Insights from PSA Comparison in Evaluation of EPR Designs

Ari Julin, Matti Lehto (STUK)
Patricia Dupuy, Gabriel Georgescu, Jeanne-Marie Lanore (IRSN)
Shane Turner, Paula Calle-Vives (ONR)
Anne-Marie Grady, Hanh Phan (USNRC)

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Outline

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- EPR PSA Comparison
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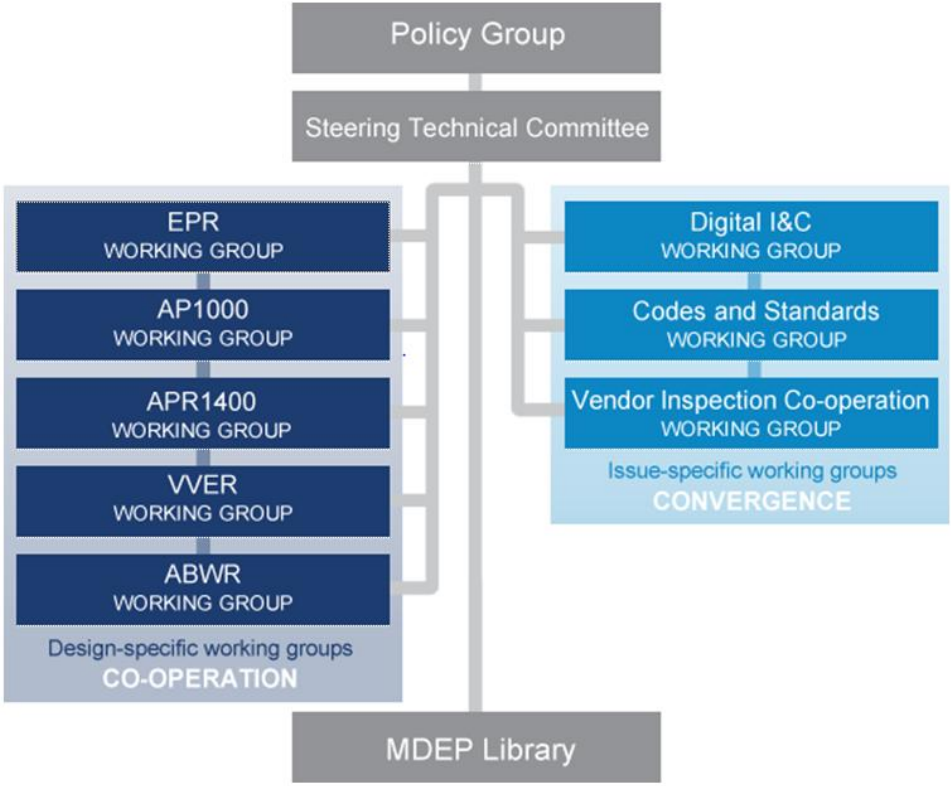
- Preliminary Conclusions

MDEP – Background (www.oecd-nea.org/mdep/)

- **M**ultinational **D**esign **E**valuation **P**rogram
- Established in 2006
- National regulatory authorities, OECD NEA as technical secretariat
- Review of new reactor power plant designs



EPR PSA Comparison

The comparison is performed within MDEP by the Design specific Working Group on Evolutionary Power Reactor (EPR WG):

- PSA subgroup started in 2008, PSA comparison task started in 2012:

  Participants:

  - Radiation and Nuclear Safety Authority (STUK), Finland
  - Nuclear Safety Authority / Institute of Radiological Protection and Nuclear Safety (ASN / IRSN), France
  - Office for Nuclear Regulation (ONR), United Kingdom
  - United States Nuclear Regulatory Commission (USNRC), USA

  EPR designs [year of the design documentation]:

  - Olkiluoto 3 NPP in Finland (OL3) [2010]
  - Flamanville 3 NPP in France (FA3) [2010]
  - UK EPR design [2011]; Hinkley Point C Pre-Construction Safety Report [2012]
  - U.S. EPR design [2013]

Scope of EPR PSA

- The EPR design takes benefits from previous design and operating experience especially in France and Germany
- Improvements introduced to prevent and mitigate severe accidents
- Level 1, Level 2 and hazards PSA have been developed
- PSA has been used during the design process in order to optimize safety and availability (Risk Informed applications required for OL3)
- Differences exist between the studied four EPR designs due to regulations, site, operator, industry or project timing (see Sections 2 and 4 in the paper)

EPR PSA Comparison - Objectives

- To identify differences in the modeling aspects and results
  - Overall CDFs are fairly similar
  - Risk profiles are not identical
- To assess rationale for the differences
- To support safety evaluation and PSA reviews in MDEP member countries

EPR PSA Comparison – Scope

- Level 1, Internal Events

- Four Initiating Events chosen, challenging a broad scope of safety functions:
  - Medium Loss Of Coolant Accident (MLOCA)
  - Loss Of Offsite Power (LOOP)
  - Steam Generator Tube Ruptures (SGTR)
  - Loss Of Cooling Chain (LOCC)

MLOCA Comparison – Typical Sequences

- MLOCA results in depressurization of the reactor coolant system, a decrease of pressurizer level, an increase in the pressure inside containment
- Partial Cool Down (PCD, steam release from the steam generators) is required to allow Medium Head Safety Injection (MHSI)
- If PCD fails, primary circuit Feed and Bleed function is necessary (F&B)
- If MHSI is unavailable, fast secondary cool down can be manually actuated to allow Low Head Safety Injection (LHSI)

MLOCA Comparison - Findings

- Accident scenarios are similar in all compared PSAs
- Conditional core damage probability is similar in all PSAs
- Differences in reliability data, human error probabilities and treatment of digital I&C – insufficient information to review these in details
- UK EPR design differs from others regarding the need for primary F&B in case of PCD failure - due to differences in modeling assumptions and thermal-hydraulic analyses ?
- Significant difference in assumed MLOCA frequency:
  - NUREG 1829 is used in UK and US designs
  - Studies developed in France/Germany are used in FA3 and OL3

LOOP Comparison – Typical Sequences

- Following the loss of main and auxiliary grids, the plant will be transferred to House Load Operation - reactor trip is triggered in case of unavailability of the House Load Operation
- Emergency Diesel Generators (EDGs) are started and connected automatically, SG level decreases leading to automatic actuation of Emergency Feed Water System (EFWS), automatic SG level regulation
- In EPR, the EDGs are not necessarily needed during the first 2 hours after the LOOP occurs in case of Reactor Coolant Pump (RCP) seals are OK
- If EFWS is unavailable, primary circuit F&B function is necessary
- RCPs seal injection and thermal barriers cooling is required – in case of their failures, the Stand Still Sealing System (SSSS) will be automatically actuated to maintain the primary circuit integrity
- If SSSS is required and fails -> seal LOCA

LOOP Comparison - Findings

- Accident scenarios are similar in all compared PSAs
- Short (2 hours) and long (24 hours) LOOP are considered
- The overall results are very similar, main sequences leading to CD:
  - Loss of heat removal (EFWS and F&B failure) due to loss of DGs
  - Seal LOCA followed by total loss of water injection
- Differences in success criteria and strategy in degraded situations (e.g. F&B with LHSI only in U.S. EPR; SBO DGs actuation only after loss of all EDGs in FA3) – due to differences in procedures and support calculations ?
- In U.S. EPR, SBO DGs supply Chemical and Volume Control System
- Differences in the level and detail of modeling (ventilation, batteries ...)
- Significant differences in modeling and quantification of I&C (CCFs identified; account for diversified means) – due to different modeling, assumptions and design ?
- Some dominant results rely on similar assumptions in the four PSAs (adequate diversity in EDGs / SBO DGs, no CCF for all DGs)

SGTR Comparison – Typical Sequences

- SGTR causes a loss of coolant inventory from the primary to the secondary side of the SG, primary pressure decrease, a level increase in the affected SG
- Based on diagnosis, operators trip the reactor, isolate the faulted SG and initiate the cool down with the intact SGs
- If isolation of the faulted SG fails (= LOCA outside containment), MHSI actuates automatically and extends the time available for operators to cool down and depressurize the reactor cooling system
- If secondary cool down fails, operators initiate primary circuit F&B function using MHSI
- Long term cooling provided by one train of LHSI or severe accident heat removal system
- Operator aligns and initiates residual heat removal via secondary circuit before IRWST capacity is lost

SGTR Comparison – Findings

- The four PSA models are largely similar, all of them covered a single tube leak; a double tube leak included in three PSAs (all but US) and multiple tube leaks included in two PSAs (US and FA3; modeled as rupture of ten tubes in single SG following a secondary side break)
- The results for single tube rupture are quite similar, same situation in the results for multiple tube ruptures (US and FA3), but variation in the results for double tube ruptures (here OL3 has clearly the lowest CDF)
- Main CD sequences are failure to initiate fast secondary cool down and failure of primary F&B function
- OL3 has different SGTR management strategy compared to others: the aim is to minimize steam release into environment and the faulty SG will be automatically isolated at the end of partial cool down (time < 10 min), in the other designs isolation takes place around 60 minutes post fault
- Potential design changes from I&C, especially in the U.S. EPR, need to be reflected in the PSA

LOCC Comparison – Typical Sequences

- LOCC covers several failure modes of the Component Cooling Water System (CCWS) and the Essential Service Water System (ESWS). Main groups are:
  - Loss of one train; affects MHSI, RHR, LHSI
  - Loss of one common user header (CH; two CHs exist, each connected to two CCWS trains); affects operating charging pump (make-up);
  - Loss of all trains (loss of two CHs) leads to loss of RCPs cooling (e.g. thermal barrier, bearings) and loss of charging pumps
- LOCC causes a reactor trip or requires a shutdown of the reactor, and at the same time degrades one or several safety functions required for the shutdown (possible Common Cause Initiator, CCI)
- If the CH switchover to the standby train fails, loss of one train leads to a CCI (loss of one CH)
- RCP seal LOCA is possible in certain sequences

LOCC Comparison – Findings (1/2)

- In general, the plant response is similar in all EPR PSAs
- Significant differences exist in the grouping of Initiating Events (IEs)
  - OL3: only one group “loss of one CH”
  - Seven groups handled e.g. in UK EPR
- Differences in exact definition of LOCC IEs, their frequencies, data and in use of operating experience (pipe breaks and leaks)
  - Conservative modeling, choice of modeling approach and different data sources explain some of the differences
- Differences in design affect the consequences of LOCC IEs
  - OL3 has additional heat exchangers cooled by CCW in safeguard building room cooling (diversity)
  - In U.S. EPR opening/closing of CH valves (switchover to standby train) needs two divisions to succeed
    (difference in power supply to the solenoid valves)
  - RCP thermal barrier cooling is not possible from both CHs in OL3
    (others have interconnection, but adding that feature to OL3 would have no significant impact on the risk)

LOCC Comparison – Findings (2/2)

- Comparison in details was not possible due to insufficient information
- Treatment of software failures and spurious signals of I&C systems as well as their impact on the results is still under review by some regulators
- Differences were identified in the modeling of RCP seal LOCA. Before final conclusions, further study is needed on:
  - Complexity of potential failure combinations
  - Assumption related to the leakage potential of the RCP seals

Numerical Results, CDF (1/a)

IE	DESCRIPTION	FA3	UK EPR^{A*}	U.S. EPR*	OL3*
LOOP	Loss of Offsite Power	1,40E-07	2,97E-07	1,23E-07	1,33E-07
MLOCA	Medium LOCA	3,6E-08	9,2E-09	9,1E-10	3,1E-08
SGTR	Steam Generator Tube rupture(s)	1,10E-08	1,02E-08	2,63E-08	2,21E-08
LOCC	Loss of cooling chain or heat sink	8,80E-08	9,46E-08	3,61E-08	1,94E-08
	TOTAL (see Table 2 in the paper)	4,8E-07	6,2E-07	3,0E-07	4,8E-07

[A] *PSA for a UK EPR™ at Hinkley Point C [7]*
* *At power operating states*

Preliminary Conclusions

- Outcomes and lessons learnt from the EPR PSA comparison have been used and will be used to facilitate the regulatory reviews and assessment work.
- The EPR designs represent various stages of the design process, licensing process and level of modeling detail -> makes differences also in numerical results of PSAs.
- Differences in the details and assumptions related to the modeling of I&C systems explain some identified differences. Detailed design of OL3 I&C system is still under development and comprehensive fault analyses are needed.
- The available information was not sufficient for a detailed comparison (e.g. ventilation systems, modeling of CCFs).
- EPR vendors have promised to provide further information to continue and enhance our comparison study (especially related to design differences affecting the risk and details in modeling).

Thank You for Your Attention!

Contact person: Ari.Julin@stuk.fi
MDEP EPR WG, PSA subgroup