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APPLICATION OF THE UMAE METHODOLOGY FOR THE UNCERTAINTY EVALUATION IN SUPPORT TO THE EOP BACKGROUND ANALYSIS

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ABSTRACT

Validation of EOPs (Emergency Operating Procedures) relies on the best-estimate analysis of the transient scenarios. In order to cover associated uncertainties, usually limited number of sensitivity studies is performed for the development of the EOPs in order to identify possible plant states and associated parameters relevant for operator actions. Recently, developed methodologies for the uncertainty evaluation made it possible to evaluate directly uncertainties with the respect to the scenarios analyzed. UMAE (Uncertainty Methodology based on Accuracy Extrapolation) uncertainty methodology has been applied for development of function restoration EOPs. More specifically, Inadequate Core Cooling (ICC) LOCA (Loss of Coolant Accident) scenario has been analyzed using best estimate transient analysis code RELAP5/SCDAPSIM code. Time window for successful operator action has been evaluated following 4.0" cold leg break near the Reactor Pressure Vessel (RPV) in a 2-loop PWR plant.

KEYWORDS

RELAP5, EOP (Emergency Operating Procedures), UMAE (Uncertainty Methodology based on Accuracy Extrapolation), Loss of Coolant Accident (LOCA), ICC (Inadequate Core Cooling), ECCS (Emergency Core Cooling System) injection, rod temperature, core level, core cooling.

INTRODUCTION

Emergency Operating Procedures (EOPs) provide a network of predefined and prioritized event and symptom based response procedures that provide guidance the operator in management of accidents. Event related recovery and function related restoration procedures are combined to provide diagnosis and guide plant recovery to the optimal end state while ensuring explicit diagnosis and restoration of the plant safety state independent of event sequence. In function restoration EOPs diagnosis is the process used to direct the operator to the appropriate procedure(s) and procedure step(s) to address the existing plant state (symptoms) and does not require identification of the cause (event) of the symptoms.

Accident analysis is crucial step in the EOP development project because computer simulation is the most comprehensive way of knowing how the plant will respond to the recovery strategies. These analyses, of course, are additional to what is already available in the Final Safety Analysis Report (FSAR) and other available analytical documents for the nuclear power plant. In the EOP project a larger number of additional analyses may be needed to support the development of individual strategies and to better document the EOPs. The development is an undertaking that is primarily operationally oriented and requires a broad understanding of the entire plant response. Operationally oriented aspects, such as general trends of plant parameters, available symptoms, states, timing of actions, play a role in strategy development as well as verification of some safety criteria.

Typical analytical support tasks to development of EOPs are: identification of the applicability of the reference/generic method; identification of the scope of the EOP; identification of plant vulnerabilities, time and means available to operator for the success of the applied measures and specifics of plant behaviour to include instrument response under accident conditions and hazardous conditions within the plant. With this in mind special care should be given to development and validation of strategies. Because determining or justifying the strategies selected for individual EOPs, or even sometimes selecting the strategy among different possibilities, might involve a large number of best estimate analyses.

The UMAE uncertainty methodology relies on the extrapolation of measured variables in integral test facilities. In this particular case counterpart small break LOCA tests performed in LOBI, SPES, BETHSY and LSTF were used. Extrapolation consisted of deriving uncertainty values which were considered to be applicable to the significant variables that identify the Inadequate Core Cooling (ICC) scenario. Two parameters were used: accuracy and uncertainty. Accuracy is related to the comparison between calculated and experimental data, and is a measurement of the difference between experimental and calculated data. Uncertainty is related to the prediction of incidental scenario in the nuclear power plant. It is a measure of the error in the evaluation of the different parameters during the transient. The methodology, aiming at the evaluation of uncertainty, is based on the extrapolation of accuracy [1], [2].

Defining time window in which one action is successful is very important for defining and validation of procedure. Uncertainty of one event could play large role in overall plant recovery. Usually, this is estimated based on the sensitivity studies. This is where UMAE methodology has been implemented to give minimum time window for operator action which will ensure successful completion of recovery procedure in case of the ICC scenario.

The ICC scenario is the one of the most challenging scenarios from the view point of core integrity. This scenario was evaluated with the objective to determine the minimum ECCS (Emergency Core Cooling System) configuration needed to be operational time available for the operator to ensure cooldown of the core when the core is in extreme dry-out condition. Severity of this case is that there are no active SI (Safety Injection) systems and the operator starts attempts to recover them once core exit thermocouple reaches temperature of 650^0C (923.15K). After reaching that temperature it is important to estimate window during which actuation of SI systems prevents severe overheating of the core. Accumulators as passive components are available.

The trend in core level indication is used to check the effectiveness of safety injection in restoring Reactor Coolant System (RCS) inventory. If increasing, then no further action may be necessary. The core collapsed and Reactor Vessel Level Indication System (RVLIS) was used for this. The trend in core exit TC (ThermoCouple) temperatures is used to check the effectiveness of safety injection in restoring core cooling. If core exit TC temperatures are greater than predefined values and not decreasing in conjunction with a low core level, then applicable function restoration procedure must be expeditiously continued in order to perform the alternative actions for restoring core cooling.

In order to exit Inadequate Core Cooling situation, the core exit TC temperatures must be less than 923 K (650°C); two RCS hot leg temperatures must be less than 450 K (177°C) to ensure RCS pressure is less than the shutoff head of the Low-Head SI pumps (LHSI). Core cooling has been restored when the above conditions have been met and SI flow or other make-up flow has been established.

Typical success criteria for ICC are:

- Peak cladding temperature Tpeak < 1478K, [3]
- Time period of temperature excursion at T ≥ 923K before reduced (Δt) < 1800s, [4]

DESCRIPTION OF THE ANALYTICAL MODEL

"Best estimate" code RELAP5/SCDAPSIM, [5], has been applied for the development of the plant model. The plant model has been established with the predefined fidelity of the plant physical parameters (geometry, thermal hydraulic parameters, control and protection system set points, etc.) and taking into account limits of the mathematical model and related assumptions and necessary simplifications, [6]. The model has been developed to a high level of detail and includes detailed discretization of all important components of the plant primary and secondary side (Reactor Pressure Vessel – RPV and Stem Generators - SGs) and the models of the Emergency Core Cooling System - ECCS, Main Feedwater - MFW and Auxiliary Feedwater - AFW and simplified model of charging and letdown system. Protection and control system has been developed according to the plant available documentation. It is suitable for calculation of all transients and accidents for which RELAP5 can give reasonable predictions. Verified and recommended RELAP5 modeling techniques are used in preparation of RELAP5 input deck, [7]. The schematic can be viewed on Fig. 1. The nodalization features 508 volumes, 541 junctions, 351 heat structures and 2019 mesh points.

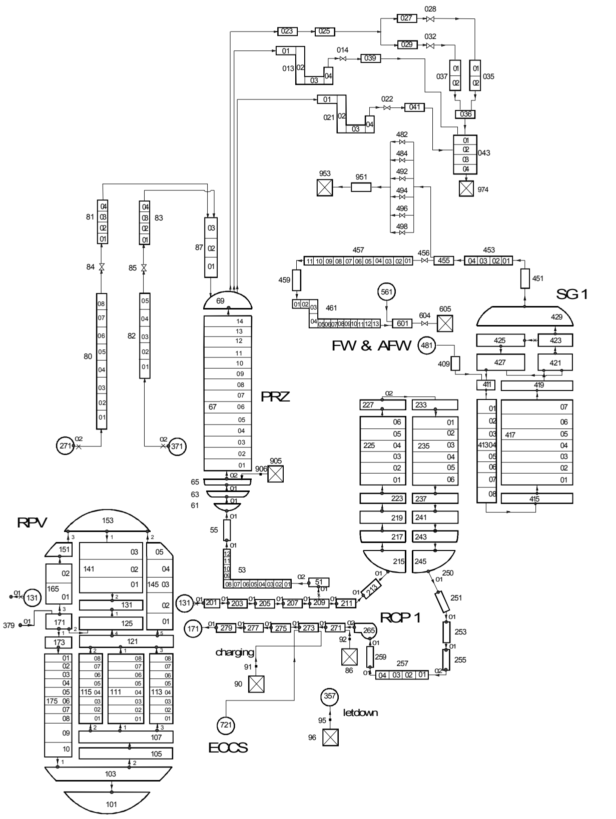

Fig. 1 RELAP5 SCHEMATIC OF THE PLANT

Steady state calculation was verified against real plant data and was found satisfactory. Nodalization has been qualified using plants tests and transients as well as "Kv scaled" calculation of ISP-27 (BETHSY Test 9.1b). In "Kv scaled" calculation all the main trends were predicted with reasonable accuracy. The plant tests used were ECCS full flow tests, accumulator discharge test and AFW system test. Good agreement between test and calculated data were obtained. Plant MSIV (Main steam Isolation Valve) event was calculated and verified against plant data. Little difference was found. Reactor trip calculation was performed to test functionality of the real plant control system model.

INADEQUATE CORE COOLING SCENARIO

Different LOCA scenarios were analyzed in order to evaluate physical parameters and phenomena relevant for the determination of background for operator actions within function restoration EOPs. The main goals of the analysis was to identify major phenomena relevant for the development of the background document for EOP, [9]. Also, analyses served to provide additional information to the expected sequence of events following different break sizes and locations on the RCS primary side occur. Additionally, analyses served to identify critical times for some operator action (minimum time since the beginning of the accident before which action needs to be employed).

Considering different break location, the LOCA transient analysis for development of EOPs have been categorized in four main subcategories with the addition of the fifth in which extreme beyond design basis failure of the ECCS have been considered in order to evaluate conditions in which inadequate core cooling threatens to damage the core: Cold Leg (CL) LOCA; Intermediate Leg (IM) LOCA; Hot Leg (HL) LOCA; Stuck open pressurizer (PRZ) Power Operated Relief Valve (PORV), as a special case of small break LOCA and Inadequate core cooling. For CL, IM and HL LOCA spectrum of breaks were analyzed ranging from leakage to double ended guillotine break. Additional sensitivity analysis was performed on CL to identify bigger and quicker discharge from the core and so larger core depletion. Influence of break location with respect to pressurizer was addressed (PRZ or non PRZ loop).

Pressure trends (Fig. 2 to Fig. 9) were used to define LOCA categories depending on approximate break size and safety related equipment status (one train versus two train operation of the ECCS):

1) Breaks with equivalent diameter at which the normal charging flow maintains RCS inventory.
2) Breaks with equivalent diameter at which minimum safety injection flow maintains RCS inventory and RCS pressure stabilizes above steam generator pressure.
3) Breaks with equivalent diameter at which maximum safety injection flow is initiated and re-pressurization of RCS occurs.
4) Breaks with equivalent diameter at which the RCS pressure decreases below steam generator pressure.
5) Double Ended RCS Pipe Break Area. RCS rapidly depressurizes to values close to the containment atmospheric pressure.

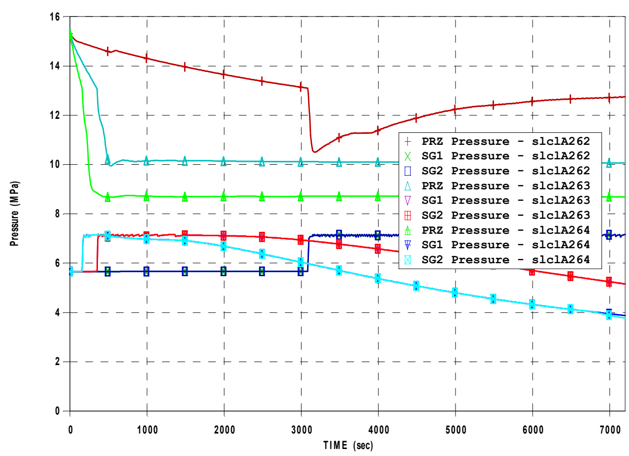

Fig. 2 PRIMARY AND SECONDARY PRESSURE FOR CL SMALL BREAK LOCA

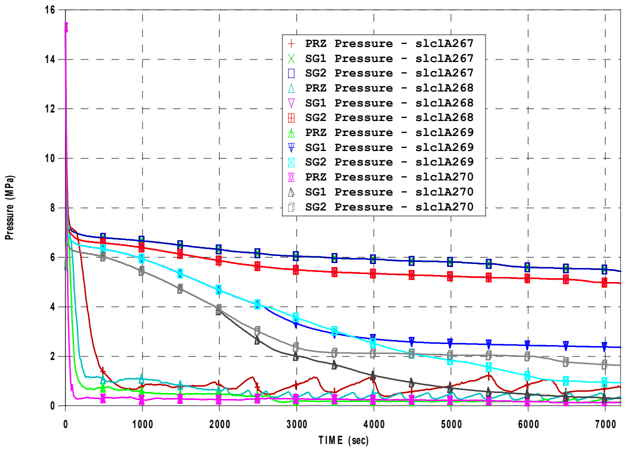

Fig. 3 PRIMARY AND SECONDARY PRESSURE FOR CL MEDIUM BREAK LOCA

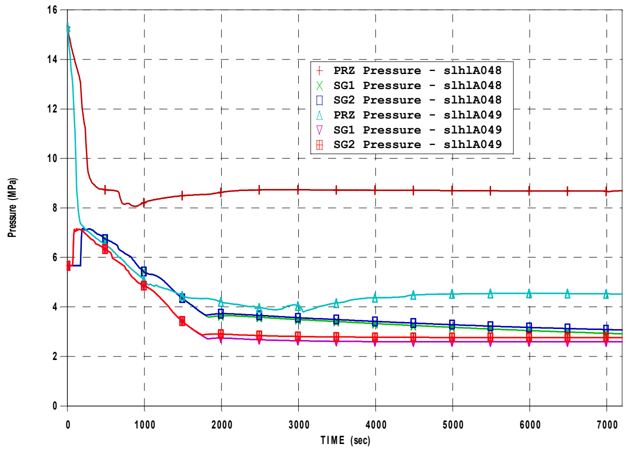

Fig. 6 PRIMARY AND SECONDARY PRESSURE FOR HL SMALL BREAK LOCA

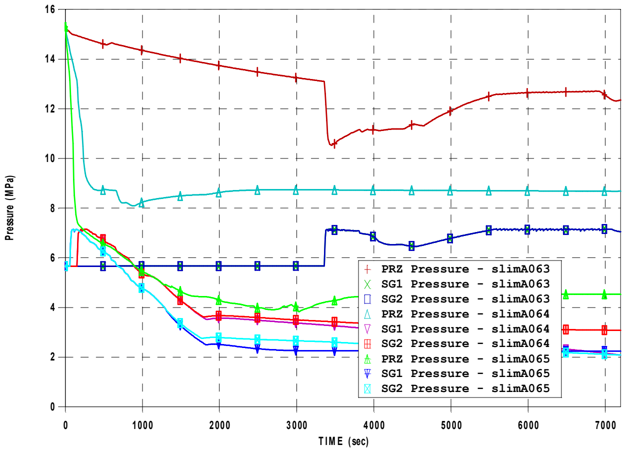

Fig. 4 PRIMARY AND SECONDARY PRESSURE FOR IM SMALL BREAK LOCA

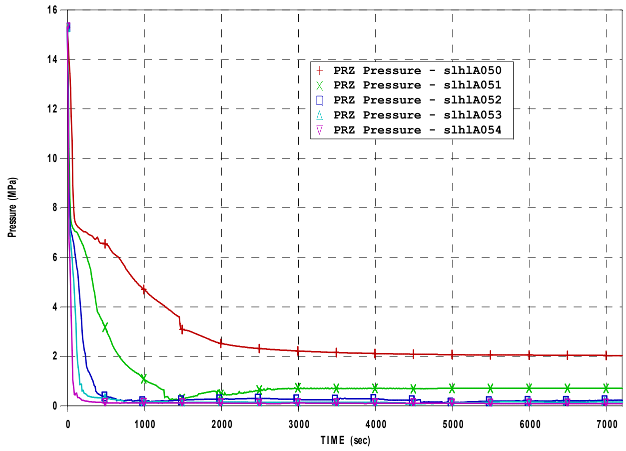

Fig. 7 PRIMARY PRESSURE FOR HL MEDIUM BREAK LOCA

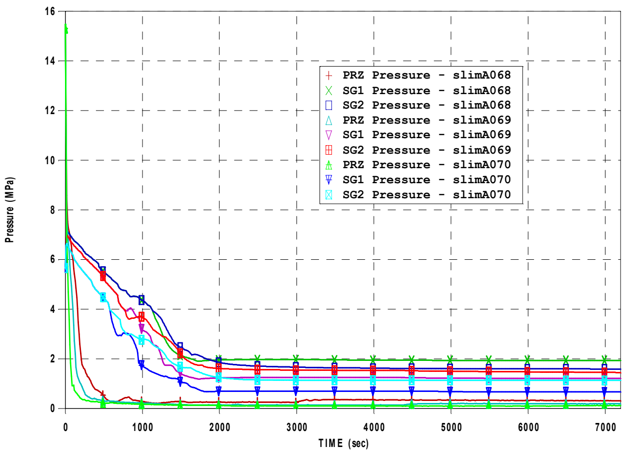

Fig. 5 PRIMARY AND SECONDARY PRESSURE FOR IM MEDIUM BREAK LOCA

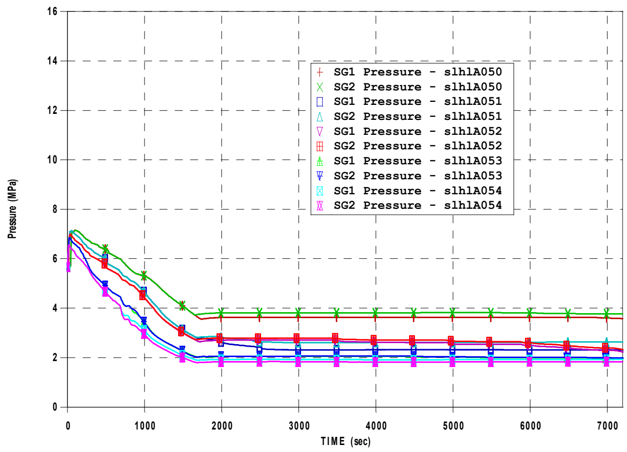

Fig. 8 SECONDARY PRESSURE FOR HL MEDIUM BREAK LOCA

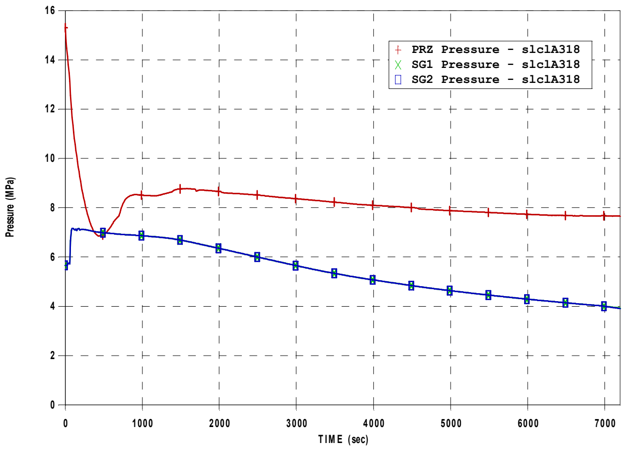

Fig. 9 PRIMARY AND SECONDARY PRESSURE FOR STUCK OPEN PRZ PORV

The most severe case from the point of the availability of the emergency systems from LOCA transient analysis is presented in hereafter, ICC 4.0" LOCA.

Initial conditions for the transient run are: 4.0" cold leg LOCA near RPV; primary pumps (RCPs) tripped on primary pressure lower than 10MPa; there are no active SI systems until core exit thermocouple reaches temperature of 650^{0}C (923.15K); one train of High Head SI (50% of HHSI pumps) is available afterwards with delay for operator action; accumulators as passive components are available from the start; MFW is tripped on SI signal; there is no AFW; SGs' PORVs (Power-Operated Relief Valves) and steam dump is unavailable.

After the opening of the break primary pressure begins its decline (Fig. 10) generating reactor trip, turbine trip and SI (but as assumed in inadequate core cooling scenarios, no injection is possible until 650^{0}C is reached with additional delay for operator action, except for ACC injection). The break is large enough to depressurize primary system to start of ACC early in transient (Fig. 11). The ACCs are soon empty, but their fluid volume is large enough to flood part of uncovered core before it begins to heat-up (Fig. 12). After the end of ACC injection further drainage from the core occurs and heat up begins. At around 2000s core level is at about 50% (Fig. 13) and abrupt temperature increase occurs (Fig. 12). After the temperature in CET reaches 923K (650^{0}C) operator establishes injection from SI system that exceeds break flow (Fig. 14) and lowers rod temperatures between 400 and 500K.

Even in the severe shown case core rod temperature is below required temperature limit (e.g. 2200^{0}F; 1478 K), [3].

The analysis demonstrated that time window for the operator to establish minimum ECCS flow is crucial to prevent core damage. Because of temperatures close to upper licensing limit, and associated uncertainty with operator action timing UMAE uncertainty analysis was performed.

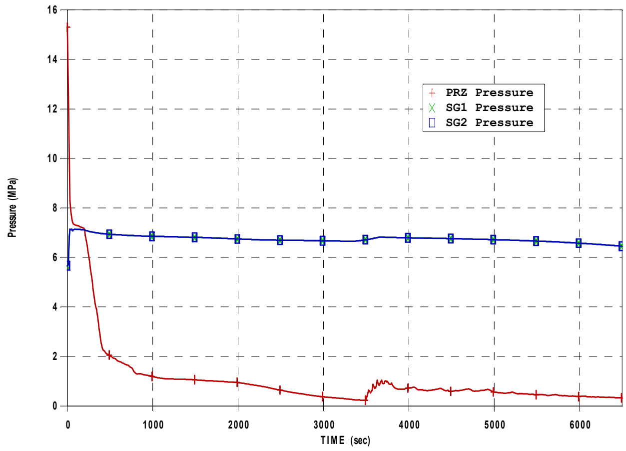

Fig. 10 PRIMARY AND SECONDARY PRESSURE FOR 4.0" ICC

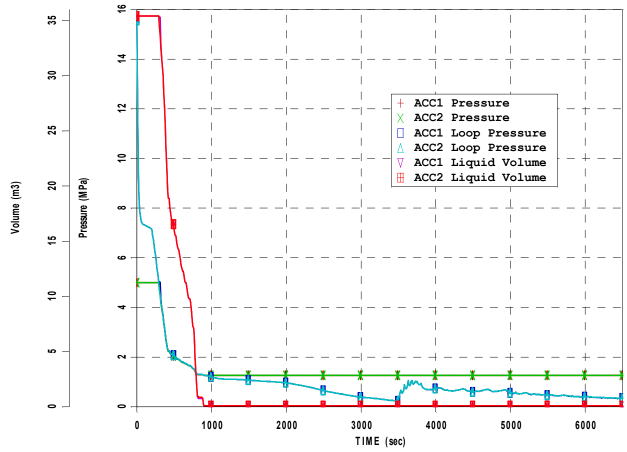

Fig. 11 ACCUMULATOR PARAMETARS FOR 4.0" ICC

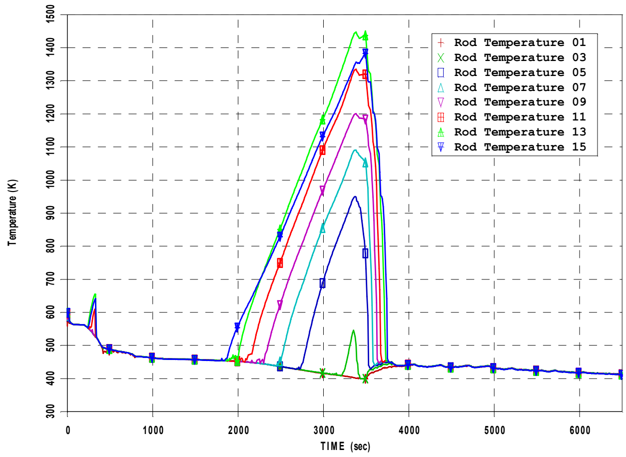

Fig. 12 ROD TEMPERATURES FOR 4.0" ICC

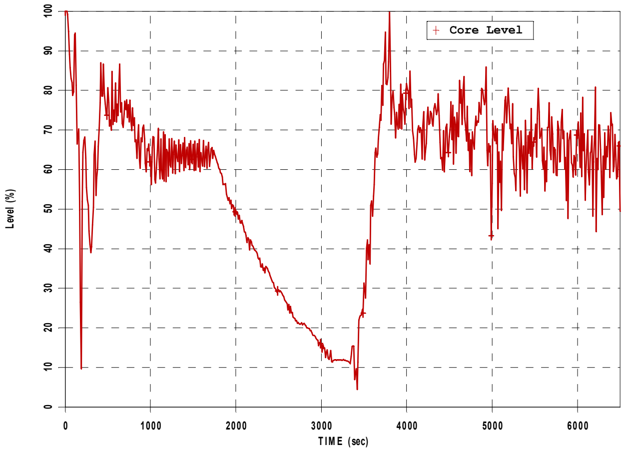

Fig. 13 CORE LEVEL FOR 4.0" ICC

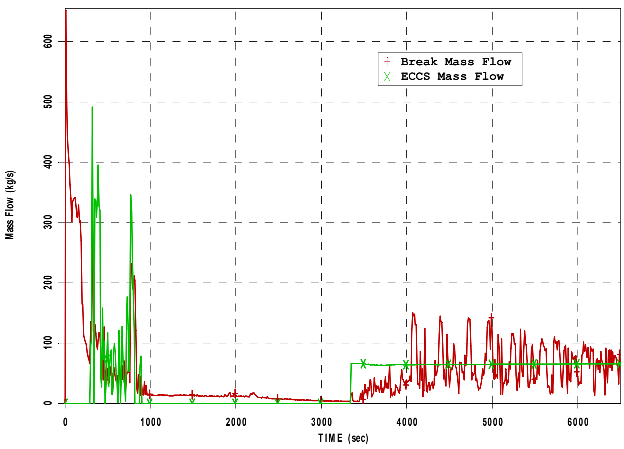

Fig. 14 BREAK AND ECCS MASS FLOW

APPLICATION OF THE UMAE METHODOLGY

Uncertainty methodology based on accuracy extrapolation (UMAE) relies on the use of the counterpart test data from the integral test facilities [1], [10] of different size. Special kinds of experiments are the so-called counterpart tests, [11]. These are similar experiments performed in differently scaled facilities aimed to evaluate the influence of the geometric dimensions of the facilities upon the evolution of a given accident. Transient scenarios measured in the experimental facilities of the reduced size cannot be directly extrapolated to plant conditions but provide sufficient evidence that calculations of the same tests are scale independent, [12]. The idea exploited is to extrapolate accuracy of the calculations for the counterpart tests performed in the integral test facilities that are scaled down replicas of the PWR plant (LOBI, SPES, BETHSY and LSTF). Two major values are used in this methodology: accuracy and uncertainty. Accuracy is related to the comparison between calculated (Y_c) and experimental data (reference measured value – Y_e), and is a measurement of the difference between experimental and calculated data, [12].

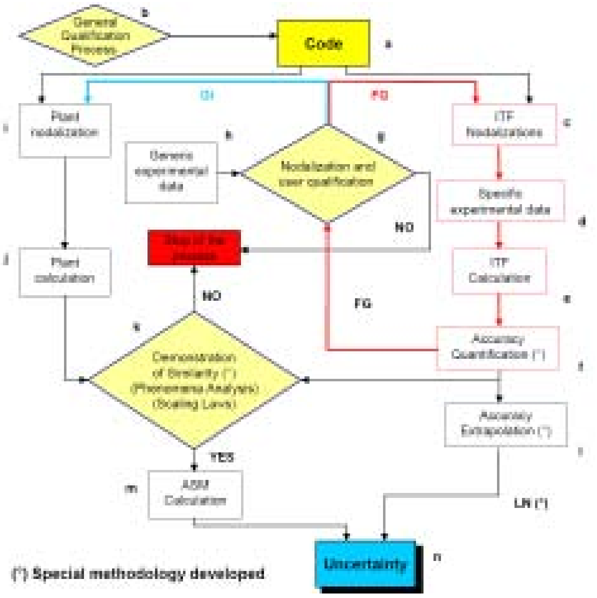

Fig. 15 UMAE FLOW DIAGRAM

Uncertainty is related to the prediction of incidental scenario in the nuclear power plant. It is a measure of the error in the evaluation of the different parameters during the transient. The overall process of UMAE is shown in Fig. 15. Focus on the work presented hereafter is on the last phase of the UMAE in which average accuracy is used to derive uncertainty values for the variables considered important to identify the time window for the successful operator action in case of ICC scenario.

Average accuracy [2] can be calculated for three groups of parameters:

1) individual single valued parameters that characterize the sequence of events;
2) parameters that have been obtained by non-dimensional analysis of phenomena such as natural circulation integral parameters.
3) integral parameters

The first group parameters were selected with reference to Relevant Thermalhydraulic Aspects (RTA) involved in the counterpart tests [7]. Each RTA was related to at least one single valued parameter. Counterpart test performed in all four facilities was a small break LOCA originated by a rupture in the cold leg without actuation of high pressure injection system and with accumulators available and as, such fully comparable to the ICC scenario.

Table 1 shows the correspondence between RTAs and selected single valued parameters for the particular transient scenario. The considered "amount" of single valued parameters identifies the relevant thermalhydraulic aspects and is suitable for fully characterizing the selected transient. Second and third parameter groups were not considered in the EOP application since the focus was on parameters that are readable from the instrumentation and available to the operator.

Table 1 CORRESPONDENCE BETWEEN RTAs AND SINGLE VALUED PARAMETERS

PHASE a) SUBCOOLED BLOWDOWN AND FIRST CORE DRYOUT REWET	
RTA	**SINGLE VALUED PARAMETER**
Pressurizer emptying	Time of pressurizer emptying
Maximum break flowrate/initial loop flowrate	Specific maximum break flowrate
Average specific break flowrate during phase a)	Average specific break flowrate during phase a)
First dryout duration	Time of first dry-out, Time of loop seal clearing, Minimum primary mass over facility volume
Period for dry out starting at bottom level (+)	Time of first dry-out, Time of loop seal clearing, Minimum primary mass over facility volume
Period for dry out starting at middle level (+)	Time of first dry-out, Time of loop seal clearing, Minimum primary mass over facility volume
Period for dry out starting at high level (+)	Time of first dry-out, Time of loop seal clearing, Minimum primary mass over facility volume
Direction of DC-UH bypass flowrate	Integral of pressure drop across DC-UH bypass in phase a)
Break two phase flow start	Time of break two phase flow start
Occurrence of loop seal clearing	Time of loop seal clearing
Occurrence of natural circulation	Average specific break flowrate during phase a), Time of first dry-out, Time of primary-to-secondary pressure reversal
PHASE b) SATURATED BLOWDOWN AND PRIMARY TO SECONDARY PRESSURE DECOUPLING	
RTA	**SINGLE VALUED PARAMETER**
Primary – Secondary pressure reversal	Time of primary-secondary pressure reversal
SGs U-tubes emptied	Time of SG U-tubes emptying, Integral of pressure drop between SG inlet plenum and U-tubes top
Direction of DC-UH bypass flowrate	Integral of pressure drop across DC-UH bypass in phase b)
Second dry out duration	Time of second dry-out, Minimum primary side mass over facility volume, Time of minimum primary side mass
Liquid hold up in steam generator	Time of SG U-tubes emptying, Integral of pressure drop between SG inlet plenum and U-tubes top in phase b)
Accumulator intervention period	Time of accumulators intervention
Average specific break flowrate during ACC intervention period	Minimum primary side mass over facility volume, Time of minimum primary side mass, Primary side mass over facility volume at time of third dry-out
Average specific break flowrate during phase b)	Average specific break flowrate during phase b)
Primary - Secondary pressure reversal	Time of primary-secondary pressure reversal
Saturation temperature decrease in SGs sec. side during phase b)	Secondary side pressure at phase b) start, secondary side pressure at phase c) start
Period for dry out starting at bottom level (+)	Time of second dry-out, Minimum primary side mass over facility volume, Time of minimum primary side mass

PHASE b) SATURATED BLOWDOWN AND PRIMARY TO SECONDARY PRESSURE DECOUPLING (cont.)	
RTA	**SINGLE VALUED PARAMETER**
Period for dry out starting at middle level (+)	Time of second dry-out, Minimum primary side mass over facility volume, Time of minimum primary side mass
Period for dry out starting at high level (+)	Time of second dry-out, Minimum primary side mass over facility volume, Time of minimum primary side mass
PHASE c) MASS DEPLETION IN PRIMARY LOOP	
RELEVANT THERMALHYDRAULIC ASPECT	**SINGLE VALUED PARAMETERS**
Third dry out duration	Time of third dry-out, Minimum primary side mass over facility volume, Time of minimum primary side mass, Primary side mass over facility volume at time of third dry-out
Minimum mass in the primary side	Minimum primary side mass over facility volume, Time of minimum primary side mass
Saturation temperature decrease in SGs sec. side during phase c)	Secondary side pressure at phase c) start, Secondary side pressure at phase d) start
Average specific break flowrate during phase c)	Average specific break flowrate during phase c)
Period for dry out starting at bottom level (+)	Time of third dry-out, Minimum primary side mass over facility volume, Time of minimum primary side mass, Primary side mass over facility volume at time of third dry-out, Time when heater rod temperature reaches 773 K
Period for dry out starting at middle level (+)	Time of third dry-out, Minimum primary side mass over facility volume, Time of minimum primary side mass, Primary side mass over facility volume at time of third dry-out, Time when heater rod temperature reaches 773 K
Period for dry out starting at high level (+)	Time of third dry-out, Minimum primary side mass over facility volume, Time of minimum primary side mass, Primary side mass over facility volume at time of third dry-out, Time when heater rod temperature reaches 773 K
PHASE d) INTERVENTION OF LOW PRESSURE INJECTION SYSTEM	
RELEVANT THERMALHYDRAULIC ASPECT	**SINGLE VALUED PARAMETERS**
LPIS intervention	Time when heater rod temperature reaches 773 K

(+) An important single valued parameter for this relevant thermalhydraulic aspect is the core level at the time of dry out but the uncertainties related to the position of the taps in the facilities and to the way it is derived in the calculation indicate it should not be used.

The "average accuracy" is calculated as the average difference from unity of the values of Y_e/Y_c (experimental-Y_e, calculated value-Y_c) provided for all four facilities (LOBI, SPES, BETHSY and LSTF):

$$\overline{A} = \frac{\sum_{i=1}^{4} \left| \left(Y_e / Y_c \right)_i - 1 \right|}{4} \qquad (1)$$

The next step was to identify the uncertainty bands directly applicable to the calculations of physical values. This represent the single valued parameters. Plant calculation was used as reference value (Y_R) and from this value the "average uncertainty" was calculated:

$$\overline{U} = \overline{A} * Y_R \qquad (2)$$

An "average accuracy" was defined by considering the dispersion of Y_e/Y_c (experimental over calculated value) for the four counterpart tests. The resulting accuracy has been applied to the ICC calculation.

Fig. 16 shows errors and their combined influences.

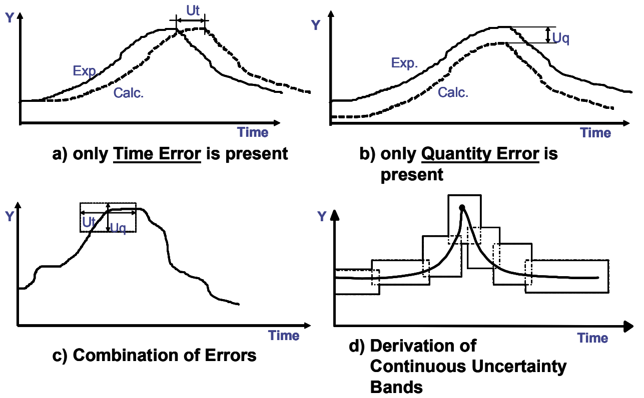

Fig. 16 SEPARATION AND RECOMBINATION OF TIME ERROR AND QUANTITY

THE RESULTS OF THE UNCERTAINTY EVALUATION

Based on the results and methodology presented in previous chapters, uncertainty evaluation has been applied to the ICC scenario.

Calculated uncertainty values are reported together with the reference values from the calculations in Table 2. Origin of the table is in [13]. Engineering judgment was used to select applicable single valued parameters, others were judged not relevant for shown variables.

Calculated uncertainties for the previously chosen single valued have been applied to establish uncertainty bands for the time trends of most interesting variables e.g. primary pressure and rod surface temperature, Fig. 17 and Fig. 18.

Table 2 CORRESPONDANCE AMONG RELEVANT THERMALHYDRAULIC ASPECTS AND SINGLE VALUED PARAMETERS

No	PARAMETER	$\overline{A}$	UNIT	ICC	
				Y_R	$\overline{U}$
1	time of pressurizer emptying	14.4%	s	20	2.9
2	peak cladding temperature at time of 1st dry-out	6.4%	K	655	42
3	time of first dry-out	13.9%	s	-	-
5	time of break two phase flow start	27.9%	s	20-80	16.7
6	time of secondary-to-primary side pressure reversal	5.6%	s	201	11.3
8	time of loop seal clearing	24.8%	s	180 (IL)	45
10	time of second dry-out	24.1%	s	240-340	24
14	time of accumulators intervention	3.9%	s	292	11.4
18	time of 3rd dry-out	15.2%	s	2000-3500	228
21	rate of temperature increase at time of third dry-out+	9.8%	K/s	0.72	0.07
22	time when heater rods temperature reaches 773 K	10.1%	s	2372	239.6
24	time when pressurizer pressure reaches 8 MPa	14.6%	s	36	5.3
25	time when pressurizer pressure reaches 6 MPa	4.8%	s	252	12.1
26	time when pressurizer pressure reaches 4 MPa	5.8%	s	333	19.3
27	time when pressurizer pressure reaches 2 MPa	19%	s	508	96.5
28	time when pressurizer pressure reaches 1 MPa	13%	s	1741	226.3
29	primary pressure at time of phase b) start	3.1%	MPa	7.10	0.22
30	primary pressure at time of phase c) start	2.4%	MPa	1.26	0.03
31	primary pressure at time of third dry-out start	15.8%	MPa	0.95	0.15

+The uncertainty about the rate of temperature increase and peak cladding temperature can not directly be extrapolated to the plant conditions because the behaviors of the electrical rods used in the test facilities and of the nuclear rods are different.

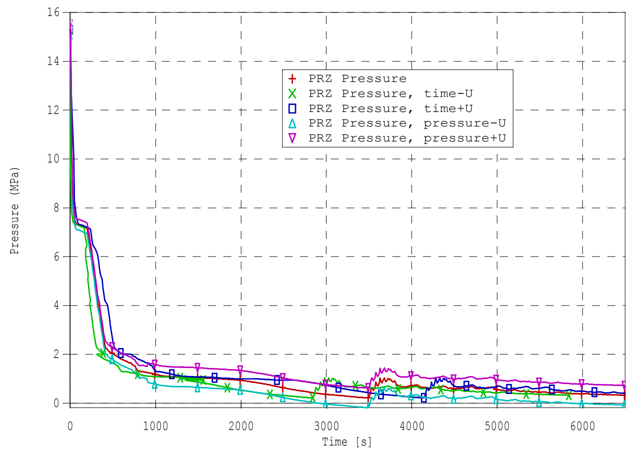

Fig. 17 PRESSURE UNCERTAINTY

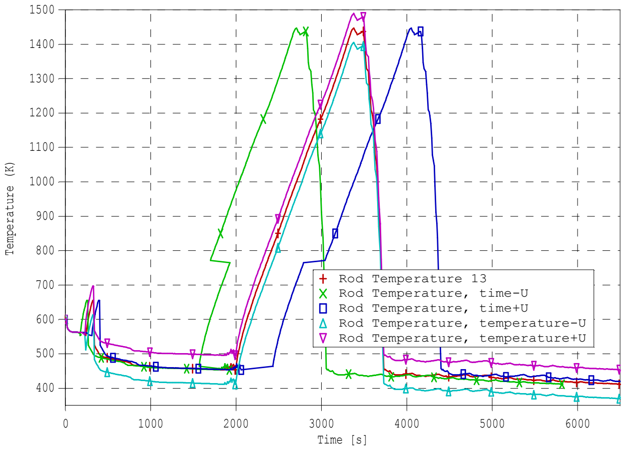

Fig. 18 ROD TEMPERATURE UNCERTAINTY

Single valued parameters related to the timing of the pressure evolution were applied to the calculated trend. Time was shifted for the value of the calculated uncertainty, e.g. if the uncertainty when pressure falls bellow 8MPa is 5.3s and the calculated value is 36s, than the trend was shifted to reach 8MPa in 30.7s and 41.3 seconds. Likewise, uncertainty of calculated value was added or subtracted to mean value to gain uncertainty region of calculated value.

Rod temperature uncertainty band was established similarly to the primary pressure trend by time shifted for the value of the calculated uncertainty, and for uncertainty of calculated value. It should be noted that uncertainties of single valued parameters provide scattered information about the uncertainty. Linear interpolation between the values for which uncertainty is known (from single valued parameters uncertainty) is not directly applicable because the time trend of a generic parameter is not linear in most of the cases.

The results show that minimum time for the operator to start emergency recovery of at least one HHSI train is within acceptable time of half and hour after the accident initiation (temperature excursion at $T \geq 650^{o}C$ with maximum of 10 minutes for the operator to establish required injection to prevent excessive heat-up of the core. The results are comparable to the values provided by similar studies preformed using sensitivity studies. Regarding the allowable duration it has been determined that uncertainty calculations provided more restrictive result for which additional evaluation might be needed regarding prolonged hydrogen generation since criterion for the reduction of the prolonged heat-up is not satisfied.

CONCLUSION

EOPs are developed with the intention to provide the protection of defense-in-depth barriers in the case of various accidents. Parameters that characterize plant states challenging these barriers are monitored in the EOPs. Generic accident analysis using sensitivity studies have been used so-far to define strategies, operator actions and define setpoint values (footnotes) in the procedures. Idea is to use time and "spatial" uncertainties verify critical time for operator action in beyond design basis situation instead of sensitivity studies. Critical scenario (ICC) with limited ECCS availability has been chosen as applicable for UMAE application.

The measured variables in SPES, BETHSY and LSTF facilities during the small break LOCA tests have been used to establish uncertainties using UMAE uncertainty methodology for the individual single valued parameters. In this way overall envelope was "developed" to cover uncertainty in calculation and to establish the minimum available time for operator action in ICC scenario. The application has provided the results comparable to the classic approach for such application using sensitivity studies. However, more conservative result has been obtained regarding the time at which core remains heated-up compared to the sensitivity studies. It should be noted that UMAE is restricted by the number of the available experimental database. However, recent development of the CIAU (Code with the capability of Internal Assessment of Uncertainty) methodology that is transient scenario much lesser restricted could be effectively used in such application.

ABBREVIATIONS

$\overline{A}$	average accuracy
ACC	Accumulator
AFW	Auxiliary Feedwater
CL	Cold Leg
CIAU	Code with the capability of Internal Assessment of Uncertainty
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
FSAR	Final Safety Analysis Report
HHSI	High Head Safety Injection
HL	Hot Leg
ICC	Inadequate Core Cooling
IM	Intermediate Leg
LHSI	Low Head Safety Injection
LOCA	Loss of Coolant Accident
MFW	Main Feedwater
MSIV	Main steam Isolation Valve
PORV	Power-Operated Relief Valve
PRZ	Pressurizer
PWR	Pressurized-Water Reactor
RCP	Reactor Cooling Pump
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RTA	Relevant Thermalhydraulic Aspects
RVLIS	Reactor Vessel Level Indication System
SG	Steam Generator
SI	Safety Injection
TC	ThermoCouple
UMAE	Uncertainty Methodology based on Accuracy Extrapolation
$\overline{U}$	average uncertainty

U_q	quantity error
U_t	time error
Y_c	calculated value
Y_e	experimental data (reference measured value)
Y_R	plant calculation (reference calculated value)

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