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Accident management following loss-of-coolant accidents during cooldown in a Westinghouse two-loop PWR

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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 h 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 4 and 5 h 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 shutdown (HS) and intermediate shutdown (IS) (nominally 4 and 5 h 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 effective-

Abbreviations: AM, accident management; LOCA, loss of coolant accident; ECCS, emergency core cooling system; KKB, Kernkraftwerk Beznau; HS, hot shutdown; IS, intermediate shutdown; LB, large break; MB, medium break; NOK, Nordostschweizerische Kraftwerke; PAR, passive autocatalytic hydrogen recombiner; PSA, probabilistic safety analyses; PSI, paul scherrer institut; PWR, pressurised water reactor; RCP, reactor coolant pump; RCS, reactor coolant system; RPV, reactor pressure vessel; SB, small break; SG, steam generator; SR5, SCDAP/RELAP5; Ssim, RELAP5/SCDAPSIM; SI, safety injection; W, Westinghouse.

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Table 1
Nominal operating parameters.

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

ness 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 (Birchley et al., 2008), and employs essentially the same modelling approach summarised there. This citation describes work in support of studies on accident management and risk evaluation modes at Beznau (Richner et al., 2008); references are provided on emergency and accident management procedures for the plant. 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.

Large break (LB), medium break (MB) and small break (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 medium and small breaks are defined as 20 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 crossover 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.

The primary objective of the study was to demonstrate the effectiveness of the nominally operational JSI-1D in recovering the core with the peak temperature within the safety criterion. To assess further the safety margin provided by the nominal injection, 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 min is regarded as necessary to rack in additional switchgear and actuate the SI pumps. The extra delay renders a much more adverse sequence when considered with the single JSI-1D pump, and its impact is judged to be considerably greater than the effect of other uncertainties within the nominal scenario. We consider this as a bounding case. Of particular significance is that the delay period provides an opportunity to actuate the additional pumps. We then consider operation of JSI-1A and 1-B in addition to 1-D.

A parallel study was performed for the less challenging intermediate break, while the small break LOCA was investigated specifically to assess the time window for avoiding core damage in the event of delayed SI. Both of these scenarios are described briefly.

3. Tools and techniques

3.1. Codes used

The primary analysis tool was the engineering-level MELCOR code (Gauntt et al., 2000), 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. MELCOR is used extensively to analyse severe accidents where the vessel and coolant loops are highly voided, such as station blackout. In such cases spatial variations in temperature and density combined with the high pressure drive heat transfer pathways which strongly affect the evolution in the longer term, and these are typically investigated by more detailed multi-node input models. However, a coarse-mesh model is employed in the present cases where those effects are much less significant. As LOCA sequences can exhibit strong interaction between the liquid water and steam phases, the MELCOR analyses are supported by SCDAP/RELAP5 simulations used with a more detailed noding. RELAP5 provides a more complete treatment of the two-phase hydrodynamics, for example the inclusion of a dynamic momentum equation for each phase, flow-regime dependent models for interphase friction, interphase and wall heat transfer, all of which have a strong bearing on the reflood behaviour. RELAP5 has been widely used for many years for design basis LOCA sequences, see for example Lee et al. (2006, 2007), and with associated model development (Choi and No, 2010) and consideration of scaling issues (Petelin et al., 2007). SCDAP/RELAP5 has been used by PSI as the code of choice for detailed pre- and post-test analysis of the ongoing QUENCH series (Sepold et al., 2007) of bundle reflood tests at Karlsruhe Institute of Technology (formerly Forschungszentrum Karlsruhe) under severe accident conditions with mainly intact geometry, where good agreement between code calculations and experimental results has been demonstrated, see for example Stuckert et al. (2009). MELCOR and SCDAP both furnish models for core degradation, from initial melting of the metallic cladding through to late melt pool behaviour. However, the present studies are aimed at situations where core temperatures are well below the onset of