THE FIVE ESSENTIAL (´KEY`) ELEMENTS OF SEVERE ACCIDENT MANAGEMENT - TO BE DEVELOPED AS PART OF A SAMG INDUSTRY STANDARD

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ABSTRACT

The Fukushima-Daiichi accident has caused a renewed interest in tools and guidelines to mitigate severe accidents. Notably, industry approaches such as by the PWR/BWR Owners Groups have been reviewed and features added from the lessons learned.
The various severe accident management approaches vary considerably: they have different measures, different priorities for the various actions, different staff responsibilities and different sorts of communication to the off-site authorities. It appears that there is no common basis from which the approaches have been developed. In this paper, the five elements are treated which the author considers essential for proper tools to terminate severe accidents and mitigate their consequences.
These five elements should be trained in well-developed drills/exercises, involving all functions of accident management.
An industrial standard, comprising these and other elements is recommended, to define a minimum common basis, to which individual approaches should adhere and so decrease the large scatter in these approaches present now.

1. Introduction

The Fukushima-Daiichi accident in 2011 has caused a renewed interest in the tools and guidelines to mitigate severe accidents. Notably, industry approaches such as by the PWR Owners Group (PWROG) and BWR Owners Group (BWROG) Severe Accident Management Guidance/Guidelines (SAMG) - originating from the USA - have been reviewed and features added from the lessons learned from the accident, [1], [2], [3]. Initiatives have been taken also in other countries, such as the Rapid Deployment Force in France, [4]. A number of measures are described in the reports following the European Union stress tests, [5]. As a whole, many countries have reviewed and strengthened their approaches in the field of severe accident management. Apart from severe accident mitigating measures - by definition measures after initiation of core damage - also considerable measures have been taken in the preventive field. Examples are the FLEX approach in the USA, [6], and similar approaches in other countries, notably dealing with the use of portable equipment, stored on-site and/or off-site, and its deployment in case of an accident. An earlier development to FLEX is the series of Extensive Damage Mitigation Guidelines (EDMG), developed in the USA after the 9/11 attacks and which include the use of portable equipment and a number of manual actions in the absence of any AC and/or DC, [7]. In various plants in Europe, independent and bunkered decay heat removal systems had been installed already at an early stage, thereby providing similar protection.

Technically, the various severe accident management approaches vary considerably: they have different measures, different priorities for the various actions, different staff responsibilities and different sorts of communication to the off-site authorities. It appears that there is no common basis from which the approaches have been developed. In this paper, the five elements are treated which the author considers essential for proper tools to terminate severe accidents and mitigate their consequences. Should these five elements be agreed upon by the SAMG developers, we will have come closer to some form of industry standard for the SAMG, which would be a major step forward.

2. The Five Basic Elements of SAMG

In the subsequent discussion we recognise three essential (´key`) elements of SAMG, plus two supportive ´key` elements. Together, they form the five essential elements of SAMG. Note that the numbering does not reflect any sequence or priority – in principle all ´Key Elements` should be in place and, where they contain actions, actively be pursued.

2.1 Restoring core/fuel cooling

Severe accidents are, by the IAEA definition, accidents that involve significant core degradation, [8]. A subclass can be assigned to accidents at the spent fuel pool, involving fuel damage. In the IAEA terminology, severe accidents are a subclass of the Design Extension Conditions, [9], also known as Beyond Design Basis Accidents.

The major characteristic of such accidents is that the cooling of the core gets lost and the fuel will be overheated, with release of the fission products to the reactor coolant system (RCS) and, if the RCS is open to the containment, e.g. by a leak, a line rupture or one or more open primary Safety Relief Valves (SRVs), the fission products will enter the containment. It is clear that the cooling of the core must be restored, by injecting sufficient coolant at one side and rejection of the fission or decay heat at the other side. Of course, shutting down the nuclear chain reaction is a prime priority to minimize heat generation and either a scram must be initialized or, if it does not work, a procedure followed to shut down the reactor manually.

Failure of the core cooling is the consequence of extensive mechanical and/or electrical failure(s) and/or serious operator errors; consequently, repairs must be initiated to restore core cooling or alternate cooling methods must be initiated. This requires people to be dispatched to damaged areas, failed components, storage of portable equipment, etc. These people must work uninterruptedly until some core cooling /fuel cooling has been restored. Where such work may be hampered by potential or actual releases, placing the teams at personal risk. Where personal safety always prevails above reactor safety, teams may be forced to interrupt their work, possibly even leading to a renewed loss of core/fuel cooling[1].

Restoring core/fuel cooling may occur only late in the accident evolution, as it depends on the amount of damage, the available resources, the available manpower and, if needed, also on off-site support. Therefore, this possibly could only happen long after fuel damage has occurred, which in the mean time may have led to core collapse and subsequent vessel meltthrough. The cooling that ultimately will be achieved, therefore, may not be more than just debris cooling.

It may happen that other damage also must be mitigated, for example, there may be one or more big fires on the site. Or debris must be removed to permit access to certain key plant areas. There may be flooding of site areas with important equipment rooms. Or people are wounded and must be evacuated to hospitals. Also intruders may have taken violent actions against the plant and parts of the site are not under plant management control any longer. EDMGs have been designed to mitigate such damage and only if these are not effective, core damage will occur.

In summary, if core cooling is interrupted, repair teams must be set up to restore core cooling, possibly together with other actions to mitigate site damage. Note that once the core cooling capability has been restored, it is the responsibility of the Technical Support Centre to recommend or decide if and how it should be used, as RCS injection can also have detrimental effects, such as excessive generation of hydrogen or steam generator tube creep rupture.

We define restoring core cooling as Key Element #1 of the SAMG.

[1] Recall the need to evacuate personnel operating fire trucks used for core cooling at Fukushima-Daiichi several times, for their personal safety.

2.2. Protection of fission product boundaries and mitigation of releases

If we are unable to restore core cooling in time, fission products will be released from the fuel, possibly travelling to the containment. A plant has a number of fission product boundaries which can be challenged, dependent on the nature of the accident. Typical threats involve high temperature failure of steam generator (SG) tubes (known as creep rupture), failure of relief valves to reclose, high temperature / pressure failure of containment penetrations, high temperature / pressure failure of the containment or containment liner, melt-through of the containment basemat or containment failure by (long-term) containment sub-atmospheric pressure. The barriers would not all fail at the same time, i.e. there is a certain chronology in fission product boundary failures. For example, it is likely that a SG tube rupture will be an early failure, whereas a basemat melt-through – from Molten Core Concrete Interaction (MCCI) - will be a late failure. An example of the chronology of fission product boundary challenges is depicted in Fig. 1.

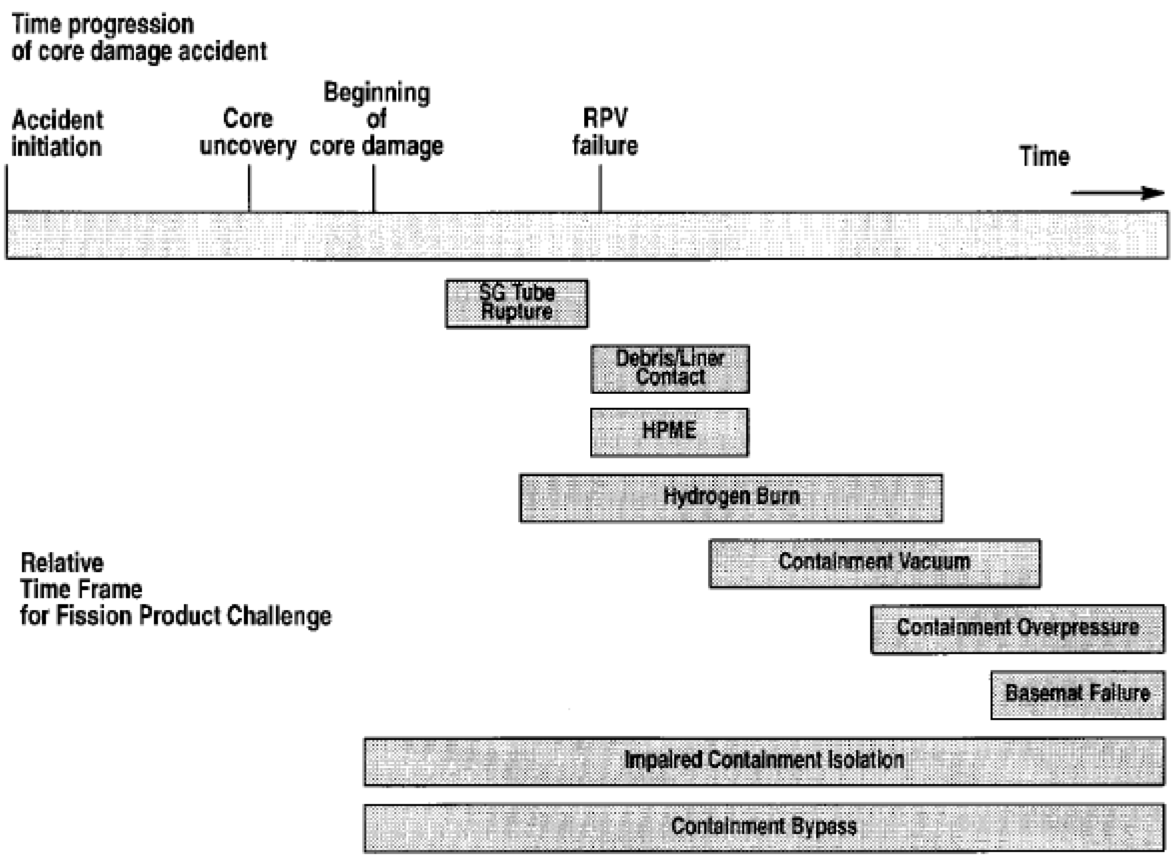

Fig. 1 – Time line for fission product boundary challenges (PWR), adapted from [10].

Similarly, not all failures will be as severe. An early containment failure will be a severe failure in terms of releases, whereas a basemat failure may be a relatively benign failure. Hence, a prime action during a severe accident is protection of the plant's remaining intact fission product boundaries and, if one or more of these already have failed, to mitigate the releases. Such releases may already be large, e.g. by an early containment liner failure, or they may be relatively small, for example from leaking relief valves. It can also happen that they do not yet occur but are anticipated to occur.

The responsibility for evaluating the accident evolution and selecting mitigative actions is often placed in the hands of a specialised body of experts, usually called the Technical Support Centre (TSC). With final decision making for mitigating measures often by the Site Emergency Director (SED), who is presiding over the plant Emergency Response Organisation (ERO). One of the tasks of the TSC is to define the priorities of the mitigative actions, as not all measures to protect fission product boundaries can be taken neither need to be taken at the same time. Similarly, it determines which ongoing or anticipated releases must be mitigated. In doing so, the TSC will be guided by their insights in the chronology as well as in the severity of the fission product boundary threats and the ongoing or anticipated releases.

It requires a substantial insight in the time scales and the processes of a severe accident to make these evaluations and prepare or make the decisions. It is not sufficient non-severe-accident experts to just follow a number of pre-defined plant parameters to be able to mitigate successfully a severe accident, as has been suggested by some SAMG developers. An insight in the nature and time line of severe accident phenomena is an absolute requirement for TSC staff. Also because certain mitigative actions in certain plant conditions can also have a detrimental effect and, hence, must be balanced against their anticipated benefit. An example: refilling an overheated core can lead to an SG tube creep rupture and, thereby, to a large release. Certain refill scenarios can also lead to a massive generation of hydrogen and subsequent explosions in the containment. Using containment spray to flood debris in the cavity can lead to a pressure surge in the containment, and so possibly fail the containment. Hence, basic guidance in the form of a set of guidelines is a helpful tool, but it does not replace TSC knowledge and insights. The EPRI Technical Basis Report (or equivalent documentation) gives here essential information and should be a key document in the training for deliberations and decision-making in the TSC, [11]. Further insights in these matters have been compiled from the Fukushima-Daiichi accident and are reported in the US Dept. of Energy study [12][2].

Insights in the plant damage states/conditions depend on instrument readings. Where these are unreliable or even lost (see detailed discussion in #4), the TSC can embark on a certain predetermined ´standard` series of actions. An example for a PWR is: first, prevent SG tube creep rupture, then prevent HPME, then prevent vessel meltthrough and/or basemat attack (by flooding the cavity, possibly high-up to take credit from the In-Vessel Retention (IVR) cooling mechanism), then protect the containment against overpressure and, later, after steam condensation, against sub-atmospheric pressure. During these actions, the TSC estimates the risk for hydrogen (and CO) combustion and takes appropriate action to mitigate it (e.g. by ignition, recombination, purging). Such a predetermined set of actions is, for example, contained in the Westinghouse Owners Group (WOG) SAMG, which even has two logic diagrams: one for the chronology and one for the severity of fission product boundary challenges, [13]. A difference with the approach as described in this article is that injection should not be an action in priority after SG tube creep rupture prevention and RCS depressurisation, but an ongoing action as described under #1, only to be decided upon after restoring the capability to cool the core by the TSC, after balancing positive and negative consequences of RCS injection, as discussed.

For a BWR, the example could be similar: first, look for a possible creep rupture challenge (e.g., unisolated steam lines), then prevent HPME, then prevent vessel meltthrough and drywell floor attack, then protect containment, as in the PWR-case. The actual strategies should also preserve the pressure suppression capability, as long as it is needed. An additional risk

[2] This author does not share the recommendation in [12], sec. 3.1, p. 15, that there is - apparently generally - a higher priority on injection of water to the reactor vessel compared to the primary containment. This statement has only a physical basis as long as the core geometry (fuel stack or debris/rubble bed) is coolable. In the later in-vessel phase, when a molten pool has formed, top cooling has not been proven to be a method to prevent vessel failure. But for a number of plants, IVR has been demonstrated to be able to prevent or at least delay such failure. Hence, in such cases, the TSC may (or even should) decide to prioritize containment injection above vessel injection. Once vessel failure has been determined to have occurred, the author concurs with the statement of [12], sec 3.1.

here is the BWR Mk I drywell liner meltthrough, well analysed in early days by F.J. Moody, [14].

Particular attention should be paid to potential fission product leakage paths from valves, drains, joints, seals, etc., as have been detected in [12].

Ways to set up guidelines for severe accident mitigation have been described extensively in the IAEA Safety Guide NS-G-2.15, [15]. It includes the search for plant vulnerabilities for severe accidents, development of strategies to mitigate the vulnerabilities, development of operating guidelines from these strategies, their verification and validation, staff responsibilities and education and training. As discussed, one of the important items is the balancing of the positive and negative consequences of proposed actions. Unfortunately, a frequent shortcoming in SAMG approaches is that not sufficient guidance is available for the TSC to make such balances; notably quantitative information about the possible positive and negative effects is usually lacking. Where such information is available, due consideration should be given to the uncertainties involved.

In summary, during a severe accident the TSC monitors the challenges to the fission product boundaries and takes mitigative action, in dependence of the time line and the severity of the challenges. The prime attention is with containment isolation and preserving the containment integrity. Techniques available are e.g. flooding and cooling the containment, and containment depressurisation. Similarly, the TSC mitigates ongoing or anticipated releases. When the capability to cool the core has been restored, they inject into the RCS in dependence of potential negative aspects of injection.

We define the protection of fission product boundaries and mitigation of releases as Key Element #2 of SAMG.

2.3 Investigation into the nature and volume of the anticipated radio-active releases

Most nuclear power plants (NPPs) have not been designed against severe accidents. This means that, even where they have a full package of SAMG installed, there is a considerable probability that they may not be able to fully manage the accident and that, hence, radioactive releases may occur. Where the situation may be aggravated by equipment malfunctions, human errors and unforeseen accident scenarios while following the SAMG. This places a responsibility on the accident management team to make an estimate of the potential release, should it fail to prevent such a release. As many plants have a Probabilistic Safety Analysis (PSA) - also called Probabilistic Risk Analysis (PRA) - a simple predictive tool is available while one compares the accident evolution with the ´model scenarios` from the PSA, from which the source terms are known. This requires an insight in the accident evolution, which may not be easy to obtain. This is an additional reason why the TSC must be severe accident experts and not just lightly trained (in terms of severe accident phenomenology) plant people.

Even where the exact cause of the accident may be unknown, it may be possible to get some insight in the evolution of the accident. One method is to try to obtain the Plant Damage Conditions (PDC), as they are described in the EPRI Technical Basis Report, [11]. Logic diagrams have been constructed to conclude to such PDCs, as e.g. is described in the Combustion Engineering Owners Group (CEOG) SAMG, [16]. These make it possible to estimate whether e.g. the core is still in the vessel or already ex-vessel, or whether the containment is intact, challenged, bypassed or ruptured. Unfortunately, there may be large uncertainties involved in such estimates. For example, after years of study, we still do not know whether the core in one or more reactors of the Fukushima-.Daiichi site has melted through the reactor vessel or not (but a high probability it has done so from plant analyses).

A newer method is to derive the accident scenario from instrument readings using various techniques. One is the use of an iterative method based on the CAMS method, developed in the OECD Halden Reactor Project and later investigated in a project by the

European Commission, [17]. Another method is the use of Bayesian Belief Networks, [18], which is one of the methods at present under study in another European Union project, [19]. Work has been performed at the GRS, Germany, and also implemented in a number of NPPs, [20]. Such methods give a spectrum of possible scenarios, with their estimated probabilities. This then gives insights in the potential source terms, which can be communicated to the off-site authorities for their actions to protect the public. The major objective is to take protective measures which correspond to the actual threat, and so avoid unnecessary evacuation. For example, in Fukushima many people were evacuated, as there were no proper tools to estimate upcoming releases. As was reported, many casualties were the consequence of the evacuation.

Even where no advanced tools are available to estimate the accident evolution and the upcoming source term, it is the task of the ERO (TSC) to do a best estimate, using engineering judgement, simple calculations, computational aids or other insights. Authorities *must* be informed about what will happen next, as their basis for protection of the public. A good example of such estimate is the assessment of the hydrogen that had accumulated in the Three Mile Island reactor pressure vessel. It was concluded that ways existed to remove it without danger to the containment. This insight led directly to the decision not to evacuate the surrounding area.

It is clear that the more knowledge is available about the evolution of the accident, the better mitigative strategies can be selected. However, the possibility of misdiagnosis does not disappear, Hence, the SAMG should never become *dependent* on recognition of the scenario.

We define the investigation into the nature and volume of the anticipated radio-active releases Key Element #3 of SAMG.

2.4 Proper understanding of instrument behaviour during severe accidents and its deviation from normal operation

The severe accident management guidelines direct the TSC to initiate, throttle and terminate the necessary actions to mitigate the accident. As discussed, these actions are directed to protect fission product boundaries and/or to mitigate releases. Whether the accident challenges fission product boundaries or whether releases are ongoing or anticipated must be derived from instrument readings. As most NPPs have not been designed to mitigate severe accidents, the instrument readings itself may also be influenced by the severe accident instrument environment. For example, an SG level measurement depends on the conditions around the reference leg. If the pressure in that environment deviates from its normal value, e.g. caused by a rising pressure in the containment, the level measurement deviates from its nominal conditions. Similarly, if the temperature in the containment is too high, the reference leg may boil - again causing a deviating pressure reading.

Some plants have dedicated accident management instrumentation, which of course is the best option, as the severe accident environment then has been considered in the instrument design basis. Even then, however, severe accident instrumentation has its limitations and does not cover all possible severe accident conditions. Current technology and economics play an important role in establishing the design conditions for severe accident instrumentation.

Instruments, therefore, should be investigated for three issues:

A. Does the instrument operate in its design range? It helps if for all relevant instruments the qualification data are known, so that the TSC rapidly can estimate the probability of failure under the prevailing conditions.

B. If the instrument has not (yet) failed, it should be known how much the instrument reading will be different from reading under nominal conditions.

C. Can the instrument also be used to obtain other information than it has been designed for?

The best-known example of the issue B is that the operators of the Three Mile Island reactor thought the RCS was full, because of the high water level in the pressuriser, whereas in reality the RCS had lost much coolant.

Many plants have qualification tables, from which it can be understood whether the instrument is still in its design range or qualified range. Which latter item means that the instrument still behaves well in a certain window outside its design range. Yet, how much an instrument will deviate under severe accident environmental conditions is often less well known or even unknown. In addition, it is always recommended to use more instruments to get one data point, in order to gain more confidence in the measured data. It is also recommended to also look for trends rather than single data points, as point values may be less reliable, whereas the trends still are, [21]. In this respect, one should observe that off-scale high or off-scale low indications do not necessarily mean the instrument has failed - it could be simply a data point outside the measuring range of the instrument.

It is also recommended to check whether measured data are in agreement with expectations regarding the accident evolution.

Another recommendation is to use, where possible, instruments for indications for which they have not been designed. For example, progress of core damage and core slumping can possibly be read from the behaviour of the Average Power Range Monitor (APRM).

The TSC should also have insights in the probability of failure of instruments. For example, instruments close to the RPV may fail due to the excessive heat and/or radiation, whereas the instruments in the containment may live much longer, as the conditions there may not deviate much from those during a Large Break LOCA (LBLOCA), for which the instruments have been designed. This effect may result in a more cautious reading of instruments close to the RPV, whereas instrumentation in the containment may be more trustworthy.

Where no plant data are available, e.g. because of failure of the instruments concerned, one can try alternate instrumentation or other techniques, as e.g. have been described in [22]. One can also try to use Computation Aids (CAs). Some SAMG developers, notably the Babcock & Wilcox Owners Group (B&WOG), [16], have developed quite sophisticated sets of such CAs. These serve also other needs of the TSC, e.g. to know how much water is needed to cool the core after X minutes. Which quantity also depends on the way one wishes to cool, i.e. with or without intentional evaporation of the injected water (detailed discussions on this topic in the EPRI Technical Basis Report, [11]). CAs, of course, can also help to gain confidence in certain instrument readings.

Apart from dedicated built-in instruments, one can also use portable instruments, e.g. portable flow meters. These must be used locally, so their suitability depends on the possibility to access the location needed and possibly the use of emergency equipment, such as a lightning bulb or a ladder to climb to a component.

If DC is lacking, one should be able to read instruments by auxiliary batteries (in an emergency also car batteries from the parking lot, as was done in Fukushima). This may also require staff to be able to dismantle cabinets, measure the voltage/current over cable ends and interpret the voltage/current by using calibration tables. Strategies to preserve DC (e.g. by load shedding) and use of recharging equipment are, of course, also beneficial. If pneumatic operation is needed but not available, one could possibly use portable air bottles. Of course, such handling requires training and experience.

Useful further information about use of instruments is contained in IAEA documentation, [21], and other literature, [22], [23]. Nevertheless, SAMG developers should create also guidance for the event no reliable instrumentation is available. This could include a ´black-start` of relevant equipment (e.g., the Reactor Core Isolation Cooling system, RCIC, for BWRs) or a ´pre-mature` containment venting - i.e. venting the containment before a substantial release of fission products to the containment has taken place - thereby gaining time for the later venting under a heavy load of radioactive substances. In the presently available SAMG, such ´black start´ had not yet been found. The matter is also discussed in [22].

Similarly, it may be wise in certain scenarios not to wait until instruments show plant parameters to change, which is the basis of various EOPs/SAMG, but to take anticipatory actions. An example is the shutdown and gradual depressurisation and cooling down upon a tsunami warning, [24]. Note: this is an extension of the existing practice in some countries (e.g. USA) to go to cold shutdown after a hurricane warning (for which there is more warning time – days, which time does not exist for tsunami warnings).

In summary, instruments are the eyes and ears of the operators and the TSC. Qualification tables are needed to estimate the reliability of the measurements. Deviation under abnormal conditions should be known. Emergency reading (via portable batteries, pneumatic) should be trained. Alternate use of instruments and use of CAs should be developed and trained. Guidelines should be prepared for the case no reliable instrument is available. In some SAMG approaches, guidelines are available that support the TSC in these tasks, mostly known under the name Technical Support Guidelines (TSG). An example of these techniques is presented in [3], [22], [28].

We define proper use of instruments, understanding of instrumentation behaviour during severe accidents and its deviation from normal operation as Key Element #4 of SAMG.

2.5 Proper command and control from a well-protected command centre, with clearly established responsibilities and in-depth severe accident expertise

Usually, the plant shift supervisor has responsibility for taking actions under all operating events, up to and including the most severe design basis accident. The procedures and guidelines to mitigate these events are usually quite prescriptive, which is possible because the evolution of the accident and the effect of countermeasures (e.g. actions by the Emergency Core Cooling System, ECCS) are well known. Such procedures are usually called Emergency Operating Procedures (EOPs) or Emergency Procedure Guidelines (EPGs).

Yet, in a severe accident situation, evaluation and decision making are far more complex tasks. Information of the plant status may be incomplete or in error, the evolution of the accident cannot be well predicted by existing computer codes, a number of potential actions has negative consequences and must be balanced against the expected benefits before making a decision. Systems that are needed for mitigative actions may be unavailable due to plant damage (e.g., no AC, no DC, no water). And it may be needed to ask for off-site help (portable equipment, additional staff, etc.). Due to these uncertainties, the guidelines are not prescriptive and it may even be needed to deviate from the guidelines and embark on improvisation.

Evaluation and decision making in such a situation is far more complex than under design basis accidents. For this reason, as described, the plant Emergency Response Organisation (ERO) takes over these tasks, usually with the help of the special body of experts, the Technical Support Centre. As these people are only available on call, the time that elapses before they have assembled on the plant site and can give their first recommendation to the control room staff must be covered by special guidelines, which should be suitable to bridge the gap between the EOPs and the SAMG, sometimes called Severe Accident Control Room Guidelines (SACRGs).

As the accident may result in radioactive releases, the Command Centre from which the ERO and TSC fulfil their duties must be well protected against such releases. Other required protection is against external events, such as flooding, fire and smoke, extreme weather, toxic gases and explosions. The Command Centre, hence, must have its own and robust AC power provisions, robust communication capabilities (satellite telephones), heating and cooling, and capabilities to lodge staff for extended time (breathing air, food, accommodation, medical care). Accidents are not ´over´ in hours or days - it can take weeks or months or longer before more or less normal conditions on-site have been re-established. Note that also plant workers need protected zones to rest, take orders, plan activities, etc.

Many plants have also an off-site Command Centre (often demanded by regulation), but this does not dilute the requirements on the on-site Command Centre. In addition, in the various

SAMG exercises which this author has observed, it was never tried to shift the command from the on-site to the off-site Command Centre in the mid of the accident.

The Command Centre should have access to plant data, possibly provided on-line, and appropriate documentation (drawings, specifications for systems and equipment). All staff functions and responsibilities should be well defined and described, and adequate training (initial and refreshing) should be available for all these functions. The functions should be distributed over available staff in adequate redundancy; no function should be allocated to a single individual (even not the highest commanding function).

In summary, command and control during a severe accident should be from a command centre that is well protected against radioactive releases and external events, and is so for prolonged time. Functions and responsibilities of the staff should be well described and trained.

We define proper command and control from a well-protected command centre, having appropriate staff including severe accident experts and with clearly established responsibilities and lines of authority as Key Element #5 of SAMG.

2.6 Overview

The five ´Key Elements` as described above are depicted in Fig. 3. The prime actions are present in the Key Elements, shortly ´Keys` #1 - #3 - to be executed largely in parallel - whereas the necessary supportive functions are in the Keys #4 and #5.

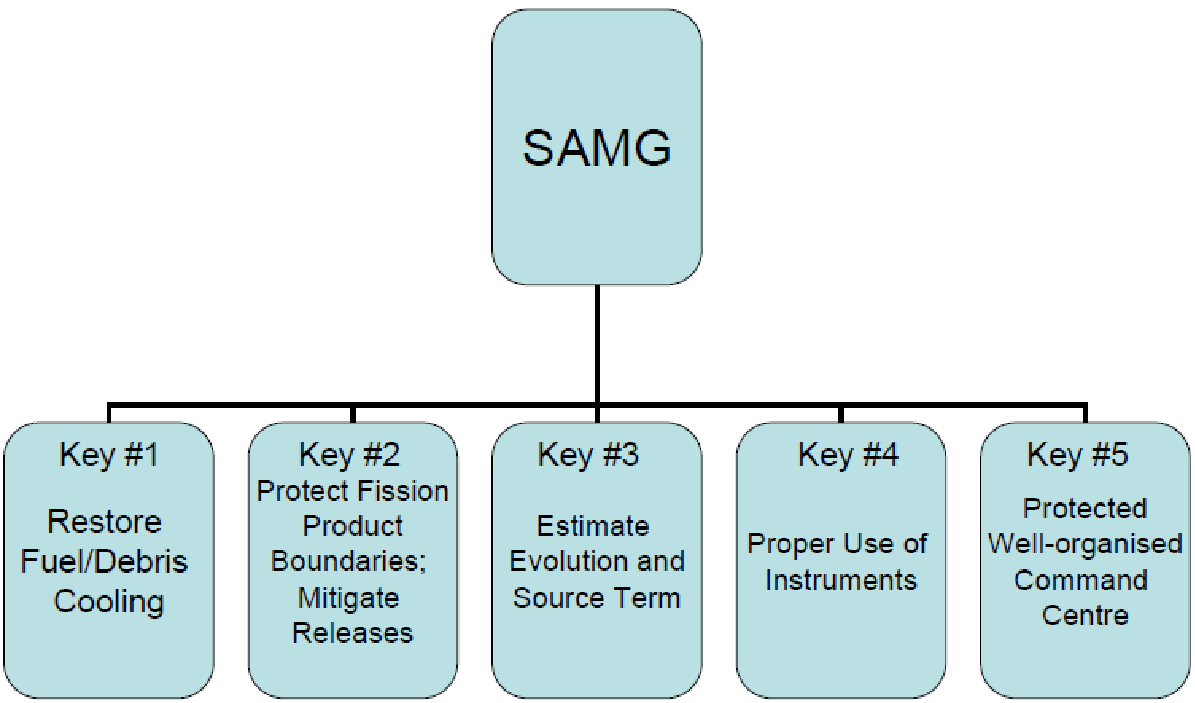

Fig. 3 – The Five ´Key Elements` of Severe Accident Management and Severe Accident Management Guidance

3. Conformance of existing practices with the five ´Key Elements`

The author has investigated a number of SAMG approaches and observed a number of plant SAMG exercises, and assessed these with respect to the above mentioned ´Key Elements`. It appeared that many plants had done an appreciable effort to develop and implement a SAMG programme, which is an important first step. Yet, a number of nonconformances with the ´Key Elements` have been observed. Examples:

3.1 In a number of PWR SAMG approaches, the very first SAMG action is to inject into the RCS, in order to terminate the progress of the severe accident, [16]. Although this sounds understandable, it does not seem to be the right way of thinking. We have progressed from some initiating event to a severe accident due to prolonged absence of core cooling, and core damage is imminent or has already occurred. It cannot be assumed that RCS injection is suddenly again available at the transition to SAMG; one should therefore shift its efforts to protecting fission product boundaries. In stead of a *sequence* of SAMG actions in terms of first trying to inject into the RCS and then trying to protect fission product boundaries, one should work on these issues *in parallel*. And certainly not wait to initiate actions to protect fission product boundaries after one has first tried to restore RCS injection and cooling. Although the SAMG approaches mentioned allow staff to enter various Severe Accident Guidelines (SAGs) at the same time, the approach of injecting into the RCS as the first SAG dilutes the attention to the main task of severe accident management, i.e. protection of the remaining intact fission product boundaries.

3.2 In some other PWR SAMG, injection into the RCS is not the first SAG, it comes after SAG-1 which directs the plant staff to inject into the steam generator (SG). Although the focus is correct, i.e. on the protection of fission product boundaries (here prevention of SG tube creep rupture), it cannot occur that efforts to inject into the RCS are of a lower priority - such efforts must start at the moment the injection is lost and be continued until the capability to inject has been restored, as before discussed under 2.1. The actions under SAG-1, injection into the SG, and - in this case, [13] - SAG-3, injection into the RCS, are, therefore, not to been seen in *sequence*, but in *parallel*. In terms of the ´Key Elements`, both Key Elements #1 and #2 must be worked on in parallel - they are not to be intermixed or merged into one Key Element, with a lower priority for restoring core cooling. Only if under all restorative work only one water source has become available, it should be balanced by the TSC where to put the water: the SG or the RCS. The TSC may then decide to inject into the SG, as injection into the RCS may result in a pressure spike, further endangering the risk for SG tube creep failure or to a hydrogen spike in the containment, possibly resulting in a damaging explosion in the containment. If such risks are deemed to be acceptable, the TSC may indeed decide to first inject into the RCS. Injection into the SG may also result in RCS depressurisation, thereby reducing the risk for an SG tube creep failure. In addition, mitigating this risk can also be achieved by RCS depressurisation, so the TSC has here various options.

3.3 Some BWR SAMG approach departs from a logic diagram which contains questions about the status of the RPV and its water content, Fig. 2, [2]. Such an approach places too much attention to the Key Element #1, whereas protection of the fission product boundaries, the Key Element #2, is apparently placed at a lower hierarchy. Also here both Key Elements are intermixed and, in the opinion of this author, even in the wrong sequence: where injection into/cooling of the RCS has been lost, prime attention must be placed on the preservation of the fission product boundaries, [25].

3.4 An additional problem here is that the first question – whether the RPV has failed and the core is ex-vessel – may be impossible to answer in, say, a matter of *minutes* or even *hours*: it took several *years* before we obtained some understanding of the location of the core in the case of the Fukushima-Daiichi reactors. Where it still has not been *determined* that the core is ex-vessel, as the logic tree asks. Similarly, the question whether under sufficient injection the core will remain inside the vessel cannot be answered if the core has lost its coolable geometry. In this respect it should be observed that logic trees should not contain questions which are difficult or impossible for the TSC to answer within an appropriate time.

3.5 In general, a number of SAMG approaches place high emphasis on such actions as injecting into the core, depressurising the RCS, filling the steam generator, which most probably already have been tried in the EOP-domain and were apparently not or not fully successful. It does not seem to be a good use of scarce resources to place emphasis early in SAMG-domain on actions that already had failed in EOP-domain, [26]. The equipment failures, operator errors or some big external event that caused the failures, have not suddenly disappeared at the transition from the EOP- to the SAMG-domain. This places a high relevance on Key Element #2, protection of fission product boundaries and mitigation of any potential releases.

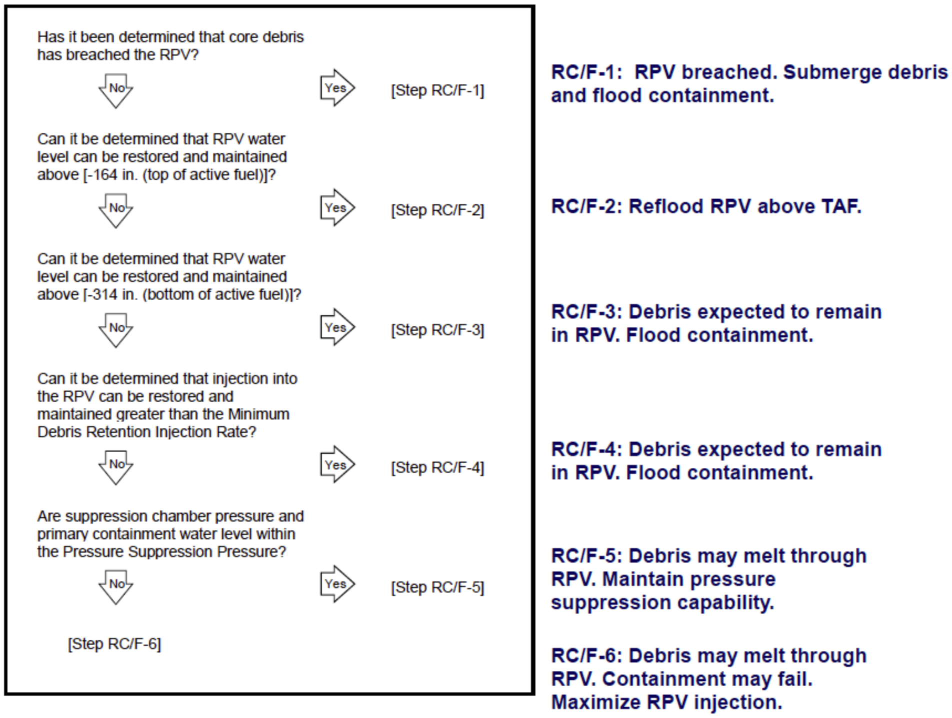

Fig. 2 – Example of a BWR Severe Accident Guidelines Decision Loop[3], [2].

3.6 Most SAMG approaches have no clearly defined transition guidance for the time that elapses between the initiation of the severe accident and the time the ERO has assembled and is able to guide the MCR staff through the accident, called above the SACRGs. In only one approach, such guidance was clearly formulated and assigned to plant staff, [3], [10], [13].

3.7 In most SAMG programmes which this author observed, it was not tried to estimate an upcoming source term, in order to inform the off-site emergency planning organisation. Usually, one waited until the first releases were measured, which information then was passed on to the authorities. Even then, an estimate of future further releases was not attempted. In view of the potentially catastrophic nature of a full and short-term evacuation of a large area, the estimate of an upcoming source term is of utmost importance. In Fukushima, many lives were lost due to such large scale evacuation, where the radiation risk had not yet been established. Note: such predictive tools have recently been developed by GRS Germany and implemented in a number of German NPPs, [20].

3.8 In observing SAMG exercises, this author saw only few occasions where plant staff questioned the proper functioning of the instrumentation which they consulted for their actions. Neither was it tried to use multiple indications to minimise the risk of faulty readings. And it was seldom questioned whether the planned actions could have any negative effects. Quite a few people did not stick to their functions. For some functions in at least one plant, there appeared to be not even a function description, not to speak about a defined training programme for these functions. Some plants had kept the responsibility with the shift supervisor, where he/she clearly lacked the necessary training for evaluation and decision making during a severe accident.

[3] In the mean time revised but main principles have been maintained, [28], [29] slide 17.

3.9 In observing plant SAMG exercises, this author has seen very few plants with an adequate protection of the Command Centre. Some TSCs were placed adjacent to the Main Control Room (MCR), which facilitates communication between the TSC and the MCR, but makes the TSC inoperable the moment the MCR must be evacuated. Other plants used normal office building facilities for their Command Centre, i.e. without any protection against radioactive releases or external events. Also no independent power sources were available, which makes the whole ERO inoperable in case of a station black-out (SBO). The TSC staff was following the accident on their laptops, without any possibility to recharge the batteries in that SBO. Again another Command Centre of a coastal plant was situated underground - the first location to get lost in a flooding accident. Not having an appropriate Command Centre is identical to *not having any SAMG in place*.

The conclusion of these examples is that, although progress in SAMG has been made, improvements are still possible and needed, as not all ´Key Elements` are yet in place. Approaches not having Key Element #5 in place should not claim to have any functioning SAMG.

4. The need to harmonise SAMG - development of an industrial SAMG standard

SAMG is in many countries not the subject of nuclear safety regulation. The consequence is a wide scatter of approaches, in varying depth and volume, and with different philosophies. Some harmonisation has been initiated for the PWR SAMG, in the new PWR Owners Group (PWROG) SAMG, [3], but still other approaches exist. The approach for most BWRs is much different from the PWRs, although the goals of SAMG should be very similar.

In the absence of regulation, it is proposed to develop an industry standard, similarly as a century ago an industry standard was developed for pressure vessels in order to terminate the series of accidents with such vessels (i.e., the ASME Boiler and Pressure Vessel Code in the USA, and similar Codes in other countries). Now the number of accidents at NPPs is, fortunately, very low, yet a robust SAMG program should exist at all plants to mitigate such an accident, in the unlikely event it occurs, as no plant is immune to severe accidents.

It is believed that the above mentioned five Key Elements are a good starting point for such standard. Additional basic documentation is believed to be in the IAEA Safety Guide NS-G-2.15, [15], and the Safety Series Report #32 on Accident Management Implementation, [27]. Industry approaches existing today, [2], [3], [28], have already many valuable elements which can be used in such a standard.

Somewhere, sometime, another severe accident will occur, similar to or different from the accident at the Fukushima plants - it should find us prepared. Hence, it is time to act.

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