A VIEW ON IDENTIFICATION OF THERMAL-HYDRAULIC PHENOMENA FOR VALIDATION OF BEST-ESTIMATE COMPUTER CODES

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ABSTRACT

Validation of best-estimate codes is a necessary step to prove their applicability to calculate accident scenarios, including the course of events. It shall demonstrate that those physical phenomena, which are important for a scenario, are calculated appropriately. A common list of 113 thermal-hydraulic phenomena is provided, based on previous reports of OECD/NEA-CSNI and IAEA, including Separate Effects Test (SET) facilities and Integral Test facilities of PWRs and BWRs, VVERs, Advanced Reactors, as well as Containment. Added objective of the activity is to show that the list of phenomena is applicable to the major number of water-cooled reactors. Twelve reactor types are considered for the characterization of 47 accident scenarios cross-linked with the identified phenomena. Focus of this paper is on the updated identification of the list of thermal-hydraulic phenomena.

1 Introduction

A necessary and important basis to perform safety analysis are best estimate computer codes. Validation of these codes is a necessary step to prove their applicability to calculate accident scenarios, including the course of events. For example, thermal-hydraulic computer codes shall demonstrate to calculate the behaviour in the primary and secondary systems of a nuclear power plant before performing safety analyses [1]. Two types of experimental facilities have been used for thermal hydraulic system code validation: Separate effects experiments and integral test facilities. Whereas integral test facilities are usually designed to follow the behaviour of a reactor system in various off-normal conditions or accident transients, separate effects tests focus on the behaviour of a single component, or on the characteristics of one thermal-hydraulic phenomenon or a limited number of phenomena.

Already in the year 1987, the OECD/NEA Committee on the Safety of Nuclear Installations (CSNI) published a document that identified systematically a set of physical phenomena and tests, which were considered to provide the best basis for the assessment of the performance of thermal hydraulic codes, "CSNI Code Validation Matrix of Thermohydraulic Codes for LWR LOCA and Transients" [2]. The report included all typical phenomena expected to occur in plant transients and LOCA analyses. After publishing this report, it turned out that continued comparison of calculations with additional separate effects test (SET) data is also necessary to consider particular applications of codes, especially where a quantitative evaluation of prediction accuracy is required for best estimate computer codes, as well as for code model improvement. Based on these needs, the OECD/NEA-CSNI issued an independent and separate two-volume report on separate effects tests validation matrix in 1994 [3]. An updated integral test validation matrix report was issued in 1996 [4], a VVER validation matrix in 2001 [5], and a containment validation report in 2014 [6]. The construction of such validation matrices is an attempt to collect in a systematic way the best sets of test data for code validation, assessment and improvement, including quantitative assessment of uncertainties in the modelling of individual phenomena for the best estimate thermal-hydraulic transient computer codes.

A recent activity in the frame of issuing a book on Thermal-Hydraulics in Nuclear Reactors [7] provides a common list of 113 thermal-hydraulic phenomena, based on the previous reports of OECD/NEA-CSNI and IAEA [2-13], including SET facilities and Integral Test facilities of PWRs and BWRs, VVERs, Advanced and Generation IV Reactors, as well as Containment with updated descriptions. Added objective of the activity is to show that the list of phenomena is applicable to the major number of water-cooled reactors. A procedure is proposed to connect phenomena, accident scenarios and measured or calculated variables. The procedure yields cross-link tables, making use of the results of reactor calculations derived from several documents available from the literature and dealing with selected accident scenarios and reactor designs. Twelve reactor types are considered for the characterization of 47 accident scenarios cross-linked with the identified phenomena. The phenomena are largely affected by geometry and boundary conditions and need suitable modelling capabilities. Two main objectives have been pursued:

a) the description of accident scenarios;

b) the connection of thermal-hydraulic phenomena to these accident scenarios.

The accident scenarios are here limited to design basis accidents before loss of core integrity. This paper focusses on the new activity of thermal-hydraulic phenomena.

2 Thermal-hydraulic phenomena

Phenomena have been established within the framework of activities performed within international institutions. They deal here with the condition Design Basis Accident (DBA), before the occurrence of the loss of geometric integrity for the core. DBAs define accident conditions against which a facility has to be designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits [14]. Here, the concerned condition comprise some Beyond Design Basis Accident (BDBA) situations, e.g. in terms of probability of occurrence of a selected Postulated Initiating Event (PIE), and in cases outside the DBA boundary, when accident management procedures are performed to prevent core degradation, e.g. fast depressurization and feed of the secondary side of the steam generators or the primary side of the reactor.

A reviewed and combined list of thermal-hydraulic phenomena in alphabetic order is set-up in Table 1. Four categories of phenomena are distinguished: Separate effect (S) including basic (B), integral effect (I) and phenomena addressing the design of ‘Advanced reactors’ (A). Reactor Coolant System (RCS) is considered within these three categories for conventional Light Water Reactors. In case of advanced reactors, both RCS and containment phenomena are considered due to close interconnection of the RCS and containment. Descriptions of these phenomena will be also provided in the “Thermal-Hydraulics in Nuclear Reactors” book [7] and the former mentioned OECD/NEA-CSNI and IAEA documents [3, 4, 5, 6, 8-13].

Apart from integral test (I) phenomena, a different group named "Basic Phenomena" (B) has been added to the SET (S) phenomena. Each phenomenon in this group can be characterised as independent of other phenomena and is of a constitutive nature for the two-phase flow. Basic phenomena are the transfer processes at the interface between different fluids, like liquid and steam or a non-condensable gas, and solid surfaces facing the fluid. They comprise momentum, heat and mass transfer, which have been divided into

- evaporation due to depressurization including at geometric discontinuities (B-3),
- evaporation due to heat input (B-4)
- condensation due to pressurization (B-2)
- condensation due to heat removal (B-1)
- interfacial friction in vertical flow (B-6)
- interfacial friction in horizontal flow (B-5)
  wall to fluid friction (B-9)
- pressure drops at geometry discontinuities (B-7)
- pressure wave propagation (B-8).

With regard to basic phenomena, the possible steam generation at abrupt discontinuities, i.e. caused by the total pressure drop is included in the phenomenon B-3. Downstream of a geometric discontinuity, e.g. in the presence of a sharp edge possibly combined with high Reynolds-number flow, the local pressure may decrease below the saturation pressure corresponding to the fluid temperature, vaporization may occur and void may appear which become sub-cooled void because of the sudden pressure recovery. Subcooled void may affect the total pressure drop and, if present, the two-phase critical flow in the downstream geometry.

In the six equation models used in some thermal-hydraulic safety computer codes, the constitutive laws are intended to model these basic transfers between phases. The method of discretization for the fluid dynamics, and the numerical solution, significantly affect the calculation of pressure wave propagation. Basic phenomena govern all the fluid behaviour during an accident sequence, so they are inevitably important for nuclear safety.

In most cases of practical interest there is no direct measurement of basic phenomena for two phase flow. They can be only evaluated on the basis of global measurements (e.g., pressure difference, pressure, temperature). Models are necessary to evaluate the basic transfers from the global measurements. Correlations are set up from these models. However, it is important to realize that the use of these correlations is dependent of the chosen model. Normally, the correlations should be used in a way, which is consistent with their derivation. Usually, the basic phenomena are treated as independent of modelling. With the present state of knowledge, models employed in codes give a large range of differences on evaluation of basic phenomena. However, the global thermal-hydraulic behaviour is nevertheless predicted quite similarly by different approaches.

Integral tests are carried out in scaled test facilities and provide data on the overall behaviour of a simulated reactor system during a LOCA or transient. These tests are being used for code assessment purposes relating to overall reactor behaviour. A definition of sets of such tests has been provided in [4]. These facilities have a complicated configuration and are expensive to operate. Compromises, for instance with respect to scaling to real plants, are inevitable. In addition, the instrumentation for measurements of parameters governing different two-phase flow phenomena is limited. This makes the integral tests less suitable for detailed investigations of specific two-phase flow phenomena. Consequently, this also requires in assessment analyses that the code user is very confident of which phenomena prevail during the course of a transient in order to avoid "good results but for the wrong reasons" (compensating errors).

Separate effects tests are employed not only to develop correlations of specific processes but also to investigate individual or localised two-phase phenomena, which in most cases are dependent on several specific processes. These kinds of tests are also used to characterise the behaviour of single components such as pumps or steam separators.

The interactions with other phenomena or components as in a full scale light water reactors are either imposed by external boundary conditions on the test, or are purposely eliminated.

In fact, an important consideration in the design of separate effects tests is to have well defined boundary conditions. In some cases, the test section in a separate effects test is locally in full scale or very near to full scale. This minimises the concern about scale effects. On the other hand, in all separate effects tests it is important to evaluate the influence of the chosen boundary conditions on measured parameters.

In separate effects tests the instrumentation for measurements of two-phase flow parameters can be quite extensive. Thus, data for assessment of details of the models used in the codes to simulate localised phenomena can be provided, and this can help in understanding the interactions of different processes that combine to produce the overall phenomena of interest. The data have also proved to be valuable when evaluating prediction uncertainties of specific phenomena at or near full scale.

Table 1 - List of phenomena

ID	PHENOMENA – TABLE 1 PART 1 OF 2	TYPE	REACTOR	DETAILS & NOTES
S-1	Accumulator behaviour	SETF		Mainly PWR
I-1	Asymmetric loop behavior	ITF	PWR	
I-2	Asymmetry due to the presence of a dam	ITF		Shutdown conditions
A-1	Behavior of check valves	Advanced Reactors		Also containment
A-2	Behavior of containment emergency systems (e.g. Passive Containment Cooling System)			
A-3	Behavior of core make-up tanks			
A-4	Behavior of density locks			
A-5	Behavior of emergency heat exchangers including Passive Residual Heat Removal and Isolation Condenser			
A-6	Behavior of large pools of liquid			
I-3	Blowdown	ITF/SETF/Basic		Phenomenological time window rather than phenomenon
I-4	Boiler condenser mode (of Natural Circulation)	ITF	PWR-OTSG	
	Boil-off, included in S-26, Heat Transfer (CHF/DNB, post-CHF) in core	ITF/SETF/Basic		See also B-4-Evaporation due to heat input
S-2	Boron mixing and transport (also A-12-stratification of boron)	SETF	PWR	Also ITF
S-3	Counter-Current Flow/ Limitation (CCF/CCFL)-Channel inlet orifice	SETF	BWR	
S-4	CCF/CCFL-Downcomer		PWR	
S-5	CCF/CCFL-HL & CL			
S-6	CCF/CCFL-SG tubes			
S-7	CCF/CCFL-Surgeline			
S-8	CCF/CCFL-Upper core tie plate		N/A	
	Centrifugal pump			See Impeller pump
I-5	Channel and bypass axial flow and void distribution	ITF	BWR	
I-6	Collapsed level behaviour in downcomer	ITF	BWR	See also phase separation
B-1	Condensation due to heat removal	Basic	N/A	

B-2	Condensation due to pressurization	Basic		
S-9	Condensation in stratified conditions-Horizontal Pipes	SETF		
S-10	Condensation in stratified conditions-PRZ		PWR	
S-11	Condensation in stratified conditions-SG-PS			
S-12	Condensation in stratified conditions-SG-SS & BWR-Pressure Suppression Pool			Also BWR wet-well
	Containment pressure and temperature	ITF		See I-18- Pressure-temp. increase & boiling due to energy and mass input and S-42-Natural convection and H2 distribution
I-7	Core thermal-hydraulics	ITF	BWR	See global multi-dimensional
	Core wide void and flow distribution	ITF		See global multi-dimensional
S-13	Control Rod Guide Tube flashing	SETF	BWR	
A-7	Critical and supercritical flow in discharge pipes	Advanced Reactors		Also CANDU and RBMK
	Critical flow			See Two-Phase Critical Flow
	Critical Power Ratio	ITF	BWR	See HT CHF
	De-entrainment			See Entrainment
	Depressurization			See Blowdown
S-14	ECC bypass/Downcomer penetration	SETF	PWR	
	ECC mixing and condensation	ITF		See liquid-vapour mixing
S-15	Entrainment/De-entrainment-Core	SETF		
S-16	Entrainment/De-entrainment-Downcomer			
S-17	Entrainment/De-entrainment-Hot leg with ECCI		PWR	
S-18	Entrainment/De-entrainment-SG mixing chamber			
S-19	Entrainment/De-entrainment-SG tubes			
S-20	Entrainment/De-entrainment-UP			
B-3	Evaporation due to depressurization (including at geometric discontinuities*)	Basic		* reversible part
B-4	Evaporation due to heat input	Basic		
I-8	Flow through openings	ITF		Shutdown conditions
S-21	Global multi-dimensional fluid temperature, void and flow distribution-Core	SETF		
S-22	Global multi-dimensional fluid temperature, void and flow distribution-Downcomer			
S-23	Global multi-dimensional fluid temperature, void and flow distribution-SG SS		PWR	
S-24	Global multi-dimensional fluid tem-			

	perature, void and flow distribution-UP			
A-8	Gravity driven reflood	Advanced Reactors		
S-25	Horizontal heated channel Heat Transfer (HT) [added phenomenon]	SETF	CANDU	Including HT below
S-26	HT (natural convection, forced convection, sub-cooled nucleate boiling, saturated nucleate boiling, CHF/DNB, post-CHF)-Core, SG, structures	SETF		Including VVER conditions
S-27	HT [radiation]-core			
S-28	HT [condensation]-SG structures			
A-9	HT condensation in containment structures, with or w/o non-condensable	Advanced Reactors		Also containment
S-29	Impeller pump behavior	SETF		External pumps
	Instability (in boiling channels)	SETF/ITF		See S-44- Parallel channel effects and instabilities
B-5	Interfacial friction in horizontal flow	Basic		
B-6	Interfacial friction in vertical flow	Basic		
I-9	Intermittent 2-phase natural circulation	ITF	PWR-OTSG	
S-30	Internal pump behavior (specific geometry) [added phenomenon]	SETF	ABWR	Also AP-1000
S-31	Jet pump behavior	SETF	BWR	
S-32	Liquid accumulation in horizontal SG tubes	ITF	PWR-VVER	
	Liquid carry-over			See Entrainment & I-24-steam binding

ID	PHENOMENA – TABLE 1 PART 2 OF 2	TYPE	REACTOR	DETAILS & NOTES
A-10	Liquid temperature stratification	Advanced Reactors		Also containment
S-33	Liquid-Vapor mixing with condensation-Core	SETF		
S-34	Liquid-Vapor mixing with condensation-Downcomer*			Also ITF. * Including cold-hot liquid mixing (3D effect)
S-35	Liquid-Vapor mixing with condensation-ECCI in HL and CL*		PWR	
S-36	Liquid-Vapor mixing with condensation-Lower plenum*			
S-37	Liquid-Vapor mixing with condensation-SG mixing chamber		PWR	
S-38	Liquid-Vapor mixing with condensation-UP			
S-39	Loop seal filling and clearance	SETF	PWR	Also ITF
S-40	Lower plenum entrainment	SETF	PWR	
S-41	Lower plenum flashing	SETF		See also Blowdown
	Mixture level & entrainment-Core, downcomer and SG SS	ITF		See Phase separation
I-10	Natural circulation (NC), 1-phase & 2-phase-PS & SS	ITF		SS only for PWR
I-11	NC core and downcomer		BWR	
I-12	NC core bypass, hot and cold bundles		BWR, CANDU*	*also RBMK
I-13	NC core, gap, downcomer, dummy elements		PWR-VVER	

I-14	NC core, vent valves, downcomer		PWR-OTSG	
I-15	NC with horizontal SG		PWR-VVER	
A-11	NC RPV and containment systems & various system configurations	Advanced Reactors		Also containment
S-42	Natural convection and H2 distribution	SETF		Inside containment
S-43	Non condensable gas effect including condensation HT in RCS	SETF	PWR	Also ITF
	Nuclear fuel behavior	SETF/ITF		See I-29- nuclear fuel feed-back
I-16	Nuclear thermal-hydraulics feedback and spatial effect (see also I-29- nuclear fuel feed-back)	ITF	BWR	Also RBMK, ABWR, etc.
	Nuclear thermal-hydraulics instabilities	ITF	BWR	See I-16 and S-44
S-44	Parallel channel effects and instabilities (PCEI)	SETF	BWR	
S-45	Phase separation at branches (including effect on two-phase critical flow)	SETF		Also ITF (T-branches)
S-46	Phase separation/vertical flow with and w/o mixture level-Core	SETF		Also ITF
S-47	Phase separation/vertical flow with and w/o mixture level-Downcomer			
S-48	Phase separation/vertical flow with and w/o mixture level-Pipes & Plena			
I-17	Pool formation in UP	ITF	PWR	See also S-8- CCF/CCFL-Upper core tie plate
B-7	Pressure drops at geometric discontinuities, including containment	Basic		Also Advanced Reactors
B-8	Pressure wave propagation	Basic		
I-18	Pressure-temperature increase & boiling due to energy and mass input	ITF		Containment & Shutdown
I-19	PRZ thermal-hydraulics	ITF	PWR	
S-49	Quench Front (QF) propagation/rewet-Fuel rods	SETF		
S-50	QF propagation/rewet-Channel walls, Water rods		BWR	
I-20	Refill including loop refill in PWR-OTSG	ITF		Phenomenological time window rather than phenomenon
I-21	Reflood			
I-22	Reflux condenser mode and CCFL	ITF	PWR	
	Return to Nucleate Boiling (RNB)			See Reflood & Quench Front
S-51	Separator behaviour (&* flooding, steam penetration, liquid carry-over)	SETF		*Mainly for BWR
I-23	SG siphon draining (SG interaction with Engineered Safety Features, including gravity driven)	ITF	PWR	Shutdown conditions
S-52	Spray effects-Core (including cooling and distribution)	SETF	BWR	
S-53	Spray effects-OTSG SS		PWR-OTSG	
S-54	Spray effects-PRZ		PWR	
I-24	Steam binding (liquid carry-over, etc.)	ITF	PWR	
S-55	Steam dryer behavior	SETF		Mainly BWR
I-25	Steam line dynamics	ITF	BWR	
S-56	Stratification in horizontal flow-Pipes (in 1-phase & 2-phase conditions)	SETF		Also ITF
A-12	Stratification of boron	Advanced Reactors		See S-2- Boron mixing and transport
I-26	Structural heat and heat losses	ITF		Scaling issue
I-27	Surge line hydraulics	ITF	PWR	
I-28	Superheating in OTSG SS	ITF	PWR-OTSG	

I-29	Thermal-hydraulics – Nuclear fuel feedback (see also I-16 Nuclear TH feed-back)	ITF	PWR	Also CANDU, Pressurized Heavy Water Reactor designed by KWU, etc.
S-57	Thermal-hydraulics of horizontal SG, PS and SS	SETF	PWR-VVER	
S-58	Thermal-hydraulics of OTSG, PS and SS	SETF	PWR-OTSG	See spray effect OTSG
S-59	Two-Phase Critical Flow (TPCF)-Breaks	SETF		
S-60	TPCF-Pipes			
S-61	TPCF-Valves			
A-13	Tracking of non-condensable gases	Advanced Reactors		See S-43-Non- condensable gas effect including condensation HT in RCS & containment
	Valve leak flow (connected with construction, operation, maintenance)	ITF		See also Two-Phase Critical Flow Valves
	Vapor (or steam) carry-under	ITF		See S-51-Separator Behaviour & I-10-Natural Circulation
	Vapor pull-through	SETF		See S-45- Phase separation at branches
I-30	Void collapse and temperature distribution during pressurization	ITF	BWR	Also basic condensation
B-9	Wall to fluid friction	Basic		
	Water accumulation in horizontal SG tubes	ITF	PWR-VVER	See liquid accumulation S-32

Three phenomena have been added which are not explicitly reported in the list of original documents; these are indicated in green lines in Table 1.

In some cases, the distinction between ITF and SETF phenomena is only formal; there should be no consequences in the application of the related information.

Table 1 includes 113 'independent-phenomena':

- 9 basic phenomena (B-1 to B-9) originated from the OECD SETF report [3]
- 61 SET phenomena (S-1 to S-61): 58 originated from the OECD SETF report [3] plus 3 added phenomena (S-25-Horizontal heated channel HT, S-30-Internal pump behavior (specific geometry) and S-42-Natural convection and H2 distribution, green lines in Table 1) within the present context and dealing with horizontal heated channels, internal pumps and convection flows inside the containment
- 30 IT phenomena (I-1 to I-30) originated from the OECD ITF report [4,5]
- 13 'Advanced Reactor' phenomena (A-1 to A-13) derived from above cited OECD and IAEA documents [8-13]; these phenomena plus the S-42-Natural convection and H2 distribution mentioned above, are assumed to characterize the containment performance, too.

The considered phenomena for the 'advanced reactor' phenomena, plus the S-42 phenomenon (already mentioned), are assumed to characterize reactor containment scenarios, too. These include full pressure and pressure suppression containment and the bubble condenser installed in some VVER-440 NPP. However, no phenomenon is related to the ice-condenser containments, which are excluded from the present framework.

3 Conclusions

Thermal-hydraulic phenomena have been collected from OECD/NEA-CSNI and IAEA documents covering design basis accidents or courses of events 'before loss of core integrity'. The same phenomena may be important in different evolutions of two-phase flows. This collection of phenomena can be used to prove the applicability and quality of best-estimate computer codes. Items of the present paper can be summarized as follows:

- 113 phenomena in Table 1.
- 12 water cooled reactor designs: PWR-U-tube steam generator, PWR-once-through steam generator, PWR-horizontal steam generator (VVER-440 and VVER-1000), BWR, CANDU, PHWR and RBMK, plus AP1000, APR1400, EPR, ESBWR, ABWR, and Small Modular Reactors, e.g. SMART MASLWR.

The phenomena are identified from experiments and expertise, which are continuously updated. Objective of this activity is to show that the list of phenomena is applicable to the entire class of water-cooled reactors. Variables as results of a computer code application to NPP analyses are to be associated to phenomena. All identified phenomena shall be modelled according to the state of the art knowledge. System thermal-hydraulics codes constitute the source of expertise associated with considered phenomena and the best and unique tool to calculate accident scenarios, including the course of events.

Notes are added to point out similarities, rather than reducing the number of phenomena. Importance is given to experimental programs, like data or information from the Bubble Condenser (containment type) facility in Russia, BETHSY, CCTF, LSTF, PKL, LOBI, ATLAS, UPTF, and PSB experiments, in addition to data measured in NPPs, e.g. Doel-2 Steam Generator Tube Rupture.

The knowledge and the understanding of phenomena is a prerequisite for performing meaningful accident analysis. The presented information can be used as part of the qualification process for system code calculations; this may constitute a guidance to formulate and to address the following issues or questions in relation to each NPP accident scenario calculation:

a. What are phenomena expected to be relevant in the scenario? The list in Table 1 can be considered; in general, all phenomena should be considered.

b. Are any of the phenomena expected to be relevant in the scenario under consideration? For instance, in case of small break LOCA with two-phase conditions occurring in the hot leg of a PWR, importance should be given to the phenomenon 'counter-current steam-water flow'. What kind of equations or equation parameters are necessary to account for those phenomena?

c. What is the qualification base for the phenomena expected to be relevant? What kind of experiments may be used to demonstrate a suitable knowledge for the phenomenon including addressing the scaling issue?

A qualified code calculation needs appropriate answers to these questions.

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