to further improve safety by enhancing the plant's capability to withstand the conditions generated by accidents that are more severe than DBAs. This concept by IAEA and WENRA (WENRA definition of DEC is consistent with IAEA definition from 2012, in which DEC with prevention of core melt is called DEC A) is not completely new, since some multiple failures have already been considered in the design of existing reactors, for example anticipated transients without scram and station blackout. The research for beyond design basis accidents with non-degraded core (i.e. DEC A) for existing reactors has been already done in 80's and 90' of the previous century. The purpose of this paper is to review that research. The tests performed include total loss of feedwater, station blackout, small break without high pressure safety injection, steam generator tube rupture with no high pressure safety injection etc. Besides review of experiments performed on integral test facilities, examples of DEC A tests, which have been analysed at Jožef Stefan Institute using RELAP5 or TRACE computer code in the last three decades, will be presented too.

Keywords: *design extension conditions, RELAP5, TRACE, safety analysis*

1 INTRODUCTION

The existing reactors were designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety during design basis accidents (DBAs). In the transient and accident analysis the effects of single active failures and operator errors were considered. There are also accident sequences that are possible but were judged to be too unlikely and therefore were not fully considered in the design process of second generation reactors. In that sense, they were considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand. Such accident sequences have been analysed in the past to fully understand the capability of a design. They were called beyond design basis accidents (BDBA). However, after Fukushima Dai-ichi accident the International Atomic Energy Agency (IAEA) adopted term design extension conditions (DEC) [1].

The term "design extension conditions" has rather long history and was introduced during the design of the reactors of third generation. The DEC was introduced to define some selected accident sequences due to multiple failures. The design extension conditions were introduced as preferred method for giving due consideration to the complex sequences and severe accidents at the design stage without including them in the design basis conditions [3]. On the other hand, the Western European Nuclear Regulators Association (WENRA) recommended a "design extension" analysis in 2007 [4] and they proposed a list of events to be analysed at minimum. By its meaning this list corresponds to DEC without core melt. WENRA reference levels (RLs) from 2014 [5] introduced DEC term. The WENRA guidance document for issue F [6] explains that DEC in WENRA RLs are consistent with the definition of DEC in IAEA SSR-2/1 [1], published in 2012. DEC are more complex and/or more severe than conditions postulated as design basis accidents [6].

The paper [7] recommends that the IAEA requirements and guidelines keep up the definition of severe accidents so that this type of accident be clearly identified, linked to the partial or complete melting of reactor core. The IAEA DEC term has been redefined in 2016 as follows: "Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits." WENRA did not follow the new IAEA DEC, which modification is significant.

The DEC concept by IAEA and WENRA (DEC with prevention of core melt is called DEC A) is not completely new, since some multiple failures have already been considered in the design, for example anticipated transients without scram (ATWS) and station blackout (SBO). Also, the research for beyond design basis accidents with non-degraded core (i.e. DEC A) for existing reactors has been already done in 80's and 90' of the previous century.

In this paper review of that research is done. Besides review of experiments performed on integral test facilities, examples of DEC A tests, which have been analysed at Jožef Stefan Institute using RELAP5 or TRACE computer code in the last three decades, will be presented too.

2 INITIATING EVENTS FOR DEC A

WENRA guidance document [6] for issue F provides the following list of DEC A (with a note that final sets of conditions selected for DEC A analysis will be plant and site specific, developed on the basis of the following non-exhaustive list, which applies mainly to pressurized water reactors (PWR) and boiling water reactors):

- Initiating events induced by earthquake, flood or other natural hazards exceeding the design basis events (see Issue T [5]);
- Initiating events induced by relevant human-made external hazards exceeding the design basis events;
- Prolonged station black out (SBO; for up to several days);
  - SBO (loss of off-site power and of stationary primary emergency alternate current (AC) power sources)
  - total SBO (SBO plus loss of all other stationary AC power sources), unless there are sufficiently diversified power sources which are adequately protected
- Loss of primary ultimate heat sink, including prolonged loss (for up to several days);
- Anticipated transient without scram (ATWS);
- Uncontrolled boron dilution;
- Total loss of feed water;

- Loss of coolant accident (LOCA) together with the complete loss of one emergency core cooling function (e.g. high pressure injection (HPI) or low pressure injection (LPI));
- Total loss of the component cooling water system;
- Loss of core cooling in the residual heat removal mode;
- Long-term loss of active spent fuel pool cooling;
- Multiple steam generator tube ruptures (PWR, pressurized heavy water reactors);
- Loss of required safety systems in the long term after a design basis accident.

The IAEA document [8] states that the list of DEC may include:

- ATWS;
- SBO;
- Loss of core cooling in the residual heat removal mode;
- Extended loss of cooling of fuel pool and inventory;
- Loss of normal access to the ultimate heat sink.

The IAEA document [8] further provides an example list of additional DECs derived from probabilistic safety assessment (PSA):

- Total loss of feed water;
- LOCA plus loss of one emergency core cooling system (either the high pressure or the low pressure emergency cooling system);
- Loss of the component cooling water system or the essential service water system (ESWS);
- Uncontrolled boron dilution;
- Multiple steam generator tube ruptures (MSGTR) (for PWRs);
- Steam generator (SG) tube ruptures induced by main steam line break (MSLB) (for PWRs);
- Uncontrolled level drop during mid-loop operation (for PWRs) or during refuelling.

When comparing the WENRA and IAEA list, first major difference is that WENRA list includes initiating events induced by earthquake, flood or other natural hazards exceeding the design basis events. However, IAEA document [8] stressed that "some Member States tend to include in the list of DECs also some external hazards that were not considered in the past (e.g. earthquake exceeding the design basis earthquake, commercial air craft impact, etc.). In the IAEA terminology, a DEC is a postulated plant state (see Table 1) that is determined by a postulated sequence of events, and for the same reasons that design basis hazards are not considered DBAs, more severe hazards are not considered DECs although they might result in a DBA or possibly in DEC." Second difference is that IAEA provides deterministically and probabilistically identified list.

3 REVIEW OF RESEARCH ON BDBA WITHOUT CORE DEGRADATION FOR EXISTING REACTORS

In this section selected BDBA experiments without core degradation are briefly described in Tables 1 and 2.

Table 1 shows tests for accident management in PWRs, in which operator actions were studied for BDBA with non-degraded core (DEC A). Experiments were mainly selected from cross-reference matrix for accident management for non-degraded core, which has been created in the frame of OECD/NEA [9].

Table 1: Accident management in PWRs for BDBA with non-degraded core (DEC A)

Test No.	Test type	Brief description
PKL Ill B1.2	Total loss of feedwater with secondary side feed and bleed	Total loss of feedwater (loss of main and auxiliary feedwater) with no core cooling systems (high and low pressure injection pumps and accumulators) was studied. Secondary side bleed and feed was performed. Injection of water was due to flashing in feedwater line and subsequent injection by a mobile pump [9].
BETHSY 5.2c2	Total loss of feedwater	During BETHSY (Boucle d'Etudes Thermohydrauliques de Systemes) test 5.2c2 [10], the emergency operating procedure (EOP) was conducted in accordance with the rules presently implemented in plant control rooms, which allow operators more time for the recovery of feedwater systems: it consisted in manually starting the high pressure injection system (HPIS) as soon as 2 SG liquid levels reached 3 m; as a consequence, primary pressure slowly increases up to 16.3 MPa, and is then maintained at this value through pressurizer power operated relief valves (PORVs) automatic operation. 30 minutes after EOP initiation, or earlier if the core outlet fluid temperature reaches 603 K, the pressurizer PORVs are actuated at full discharge capacity.
BETHSY 6.2TC	6” cold leg break without HPIS and LPIS	BETHSY 6.2TC test was a 15.24 cm (6 inch) cold leg break in the loop one without available high pressure and low pressure safety injection system. Accumulators were available in the intact loops. The main aims of this test were to compare the counterpart test data from BETHSY and Large Scale Test Facility (LSTF) facilities and qualification of CATHARE 2 computer code.
BETHSY 9.1b	2" cold leg break without HPIS and with delayed ultimate procedure	BDBA involves two failures: a break on the cold leg together with a complete failure of the HPIS, combined with a human error regarding the conditions in which the operators start the Ultimate Operating Procedure (UOP). The UOP then consists in depressurizing the primary circuit by means of a full opening of the 3 SG atmospheric steam dumps.
BETHSY 9.3	SGTR with HPIS and AFW unavailable	The simultaneous failure of the high pressure safety injection and auxiliary feedwater systems is a Beyond Design Basis Accident, which leads to core heat up, if no additional measures are taken. During the test 9.3 the efficiency of both the steam generator atmospheric steam dump and the depressurization of the primary circuit via the pressurizer relief valve is investigated [11].
LSTF	PWR Cold-Leg small-break LOCA with total HPI failure	Cold-leg break tests were conducted at the LSTF for five break areas 0.5%, 1%, 2.5%, 5 and 10% of the scaled cold-leg flow area, with totally failed HPI [12].
LSTF	0.5% cold leg small-break LOCA total failure of the HPI and auxiliary feedwater (AFW) systems	The depressurization procedure was simulated in a 0.5% cold-leg break LOCA experiment [13].
LSTF SB-SG-11	SGTR concurrent with secondary break	In this experiment, the pressure difference between the primary side of the steam generator (SG) and the secondary side of SG is kept so high that the two-phase critical flow is observed for a long time. The secondary break was simulated for the feedwater line because this was the only line which can be connected to the break catch tank (ST). The secondary initial level for the affected SG was lowered to 4.3m to scale the inventory. The recovery action was simulated by depressurizing, starting 600 s after scram. Also, the pressurizer auxiliary spray was activated subsequently [9].

New design includes design features aimed at preventing the onset of a severe accident, including severe accident precursors identified in SECY-90-016 [18] and SECY-93-087 [19]: ATWS, mid-loop operation, station blackout (SBO) event, fire, and an intersystem loss of coolant accident (ISLOCA). Similarly WENRA [20] provides examples of multiple failure scenarios (DEC A) to be prevented in new designs: LOCA, station blackout, total loss of feedwater and ATWS.

Table 2 shows typical multiple failure scenarios (ATWS, mid-loop operation, SBO and LOFW followed by small break LOCA).

Table 2: BDBAs with non-degraded core (DEC A), which typically need to be prevented

Test No.	Test type	Brief description
LSTF TR·LF-06	Pump seal leak following station blackout	The test simulated a pump seal leak following a station blackout (or TMLB', where T = transient event, M = failure of the secondary system steam relief valves and power conversion system, L = failure of secondary system steam relief valves and auxiliary feedwater system, and B' = failure to recover onsite and offsite AC power) transient. The test was initiated with an "accelerated transient" which was designed to obtain primary and secondary states including: steam generator (SG) secondary sides dried out; primary side reached saturation at the pressurizer power operated relief valve (PORV) opening setpoint pressure. After these states were reached, at a scaled core power of 1.2%, a cold leg break, with an area of 0.1% of the scaled cold leg cross-sectional area, was opened to simulate a pump seal leak [9].
LOBI A2-90	SBO-ATWS	LONOP-ATWS or "SBO" anticipated transient caused by loss of offsite and normal onsite electrical power (LONOP) with failure to scram [17].
BETHSY 6.9c*	Loss of RHR at mid-loop operation with pressuriser and SG manways open	The test includes a loss of residual heat removal (RHR) system during mid-loop operation at 0.5% of nominal value core power. Initial liquid level in reactor coolant system (RCS) was at horizontal axis of the hot legs. Pressurizer and steam generator manways were opened 1 s after the transient was initiated [9].
LOFT L9-1 / L3-3	Total loss of feedwater (LOFW) accident followed by small break LOCA	Experiment L9-1 was the first anticipated transient with multiple failures performed at Loss-of-Fluid-Test (LOFT), and consisted of a simulated LOFW accident with delayed reactor scram and no auxiliary feedwater injection. Experiment L3-3 simulated two independent recovery procedures from the LOFW accident L9-1, without engaging the emergency core coolant (ECC).
LOFT L9-3	Loss of feedwater without reactor trip	Experiment L9-3 conducted in the LOFT facility was a unique one simulating an ATWS event in pressurized water reactor. The experiment simulated a loss of feedwater induced ATWS in a commercial plant. The experiment consisted of two parts: the ATWS itself, which lasted about 600 s, and the plant recovery [15].
LOFT L9-4	Loss-of-offsite-power accident without reactor trip	This was an anticipated-transient-without-scram test initiated from typical commercial PWR operating conditions in which the primary coolant and main feedwater pumps, the steam generator main feedwater discontinued, and the main steam-outlet valve closed. Auxiliary feedwater was initiated after a delay of 10 s to simulate the start-up time of the diesel generators, and the pressuriser PORV and spray were both inoperative throughout the transient [16].

* - low power operation

4 REVIEW OF BDBA (DEC A) SIMULATIONS AT JOŽEF STEFAN INSTITUTE

Results of selected simulations of experiments described in Tables 1 and 2 are presented. This includes BETHSY 9.1b, 6.2TC and 6.9c tests, and LOFT L9-1/L3-3 test. The scenarios with multiple failures simulated for Krško Nuclear Power Plant are not in the scope of this paper (e.g. references [21] through [27]).

BETHSY was an integral test facility, which was designed to simulate most pressurized water reactor accidents of interest, study accident management procedures and validate the computer codes. It was a scaled down model of three loop Framatome (now AREVA NC) nuclear power plant with the thermal power 2775 MW.

The LOFT facility was a 50 MW_{th} two-loop PWR, which was designed to study the thermo-hydraulic response of the system to a variety of simulated LOCA scenarios. The facility incorporated similar hydraulic components to those in commercial PWRs, although the components were volumetrically scaled by a ratio of 1/60 in comparison to a full-scale commercial PWR with a power of 3000 MW_{th}. Inherent in the scaling are some compromises in the geometric similarity. In particular, the 1.7m-long LOFT reactor core was around half the length of that of a commercial PWR, but the Emergency Core Coolant (ECC) system was designed to inject a similar amount of core coolant in the event of an LOCA.

4.1 Simulation of BETHSY 9.1b

The Bethsy 9.1.b test is a scaled 5.08 cm cold leg no. 1 break without high pressure safety injection and with delayed operator action for secondary system depressurization. Due to core heatup the operator depressurized the secondary side by atmospheric relief steam dump valves. In the simulation this operator action was delayed. The test was analyzed in the frame of international standard problem 27 (ISP-27) performed to validate the thermalhydraulic computer codes. The test scenario was the following: break was opened in the cold leg no. 1 (initiation of the transient). When the maximum heater rod cladding temperature reaches 723 K, the ultimate procedure was started by opening three steam line dumps to atmosphere. When pressurizer pressure dropped below 4.2 MPa accumulators started to inject and they stopped to inject below 1.46 MPa. The low pressure safety injection system was activated when the primary pressure was below 0.91 MPa. When stable residual heat removal system operating conditions prevail, the transient was terminated.

The aim of the study [28] was to perform calculations with to Jožef Stefan Institute (JSI) available RELAP5 versions using as much as possible the same input model in order to see the differences between the code versions. As it is difficult to compare so many calculations, line colors are selected in such way that MOD3.3 versions have green color palette, MOD3.2 are in red and pink and MOD3.1 has blue palette. Pressurizer pressure and maximum heater rod temperature are shown in Figure 1 and Figure 2, respectively. As high pressure injection is not available, the core starts to uncover and when maximum heater rod cladding temperature reaches 723 K, the ultimate procedure was started by opening three steam line dumps to atmosphere, in the calculations a bit earlier than in the experiment. This causes secondary pressure decrease, followed by primary system pressure decrease. When primary system dropped below the accumulator injection setpoint, the injection started and soon the clad temperature started to decrease. Again the heatup in the calculation is earlier than in the experiment. Later the accumulators are emptied, however cooling is established through the secondary side, and therefore the primary pressure is decreasing. When reaching the low pressure injection system setpoint, the low pressure injection started and the experiment lasted until the stable residual heat removal system conditions were reached. From results it may be seen that secondary side depressurization prevented core heatup as primary pressure drops below accumulator injection.

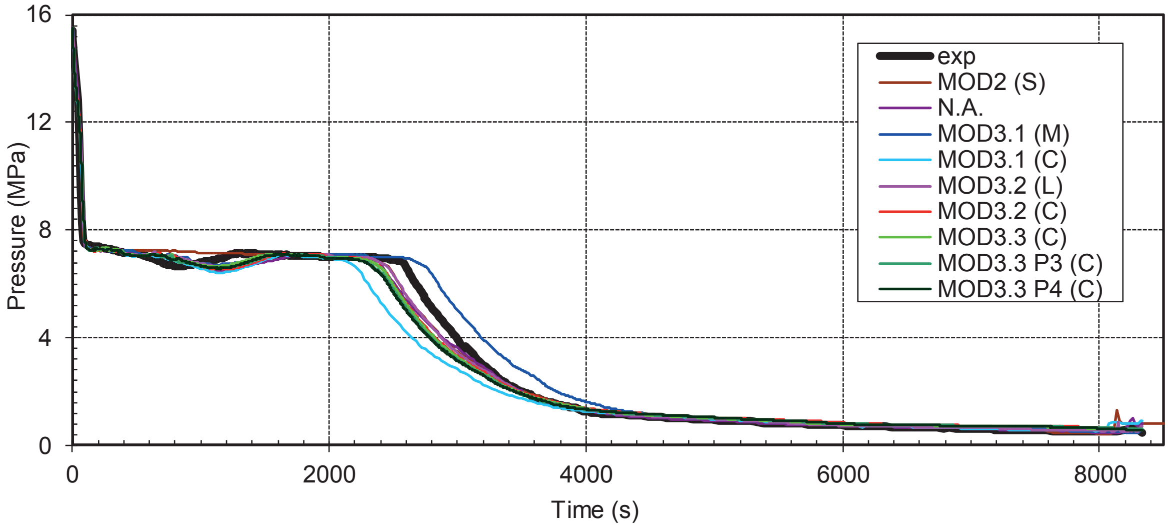

Figure 1: Pressurizer pressure – BETHSY 9.1b

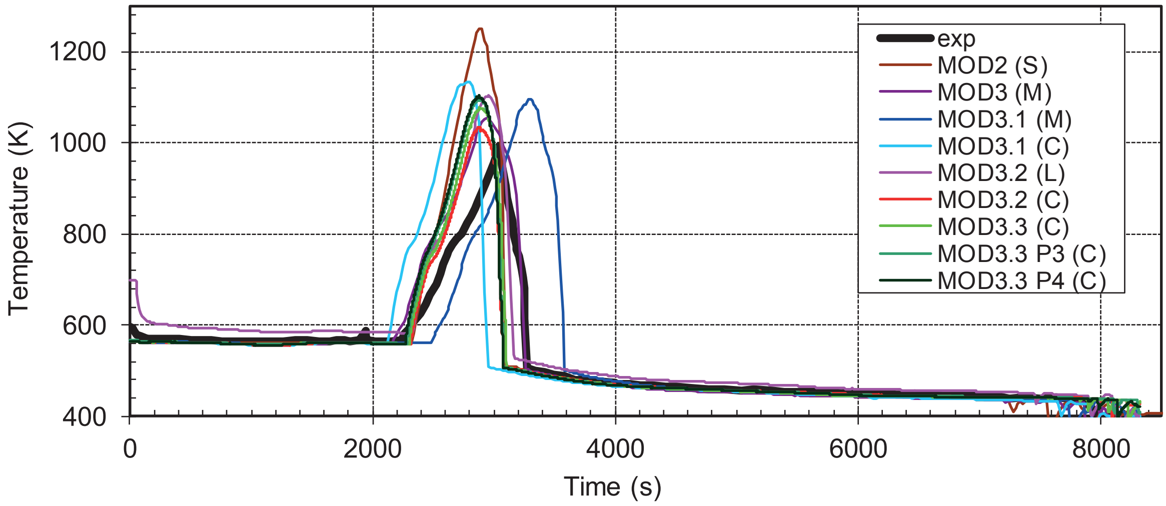

Figure 2: Maximum heater rod cladding temperature – BETHSY 9.1b

4.2 Simulation of BETHSY 6.2TC

BETHSY 6.2TC test was a 15.24 cm (6 inch) cold leg break in the loop no. 1 without available high pressure and low pressure safety injection system. The experiment started with opening of the valve simulating the break in the cold leg no. 1 at the time 0 s. Sudden primary pressure drop caused scram signal when pressure was below 13.0 MPa and safety injection (SI) signal was generated, when primary pressure was below 11.7 MPa. At scram signal all three primary pumps were stopped and natural circulation regime took over the primary system. The hot parts of the primary circuit (upper head, upper plenum, SG U-tubes) started to boil. The formation of loop seal caused the core level depression. The drop in the core collapsed liquid level was stopped at 134 s by loop seal clearance on the three loops. The loop seal clearance occurred at the same time on all three loops. After loop seal clearance the core liquid level rose again due to pressure balances and then started to drop again due to inventory loss through the break. When primary pressure dropped below 4.2 MPa, the accumulator injection started, which recovered the core. The accumulator injection was stopped on the basis of low level criterion. After it stopped, in the absence of high pressure injection, the primary circuit emptied through the break and third core uncovery occurred. The low pressure injection was not activated by assumption. The test was ended when the primary pressure dropped below 0.7 MPa.

The results of simulation [29] are shown for pressurizer pressure and maximum heater rod temperature in Figure 3 and Figure 4, respectively. For RELAP5 original Ransom-Trapp break flow model the values of 0.85, 1.25 and 0.75 were used for subcooled, two phase and superheated discharge coefficient, respectively. For TRACE break model the values of 0.8 and 0.9 were used for subcooled and two phase discharge coefficients, respectively. The pressure drop (see Figure 3) is faster in case of TRACE calculation than in the experiment, while in the case of RELAP5 is slower. In the case of heater rod surface temperature (see Figure 4) the timing of heatup prediction was better in the case of TRACE, while heatup rate was better predicted in the case of RELAP5. It may be seen that due to unavailability of high and low pressure injections systems the core heatup would continue, if test would not be ended. In such a case new engineered safety feature for primary injection would be needed to prevent core heatup.

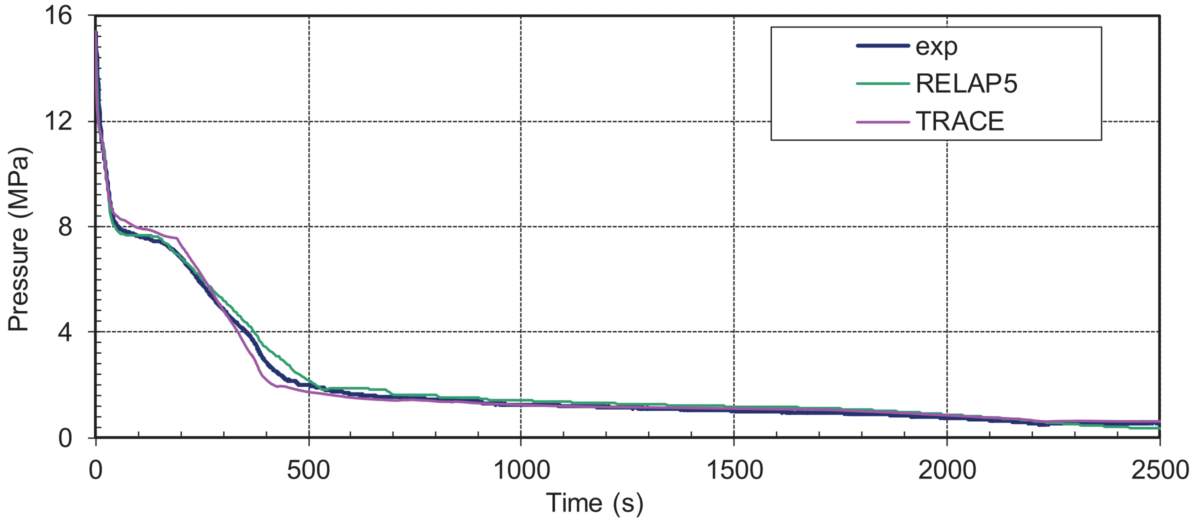

Figure 3: Pressurizer pressure – BETHSY 6.2TC

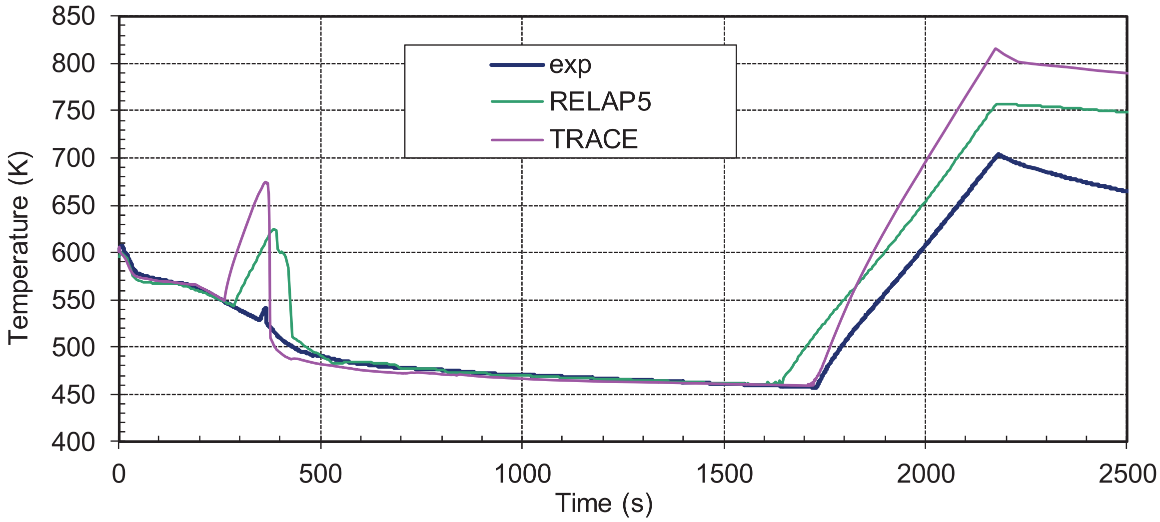

Figure 4: Maximum heater rod cladding temperature – BETHSY 9.1b

4.3 Simulation of BETHSY 6.9c

Test 6.9c OECD ISP-38 includes a loss of RHR system during mid-loop operation at 0.5% of nominal value core power. Initial liquid level in RCS was at horizontal axis of the hot legs. Pressurizer and steam generator manways were opened 1 s after the transient was initiated. Boil away and liquid entrainment through manways are in that case the physical phenomena which mainly determine the RCS behaviour while both the presence of non condensable gas above the liquid level and heat removal by SG's play a minor role. The initial conditions for this tests are: RCS at atmospheric pressure with a liquid level at mid height of hot legs, fluid and structure temperatures close to 373 K in the whole RCS (the liquid heat up phase was ignored in test), and the SG secondary sides are filled with air and isolated. Manways are simulated by geometrically scaled orifices with the same form loss coefficient.

At the start of the test the water in the primary circuit was at the centre line level of the hot legs and very close to the saturation temperature. The manways in the pressuriser and steam generator were opened and the core power increased to 140kW. Boiling occurred almost immediately.

Over the first 3000 s of the transient, water was entrained·into the surgeline and then carried on up into the pressuriser by the high steam flow rate. It accumulated in the pressuriser. The accumulated water was not held there continuously, but twice it flowed back to the hot leg and then partially refilled. Finally as the mixture level in the vessel fell the pressuriser and surge line emptied completely.

During this period also there was water entrained into the vertical part of the hot leg and the steam generator inlet plenum and tubes. These also emptied when the mixture level fell below the hot legs.

After about 6000 s the level fell sufficiently for the core to become uncovered and the temperature of the fuel rod simulators to rise. When the temperature rose above 523 K, emergency core cooling was initiated by a (simulated) gravity driven feed. This was sufficient to halt the core heat-up and re-establish the primary circuit inventory. The test was stopped when the level in the vessel reached the mid loop condition. The total test time was nearly 10000 s. The simulated mass calculated by different computers is shown in Figure 5. The experimental line is blue.

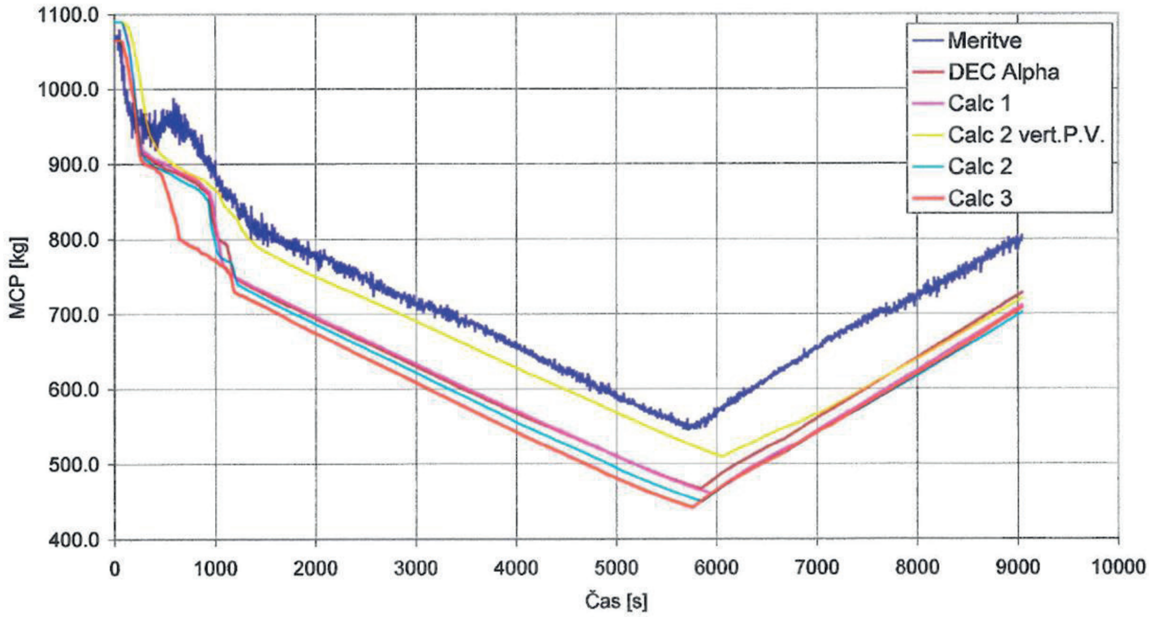

Figure 5: Mass in primary system – BETHSY 6.9c

The calculations were performed on DEC Alpha and SUN Sparc workstation (labelled “DEC Alpha” and “Calc1”) using base input deck and modified input models (“Calc 2 vert.P.V”, “Calc 2” and “Calc 3”). From the experiment and simulated results it may be seen that without gravity driven injection the core would continue to uncover. By GL 88-17 [30] the following enhancements have been recommended for mid-loop operation such as training, temperature and level indications, implementation of procedures and controls, and at least two available or operable means of adding inventory to the RCS that are in addition to pumps that are a part of the normal RHR systems.

4.4 Simulation of LOFT L9-1/L3-3

The LOFT experiment L9-1/L3-3 tested the system response to an anticipated transient with multiple failures (L9-1) followed by a small-break LOCA (L3-3) due to the failure of a power-operated relief valve (PORV).

Experiment L9-1 was the first anticipated transient with multiple failures performed at LOFT, and consisted of a simulated LOFW accident with delayed reactor scram and no auxiliary feedwater injection. The LOFW accident was initiated due to the failure of the main feedwater pump, leading to the loss of coolant through the PORV, which resulted in a LOCA.

Experiment L3-3 simulated two independent recovery procedures from the LOFW accident L9-1, without engaging the emergency core coolant (ECC). The first recovery mode involved latching open the PORV to depressurize the primary system whilst simultaneously turning off the primary coolant pumps. The second mode consisted of refilling the steam generator (SG) and removing excess decay heat through a feed-and-bleed operation of the SG secondary side.

In short, the LOFT experiment L9-1/L3-3 was conducted to evaluate the effectiveness of the PORV cycling and the subsequent feed-and-bleed operation using the secondary side for removal of decay heat.

The simulation was performed by RELAP5/MOD3.3 Patch 04. The transient conditions at 1690 s is displayed in Figure 6, clearly indicating that the pressurizer has completely filled with fluid when the PORV cycling is initiated. Figure 7 depicts the system at 7050 seconds, upon the initiation of the feed-and-bleed operation in the secondary loop.

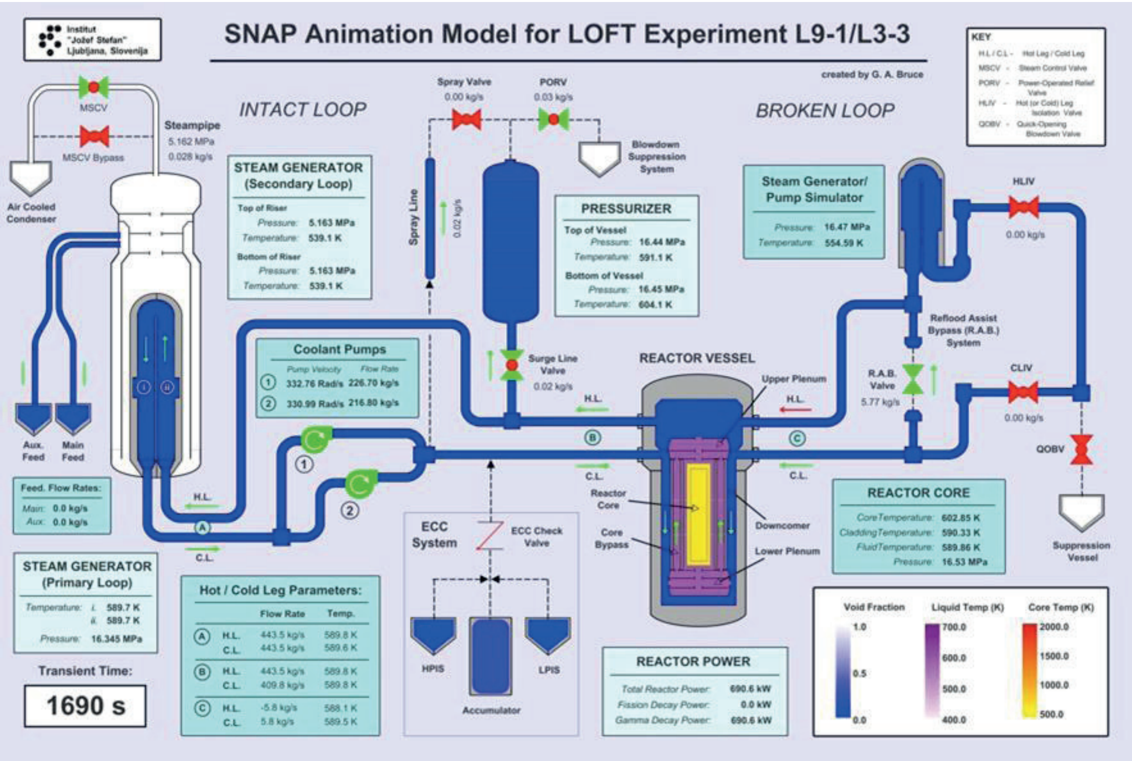

Figure 6: Spray valve closed and PORV cycling initiated (t=1690 s)

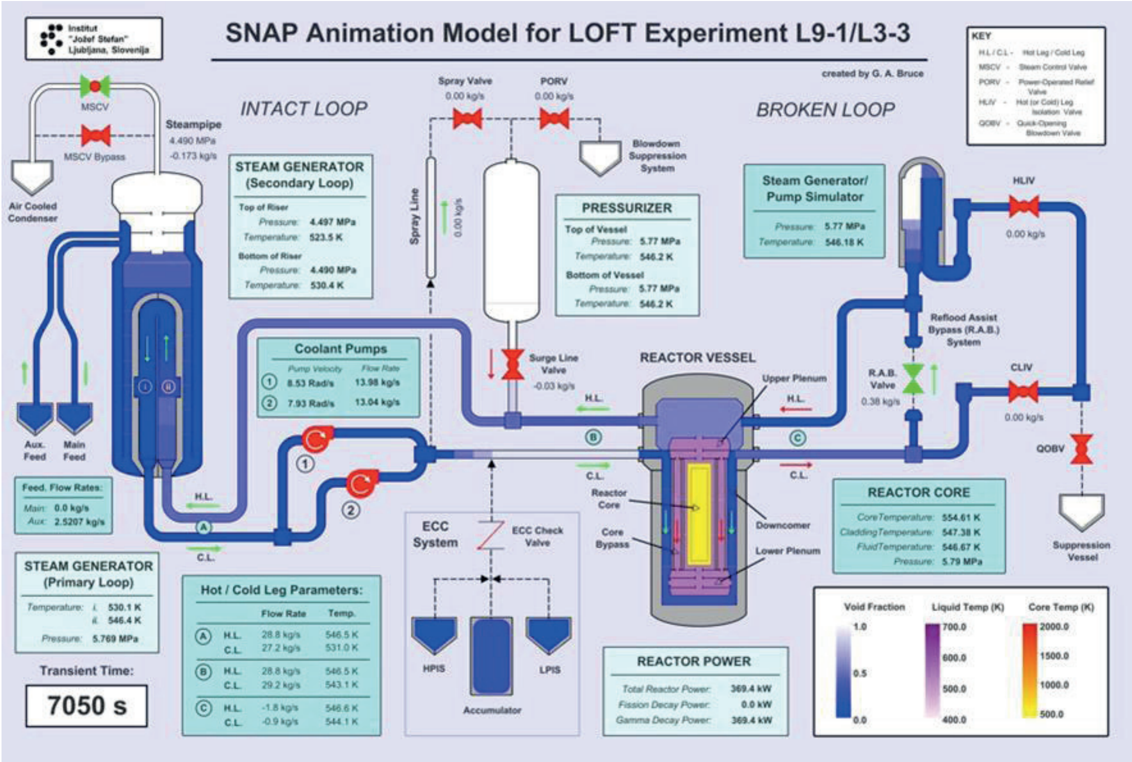

Figure 7: SG secondary refilled, feed-and-bleed operation initiated (t=7050 s)

5 CONCLUSIONS

The requirements to analyse design extension conditions for existing reactors have been introduced after Fukushima Dai-ichi accident. The purpose of considering design extension conditions (DEC) is to further improve safety by enhancing the plant's capability to withstand the conditions generated by accidents that are more severe than design basis accidents (DBAs). The paper first provides example lists of DEC proposed by International Atomic Energy Agency (IAEA) and Western European Nuclear Regulators Association (WENRA). Then, research for beyond design basis accidents with non-degraded core (i.e. DEC A) for existing reactors done in 80's and 90' of the previous century is presented. The tests performed include total loss of feedwater, station blackout, small break without high pressure safety injection, steam generator tube rupture with no high pressure safety injection etc. Finally, simulations of few experiments (representing DEC A) tests performed on integral test facilities, which have been analysed at Jožef Stefan Institute using RELAP5 or TRACE computer code in the last three decades, have been presented. The review of beyond design basis accidents performed on integral test facilities and simulations suggest that selected DEC scenarios were studied well before the requirements on DEC analyses have been made. Also, before the Fukushima Dai-ichi accident several existing plants have already implemented certain measures to prevent severe accidents from multiple failures (e.g. station blackout or anticipated transients without reactor scram).

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