1. Introduction

Accident Management, and related terms ‘procedures’ and ‘strategies’ (instead of Management), constitutes a branch of the nuclear reactor safety. The understanding of the meaning and of the objectives of the AM branch requires the knowledge of the safety/licensing concepts like Design Basis Accident (DBA), Beyond DBA (BDBA), and Severe Accident (SA), as well as Probabilistic Safety Assessment (PSA) and role of Human Factors (HF) within nuclear reactor safety. Based on this terminology, the AM branch occupies a virtual region before or upstream the SA area and aims at forming an additional boundary to the progression of accidents that eventually escaped the DBA boundary. This is done consistently with findings and requirements of the PSA branch, taking into account of the HF and of the available NPP components and systems and of their actual status.

The purpose of this paper is to summarize the main findings from the execution of a wide range analysis focused to AM in VVER-1000 (within a European Commission funded Project) with main regard to the qualification of computational tools and the proposal for an optimal AM strategy for this kind of NPP.

2. Key-definitions and background

The following key definitions are at the basis of the AM activity performed within the Project [IAEA Safety Report Series No 23, 2002]:

1) **AM** – The taking of a set of actions during the evolution of an event sequence to a BDBA: a) to prevent the escalation of the event into a severe accident (preventive accident management measures); b) to mitigate the consequences of a severe accident (SAMG), c) to return the plant to a long term safe stable state (Accident Termination Procedures, ATP).

2) **EOP** – A set of documents describing the detailed actions to be taken by response personnel during an emergency. The plant specific procedures contain instructions to operating staff for implementing preventive accident management measures for both DBA and BDBA.

3) **SAMG** – A set of guidelines containing instructions for actions in the framework of severe accident management (SAM) where SAM is a subset of AM measures that: a) terminate core damage once it has started, b) maintain the capability of the containment as long as is possible, c) minimize on-site and off-site releases, d) return the plant to a controlled safe state.

Furthermore, it is considered that suitable EOP are part of any NPP operator book. Deviations from the EOP lines (e.g. failures outside the DBA boundaries or multiple failures), established in the operator book bring the system constituted by the NPP, the operators and by whatever connected (e.g. control room, logics of actuation, etc.) into the AM. In particular, the preventive AM Procedures, or Measures, or Strategies aim at avoiding the loss of structural integrity for the core as an ensemble (i.e. preventing the extraction of fuel bundles or the insertion of control rods) and the damage of a significant number of fuel rods (typically > 10% of the total number in the core). The above core status is also referred as “in-a-BDBA-situation-before-extended-degradation. The terms “preventive AM Strategies” better identify and characterize the activities of concern in this paper even though it is recognized that the terms Accident Management are used as a synonymous of preventive AM.

The background preparation for the AM study, other than the state of the art analysis for AM implied the availability of computational tools (codes, nodalisations, boundary conditions), the design and the execution of experiments, the demonstration of code-nodalisation quality, the availability of reference PSA studies (though PSA was not a primary interest for the Project) and the availability of reference NPP information.
The AM study starts from the observation that the investigation area is very broad: a large number of actions can be taken by operators utilizing several components and systems ranging from the Main Coolant Pumps (MCP), to the boron tanks to the pressurizer heaters and the spray lines, to the gas removal system to the fire-work pumps, to the water stored in the FW lines, to the BRU-A and BRU-K discharge. In order to limit the scope of the investigation, the attention was focused toward Station Blackout situations and the use of non-energized (at least non large-energy consuming) equipments. Therefore, the depressurization was selected as main strategy to be pursued following a multiple failure event, to bring the NPP to low pressure keeping the core geometric integrity. When at low pressure, i.e. below the set-point for actuation of the Low Pressure Injection System (LPIS), two alternative targets were fixed depending upon the availability or less of suitable LPIS flow rate: a) in the former case, the target is to show that primary system pressure remains at low values notwithstanding the injection and that stable cooling conditions (including non rising pressure) are established; b) in the latter case, the target is to delay the time of occurrence of significant core degradation.

The final step of the activity consisted in optimizing the strategy including the definition of key-operator actions and building up an AM strategy suitable for implementation in NPP after having demonstrated the effectiveness. Secondary system depressurization followed by primary system depressurization constituted the skeleton of the selected AM strategy. The availability of the coolant in the feed-water lines including the deaerator tanks was assumed. The demonstration of capability of computational tools to deal with expected phenomena, including data availability, was achieved. The step was completed through performing the following activities:

- Developing a procedure to optimize the design of an AM strategy.
- Optimizing the selected strategy that typically includes i) steam generator depressurization, ii) passive injection of coolant from the feed-water lines, iii) primary system depressurization owing to heat removal from steam generators, iv) continued primary system depressurization caused by delivering of accumulator water, v) continued primary system depressurization caused by the opening of pressurizer relief valve (PORV) till the LPIS set-point, vi) achieving the targets a) or b) of the previous paragraph.
- Applying the procedure with minor variants to the analysis of three BDBA scenarios assumed in Balakovo Unit 3 VVER-1000 NPP also establishing the basis for passing from an Accident Management Strategy to a Procedure.

The designed strategy and the related AM procedure outline well accepted by Balakovo 3 NPP.

2. The qualification methods and the role of uncertainty

A key feature of the activities performed in nuclear reactor safety technology is constituted by the necessity to demonstrate the qualification level of each tool adopted within an assigned process and of each step of the concerned process, [1]. Computational tools are used within the present context that include (numerical) codes, nodalizations and procedures. Furthermore, the users of those computational tools are part of the play and need suitable demonstration of qualification.

The ‘global’ qualification approach proposed by the University of Pisa, based on the UMAE methodology, has been adopted including the tool (FFTBM = Fast Fourier Transform Based Method) ‘to measure’ or to quantify the quality of a calculation in specified situations. The demonstration of code qualification implies the availability of qualified nodalizations (and qualified users). Criteria and thresholds of acceptability for calculation results at steady-state and at ‘on-transient’ level are introduced to this aim. Code-user effect and scaling issue are relevant in this connection [1].

The application of this methodology guarantees a suitable qualification level of all the tools invoked in the AM strategy investigation. As stated before to study a complex system like a NPP a set of tools are necessary that at the end constitutes the main instrument of the safety analysts. All those parts should be qualified following a method able to ensure the reliability of the obtained results.
The code have been qualified against experiments performed in a test facility, PSB-VVER, which its correct scaling have been addressed following the procedure reported in section 4.4.6 and demonstrated by the comparison between experimental evidences (section 6.6). It should be mentioned that also the experimental database against which the codes are usually validated needs to be qualified. Within the present Tacis Project the qualification of the experimental data has been proven and discussed in section 6.5. Emphasis should be put to the steps that brought to the definition of the test matrix (see section 5.2). A the top level scientists brainstorming is at the basis of the design of all experiments trying to define: suitable tests for the code qualification and experiments of major interest for the AM point of view.

The uncertainty (i.e. the process needed to associate errors to the prediction of best estimate codes engaged in accident analysis) and its evaluation (i.e. the capability to establish those errors) play a crucial role when a BE approach in code application is followed. The BE approach means use of a BE code (Relap in the present contest) and use of BE boundary and initial conditions.

At University of Pisa a specific tool for the uncertainty evaluation named CIAU has been developed (e.e. ref. [1]). The CIAU is based on the accuracy extrapolation from a database which contains a large number of code runs. Such code calculations have been validated against experimental data and used to ‘fix’ the error expected when a NPP transient scenario is calculated.

A necessary condition for the estimation of uncertainty by the use of CIAU is constituted by the availability of qualified experimental data.

3.    The experimental database

In the area of system thermal-hydraulics the PSB-VVER is one of the largest facilities (ITF = Integral Test Facility) put into operation with a power and volume scaling factor equal to 1/300. Data from ITF are necessary to identify phenomena expected in case of accidents in water cooled nuclear reactors and to demonstrate the qualification level of system codes. In addition, the PSB-VVER is the most qualified facility for the study of the VVER-1000 and the only one in operation. The created experimental database, applied to the AM study, consists of four key parts: a) the ITF description including test specific configuration and description of components added for the execution of individual experiments; b) the results from the characterization or shake-down tests (pressure drops, heat losses, volume vs height, etc.); c) the logic of imposed events in each experiment; d) the resulting sequence of main events and the time trends of a significant number of quantities.

Sixteen (16) experiments are part of the database. The actual quality of the database should be evaluated considering that for each experiment at least one pre-test and one post-test analysis have been performed and are documented. In connection with the number of time trends, about forty quantities are considered sufficient to identify any scenario in ITF and more than two-hundred time trends have been recorded and are available for each PSB-VVER experiment.

An outline of the database can be derived from Table 1, [2]. The detailed description of all the tests is beyond the purposes of the paper.

4.    The key results

The proposed strategy suitable for implementation in existing VVER-1000 NPP is based upon:

i) depressurization of the steam generators through the BRU-A and BRU-K, if needed;
ii) delivery of the coolant stored in the deaerator tanks to the steam generator(s) exploiting the driving force constituted by the vaporization subsequent to the depressurization;

| No | Test  | Id | Test type | Additional failure | AM strategy | AM set point | Note |
|----|-------|----|-----------|--------------------|-------------|--------------|------|
| 1  | Test 1| LFW-25 | LOFW | SS depressurization by SG1 & SG 4 BRU-A opening aiming at water injection from external | T core exit = 350°C |
| Test  | Activity Description                                                                 | Source                                                                 | T core exit      | T core exit      | Zaporozhye accident replication |
|-------|--------------------------------------------------------------------------------------|------------------------------------------------------------------------|------------------|------------------|---------------------------------|
| 2     | Test 2 LFW-28 LOFW                                                                   | 1) SS depressurization by SG1 & SG 4 BRU-A opening aiming at water injection from external source. 2) PS depressurization by PORV opening | T core exit = 350°C | T core exit = 300°C & PS pressure < 16 MPa |                                 |
| 3     | Test 3 PrzVS-01 SB LOCA                                                              | PORV stuck open HPIS intervention                                       | PS pressure = 8.8 MPa |                  | Zaporozhye accident replication |
| 4     | Test 4 CL-0.7-08 SB LOCA                                                            | SS depressurization by SG2 & SG 3 BRU-A opening.                         | T. rod surface = 450°C |                  |                                 |
| 5     | Test 5 SL-100-01 SL break + PRISE                                                     | PS depressurization by PORV opening & SS cool down procedure at 60 K/h. | After 30 min     |                  |                                 |
| 6     | Test 6 LFW-27 LOFW                                                                  | PS feed and bleed procedure by PORV opening, HPIS and LPIS injection.   | After 30 min     |                  |                                 |
| 7     | Test 7 BO-05 SBO                                                                    | SS depressurization by SG1 & SG 4 BRU-A opening at water injection from external source | T. rod surface = 350°C |                  |                                 |
| 8     | Test 8 CL-0.5-03 SB LOCA                                                            | PS feed and bleed procedure by PORV opening and make-up system injection. | T. rod surface = 450°C |                  |                                 |
| 9     | Test 9 PSh-1.4-05 PRISE                                                              | SS cool down procedure with a rate of 60 K/h.                           | After 30 min     |                  |                                 |
| 10    | Test 11 NC-6 NC                                                                     | -                                                                      |                  |                  |                                 |
| 11    | Test 12 CL-0.7-12 SB LOCA                                                            | 1) SS cool down procedure with a rate of 30 K/h 2) 1 HPIS recovery.     | After 30 min     | T core exit = 350°C |                                 |
| 12    | Test 12-2* CL-0.7-10 SB LOCA                                                         | SS cool down 30 K/h make-up system.                                     | After 30 min     | T. rod surface = 300°C | Test repetition                |
| 13    | Test 13* CL-0.7-10 SB LOCA                                                           | SS cool down 30 K/h make-up system.                                     | After 30 min     | T. rod surface = 300°C | Test repetition                |
| 14    | Test 13* BO-06 SBO                                                                  | SS depressurization by SG1 & SG4 BRU-A opening at water injection from external source. | T. rod surface = 350°C |                  | Single variant of test No 7.   |
| 15    | Test 14* PSh-1.4-07 PRISE                                                            | SS cool down procedure with a rate of 60 K/h.                           | After 30 min     |                  | Single variant of test No 9.   |
| 16    | Test 15* CL-0.7-13 SB LOCA                                                           | 1) SS depressurization by SG2 & SG 3 BRU-A opening 2) PORV opening.     | T core exit = 450°C after 30 min |                  | Single variant of test No 4.   |

Table 1 – Overview of the experimental activities, the use of the experimental data and the connection with the AM.

iii) depressurization of the primary system through the PORV and the gas removal system; iv) cooling of primary system ensured (for a period) by accumulators.

All of this implies no hardware changes in NPP and only introduction of suitable control logic for the involved components (BRU-A, BRU-K, PORV, gas removal system and FW line valves). Therefore,
assumed as 1 MEURO the daily operational cost of the NPP, the cost for the implementation of the procedure are negligible. The result of PSA studies performed in relation to PWR, demonstrate that the consideration of ‘passive’ depressurization has the potential to reduce the risk of core melt for a factor ten. However, the uncertainties for evaluating the risk shall be considered and specific PSA analyses for VVER-1000 should be completed (both of these activities are outside the Project boundaries).

Within the Project activities, [2], it has been demonstrated that the ‘grace period’ (i.e. the time period between the start of the accident and before a substantial core degradation occurs) following a station blackout, including the failure of diesel generators, for the selected VVER-1000 Balakovo NPP changes from about two hours to more than ten hours provided the considered AM strategy is implemented. The key results from the application of the procedure are illustrated in Fig. 1: the time when loss of geometric integrity occurs for the core is reported as a function of the pressure at which the event happens. Each of the dots in Figure 1 is the result of a ‘AM’ optimization calculation performed with reference to the Balakovo Unit 3 VVER-1000 NPP.

*Figure 1 – ‘AM map’ for VVER-1000 NPP following Station Blackout.*

Following a station blackout event including failure of emergency feedwater, any NPP has a ‘survival period’ that is typical of the order of two hours. This corresponds to the yellow-region in the left part of the Figure 1. When AM procedures are applied, the grace period (i.e. the survival time for the core) moves toward the right part of the diagram. Furthermore, the failure may occur at low or at high pressure, with the former situation being the preferable one from the safety stand-point.

Therefore the scenarios that are identified by bullets in the bottom right part of the diagram are the preferable ones and imply the demonstration of optimisation for the selected AM procedure. Additional information available from the Fig. 1 includes: a) uncertainty in predicting failure time is represented by horizontal bars; b) violet bullets reported in the bottom and top right of the diagram are ‘virtual’ scenarios end-points. These are obtained assuming that all the coolant stored in the NPP at the beginning of the transient is available to cool-down the core.
The final result can be summarized as follows:

- NPP failure time is expected at about 10000 s, that is a bit less than three hours, without AM.
- The use of AM (without any energy needed, apart for proper actuation of valves) ‘moves’ the NPP failure time at about 45000 s, that is at about 13 hrs.
- The maximum theoretical failure time is estimated at about 65000 s, that is about 18 hrs.

6. Conclusions

Various AM strategies were investigated. However, the reference strategy for the Project is constituted by the depressurisation of the steam generators followed by the primary system depressurisation that is actuated at an optimised time when the conditions of ‘maximum sub-cooling’ in the loop are achieved. In those conditions, coolant loss from the primary system and depressurisation rate are respectively minimized and maximized.

In between and as a consequence of the two AM depressurisation actions, ‘passive bleed’ of steam generators and of primary loop occurs from deaerator tanks and from accumulators that is sufficient to keep the core cooled with a suitable margin to DNB.

The objective of the selected AM strategy is two-fold: a) to keep cooled the core for the longest time without the availability of external energy sources, b) to keep the primary system pressure at the lowest level consistent with the first objective and with the overall strategy to minimize the risk of primary system failure at high pressure. The optimized AM strategy has been applied to the analysis of SBO, LOFW and PRISE scenarios, the last one originated by two (very) different breaks.

7. References

[1] D'Auria F., Bousbia-Salah A., Petruzzi A., Del Nevo A. "State of the Art in Using Best Estimate Calculation Tools in Nuclear Technology" J. Nuclear Engineering and Technology, Vol 38, No 1, February 2006, pages 11-32, ISSN 1738-5733

[2] D'Auria F. (Editor), Melikhov O., Suslov A., Bykov M., Elkin I., Araneo D., Cherubini M., Del Nevo A., Giannotti W., Muellner N. (Lead Authors) “Accident Management Technology in VVER-1000”. University of Pisa, 2006, ISBN 88-902189-0-9, pp. 1-1250