International Conference
Nuclear Energy for New Europe 2007
Portorož /Slovenia /September 10-13

port2007@ijs.si
www.nss.si/port2007
+386 1 588 5247, fax +386 1 588 5376
PORT2007, Nuclear Society of Slovenia, Jamova 39, SI-1000 Ljubljana, Slovenia

MELCOR and SCDAP Analyses of Loss-of-Coolant Accidents During Cooldown in a Westinghouse 2-Loop PWR

T Haste, J Birchley
Paul Scherrer Institut
5232 Villigen PSI, Switzerland
tim.haste@psi.ch, jonathan.birchley@psi.ch

M Richner
Nordostschweizerische Kraftwerke (NOK)
Kernkraftwerk Beznau, CH-5312 Döttingen, Switzerland
martin.richner@nok.ch

ABSTRACT

Operation of pressurised water reactors involves shutdown periods for refuelling and maintenance. In preparation for this, the reactor system is cooled down, depressurised and partially drained. Although reactor coolant pressure is lower than during full-power operation, there remains the possibility of a Loss-of-Coolant Accident (LOCA), with a certain but low probability. While the decay heat to be removed is lower than that from a LOCA at full power, the reduced availability of safety systems implies a risk of failing to maintain core cooling, and hence of core damage. This is recognised though probabilistic safety analyses (PSA), which identify low but non-negligible contributions to core damage frequency from accidents during cooldown and shutdown.

Analyses are made for a typical two-loop Westinghouse PWR of the consequences of a range of LOCAs during hot and intermediate shutdown, 4 and 5 hours after reactor shutdown respectively. The accumulators are isolated, while power to some of the pumped safety injection systems (SIs) is racked out. The study assesses the effectiveness of the nominally assumed SIs in restoring coolant inventory and preventing core damage, and the margin against core damage where their actuation is delayed.

The calculations use the engineering-level MELCOR1.8.5 code, supplemented by the SCDAPSIM and SCDAP/RELAP5 codes, which provide a more detailed treatment of coolant system thermal hydraulics and core behaviour. Both treatments show that the core is readily quenched, without damage, by the nominal SI which assumes operation of only one pump. Margins against additional scenario and model uncertainties are assessed by assuming a delay of 900 s (the time needed to actuate the remaining pumps) and a variety of assumptions regarding models and the number of pumps available in conjunction with both MELCOR and versions of SCDAP.

Overall, the study provides confidence in the inherent robustness of the plant design with respect to LOCA during cooldown to cold shutdown, and in the validity of a two-tier calculational method. The results have been directly used in updating the plant shutdown PSA, by changing the success criteria for core cooling during cooldown of the plant and showing a reduction in overall risk.

1 INTRODUCTION

Operation of a Pressurised Water Reactor (PWR) includes periods of shutdown, for refuelling and maintenance. In preparation for this, the coolant temperature is progressively reduced by operation of the Steam Generators (SGs), while maintaining both inventory and subcooling via appropriate use of the make-up system and pressuriser heaters. Two states are identified: hot and intermediate shutdown, which are reached at approximately four and five hours respectively after reactor shutdown. At this stage the RCS is still at pressure and heat is removed by the Reactor Coolant Pumps (RCPs) and SGs. The possibility of a Loss-of-Coolant-Accident (LOCA) exists, as it does during power operation, but during shutdown not all the Emergency Core Cooling Systems (ECCS) are operational: The accumulators are necessarily isolated, while power distribution to some of the pumped Safety Injection (SI) systems is racked out.

This paper summarises analyses of postulated LOCAs in the Beznau (KKB) PWR, occurring during hot (HS) and intermediate (IS) shutdown (nominally 4h and 5h after shutdown respectively), concentrating on large break LOCAs during hot shutdown, as these pose the greatest challenge to the plant safety systems. The objectives of the study are to demonstrate effectiveness of the nominally assumed SI in restoring coolant inventory, restricting core temperatures and preventing core damage, and to assess the additional margin against core damage in cases where SI actuation is delayed. The temperature criterion used is taken from the USA 10CFR50 Appendix K limit of 1204 °C (1477 K), which is commonly adopted for licensing of Western PWRs. For the relatively rapid transients investigated here, it is considered that the limit of 1204 °C conservatively bounds the PSA success criterion.

This paper is a companion to one on loss of residual heat removal in mid-loop operation [1], and employs essentially the same modelling approach summarised there. The following section briefly describes the plant and assumed accident sequences, concentrating mainly on aspects specific to the present study. Sections 3 and 4 summarise respectively the analytical tools and calculated results. The main conclusions are presented in section 5.

2 BRIEF DESCRIPTION OF PLANT AND POSTULATED SEQUENCES

The reference plant is a Westinghouse (W) two-loop PWR of which two identical units are operated at Beznau, Switzerland. Since the start of operation, the Steam Generators (SGs) were replaced by Framatome units of greater capacity, while certain other engineered safeguards, in particular the Safety Injection (SI), were uprated to provide additional redundancy and capacity. Passive autocatalytic recombiners (PARs) were also installed with the objective of avoiding a hydrogen burn and the associated loading on the containment. The nominal plant operating parameters are given in Table 1.

Table 1: Nominal operating parameters

Parameter	**Value**	**Parameter**	**Value**
Core power	1130 MW	Primary coolant flow	6640 kg/s
Reactor coolant system pressure	15.5 MPa	Pressuriser level (above hot leg centreline)	8.5 m
Hot leg temperature	585.9 K	Secondary side pressure	5.55 MPa
Cold leg temperature	554.6 K	Steam flow rate	604 kg/s

Large (LB), intermediate (IB) and small (SB) LOCA sequences were calculated. The large break is defined, as usual, as a 200% offset shear in one of the cold legs such that the flow from each side of the opening does not affect the flow from the other side. The intermediate and small breaks are defined as 20 cm and 3 cm downward-facing breaches respectively. In each case the location is assumed to be between the SI location and the Reactor Pressure Vessel (RPV), which is believed to be the most penalising as regards spillage of injected coolant.

The RCPs are running at nominal speed at the time of LOCA initiation. It is assumed in most cases that the RCPs are tripped on reduced collapsed level in the intermediate leg; a few sensitivity studies were made with the RCPs assumed running. As indicated above, there is restricted ECCS availability during the approach to cold shutdown. The accumulators are isolated, while the breakers which deliver power to injection pumps JSI 1-A, 1-B and 1-C, which inject to the cold legs and upper plenum, are racked out. JSI 1-D remains available, however, to inject coolant to the cold legs. A 30 s delay on start-up of the SI pumps is assumed in all cases, over and above any additional delay assumed. The nominal scenario, therefore, is LOCA followed by early pump trip and injection via JSI 1-D after 30 s. Despite the significantly reduced injection compared with a LOCA at full power, the lower decay heat level and absence of significant stored heat in the fuel means that the depletion would not lead to such an acute heating of the core.

To assess further any extra margin against core damage, more pessimistic scenarios were considered in which JSI 1-D is assumed unavailable at the time of LOCA initiation. SI was further delayed on the basis that, in the event of failure of the power distribution system or of JSI 1-D itself, a period of ca. 15 minutes is regarded as necessary to rack in additional switchgear and actuate the SI pumps. One study of the small break LOCA was performed to assess the time window for avoiding core damage in the event of delayed SI.

3 TOOLS AND TECHNIQUES

3.1 Codes used

The primary analysis tool was the engineering-level MELCOR code [2], which is established in Switzerland as the main code for beyond-design-basis accident analysis, employing the production version at the time, 1.8.5QZ. The code comprises, typically, simple empirical correlations or parametric statements, and is frequently used in conjunction with coarse-mesh input models. As LOCA sequences can exhibit strong interaction between the liquid water and steam phases, the MELCOR analyses are supported by SCDAP simulations which provide a more complete treatment of the two-phase hydrodynamics. Different versions of SCDAP are used here, namely SCDAP/RELAP5/MOD3.2 [3] and the direct descendent, SCDAPSIM [4]. These provide similar models of the hydrodynamics, but with differences in numerical implementation. Both were used here, to check for code dependency, using versions MOD3.2ia and MOD3.4bi3 respectively.

3.2 Plant models

The basic reference data are given in report [5]. The MELCOR input model is a coarse node representation of the hydraulic system and structures, comprising about 40 fluid cells for the entire system including the secondary sides and containment, of which about 20 cells are used for the RCS. The containment noding comprises the cavity, lower and upper compartments, and annular compartment. The RPV downcomer, lower plenum, core, core bypass and upper plenum (including the upper head) are each represented by a single node. Single nodes are used for each of the hot legs, SG up- and down-sides and the crossover legs. The cold leg in loop A is subdivided into two hydraulic nodes to allow representation of the double-ended cold leg break. The RCS is isolated from the secondary sides, but is connected to the containment via the pressuriser vent line and the pressuriser Safety/Relief Valves

(SEBIM-Valves). The reactor core components include the fuel and control rods, spacer grids, the upper and lower structures, which occupy the hydraulic nodes that represent the core and lower plenum. The active, 3.05 m long core is subdivided into seven equi-length axial nodes, while the non-active components, mainly the lower grid plate and support structures are divided into four axial nodes. The entire core is divided radially into three zones. Further details are given in [1].

The SCDAP model, Figure 1, is more detailed than that used for MELCOR, in the spirit of the more mechanistic nature of the physical treatment. About 160 thermal hydraulic volumes are used in all, about 55 in the vessel region. The containment is specified as a boundary condition, using time-dependent volumes. For the vessel, a three-channel model is employed, representing a central zone, an intermediate zone and an outer zone, with 37, 52 and 32 assemblies represented respectively. Eleven axial nodes are used for the thermal hydraulic representation here, with nine nodes for the downcomer and six for the core bypass. The core channels are connected by crossflow junctions for representation of internal circulation within the vessel.

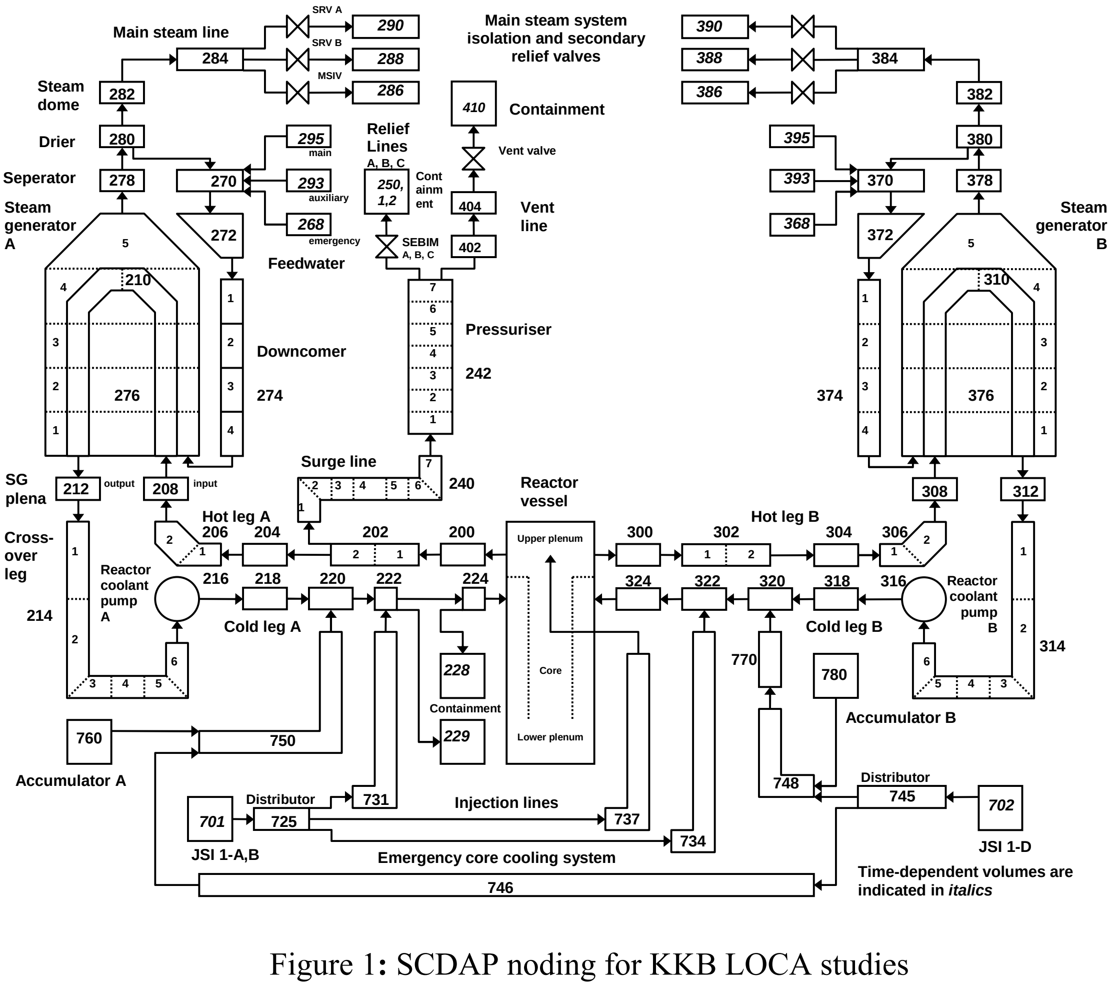

Figure 1: SCDAP noding for KKB LOCA studies

A feature of the plant state during shutdown is that the cooldown is taking place and may be ongoing at the time of the LOCA. The simulation is more complicated to control and interpret if the transient is initiated from a non-steady state. An approximation was made for the large and intermediate break cases in that the secondary side conditions were frozen for a period prior to the break opening, thus running a null transient with constant boundary conditions until a steady state was reached (typically after 2000 s). The timings of LOCA initiation correspond to the same decay heat as would apply at the nominal conditions that would occur following the actual operation in which the reactor fission power is gradually reduced to zero. Initial conditions are summarised in Table 2.

Table 2: Initial conditions for hot and intermediate shutdown LOCA

Parameter	**Hot**	**Intermediate**
Primary system (hot leg) pressure/ temperature (bar/K)	70.5/520.3	70.5/450.1
Subcooling margin, ΔT_{sub} (K)	39.2	100.7
Primary system (cold leg) temperature (K)	520.0	449.7
Secondary system pressure/ temperature (bar/K)	37.0/518.9	9.1/449.2
Primary loop flow (combined) (kg/s)	6616	6925
Decay heat level (MW)	10.36 (9.17%)	9.77 (8.65%)

3.3 Cases run

A large number of cases were run with both codes, the main purpose to investigate sensitivities concerning the availability of safety systems, but also to establish sensitivities concerning such factors as thermal hydraulic modelling assumptions (MELCOR), RCP operation, containment pressure feedback (SCDAP), timestep size and integration method (for SCDAP), Zircaloy oxidation kinetics and axial power profile. These helped to establish that the overall conclusions were robust against the inevitable uncertainties in physical modelling and plant state. The following sections summarise the hot and intermediate shutdown studies respectively, illustrating the results with a few examples.

4 ANALYSIS

The hot shutdown large break LOCA provided the greatest challenge to the safety injection, due to the rapid voiding of the core. Nevertheless, MELCOR showed the nominal injection was sufficient to recover the core and restrict temperatures to less than 1000 K; pressure and cladding temperature responses are shown in Figure 2 and Figure 3 respectively. There was some dependence on timestep size, containment pressure and limit on level swell in the core, but the peak temperature remained comfortably below the safety guideline in all cases. Despite the modest core temperatures, there was only a small margin of coolant availability to fill the core. Due to the simplified two-phase hydrodynamic models of MELCOR, including the neglect of certain cooling mechanisms, SCDAP simulations were performed to examine the transient evolution in more detail. The latter generally exhibited a more efficient core refilling and cooling, see Figure 4 which also shows the sensitivity to including the pressure feedback from the containment volume. The greater efficiency is due in part to the smaller fraction of the injected coolant lost via the break. SCDAP is thought to be more realistic due to the more detailed and more extensively validated hydrodynamic models.

The more challenging situation in which injection is delayed for a further 900 s implies a high probability that the safety criterion of 1204 ˚C may be approached by the time injection starts. Analyses were performed to determine if the core could still be recovered without a severe thermal excursion or damage occurring. Results of the MELCOR calculations indicated that injection from JSI 1-D alone might not be sufficient, since the core temperatures at the start of core refilling are already high enough for oxidation to have begun and thermal runaway is imminent. Since the time delay would normally be sufficient to provide power to the JSI 1-A, 1-B and 1-D pumps, simulations were also performed assuming injection from three JSI pumps in total. The MELCOR cases showed the additional injection was sufficient to refill the core but high temperatures occurred during reflood, driven by oxidation excursion. By contrast SCDAP indicated effective cooling almost immediately after the start of injection with temperatures restricted to tolerable levels, below the criterion of 1204 ˚C, even with only one of pumps JSI 1-A, 1-B or 1-D operating, and conservative assumptions about the fluid pathways in the RPV. A comparison of MELCOR and SCDAP results is shown in Figure 5, with liquid levels calculated by MELCOR in Figure 6. However, with the assumption that only JSI 1-D is operating, there is no margin.

Cooling appears to be sensitive to modelling assumptions if the core temperatures already exceed 1200 K when injection starts. In addition to other conservatisms, the MELCOR oxidation correlation is outside its validity range below 1500 K and is generally believed to overestimate the kinetic rate at those temperatures. Sensitivity studies using the same oxidation kinetics as SCDAP (Cathcart-Pawel) showed cooling was achieved readily, indicating that the overly conservative kinetics was mainly responsible for the oxidation excursion. Overall, the SCDAP-based models are believed to provide a more reliable assessment in these marginal cases; sensitivities to oxidation modelling, injection point and pump availability shown in Figure 7 show cooling in all cases.

Similar scenario assumptions were made for the intermediate break LOCA. The analyses showed the reactor to be readily cooled in all cases, for example see Figure 8, and the results are not discussed further here. Analyses showed that a small (3 cm, downward-facing) break LOCA during hot shutdown also poses no safety challenge unless safety injection is delayed excessively. With the secondary side isolated, causing the pressure to steadily rise, MELCOR showed core uncovery to start at about 2100 s. With injection from JSI 1-D assumed to start at 4200 s, when the maximum core temperature was about 1500 K, a severe excursion could not be avoided. With the secondary side kept at constant pressure, nominally 37 bars during hot shutdown, core uncovery was delayed for a further 600 s and the core was readily cooled by injection from JSI 1-D starting at 4200 s (1500 s after the start of uncovery), Figure 9. Peak temperatures were about 1050 K. It is concluded that the core can be recovered by this mode of injection initiated at any time within one hour of break opening, or within 1500 s of the start of uncovery.

The study of large break LOCA during intermediate shutdown focussed on cases of delayed injection. As expected, recovery was achieved more readily than during hot shutdown. MELCOR shows recovery by the combined injection of three pumps out of JSI 1-A,-B,-C,-D after 930 s delay, after peak temperatures reached about 1500 K and a small amount of oxidation. Sensitivity calculations indicate that 930 s is close to the maximum for which recovery can be conservatively demonstrated using MELCOR. SCDAP results showed that there might be additional time margin.

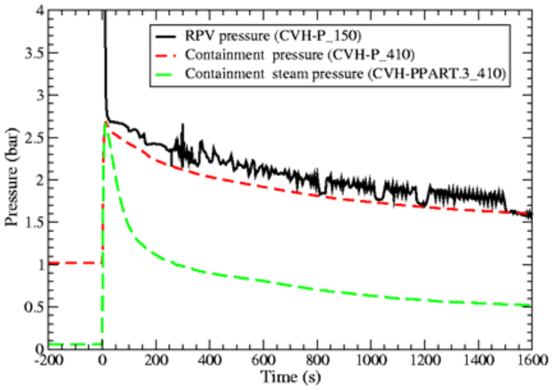

Figure 2: RCS and containment pressure (HS, LB, no SI delay, MELCOR, base)

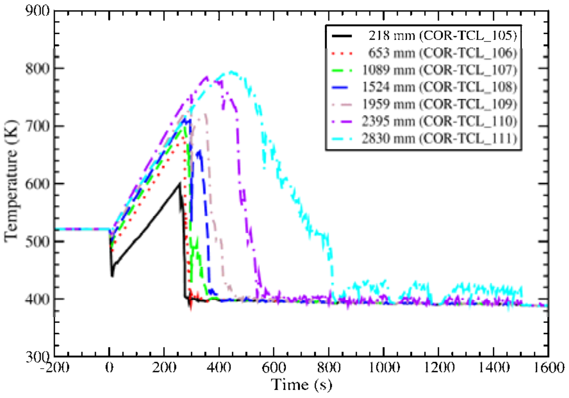

Figure 3: Clad temperatures in central zone (HS, LB, no SI delay, MELCOR, base)

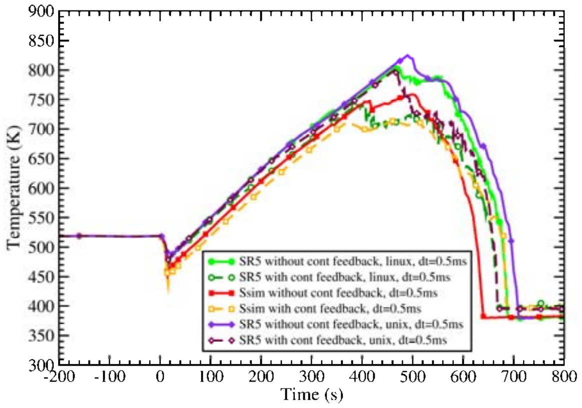

Figure 4: Maximum clad temperature (HS, LB, no SI delay, SCDAP: effect of code and containment pressure)

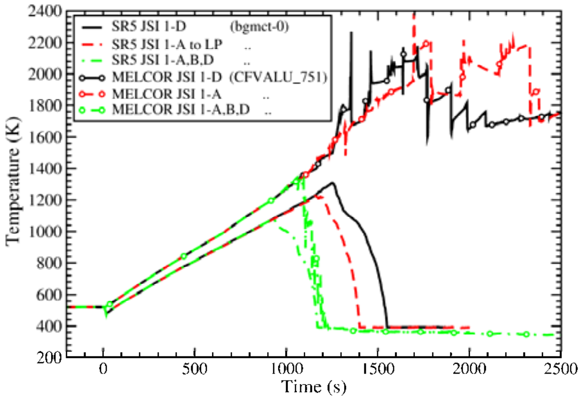

Figure 5: Maximum cladding temperature (HS, LB, SCDAP/MELCOR comparison: 930 s delay on JSI, effect of JSI availability and injection point)

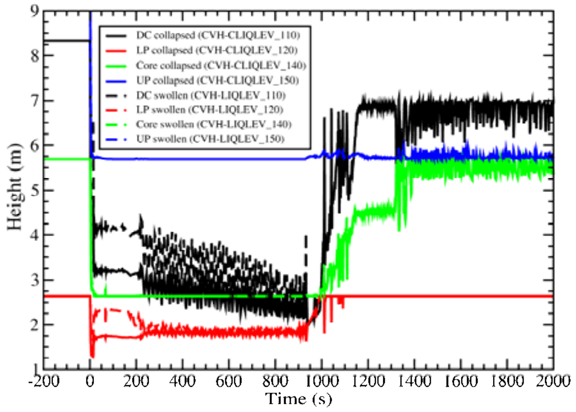

Figure 6; Liquid levels in RPV volumes (HS, LB, 930 s delay on JSI 1-A, B, D, MELCOR, base)

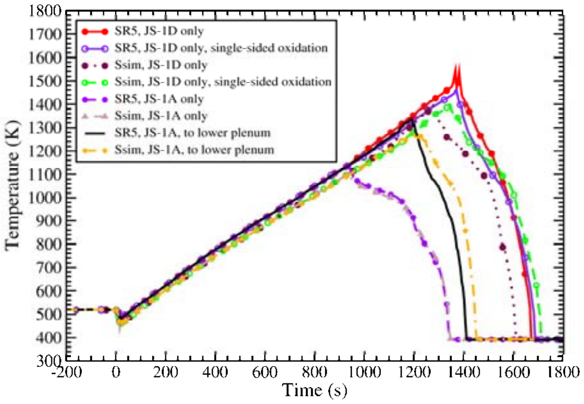

Figure 7: Maximum clad temperature (HS, LB, SCDAP: 930 s delay on JSI, effect of JSI availability and injection point, and code version)

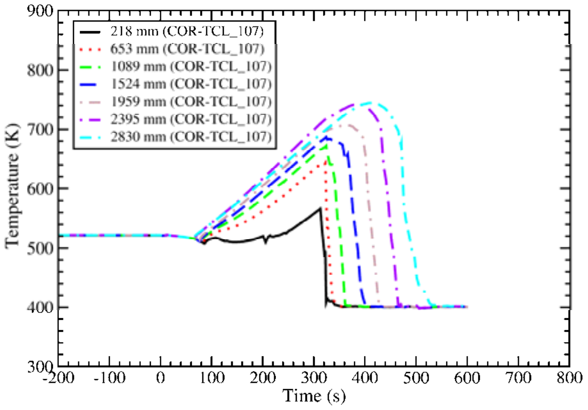

Figure 8: Clad temperatures in central zone (HS, IB, no delay on JSI 1-D, MELCOR, base)

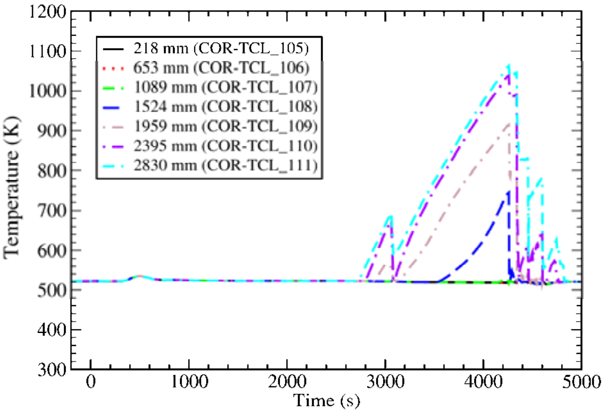

Figure 9: Clad temperatures in central zone (HS, SB, 4200 s delay on JSI 1-D, MELCOR, SG at constant pressure)

5 CONCLUSIONS

Analyses were performed for large, intermediate and small break LOCA in the KKB plant during hot shutdown (4 hours after reactor shutdown) and for large and intermediate break LOCA during intermediate shutdown (5 hours after shutdown). The purpose was to demonstrate core cooling and recovery by means of the coolant injection, no core damage, and temperature below the safety criterion of 1204 ˚C (1477 K), and to assess the margin against delayed SI availability and model uncertainties. Analyses were performed primarily using MELCOR 1.8.5; confirmatory calculations using SCDAP/RELAP5 and SCDAPSIM concentrated on cases where cooling was only marginally demonstrable, or was not demonstrated using MELCOR.

Both treatments showed that in the limiting large break case the core is readily quenched, without damage, by the nominal SI which assumes operation of only one pump. However, operation of a single SI pump after a delay of 900 s (the time needed to actuate the remaining pumps) might be insufficient. Delayed injection by all three SI pumps would be sufficient to recover the core if injection were so delayed, although the safety margin is possibly not enough to cover all the conservatisms inherent in the MELCOR treatment. The more mechanistic SCDAP-based calculations demonstrate that a larger margin exists, with recovery being possible even if only one pump operates after such a delay.

Overall, the study provides confidence in the inherent robustness of the plant design with respect to LOCA during cooldown to cold shutdown, and in the validity of a two-tier calculational method. The results have been directly used in updating the plant shutdown PSA, by changing the success criteria for core cooling during cooldown of the plant and showing a reduction in overall risk.

ACKNOWLEDGMENTS

The authors are pleased to acknowledge the financial support and technical input from Nordostschweizerische Kraftwerke (NOK) AG, Kernkraftwerk Beznau (KKB) on whose behalf the present study was performed.

REFERENCES

[1] J Birchley, T Haste and M Richner, "MELCOR analysis of loss of residual heat removal during mid-loop operation in a Westinghouse two-loop PWR", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP07), Nice, France, May 13-18, 2007.

[2] R O Gauntt et al., "MELCOR code manuals – version 1.8.5", NUREG/CR 6119 Rev 2, 2000.

[3] L Siefken et al., "SCDAP/RELAP5/MOD3.2 code manual", NUREG/CR-6150 Rev. 1, 1997.

[4] Innovative Systems Software, "RELAP5/SCDAPSIM/MOD3.4 code manual", 2003.

[5] NOK AG, Kernkraftwerk Beznau, "Final safety report", 2003.