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Review of current Severe Accident Management (SAM) approaches for Nuclear Power Plants in Europe

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Abstract
The Fukushima accidents highlighted that both the in-depth understanding of such sequences and the development or improvement of adequate Severe Accident Management (SAM) measures are essential in order to further increase the safety of the nuclear power plants operated in Europe. To support this effort, the CESAM (Code for European Severe Accident Management) R&D project, coordinated by GRS, started in April 2013 for 4 years in the 7th EC Framework Programme of research and development of the European Commission. It gathers 18 partners from 12 countries: IRSN, AREVA NP SAS and EDF (France), GRS, KIT, USTUTT and RUB (Germany), CIEMAT (Spain), ENEA (Italy), VUJE and IVS (Slovakia), LEI (Lithuania), NUBIKI (Hungary), INRNE (Bulgaria), JSI (Slovenia), VTT (Finland), PSI (Switzerland), BARC (India) plus the European Commission Joint Research Center (JRC).
The CESAM project focuses on the improvement of the ASTEC (Accident Source Term Evaluation Code) computer code. ASTEC,, jointly developed by IRSN and GRS, is considered as the European reference code since it capitalizes knowledge from the European R&D on the domain. The project aims at its enhancement and extension for use in severe accident management (SAM) analysis of the nuclear power plants (NPP) of Generation II-III presently under operation or foreseen in near future in Europe, spent fuel pools included.
In the frame of the CESAM project one of the tasks consisted in the preparation of a report providing an overview of the Severe Accident Management (SAM) approaches in European Nuclear Power Plants to serve as a basis for further ASTEC improvements. This report draws on the experience in several countries from introducing SAMGs and on substantial information that has become available within the EU "stress test".
To disseminate this information to a broader audience, the initial CESAM report has been revised to include only public available information. This work has been done with the agreement and in collaboration with all the CESAM project partners. The result of this work is presented here.

Contents

Abbreviations and acronyms

AM	Accident Management
AMP	Accident Management Programme
AUX	Auxiliary Building
B/C	Bubbler/Condenser (tower - case of VVER-440 unit)
BDBA	Beyond Design Basis Accident
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Uranium (Reactor)
CET	Core Exit Temperature
CFD	Computational Fluid Dynamics
CHLA	Candidate High Level Action
CMT	Containment
DBA	Design Basis Accident
DCH	Direct Containment Heating
DFC	Diagnostic Flow Chart
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
EOP	Emergency Operating Procedures
EPR	European Pressurized Reactor
FA	Fuel Assembly
FCVS	Filtered Containment Venting System
FCI	Fuel Coolant Interaction
FP	Fission Products
GIAG	Severe Accident Intervention Guide
HHSI	High-Head Safety Injection
HP	High Pressure
IVR	In-Vessel Retention
LHSI	Low Head Safety Injection
LP	Low Pressure
LWR	Light water Reactor
MCCI	Molten-Core-Concrete-Interaction
MCR	Main Control Room
MOX	Mixed Oxide (fuel)
NPP	Nuclear Power Plant
OG	Owners Group
PAMS	Post Accident Monitoring System
PAR	Passive Autocatalytic Recombiner
PORV	Pilot Operated Relief Valve
PSA	Probability Safety Assessment
PSR	Periodic Safety Review
PSV	Pressurizer Safety Valves
PWR	Pressurized Water Reactor
RCS	Reactor Cooling System
RCP	Reactor Coolant Pump
RPV	Reactor Pressure Vessel
RWST	Refuelling Water Storage Tank
SA	Severe Accident
SAM	Severe Accident Management

SAMG	Severe Accident Management Guidelines
SARNET	Severe Accident Research NETwork
SBEOP	Symptom-Based Emergency Operating Procedures
SBO	Station Black-Out
SCST	Severe Challenge Status Tree
SFP	Spent-Fuel Pool
SG	Steam Generator
SGSV	Steam Generator Safety Valve
SI	Safety Injection
SPDS	System Parameter Display System
TMI	Three Mile Island
TSO	Technical Support Organisation
TSC	Technical Support Centre
TG	Task Group
VVER	Water-Water Energetic Reactor (or WWER)
WENRA RL	WENRA Reference Levels
WOG	Westinghouse Owners Group

1 Introduction

The Fukushima accidents highlighted that both the in-depth understanding of such sequences and the development or improvement of adequate Severe Accident Management (SAM) measures are essential in order to further increase the safety of the Nuclear Power Plants (NPPs) operated in Europe. To support this effort, the CESAM (Code for European Severe Accident Management) R&D project, coordinated by GRS, started in April 2013 for 4 years in the 7th EC Framework Programme of research and development of the European Commission. It gathers 19 partners from 12 countries: IRSN, AREVA NP SAS and EDF (France), GRS, KIT, USTUTT and RUB (Germany), CIEMAT (Spain), ENEA (Italy), VUJE and IVS (Slovakia), LEI (Lithuania), NUBIKI (Hungary), INRNE (Bulgaria), JSI (Slovenia), VTT (Finland), PSI (Switzerland), BARC (India) plus the European Commission Joint Research Center (JRC).

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In the frame of the CESAM project one of the tasks consisted in the preparation of a report providing an overview of the SAM approaches in European Nuclear Power Plants to serve as a basis for further ASTEC improvements. This report draws on the experience in several countries from introducing SAMGs and on substantial information that has become available within the EU “stress test”.

To disseminate this information to a broader audience, the initial CESAM report has been revised to include only public available information. This work has been done with the agreement and in collaboration with all the CESAM project partners. The result of this work is presented here.

The present report reviews current SAM approaches in Europe to identify the SAM strategies and actions that are foreseen in reality. This source of modelling requirements covers systems that are part of SAM measures and the actions carried out with reactor systems during SAM. Chapter 2 of this report gives an overview of the IAEA Safety Guide and Safety Reports on Accident Management and present especially the concept of "preventive" and 'mitigative" measures for accident management.

Chapter 3 of this report discusses the transition between "preventive" and "mitigative' domain for accident management to start illustrating some of the differences existing in Europe regarding SAM approaches and the need for ASTEC to be able to reproduce well these differences.

Chapter 4 gives an overview of the main SAM strategies used for the different types of EU NPPs (PWR, VVER and BWR). This chapter is complemented by 15 detailed annexes presenting SAM strategies in the different EU countries which operate NPPs.

Chapter 5 provides a brief summary of the outcomes (mainly the recommendations and suggestions) coming for the EU "Stress Tests" regarding Severe Accident Management.

2 IAEA Safety Guide and Safety Reports on Accident Management

The IAEA has issued during the last decade three main safety documents covering the topic of Accident Management. These three reports are:

[IAEA04] Implementation of Accident Management Programmes in Nuclear Power Plants, IAEA Safety Report Series No. 32, Vienna, 2004.

[IAEA06] Development and review of Plant Specific Emergency Operating Procedures, IAEA Safety Report Series No. 48, Vienna, 2006.

[IAEA09] Severe Accident Management Programmes for Nuclear Power Plants, IAEA Safety Guide, NS-G-2.15, Vienna, 2009.

These reports present in detailed manner what is supposed to be included in an Accident Management Programme (AMP) and how this programme should be developed and implemented. They provide also information about the different Accident Management Strategies in place in several countries.

As starting point, it is important to recall what are the key concepts of accident management (as per [IAEA04]), because they are important in view of ASTEC development for NPP applications.

2.1 Concept of accident management

An accident management programme should be developed for all NPPs, irrespective of the total core damage frequency and fission product release frequency calculated for the plant.

A structured top-down approach should be used to develop the accident management guidance. This approach should begin with the objectives and strategies, and result in procedures and guidelines, and should cover both the preventive and the mitigatory domains. (…)

At the top level, the objectives of accident management are defined as follows:

- Preventing significant core damage;
- Terminating the progress of core damage once it has started;
- Maintaining the integrity of the containment as long as possible;
- Minimizing releases of radioactive material;
- Achieving a long term stable state.

To achieve these objectives, a number of strategies should be developed.

From the strategies, suitable and effective measures for accident management should be derived. Such measures include plant modifications, where these are deemed important for managing beyond design basis accidents and severe accidents, and personnel actions.

Appropriate guidance, in the form of procedures and guidelines, should be developed for the personnel responsible for executing the measures for accident management.

When developing guidance on accident management, consideration should be given to the full design capabilities of the plant, using both safety and non-safety systems, and including the possible use of some systems beyond their originally intended function and anticipated operating conditions, and possibly outside their design basis.

For any change in the plant configuration, or if new results from research on physical phenomena become available, the implications for accident management guidance should be checked and, if necessary, a revision of the accident management guidance should be made.

2.2 Preventive and mitigative domains

The accident management concept is based on both "preventive" and 'mitigative" measures, the link between these types of measures and the associated accident management procedures and guidelines are described in detail in reference [IAEA04], [IAEA06] and [IAEA09].

The relationship between different components of an AMP is illustrated in Figure 1.

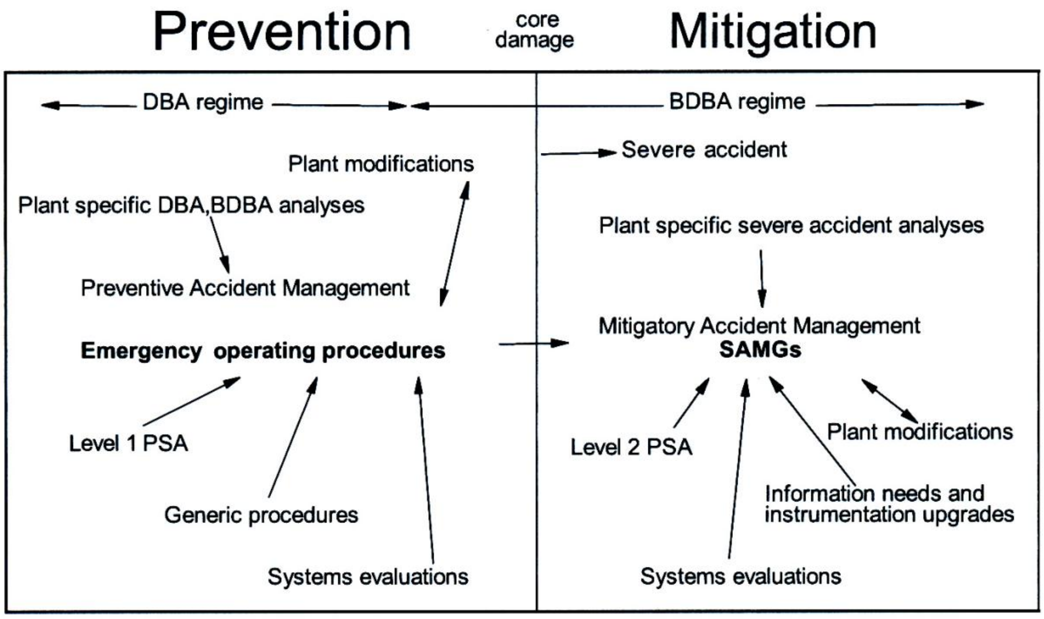

Figure 1. Relationship between different components of an AMP [IAEA04].

In the preventive domain, the guidance should consist of descriptive steps, as the plant status will be known from the available instrumentation and the consequences of actions can be predetermined by appropriate analysis. The guidance for the preventive domain, therefore, takes the form of procedures, usually called emergency operating procedures, and is prescriptive in nature. EOPs cover both design basis accidents and beyond design basis accidents, but are generally limited to actions taken before core damage occurs. EOPs are typically used by the shift.

In the mitigatory domain, uncertainties may exist both in the plant status and in the outcome of actions. Consequently, the guidance for the mitigatory domain should not be prescriptive in nature but rather should propose a range of possible mitigatory actions and should allow for additional evaluation and alternative actions. Such guidance is usually termed severe accident management guidelines (SAMGs) and is typically used by the crisis team respectively the technical support organization (TSO).

It is important to highlight these two domains and to put in perspective the different accident management approaches used in EU countries for Nuclear Power Plants, because some countries have put more emphasis on the preventive domain than others (for example in Germany) in their accident management strategies. Also, for both domains hardware modifications or additions may have been made to the plants' original design, which vary significantly between the different countries.

These differences have an impact on the requested capabilities of SA codes regarding plant applications including SAM measures, the reason being that even though such codes focus on the SA phenomena, the accident scenarios to be calculated will have to include all type of measures (both preventive and mitigative) including sometimes specific components (venturi scrubbers) and features (cooling of the melt inside the RPV or on the cavity floor).

3 Transition from EOP to SAMG

For plant applications, the question of transition from EOP to SAMG (including transfer of responsibilities) is one of the key points of any accident management strategy. A detailed analysis of transition criteria from EOP to SAMG has been presented during an OECD/NEA conference in 2009 [PRI09]. A short summary of this analysis is presented below to illustrate the differences existing in Europe regarding SAM approaches and the need for an SA code to be able to reproduce well these differences.

In practice, the transition between preventive and mitigative domain requires that a symptom be used which identifies the onset or imminent onset of core damage. A 'symptom' in this context refers to a measurable plant parameter. The choice of this symptom and its set point(s) is important due to the change in priority of actions that will result. An indication is needed which:

- is unambiguous,
- is easily used,
- is representative in a known way of the conditions in the core being characterized, and
- provides for a timely transition to SAMG (neither too early nor too late).

While various possible plant parameters have been considered in the past, most PWR and VVER type reactors (though not all) use the core exit coolant temperature. In some cases, CET is used in a combination with primary system pressure, reflecting its strong role in the correlation of CET and clad temperature.

The CET is measured by the core exit thermocouple system; although most use this instrument, there is at first sight a large range of temperature-pressure functions used, depending on the approach.

Figure 2 shows a comparison of EOP-SAMG transition criteria used by a number of widely applied SAMG approaches, especially the ones from the US owners groups (CEOG, B&WOG, WOG), the Finnish approach (Loviisa) and the French approach (GIAG). According to [RAI13], the criterion selected for the French EPR at Flamanville is a constant temperature T = 650°C and not the pressure dependent function (OSSA) in the paper [PRI09] of 2009. In principle, once the core exit temperature shown is exceeded, the transition is made away from EOPs and into SAMG. It is important to note that the comparison is simplified (see detailed explanation in the analysis).

In fact, the choice of the core state, and of the symptom and set point used to detect it, depends very much on the other characteristics of the SAM approach, and in particular the scope of EOP and SAMG actions.

Two factors are most important:

- Whether simultaneous usage of EOP and SAMG is permitted/intended, or whether at transition, EOP use is terminated and SAMG are used alone.
- Scope of coverage of EOPs versus SAMGs.

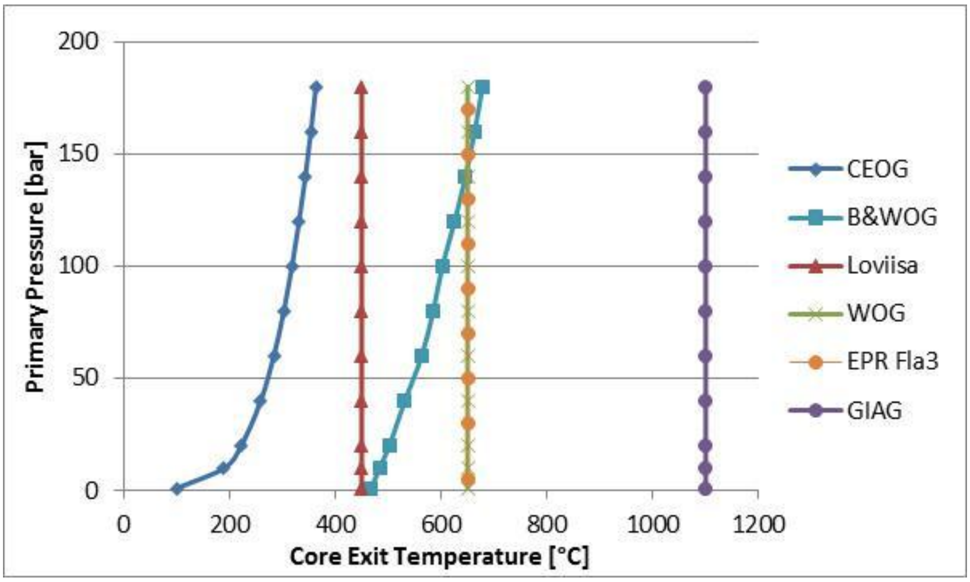

Figure 2. A comparison of SAMG entry criteria (from [PRI09] and [RAI13])

One of the key aspects of selecting the transition is that it is very important not to change too early priorities from core cooling to maintaining containment integrity and to retaining fission products. All possible attempts to re-establish core cooling (within EOPs) should be prioritized as long as possible. Thus the choice of transition is a balance between maximizing time available to restore core cooling, and ensuring that fission products are retained once releases have begun.

Most approaches do not allow the simultaneous application of EOP and SAMG. For these, it is critical to choose a transition criterion which does not cause an unnecessarily early transition. The temperatures used by these approaches are therefore higher, as indicated in the figure.

The other important aspect here is the scope of coverage, particularly of EOPs. Key questions are:

- how far into the severe accident regime do the EOPs go?
- at what point after entry to SAMG would one really do something different (than what was already being attempted in EOPs)?
- at which time is the crisis team (or TSO) alerted and when is it expected to be ready to take over the responsibility from the shift?

Severe accident phenomena such as hydrogen generation, accumulation and combustion, high pressure core melting and creep-induced structural failures are not addressed in EOPs since they just do not occur in the regime of EOP applicability. Even so, the use of an 'ultimate' EOP may allow coverage of some SA phenomena which occur in the early stages of a SA, and therefore allow a later formal and complete transition. This approach is adopted by the French GIAG, and this explains to a large extent why the transition is, apparently, so different from the others in Figure 2.

The target here is not to describe in detail all the aspects which influence the choice for the transition (this is done in detail in the analyses mentioned above), but to illustrate one aspect of the differences existing in Europe regarding SAM approaches and the need for ASTEC to be able to reproduce well these differences.

4 Severe Accident Management main approaches in the European Union

Severe Accident Management in the EU, and worldwide, is mainly based on the work initiated after the core melt accident at the TMI-2 nuclear power plant in 1979 [TMI79]. Key components of this work are:

- International research on physical phenomena during a severe accident.
- The Technical Basis Report of the US Electric Power Research Institute EPRI [EPRI92] that describes (i) plant damage states to support a symptom-based evaluation of the situation at a given moment and (ii) the positive and/or negative effects of counter measures ('Candidate High Level Actions') conceived to stop core melting and mitigate the accident.
- SAMG formulated by vendor and operator companies (typically organized as Owner Groups). It is important to note that a generic SAMG approach always needs to be turned into specific SAMG when implementing it for a given power plant.

This approach has led to a situation where several generic SAMG approaches exist. EU member states vary significantly in

- the time when they started introducing SAM concepts and/or SAMGs to their power plants
- the SAM approach they based their SAMGs on
- the modification to SAMGs and to plant hardware that was applied to take into account national particularities, legal concepts, and interpretation of SA research.

However, it can be seen for the annexes presenting the SAM practices in the EU countries that despite the differences a few main approaches regroup most of the NPPs.

Among these approaches the SAMG approach developed by the Westinghouse Owners Group has been adopted by many EU utilities for PWR or VVER reactors (in Belgium, the UK, Spain, The Netherlands, Switzerland, Czech Republic, Hungary, Slovak Republic and Slovenia). A very detailed presentation of the Westinghouse Owners Group approach for SAM is presented in the Annex for Slovenia. This approach is also used for CANDU reactor in Romania.

For PWR, the second main SAM approach is the one developed in France by EDF. Regarding the international practice, the severe accident guidelines for the French PWRs may appear singular because they give a very high importance on the prevention of early containment failure and conduct to limit the possibility of core cooling when the water injection is prohibited (see Annex for France). This approach was also adopted in Bulgaria following the PHARE BG01.10 project.

Last but not least for PWR, the approach followed in Germany for all PWR reactors which put a large emphasis on the preventive domain with selected mitigative measures including significant hardware modifications for both is also used in Spain. In Germany, no fully developed SAMG approach was used in the past; as one of the actions following Fukushima, it is currently under final implementation. In contrast, the emergency operating manual ("Notfallhandbuch") of the KWU PWR in Switzerland was extended by SAMG in 2005.

Regarding BWR, the SAMG approaches are also based to a certain extent on the one developed by user groups according to the reactor type (for example GE BWROG, in Spain or Switzerland). But some more specific approaches have been developed in countries like Sweden, Switzerland, Finland or Germany. In Germany the principles are the same as for PWR.

While important differences exist it is, however, clear that SAM strategies for one type of reactor, e.g. PWR, will be identical in the goals to be achieved and in the principal means used.

For this reason, user groups according to the reactor type (PWROG, BWROG) are today relevant. The post-Fukushima review, in the US and other countries, of SAM approaches and of the hardware improvements needed or practicable, seems to reduce differences in SAMG approaches further still by aiming at "consolidated generic PWR SAMG that reflect the best practices of the (…) current SAMG" [LUT13].

Finally, it should be pointed out that the development of SAMG and preventive measures for shutdown states and, following the experience of the Fukushima accident, for the spent fuel pool, is quite separate from the development of SAMG for operation states described above. It has had in the past the character of being propagated by isolated utilities or national initiatives, but has recently gained a status that is likely to see shutdown-state and SFP SAM measures implemented across the world.

As a pretext to the following sections, it is noted that commonalities exist between all SAMG approaches in the sense that they are typically based on

- the EPRI Technical Basis Report [EPRI92]
- CHLAs identified in [EPRI92],

while differences exist in terms of

- Organizational issues: responsibility, decision-taking, expert knowledge
- SAM entry criteria, linked to application domain of EOP/SAMG
- Methodology of assessing damage states
- Key CHLA for different concepts
- Different judgement on some results from Severe Accident research (e.g. reflooding of a damaged core)
- Computational aids

Further it should be noted that not all of the discussed strategies may be implemented into each NPP as it strongly depends on the plants capabilities and design features which can vary significantly.

4.1 SAM strategies for PWR

In the SAM strategies for PWR, strategies are grouped in terms of the overall SAM goals that they try to achieve. While the presentation might be different from one approach to another, the logic of SAM goals to be set is applicable to all:

- mitigate or terminate fission product releases by
    - reduction of FP release rates and
    - reduction of FP inventories available for releases
- maintain the integrity of the containment by

  - preventing DCH
  - preventing pressurisation
  - preventing base mat melt-through
  - preventing vacuum
- maintain integrity of reactor coolant system
- preventing RPV failure
- preventing steam generator tube rupture
- preventing hot legs/ pressuriser surge line creep rupture.

During their developments many of these SAM strategies or procedures/guidelines have been developed in parallel to plant upgrade (installation of PAR, FVS, special safety valves, special power backup, etc…).

Table 3 to Table 5 relate the goals defined above to strategies, of which many correspond to CHLAs investigated in [EPRI92]. The strategies are admittedly quite generic, still the tables indicate several important points:

- Many of the strategies have the potential to support more than just one goal, which will give a priority when choosing between different strategies;
- It cannot be deduced from the tables which strategies are compatible with each other and which are not. SAMG have to account for such (in-)compatibilities and define more precisely conditions and timing of strategies. Application of strategies within ASTEC will have to be logic in the same way.

Table 3. Goal "Mitigate or terminate fission product releases"

Goal	Location	Strategy
Reduction of FP release rates	CMT	Injection into containment
	CMT	Operation of ventilation
	SG	Injection into steam generator
	SG	Isolation of steam generator
	SG	Dumping steam to condenser
	AUX	Reduction/isolation of recirculation
	AUX	Isolation of penetration
Reduction of FP inventories available for releases	RCS	Injection into RCS
	CMT	Injection into reactor pit
	CMT	Injection into containment

Table 4. Goal " Maintain the integrity of the containment "

Goal	Location	Strategy
Prevent DCH	RCS	RCS depressurisation
Prevent pressurisation	CMT	Containment venting
	CMT	Containment heat removal
	CMT	Control of hydrogen flammability
Prevent base mat melt-through	CMT	Water injection into reactor pit
Prevent vacuum	CMT	Containment pressurisation

Table 5. Goal " Maintain integrity of reactor coolant system "

Goal	Location	Strategy
Prevent RPV failure	RCS	Water injection into RCS/RPV
	RCS	RCS depressurisation
	SG	Water injection into SG
	CMT	External cooling of RPV
	RCS	Restart RCP
Prevent steam generator tube rupture	SG	Water injection into SG
	RCS	RCS depressurisation
Prevent hot legs/ pressuriser surge line creep rupture	RCS	RCS depressurisation

Combining the information from these tables, the SAM strategies which may be applied to western-style PWR reactors, and the main systems required to implement them, are provided in the table below. The SAM strategies strongly depend on the plant's capabilities and systems existing or foreseen to be used in SAM approaches. Not all strategies may be applicable to all PWR plants.

Table 6. Main systems used to achieve SAM strategies in PWR

Strategy	**Main systems used to achieve the strategy**
Injection into SG	- main feed water pumps, emergency feed water pumps or back-up feed water pumps (source: main feed water line) - emergency feed water pumps (source: demineralized water tank) - water extraction system pumps, raw water system pumps, fire water supply system pumps (source: other tanks) - fire brigade pumps via RS system (source: external water line)
Isolation of steam generator	- steam line isolation valves
Dumping SG steam	- Steam dump to the condenser - Steam dump to the atmosphere
RCS depressurisation	- secondary side relief valves and safety valves - valve opening turbine bypass to condensers - valve opening secondary side steam removal direct to the feed water tank - turbine driven emergency feed water pump (secondary side steam removal) - pressuriser spray system from (i) volume control system, (ii) backup coolant makeup system, or (iii) primary water system - pressurizer letdown system - pressuriser relief valves and safety valves - vent line of (i) reactor vessel head, (ii) pressuriser (to relief tank), (iii) volume control tank
Injection into /cooling RCS	- HHSI/LHSI pumps of the ECCS (sources: ECCS storage tanks, recirculation from containment sump) - Charging Pumps, volume control system pumps, back-up coolant make-up system pumps (source: main coolant storage tanks) - accumulators - LHSI ECCS pumps, residual heat removal system pumps (sources: volume control system tanks, spent

	fuel pool, recirculation from containment sump, ECCS storage tanks) - volume control system pumps, back-up coolant make-up system pumps (sources: tanks mentioned before, boric-acid storage tank) - natural drain from ECCS storage tank - restarting of main coolant pumps - emergency injection system pumps, containment spray system pumps, chemical control system pumps, special means
Injection into reactor pit	- HHSI and LHSI pumps, emergency injection pumps, containment spray pumps, water and boron feed pumps, pumps of chemical and volume control system - comment: if reactor pit cannot be flooded from containment side, the flooding path will be from primary system through the damaged RPV lower head
Injection into containment	- Containment spray - Direct water injection by pumps or gravity drain - Fire-fighting system - sources: internal tanks, external supply NB: influence water chemistry (add soda) to bind FP
Isolation of penetration in auxiliary building	- containment-side valves of the line from containment to auxiliary building(s)
Operation of ventilation in the containment or annulus	- fan coolers, recirculation filters
Containment venting	- valves of the venting line - filter system
Containment heat removal	- containment spray - fan coolers, recirculation filters - biological barrier coolers, other coolers (NL) - annulus coolers

Containment depressurisation	- Containment Filtered Venting System, non-filtered venting is also possible
Containment pressurization (to avoid vacuum)	- (need to stop) systems for containment heat removal - RCS PORV - valves of system injecting non-condensable gas into containment
Control hydrogen flammability	- hydrogen PARs/ignition system - systems linked to measures if PARs fail (see [BOR11])

The appendices of this report provide information on SAM strategies applied in the different countries in Europe.

4.2 SAM strategies for VVER

While VVER are Pressurised Water Reactors, they have significant differences in reactor design and systems compared to Western style PWR (this is especially true for VVER440, VVER-1000 plants have more features in common with Western-style NPP).

For standardised VVER 440-213, other than Loviisa NPP with its cylindrical steel containment, one of the most important aspects regarding Severe Accident Management is the fact that the "containment" of the VVER-440 is a hermetically sealed zone in the reactor building, with limited capacity of withstanding elevated pressures and with a leakage rate of several per cent per day under operational conditions. In the VVER-440-213, overpressure is managed by a system of bubbler tanks (trays filled with water) for condensing steam, and by air traps that capture uncondensed gases and air and separate them from the containment environment. The low specific power of VVER440, in combination with a large water volume for the core and the horizontal SGs also influence the SAM strategies.

For VVER1000, the situation is more similar to Western PWRs.

SAM programmes for VVER have been mostly developed on the basis of the US-driven approach after the TMI-2 accident, with the Finnish NPP at Loviisa being the first to see SAMG applied. In the framework of PHARE and TACIS programmes, many of the VVER owners have followed. In most cases, SAMG have been based on WOG SAMG and associated to NPP upgrade.

Table 7 to Table 9 relate the goals defined, while the Table 10 defines the main systems used to achieve SAM strategies in VVER.

Table 7. Goal "Mitigate or terminate fission product releases"

Goal	Location	Strategy
Reduction of FP release rates	CMT	Injection into containment
	SG	Injection into steam generator
	SG	Isolation of steam generator
Reduction of FP inventories available for releases	RCS	Injection into RCS
	CMT	External cooling of RPV
	CMT	Injection into reactor pit
	CMT	Injection into containment

Table 8. Goal " Maintain the integrity of the containment "

Goal	Location	Strategy
Prevent DCH	RCS	RCS depressurisation
Prevent pressurisation	CMT	Containment venting
	CMT	Containment heat removal
	CMT	Control of hydrogen flammability
Prevent base mat melt-through	CMT	Water injection into reactor pit
Prevent vacuum	CMT	Containment pressurisation

Table 9. Goal " Maintain integrity of reactor coolant system "

Goal	Location	Strategy
Prevent RPV failure	RCS	Water injection into RCS/RPV
	RCS	RCS depressurisation
	SG	Water injection into SG
	CMT	External cooling of RPV
	RCS	Restart RCP
Prevent steam generator tube rupture	SG	Water injection into SG
	RCS	RCS depressurisation
Prevent hot legs/ pressuriser surge line creep rupture	RCS	RCS depressurisation

Combining the information from these tables, the SAM strategies applied to VVER, and the systems required to implement them, are provided in the table below.

Table 10. Main systems used to achieve SAM strategies in VVER

Strategy	Main systems used to achieve the strategy
Injection into SG	- main feed water pumps, emergency feed water pumps or back-up feed water pumps (source: main feed water line) - emergency feed water pumps (source: demineralized water tank) - water extraction system pumps, raw water system pumps, fire water supply system pumps (source: other tanks) - fire brigade pumps via RS system (source: external water line)
Secondary circuit depressurisation	- SGSVs - feed and bleed: feedwater pumps, blow-off valves; steam dump systems
Primary circuit depressurisation	- pressuriser safety valves - pilot operated relief valves PORV - primary circuit gas removal system
Injection into /cooling RCS	- HP/LP pumps of the ECCS (sources: ECCS storage

	tanks, recirculation from containment sump) - volume control system pumps, back-up coolant make-up system pumps (source: main coolant storage tanks) - accumulators - LP ECCS pumps, residual heat removal system pumps (sources: volume control system tanks, spent fuel pool, recirculation from containment sump, ECCS storage tanks) - volume control system pumps, back-up coolant make-up system pumps (sources: tanks mentioned before, boric-acid storage tank) - natural drain from a storage tank (comment: for VVER-440/V213 reactor design HP and LP ECCS tanks are placed below the confinement floor, so gravity driven drain from there is not possible). I - injection from external tank: in the frame of SAM project special external SAM coolant tank+ piping + pump powered from special SAM diesel was installed in the Slovak NPPs (Bohunice V2 and Mochovce1). Injection can be rerouted either to RCS or confinement (via spray nozzles) or into SFP. - restarting of main coolant pumps - emergency injection system pumps, containment spray system pumps, chemical control system pumps, special means
External cooling of RPV by cavity flooding	- spray pumps, LP auxiliary systems pumps (source: ECCS tanks, bubbler trays)
Isolation of penetration in containment	- containment-side valves of the lines from containment
Containment venting	- valves of the venting line - filter system (aerosols, iodine)
Containment heat removal	- containment spray - fan coolers, recirculation filters
Containment	- valves of containment filtered venting system

depressurisation	- bubbler condensation system - incondensable gas/air trap system - containment spray pumps
Control hydrogen flammability	- hydrogen PARs (comment: all VVER-440/V213 NPPs operated in Central Europe are (or will be soon) equipped with large capacity PARs only, i.e. without igniters (AREVA's type in Czech Republic and Slovakia, NIS type in Hungary)) - systems related to RCS depressurization (steam inertisation, only before core damage) - comment: bubbler cavity has low steam concentration due to condensation, thus low inertisation and raised risk of combustion - comment: a strategy with switching of confinement spray (when available) based on the symptom of increase in CET was adopted in Bohunice 2, Mochovce 1,2 and Dukovany NPPs with the aim to recombine bulk of SA hydrogen at elevated or inert steam conditions in confinement atmosphere; adoption of IVR strategy limits the hydrogen source to in-vessel phase and results in steaming of confinement atmosphere

4.3 SAM strategies for BWR

With the above parts of Chapter 3 more related to PWR SAM approaches rather than BWR, it may be useful to quote the following section from Appendix IV of [IAEA09]:

"The US BWROG has grouped its CHLAs into three guidelines which respond to deteriorating conditions in the vessel and the containment. The major guideline is an integrated RPV and containment flooding guideline that defines responses to the core degradation process in its increasing severity until vessel melt-through, while keeping track of the degree of damage to the containment with emphasis on protection of the pressure suppression function. In European approaches the distinction between the different plant damage states is less explicit, but the countermeasures envisaged fulfil the same basic objectives."

The point intended to be made by means of the text is that the SAMG for a BWR is less complex than for a PWR, owing to the simpler design and thus to the smaller number of strategies available. Meanwhile, the SAM goals are the same as those for PWR.

Table 11 to Table 13 below indicate which SAM strategies may be selected in connection with different goals.

Table 11. Goal "Mitigate or terminate fission product releases"

Goal	Location	Strategy
Reduction of FP release rates	CMT	Injection into containment
	CMT	Operation of ventilation
	AUX	Reduction/isolation of recirculation
	AUX	Isolation of penetration
Reduction of FP inventories available for releases	RCS	Injection into RCS/RPV
	CMT	Water injection into lower drywell
	CMT	Injection into containment

Table 12. Goal "Maintain the integrity of the containment "

Goal	Location	Strategy
Prevent DCH	RCS	RCS depressurisation
Prevent pressurisation	CMT	Containment venting
	CMT	Containment heat removal (fan coolers, spray)
	CMT	Control of hydrogen flammability (inertisation)
Prevent base mat melt-through	CMT	Water injection into lower drywell
Prevent vacuum	CMT	Containment pressurisation

Table 13. Goal "Maintain integrity of reactor coolant system "

Goal	Location	Strategy
Prevent RPV failure	RCS	Spray into RPV
	RCS	RCS depressurisation
	RPV	External cooling of RPV (IVRM)

Combining the information from these tables, the SAM strategies which may be applied to BWR reactors are:

Table 14. Main systems used to achieve SAM strategies in BWR

Strategy	**Main systems used to achieve the strategy**
RCS depressurisation	- safety valves - pilot operated relief valves PORV - primary circuit gas removal system
Injection into RCS	- HP/LP pumps of the ECCS - steam driven turbo pump - core spray system - sources: pressure suppression pool, condensate storage tank
Injection into containment	- containment spray system
Flooding of the drywell	- ECCS, condensation pool valves - sources: external (fire water reservoir), condensation pool
Isolation of penetration in containment	- containment-side valves of the lines from containment
Containment venting	- valves of the venting line - filter system (multi Venturi scrubber system)
Containment heat removal	- containment spray - fan coolers, recirculation filters
Containment depressurisation	- valves of containment filtered venting system - containment spray pumps
Control of hydrogen flammability	- nitrogen inertisation system - hydrogen PARs/ignition system

5 Main results and recommendations from the EU 'Stress Tests" in the field of SAM

Following the EU Stress Tests which took place from mid 2011 to mid 2012 in Europe, ENSREG issued a Compilation of Recommendations and Suggestions from the Review of the European Stress Tests (26/07/2012).

The compilation of recommendations addressed to national regulators is made up of the main recommendations found in the conclusion of the stress test report (Chapter 8) as well as the items to be considered that are found in the other parts of the report.

These items are important to be taken into account in the frame of the revision of SA codes because large efforts will be made in the coming years all over Europe to satisfy these recommendations and suggestions. Computer codes like ASTEC will be extensively used by operators and safety authorities/TSOs to demonstrate and validate the success of the SAM strategies proposed to satisfy these recommendations and suggestions.

The first key point concerns the incorporation of the WENRA reference levels related to SAM into the national legal frameworks, and their implementation in the installations as soon as possible. The detailed Reference Levels are defined in [WEN08] and are given in Appendix in Chapter 8. As a reaction to the Fukushima accident, a major update has been proposed and published [WEN13] that is yet to be officially endorsed and issued as the new Reference Levels.

WENRA RL [WEN08] would include:

- Hydrogen mitigation in the containment - Demonstration of the feasibility and implementation of mitigation measures to prevent massive explosions in case of severe accidents.
- Hydrogen monitoring system - Installation of qualified monitoring of the hydrogen concentration in order to avoid dangerous actions when concentrations that allow an explosion exist.
- Reliable depressurization of the reactor coolant system – Hardware provisions with sufficient capacity and reliability to allow reactor coolant system depressurization to prevent high-pressure melt ejection and early containment failure, as well as to allow injection of coolant from low pressure sources.
- Containment overpressure protection - Containment venting via the filters designed for severe accident conditions.
- Molten corium stabilization - Analysis and selection of feasible strategies and implementation of provisions against containment degradation by molten corium.

On top of that, the extension of existing SAMGs to all plant states (full and low-power, shutdown), including accidents initiated in SFPs is also recommended.

Finally, and without being exhaustive, the performance of further studies to improve SAMGs is also recommended on the following topics:

- Accident timing, including core melt, reactor pressure vessel (RPV) failure, basemat melt-through, SFP fuel uncovery, etc.
- PSA analysis, including all plant states and external events for PSA levels 1 and 2.

- Radiological conditions on the site and associated provisions necessary to ensure MCR and ECR habitability as well as the feasibility of AM measures in severe accident conditions, multi-unit accidents, containment venting, etc.
- Core cooling modes prior to RPV failure and of re-criticality issues for partly damaged cores, with un-borated water supply.
- Phenomena associated with cavity flooding and related steam explosion risks.
- Engineered solutions regarding molten corium cooling and prevention of basemat melt-through.
- Severe accident simulators appropriate for NPP staff training.

6 Conclusions

In the frame of the Euratom FP7 CESAM project (2013-2017) one of the tasks consisted in the preparation of a report providing an overview of the Severe Accident Management (SAM) approaches in European Nuclear Power Plants to serve as a basis for further ASTEC improvements. This report draws on the experience in several countries from introducing SAMGs and on substantial information that has become available within the EU "stress test".

To disseminate this information to a broader audience, the initial CESAM report has been revised to include only public available information. This work has been done with the agreement and in collaboration with all the CESAM project partners.

The target of this report was to provide a review as complete as possible (but only based on public available information) of the main SAM strategies used in PWR, VVER and BWR throughout the EU member states and Switzerland.

The vision of this effort is that a harmonized approach to SAM in Europe could be established that takes account of both the variety of countries utilizing nuclear power and of the large range of different NPP types used in these countries.

7 References

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8 APPENDIX: WENRA Reference Levels related to SAM

8.1 WENRA Reference Levels F.4: Design Extension of Existing Reactors

F 4. Protection of the containment against selected beyond design basis accidents

4.1 Isolation of the containment shall be possible in a beyond design basis accident. However, if an event leads to bypass of the containment, consequences shall be mitigated.

4.2 The leak-tightness of the containment shall not degrade significantly for a reasonable time after a severe accident.

4.3 Pressure and temperature in the containment shall be managed in a severe accident.

4.4 Combustible gases shall be managed in a severe accident.

4.5 The containment shall be protected from overpressure in a severe accident .

4.6 High pressure core-melt scenarios shall be prevented.

4.7 Containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable.

8.2 WENRA Reference Levels LM: Emergency Operating Procedures and Severe Accident Management Guidelines

LM 1. Objectives

1.1 A comprehensive set of emergency operating procedures (EOPs) for design basis accidents (DBAs) and beyond design basis accidents (BDBAs), and also guidelines for severe accident management (SAMG) shall be provided.

LM 2. Scope

2.1 EOPs shall be provided to cover Design Basis Accidents. These EOPs shall provide instructions for recovering the plant state to a safe condition.

2.2 EOPs shall be provided to cover Beyond Design Basis Accidents up to, but not including, the onset of core damage. The aim shall be to re-establish or compensate for lost safety functions and to set out actions to prevent core damage.

2.3 SAMGs shall be provided to mitigate the consequences of severe accidents for the cases where the measures provided by EOPs have not been successful in the prevention of core damage.

2.4 EOPs for Design Basis Accidents shall be symptom-based or a combination of symptom based and event based procedures. EOPs for Beyond Design Basis Accidents shall be only symptom based.

LM 3. Format and Content of Procedures and Guidelines

3.1 EOPs shall be developed in a systematic way and shall be supported by realistic and plant specific analysis performed for this purpose. EOPs shall be consistent with other operational procedures, such as alarm response procedures and severe accident management guidelines.

3.2 EOPs shall enable the operator to recognise quickly the accident condition to which it applies. Entry and exit conditions shall be defined in the EOPs to enable operators to select the appropriate EOP, to navigate among EOPs and to proceed from EOPs to SAMGs.

3.3 SAMGs shall be developed in a systematic way using a plant specific approach. SAMGs shall address strategies to cope with scenarios identified by the severe accident analyses.

LM 4. Verification and validation

4.1 EOPs and SAMGs shall be verified and validated in the form in which they will be used in the field, so far as practicable, to ensure that they are administratively and technically correct for the plant and are compatible with the environment in which they will be used.

4.2 The approach used for plant-specific validation and verification shall be documented. The effectiveness of incorporating human factors engineering principles in procedures and guidelines shall be judged when validating them. The validation of EOPs shall be based on representative simulations, using a simulator, where appropriate.

LM 5. Review and updating of EOPs and SAMGs

5.1 EOPs and SAMGs shall be kept updated to ensure that they remain fit for their purpose.

LM 6. Training

6.1 Shift personnel and on-site technical support shall be regularly trained and exercised, using simulators for the EOPs and, where practicable, for the SAMGs.

6.2 The transition from EOPs to SAMGs for management of severe accidents shall be exercised.

6.3 Interventions called for in SAMGs and needed to restore necessary safety functions shall be planned for and regularly exercised.

9 Annex 1: Severe Accident Management in Belgium

The information provided in this annex is taken from the reference [BEL11]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [BEL11]. Therefore the strategies presented here reflect mainly the situation end of 2011.

A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Belgian NPPs to provide guidance for further ASTEC development.

Table 15. Characteristics of Doel/Tihange site units

Table 1: Characteristics of Doel site units

Units	Type	Thermal power (MWth)	Date of first criticality	Containment building characteristics	Steam generator replacement	Fuel storage pool capacity	Designer
Doel 1	PWR (2 loops)	1 312	1974	Double containment (steel and concrete)	2009	664 positions	Westinghouse
Doel 2	PWR (2 loops)	1 312	1975	Double containment (steel and concrete)	2004		Westinghouse
Doel 3	PWR (3 loops)	3 064	1982	Double containment with inner metallic liner	1993	672 positions	Framatome
Doel 4	PWR (3 loops)	3 000	1985	Double containment with inner metallic liner	1997	628 positions	Westinghouse
SCG building	Spent fuel dry storage	-	-	-	-	165 spent fuel containers	Tractebel Engineering

Table 2: Characteristics of Tihange site units

Units	Type	Thermal power (MWth)	Date of first criticality	Containment building characteristics	Steam generator replacement	Fuel storage pool capacity	Designer
Tihange 1	PWR (3 loops)	2 873	1975	Double containment with inner metallic liner	1995	324 positions + 49 removable positions	Framatome / Westinghouse
Tihange 2	PWR (3 loops)	3 054	1982	Double containment with inner metallic liner	2001	700 positions	Framatome
Tihange 3	PWR (3 loops)	2 988	1985	Double containment with inner metallic liner	1998	820 positions	Westinghouse
DE building	Spent fuel wet storage	-	-	Bunkered building	-	3 720 positions + 30 temporary positions	Tractebel Engineering

The Tihange SAMG guides consist of an adaptation for each unit of the generic Westinghouse Owners Group (WOG) guides issued in 1994, which were validated when published.

The Doel SAMG "BKprocedures" are inspired by the philosophy of the WOG SAMG, further completed with specific procedures.

The SAMG at Tihange and Doel include diagnostic guides which call for the use of a limited number of key parameters representing the different strategies and whereby severe accidents can be managed. They are as follows:

- core exit temperature;

- pressure in the primary circuit;
- water level in the steam generators;
- pressure in the containment building;
- water level in the containment building sumps;
- dose rates on the site;
- hydrogen concentration in the containment building (Doel NPP).

Table 16. Strategy for the management of SA and means involved at Tihange NPP.

Strategy	Objectives	Means	Power supplies
Injection of water into the steam generators	Remove the residual heat and prevent the rupture of the reactor vessel	Normal feedwater pumps	AC power supply is not backed up
		Water extraction pumps	
		Raw water pumps	AC power supply is backed up by the "GDS" first level diesel generators
		Safety feedwater pumps (Tihange 1) Auxiliary feedwater pumps (Tihange 2 and 3)	
		Emergency feedwater pumps (Tihange 2 and Tihange 3)	AC power supply is backed up by the "GDU" second level diesel generators
Depressurization of the primary circuit	Maximise the number of water sources able to supply the primary circuit	Steam generators relief valves (bypass) to the condenser	AC power supply is not required
		Steam generators relief valves (bypass) to the atmosphere	
		Auxiliary spray of the pressurizer	
		SEBIM valves (Tihange 1) Relief valves of the pressurizer (Tihange 2 and Tihange 3)	DC power supply is required at Tihange 1 and is backed up by batteries and GDS diesel generators AC power supply is not required for the relief valves at Tihange 2 and Tihange 3
		Bleedoff of the chemical and volume control circuit ("CCV")	Power supply is not required at Tihange 1 Power supply is backed up by the first and second levels at Tihange 2 and Tihange 3
		Emergency spray of the pressurizer at Tihange 2 and Tihange 3 through the emergency injection circuit	AC power supply is backed up by the "GDU" diesel generators
Injection of water into the primary circuit	Remove the energy accumulated in the core	Low pressure safety injection pumps	AC power supply is backed up by the "GDS" diesel generators
	Provide a means of cooling down the core	High pressure safety injection pumps	
	Prevent or delay the breach in the reactor vessel	Containment spray pumps connected to the safety injection system	
		Charging pumps of the "CCV" circuit	

		Boric acid pumps	
		Deaerated demineralized water pumps	
		Emergency injection pumps (Tihange 2 and Tihange 3 units)	AC power supply is backed up by the "GDU" diesel generators
		Primary pumps	AC power supply is not backed up
Injection of water into the containment building	Guarantee sufficient NPSH to enable recirculation	Containment spray pumps	AC power supply is backed up by the "GDS" diesel generators
		Low pressure safety injection pumps connected to the containment sray system	
Injection of water in the reactor pit	Cool down the debris of the core present in the reactor pit	Low pressure safety injection pumps	AC power supply is backed up by the "GDS" diesel generators
	Cool down the debris of the core remained in the reactor vessel	High pressure safety injection pumps	
		Containment spray pumps connected to the safety injection system	
		Boric acid pumps	
		Deaerated demineralized water pumps	
Reduction of the pressure inside the containment building	Keep the containment building integrity	Containment spray pumps	AC power supply is backed up by the "GDS" diesel generators
	Facilitate low pressure injection	Low pressure safety injection pumps connected to the containment spray system	
	Prevent hydrogen explosion	Autocatalytic recombiners	Passive

Table 17. Strategy for the management of SA and means involved at Doel NPP

Strategy	Objectives	Means	Power supplies
Depressurization of the primary circuit	Prevent a high pressure rupture of the reactor vessel Allow low pressure injection	Steam generators relief valves to the condenser	AC power supply is not required
		Steam generators relief valves to the atmosphere	
		Auxiliary spray of the pressurizer	AC power supply is backed up by safety diesel generators
		Emergency spray of the pressurizer	AC power supply is backed up by safety or emergency diesel generators
		SEBIM valves (Doel 1/2)	DC power supply is required and is backed up by batteries and emergency diesel generators
		PORV valves (Doel 3 and Doel 4)	Besides the direct current, compressed air is also required
		MORV valves (Doel 3 and Doel 4)	380 AC power is needed

Injection of water into the primary circuit	Prevent or delay the breach in the reactor vessel	High-pressure safety injection pumps	AC power supply is backed up by safety or emergency diesel generators
		Low-pressure safety injection pumps	AC power supply is backed up by safety or emergency diesel generators
		Emergency make-up water pumps of the primary circuit (Doel 3 and Doel 4)	AC power supply is backed up by safety or emergency diesel generators
	Mitigate the consequences of the corium- concrete interactions in the reactor pit	Containment spray pumps (Doel 3 and Doel 4)	AC power supply is backed up by safety or emergency diesel generators
		Charging pumps	AC power supply is backed up by safety or emergency diesel generators
Injection of water in the reactor pit	Mitigating the consequences of the corium- concrete interactions in the reactor pit	Gravity filling from the RWST	Passive
Reduction of the pressure inside the containment building	Keep the containment building integrity	Containment spray pumps	AC power supply is backed up by the safety or emergency diesel generators
		Low pressure safety injection pumps	AC power supply is backed up by the safety or emergency diesel generators
		Ventilation cooling batteries of the reactor building	AC power supply is backed up by the safety or emergency diesel generators
		Venting of the reactor building (if no other possible solutions left)	
	Prevent hydrogen explosion	Autocatalytic recombiners	Passive

List of acronyms: Safety diesel group (GDS), Emergency diesel group (GDU)

9.1 Accident management measures to restrict the radioactive releases

9.1.1 Mitigation of Fission Products releases

This strategy comes into effect when the amount of radioactive materials inside the containment building reaches a level at which atmospheric releases can reasonably be expected and require protective measures for the population outside the site. The mitigative measures identified depend on the origin of the release:

- For the releases from the containment building, the measures identified are:
    - the spraying inside the reactor building
    - the ventilation/filtration of the annulus space,
    - and the addition of soda to the water in the sumps of the containment building (to trap iodine);
- For the releases from the steam generators, the measures identified are:
    - the filling of the steam generator with water,
    - the isolation and the release of the steam from the affected steam generator to the condenser rather than to the atmosphere,
    - and other alternative measures (such as spraying the released steam with fire hoses);
- For the releases from the nuclear auxiliary building, the measures identified are:
    - the operation of ventilation/filtration systems,

- in the event of containment penetration failure, the isolation of the containment penetration causing the release,
- and in the event of a leak in one of the recirculation loops, the reduction of the flowrate in the loop in question, the isolation of the loop, and the use of other recirculation loops.

9.1.2 Injection of water into the SGs to trap the Fission Products leaking from damaged SG tubes

This strategy comes into effect when there is an inadequate level of water in the steam generators.
The measures identified are, when the pressure in the steam generators is high: the auxiliary feedwater pumps, the emergency feedwater pumps and the normal feedwater pumps.
When the steam generators are at low pressure, the pumps of the water extraction system and those of the raw water system are used.

9.1.3 Injection of water into the Primary System to trap Fission Products released from the core debris

This strategy comes into effect when the core exit temperature indicates an uncovered core.
The measures identified are

- the pumps of the high pressure and low pressure safety injection systems,
- the pumps of the containment spray system,
- the pumps of the emergency injection system
- the pumps of the chemical and volume control system,
- the water and boron makeup pumps,
- the primary pumps and the special means in shutdown state.

9.1.4 Injection of water into the containment building

The aim of this strategy is to guarantee a sufficient NPSH to allow recirculation phase and to ensure, as a preventive measure, that there is a sufficient quantity of water in the containment building to trap the fission products resulting from the debris located outside the reactor pit. It comes into effect when the water level in the containment sumps is too low.
The measures identified are the containment spray pumps, the low pressure safety injection pumps and the addition of soda to the water in the containment building sumps.

9.1.5 Monitoring containment building conditions

The aim of this strategy is to reduce the concentration of fission products and to trap the fission products released from the containment building. It comes into effect when the containment building pressure no longer corresponds to a stable and controlled state in the long term.
The measures identified are the containment spray pumps, the low pressure safety injection pumps and the addition of soda to the water in the containment building sumps.

9.1.6 Injection of water in the reactor pit

The aim of this strategy is to trap the fission products released by the debris located in the reactor pit and in the reactor vessel. This strategy comes into effect when the core exit temperature indicates that the core is still deteriorating.

The measures identified are the high pressure and low pressure safety injection pumps, the emergency injection pumps, the containment spray pumps, the water and boron feed pumps and the pumps of the chemical and volume control system.

9.2 References

[BEL11] "Belgian stress tests National report for nuclear power plants" FANC-Bel V, 2011-12-23.

10 Annex 2: Severe Accident Management in Bulgaria

The information provided in this annex is taken from the reference [KOZ11], [KOS07] and [QBG13]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [KOZ11]. Therefore the strategies presented here reflect mainly the situation end of 2011.
A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Bulgarian NPP at Kozloduy to provide guidance for further ASTEC development.
Units 1 and 2 are in decommissioning and are not considered here. Units 3 and 4 are finally shut down, but their cores are still stored in their fuel ponds. Units 5 and 6 are in operation. SAMG exist only for units 5 and 6; for this reason, this report is limited to these units.

Table 18. Characteristics of operating units at Kozloduy NPP

	Unit 5	Unit 6
Type	VVER-1000, V320	
Thermal power	3000 MW	3000 MW
Commissioned	1987	1991
Containment building characteristics	Pre-stressed reinforced concrete cylinder with spherical dome, 8mm-thick inner steel liner [2]	
Steam generator replacement	replacement of SGSVs in 2002	
Fuel storage pool capacity	612 FA	612 FA

For Kozloduy units 5 and 6, SBEOP were developed on the basis of protection of fundamental safety functions with an approach similar to the Westinghouse one, as step-by-step procedures with description of main and alternative operator actions in two-column format. SBEOP are controlled and maintained adequate to existing power capacity through strict on-site rules on verification and validation which envision multiple checks before they are introduced into operation. Analytic validation is conducted on an original technique provided by Department of Energy (DOE USA).
For Units 5 and 6, three sets of SBEOP are envisioned:

- SBEOP for power operations;
- SBEOP for shut down reactor with pressurized primary circuit;
- SBEOP for shut down reactor with depressurized primary circuit.

The SBEOP structure contains:

- procedures on diagnostics of condition;
- procedures on optimal recovery;
- procedures on recovery of CSFs (critical safety functions);

- procedures of ADP type (Accidents with destruction of protection).

SAMG have been developed which follow the format of SBEOP. Certain criteria are defined for the transition from SBEOP to SAMG. SAMG will undergo internal procedures of verification and validation in NPP, followed by training of operators to work with them. They were scheduled to be implemented at the end of 2012.

SAMGs for Units 5 and 6 were developed on the basis of system analysis of the processes and phenomena during severe accidents (PHARE Project BG 01.10.01, Phenomena investigation and development of SAMG). The purpose of system analysis is to define the basis for knowledge of processes and phenomena, occurring at progress of severe accidents, and all the technical means (equipment and systems), which allow for reaching of the set purposes in the course of SAM.

SAM strategies identified are the following:

- Pressure reduction in the primary circuit;
- Pressure reduction in the secondary circuit;
- Water injection to the primary circuit;
- Water injection to the secondary circuit;
- Pressure reduction in the containment.

Units 5 and 6 have informational systems to support the operators in emergency conditions, such as the SPDS – for control of the CSFs – and PAMS.

No SAMGs exist for severe accident during shutdown states of the units, or for events in the spent fuel pools.

10.1 Maintaining the integrity of the containment after significant fuel damage

10.1.1 Prevention of fuel damage/vessel failure at high pressure

A strategy for pressure reduction in the reactor vessel was developed within the SAMGs. The main technical means for pressure reduction at accident progress before severe phase is the safety valves of the primary circuit and the system for emergency gas removal from the primary circuit.

The system includes 3 independent units, installed in parallel to each other on a line connected to the pressurizer top. Usually the system is mentioned as Pressurizer safety valves (PRZSV). The type of valves is “SEMPEL”, with a nominal capacity of 50kg/s. Each of the operational unit consists of one control valve and two operative valves (YP21,22,23S01).

The set points for opening the valves are:

- YP21S01 – 189 kgf/cm2
- YP22S01 – 193 kgf/cm2
- YP23S01 - 197 kgf/cm2

The set points for closing the valves are:

- YP21S01 – 179 kgf/cm2
- YP22S01 – 183 kgf/cm2
- YP23S01 - 187 kgf/cm2

As an additional possibility to reduce pressure in the reactor vessel, valves can be used on the sealing water drainage lines of the main circulation pumps. For ensuring the availability of the system for emergency gas removal from the primary circuit in conditions of an evolving severe accident, modifications were performed on the power supply to system valves, and redundancy provided for power supply of the correspondent valves from the accumulator batteries.

10.1.2 Corium cooling during in-vessel phase and after vessel failure

The feasibility of external-to-vessel cooling of the reactor was analysed, and it was found that in some cases of severe accident progress damage to the vessel cannot be avoided.
The SAMG envision actions for restoration of coolant supply to the reactor. If the operators manage to deliver sufficient coolant to the core during the in-vessel phase of a SA, there is a chance to cool down fragments and melt from the reactor core and to localize this in the reactor vessel.
As a means to supply coolant to the primary circuit, available trains of the safety systems (SS) can be used – Emergency core cooling system - high pressure (ECCS–HP), low pressure (ECCS-LP), make-up and blowdown system trains (system TK).
Details regarding the different systems available to supply coolant to the RCS are provided below.
Hydroaccumulators system: This is a passive emergency core cooling system that discharges boron solution to the primary circuit when the primary pressure decreases below 5.9 MPa. During LOCA the system should discharge to primary circuit borated water with concentration of boric acid 12(16)g/kg and temperature > 200C. There are 4 hydroaccumulators with total volume of 60 m3.
High pressure injection system: It consists of three pumps, TQ13D01, TQ23D01, TQ33D01 with a capacity 160 m3/h, and three independent tanks (TQ13B01,TQ23B01,TQ33B01) with capacity 15 m3 each, boron concentration 40g/kg and temperature >200C. The pumps start to inject if primary pressure falls below 11.0MPa.
Low pressure injection system: It consists of three pumps TQ12,22,32D01 with capacity 250-300 m3/h each if primary pressure is 2.16MPa and 700-750 m3/h if primary pressure is 0.098MPa. The boron solution tank contains 560 m3 water with boron concentration 12g/kg.

10.1.3 Prevention of high pressure in the containment

Technical means for preventing inadmissibly high pressure peaks in the containment are a hydrogen reduction system of PARs. The analyses show that their number is sufficient to avoid explosive concentrations during the in-vessel phase of a SA. SAMG and EOP procedures envision actions for monitoring, assessment and predicting of hydrogen concentration within of containment premises. Actions are described for control of spray systems depending upon concentration of hydrogen.
Independent of the hydrogen issue, the containment spray is automatically actuated when containment pressure exceeds given limits and will reduce pressure by cooling and condensation.
The units feature also pool-scrubbing filtered venting systems that are actuated by a (passive) rupture membrane.
Details regarding containment spray system are provided below:
Containment spraying system functions are:

- Spray borated water in the hermetic volume to avoid over pressurization.
- Absorb radioactive Iodine in the steam air mixture by adding a special solution to the suction side of the pumps.
- Cool the spent fuel pool in emergency cases.

The containment spraying system consists of three independent trains, each of them including:

- Centrifugal pump TQ11(21,31)D01 with capacity 700 m3/h
- Ejector pump for injection of special solution to the suction side of the pump TQ11(21,31)D02 with capacity 50 m3/h

- Special solution tank TQ11(21,31)B01 with boron concentration of water 150-160 g/kg and volume 6.0 m3

The set points for spray actuation are 10K subcooling margin and pressure in the containment higher than 0.1275MPa (1.3 kgf/cm2).

10.1.4 Management of H2 inside the containment

Units 5 and 6 are provided with Hydrogen reduction system in the containment which consists of 8 PARs, located within the containment structure. The system is designed with a capacity to handle hydrogen generated at design-basis accident with maximum leak of the primary circuit.

10.2 Mitigation of releases of radioactive substances

The localizing systems have been executed in the design of the unit which ensure meeting the set criteria for localization of releases of the radioactive substances to the environment.
To execute the localizing functions, systems and means for controlling containment environmental parameters are installed in the containment, for isolating the containment and for reduction of concentration of radioactive substances of fission, hydrogen, other substances which may release to the containment environment during and after design and severe accidents. For performing of these safety functions the following systems have been installed:

- Containment system (CS);
- Spray system;
- Filter venting system;
- Filtering pressure reduction system;
- Hydrogen recombination system.

Kozloduy SAMG take credit of the containment spray system for reducing fission products concentrations in the containment atmosphere. Also, the pool scrubbing of the containment venting system will reduce FP in releases.
A modernization program has introduced the following measures related to containment:

- Improvement of containment testing procedure;
- Qualification of cable penetrations and planning of their replacement;
- Installation of filtered ventilation;
- Development and introduction of severe accident radiation monitoring systems.

10.3 Fuel uncoverage in Spent Fuel Pond

The SFPs of Units 5 and 6 are located within the containment. Each SFP consists of 4 compartments physically separated by partition walls up to elevation 28.93m. Above elevation 28.93m up to 36.2m the pond volume is common. Three compartments are allocated for immediate storage of the spent assemblies, while the fourth compartment is used for transport and handling operations with fresh and spent fuel. Each SFP has a load capacity of 612 fuel assemblies and shall assure storage of the spent fuel assemblies not shorter than three years.
The spent fuel pool and the entire system are filled with boric acid solution, with a concentration of 16 g/kg. The fuel storage compartments do not have service drains at the bottom part which guarantees they cannot be drained empty and leave the SNF without coolant.
Hydrogen generated at possible damage will spread within the containment and will be recombined by the PAR.
Likewise, radioactive substances released will be maintained within the containment while it is intact. The emergency instructions treating the IEs (initiating events) in the SFP foresee actions regarding evacuation of the personnel in the containment and isolation of the containment.

10.4 References

[KOZ11] "SUMMARY REPORT for Conduct of Kozloduy NPP Stress Tests", “Kozloduy NPP” PLC, October 2011.

[KOS07] M. Kostov, T. Todorova, A. Andonov, Ultimate Capacity Assessment of VVER 1000 Containment Structure, SMiRT 19, Toronto, 2007.

[QBG13] Replies to CESAM WP40 questionnaire by INRNE, 2013.

11 Annex 3: Severe Accident Management in the Czech Republic

The information provided in this annex is taken from the reference [CZE11a]. Much more information regarding systems to cope with severe accident can be found especially in this reference document [CZE11a]. Therefore the strategies presented here reflect mainly the situation end of 2011.

A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Czech Republic NPPs to provide guidance for further ASTEC development.

Table 19. Nuclear Power Plant units in the Czech Republic.

NPP	First connection to the grid	Power [MWe]	Reactor Type
Dukovany Unit 1	1985	427	VVER440/213
Dukovany Unit 2	1986	427	VVER440/213
Dukovany Unit 3	1986	427	VVER440/213
Dukovany Unit 4	1987	427	VVER440/213
Temelin Unit 1	2002	1000	VVER1000/320
Temelin Unit 2	2003	1000	VVER1000/320

SAMG was first implemented in the Czech Republic during the activation of the Temelín NPP, using the experience of Westinghouse. Most of the requirements made for SAMG are based on [IAEA09].

11.1 Severe accident management measures for Dukovany NPP (EDU) and Temelín NPP (ETE)

The concept of managing technological accidents at EDU and ETE is based on a symptomatic approach. At present, the following strategies are prepared for EDU and ETE for the solution to the above-project and severe accidents.

- Symptom-based emergency operating procedures (EOPs) for power modes.
- Symptom-based emergency operating procedures for shutdown modes, including cases of threats to the heat removal from the spent fuel stored in SFP (SBEOPs).
- Manuals for decision making by Technical Support Centre (TSC).
- Severe Accident Management Guidelines (SAMG).

All the mentioned procedures and manuals were developed and are updated in cooperation with Westinghouse plc.

The procedure to treat an emergency situation beyond the scope of EOPs is in accordance with the internal emergency plan (announcement of the level of an extraordinary event).

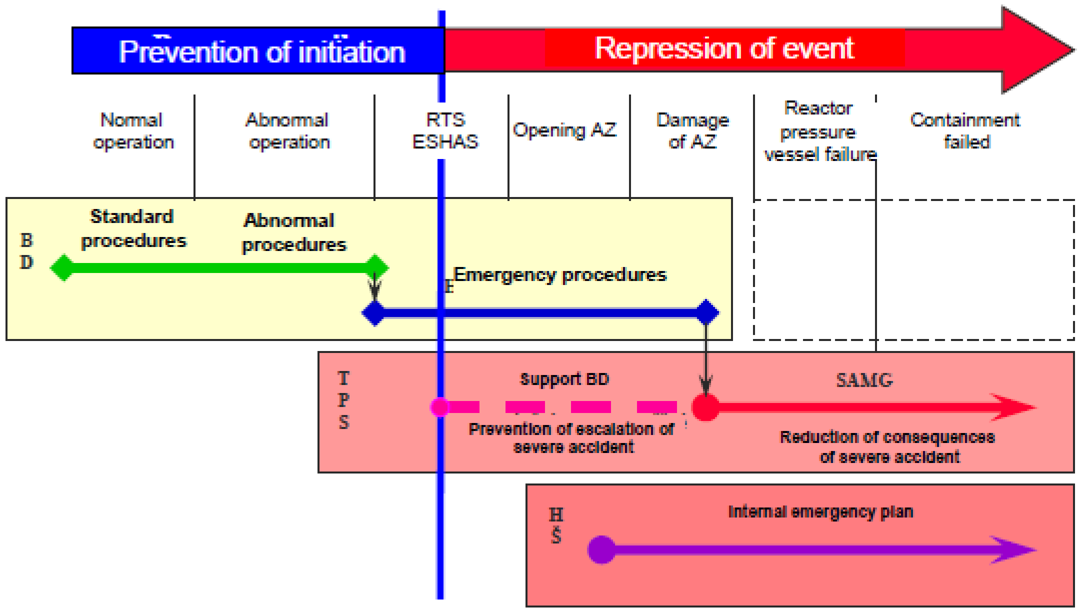

Figure 3. Regions of application of EOPs and SAMGs , from [CZE11b]

For emergency conditions in the preventive phase, strategies are prepared that are included in the EOPs. For managing severe accidents, strategies are prepared which are included in the SAMGs. The basic condition for executing the activity according to emergency procedures is that the status of the RCS enables cooling, i.e. the RCS is a geometric configuration that can be cooled. In the case of irrevocable damage, emergency procedures need not provide the optimal manual for the solution of the emergency situation and it is necessary to proceed according to SAMG. At this time, the main priorities are changed. In EOPs the main priority is the restoration of the heat removal from RCS and the prevention of damage to the 1st barrier against the release of fission products (covering the fuel), whereas in SAMG the main priority is the prevention of damage to the 3rd barrier against the release of fission products (containment), which at this time is the last integrated barrier.
The objective of interventions described within EOPs, which the MCR operating personnel will use for the solution of the project and above-project emergency events, is to ensure sufficient cooling of Reactor Core and prevent irrevocable damage to Reactor Core and also minimize the consequences of any release of radioactive substances outside the NPP.
In the case of the development of an event in the area of the severe accident, a further procedure is selected to at least ensure the remaining barriers against the release of radioactivity. The loss of integrity and geometry of the fuel in Reactor Core means a serious threat to the ability to remove heat from Reactor Core. Under these conditions it is not possible to proceed according to EOPs. SAMG for achieving stabilized status are prepared for this phase of the accident.
The transfer to SAMG takes place in the case that irrevocable damage to Reactor Core is ascertained. In this case activities according to EOPs are terminated and the transfer to SAMG takes place. The only entry point into SAMG is the manual SACRG-1, ACTIVITY MCR WITHOUT TSC.
There are three possible transfers from EOPs into SAMG:

- FR-C.1 Loss of cooling Reactor Core
- FR-S.1 No shutdown of reactor (ATWS)
- ECA-0.0 Loss of electricity supply – Blackout

These three transfers for emergency statuses into SAMG are sufficient and cover all possible severe accident scenarios. To decrease the consequences of severe accidents, the following objectives must be fulfilled:

- Primary objectives of SAMG:
    - Restoration of the heat removal from Reactor Core or restoring the source of the development of heat to a stable status that can be controlled.
    - Keeping the integrity of the containment as the last barrier against the release of Radioactive substances into the surroundings, ensuring the status of containment that can be controlled
    - Terminate the release of Radioactive substances into the surroundings
- Secondary objectives of SAMG
    - Minimizing the release of Radioactive substances while fulfilling the primary objectives
    - Ensuring the maximum operating capability of equipment during the fulfilment of the primary objectives

11.2 Dukovany NPP: SAM measures at the various stages of the scenario for loss of the core cooling function[CZE11b]

11.2.1 Prior to fuel damage in the RPV

The basic reason for severe accidents is the insufficient collection of residual heat released from the field in the Reactor Core. Damage of the Reactor Core is considered when the temperature of the fuel cladding locally exceeds 1200 °C, when a steam-zircon reaction develops. Due to the impossibility to measure this parameter, the setpoint for transfer into SAMG was set for the value of the coolant exit temperature from the Reactor Core 550 °C. Exceeding of 1200 °C in the wider area leads to an intensive steam-zircon reaction which is exothermic. A greater volume of heat is released than the residual heat; this heat contributes to the development of the accident because it is mostly accumulated inside Reactor Core.

The restoration of the heat removal from Reactor Core on the part of Secondary Circuit by alternative means is performed in EOPs i.e. before the transfer to SAMG. In addition, activities are performed related to the de-pressurizing of Primary Circuit with the aim to enable injection of low-pressure pumps into Primary Circuit.

There are two ways of stopping the development of the loss (cooling of Reactor Core up to a severe accident):

- The restoration of the heat removal through SG (alternative filling of SG with lowpressure sources, including adding water by means of Fire brigade).
- Heat removal by adding coolant into Primary Circuit and discharging from the exit hole in the primary system (at LOCA) or by open PZR valves (feed&bleed).

EOPs also include alternative strategies:

- De-pressurizing of the primary system or cooling on the part of Secondary Circuit, which may lead to the enforcement of a hydraulic accumulator or, even, low-pressure emergency or alternative sources.
- The restoration of the operational ability of high-pressure systems for emergency filling or alternative High Pressure systems for emergency filling Primary Circuit.
- Use of the remaining coolant in loops enforced by the start of RCP even at the cost of its destruction.

11.2.2 After the occurrence of fuel damage in the RPV

The basic strategy that is used in SAMG in this phase of the accident is to decrease the pressure in the primary system either due to the decrease in the production of hydrogen, as well as the prevention of creep breaking of the bottom and high pressure expulsion of the melt from the vessel. In accordance with generic manuals, the value of pressure Primary Circuit is required to be under the value 2 MPa, in particular the value 1 MPa is stated for EDU units.
The risk of the failure of the vessel would be significantly decreased by the implementation of the strategy for cooling the vessel from the outside by flooding the reactor shaft.
Construction VVER-440/213 is recommended in terms of holding the melt inside the vessel of the reactor by cooling from the outside even if the original project did not consider this measure. In particular, the residual output of the reactor is very low which ensures low thermal flows on the outside surface of the vessel in the area of bubbler boiling with a large reserve to the boiling crisis. The vessel does not have any penetrations in the lower part. The reactor shaft is the lowest place in the containment and in the case of the loss of water for emergency cooling it is sufficient for discharging of the bubbler channel for its flooding.
The success of this strategy was analytically confirmed. Within the implementation of the technical solution for the modification of piping lines for the air ventilation system into the reactor shaft, glow holes were prepared from the floor of the SG box enabling the termination of the flooding of the room in the reactor shaft.

11.2.3 After failure of the reactor pressure vessel

If the accident cannot be stopped inside the RPV, there would be the failure of its lower part and the interaction of the melted fuel with the concrete.
The penetration of the melt through the wall of the shaft is more serious than penetration through the bottom of the shaft because:
The penetration of the melt in a radial (horizontal) direction is faster than penetration in the axial (vertical) direction. The wall is 2.5 m thinner than the bottom 3.1 m. The wall of the shaft represents the border of the containment; debris can penetrate through the bottom into the foundation slab (bed) where fissile products are kept.
For reactor VVER-440 the door in the shaft would be protected for a certain period against contact with liquid debris by solid debris or the shell. However, the crust has low thermal conductance. Due to the forming of the melt and the low temperature of the steel melt, it is most probable that there would be melting of the lower part of the door and flowing of part of the melt through a lane to the second door which would fail after some time. In each case it is not possible to exclude minor damage of the containment shortly after the failure of the bottom of the RPV due to the failure of the rubber sealing on the door.
The strategy for cooling the melt is part of the SAM guideline for flooding the shaft. The existing configuration of the power plant provides the possibility to flood the shaft by pouring; for this, it is necessary to have water from two LP ECCS tanks and from B/C tower trays filled with water. Therefore, the manual considers discharging of the B/C tower trays, including the inspection of the closing of the air conditioning system (B/C tower air-traps). For pumping of water from the LP ECCS tanks, it would be possible to use shower sprinkler system pumps, alternatively Turbo Generator and TM system pumps. The strategy also considers the use of the water reserve from the surrounding units.
The main benefit of the strategy of flooding the debris in the reactor shaft is the cooling of the steel door located on the side wall of the reactor shaft and confinement of the fission products released during the interaction of the melt with the concrete.

11.2.4 Management of Hydrogen risks inside the containment

EDU has only a limited number of hydrogen recombiners. The current analyses and experience from other VVER confirmed that such a system consisting of performance re-combiners (approx. 30) completed with burners in the case of functioning showers, can restrict the risk of the flame spreading and exclude the risk of the transfer to detonation.
For inertization of the containment it is possible to discharge the nitrogen from the hydro-accumulators; for effective inertization in the existing status of the project, it is possible to use water steam which postpones the risk of burning at a higher concentration of the hydrogen. However, with the high probability there will be the burning of the hydrogen by the existing re-combiners as long as its concentration does not exceed 10% at the point of installation. The existing re-combiners do not resolve the risk of the hydrogen during a severe accident because they can remove only several kg of the hydrogen in the early phase of the accident.
In the case of de-pressurizing the RCS before the damage of the reactor core (which is performed within EOPs) and during the continuation of this procedure after the damage of the reactor core, the risk of detonation occurs later and is localized only in the bubbler shaft.

11.2.5 Prevention of containment overpressure

The system for the suppression of pressure in the containment consists of two parts:

- Vacuum-bubbler system containing passively functioning bubbler channels (trays with water) which condensate water steam and consequently ensure passive spraying of the containment. Non-condensed gases and air from the containment area are located in the catchment gas tanks (B/C tower air-traps) which are consequently automatically separated from the environment of the containment.
- Shower system with three active shower pumps.

The vacuum-bubbler system condenses the steam and creates at the start of the accident the conditions for under-pressure in the containment at the costs of certain pressurizing of its parts - gas tanks (air-traps).
The total volume of the containment, including the gas tanks, compared with the residual output, is relatively high, approximately 50,000 m3.
The relatively high operating leakage of the containment of a few percent of weight of gas/day at the projected pressure supports the decrease of the pressure.

11.3 Temelín NPP: SAM measures at the various stages of the scenario for loss of the core cooling function [CZE11c]

11.3.1 Prior to fuel damage in the RPV

The strategies for dealing with accidents involving loss of cooling in the core are described in the EOP (phase before the fuel in the core is seriously damaged) and in the SAMG (phase after the fuel in the core has been seriously damaged). A thorough symptomatic approach of the SAMG is expedient for achieving the primary objective – protecting the containment from damage.
Cooling of the core in the phase before the fuel is seriously damaged can be restored by carrying out the activities described in the EOP. The following strategies are defined for restoring cooling in the reactor core:

- Restoring high-pressure makeup for the Primary System (high-pressure emergency supply, emergency boroning, normal makeup) to restore the core cooling function.
- Depressurizing of the SG;

- Depressurizing of the Primary System. Although this method is efficient, the disadvantage is a further decrease in the already low amount of coolant in the Primary System if the purpose of depressurizing (to activate low-pressure systems) is not achieved. In any case, depressurizing allows bringing cold coolant from the hydro-accumulators;
- Using the residual amount of coolant in the Primary System – equalizing levels between the core and the lower mixing chamber by connecting the upper parts of the Primary System and cold branches of circulation loops, attempting to activate the main circulation pump (MCP) for possible cooling of the core by bringing water from the hydro valve in the part of the cold branch near the suction opening of the MCP. Activation of the MCP will allow the overheated steam to circulate.

The reduction of pressure in the Primary System is one of the highest priorities when coping with severe accidents. There are several methods of depressurizing the Primary System:

- Using the system for emergency bleeding of the Primary System;
- Pressuriser relieve valve;
- Normal injection into the pressuriser;
- Depressurizing the SG.

11.3.2 After the occurrence of fuel damage in the RPV

All strategies are based on the principle of cooling the damaged fuel from the inside of the RPV, i.e. by the addition of water into the primary circuit. With respect to the thermal output of the reactor and due to the design of the concrete reactor shaft, the studies conducted so far have highlighted the difficulties to cool the RPV in the VVER 1000 units with V320 reactors from the outside. However new studies are still ongoing on this topic.

Cooling of the core in the phase after the fuel has been severely damaged can be restored by the methods described in SAMG. The following strategies were defined for restoring cooling in the reactor core:

- Supplying water into the hot, dry reactor core, which always has a positive impact on the development of an accident;
- Another method that can be used after the fuel has been severely damaged is depressurizing of the primary system.

As a preventive measure in case of a severe accident, water is supplied into the containment. The corresponding strategy in the SAMG provides instructions for flooding the containment with water up to the maximum possible level, which serves two purposes: It protects the concrete at the bottom of the containment in case that debris of the reactor core releases from the RPV into the containment, and it effectively washes away fission products escaping from the melt.

11.3.3 After failure of the RPV

All strategies for cooling the melt that has fallen to the bottom of the containment are based on the principle of pouring water on the melt from above. Flooding the debris of the reactor core outside the RPV will transfer heat from this debris and thus reduce the speed of interaction with the concrete.

11.3.4 Management of Hydrogen risks inside the containment

The containments of units in the Temelín NPP are equipped with a post-accident hydrogen liquidation system, designed for design basis accidents. This system contains passive autocatalytic recombiners and it is able to dispose of hydrogen released during accidents for a long period of time.

The last resort preventing damage to the integrity of the containment described in the strategies implemented in the SAMG is controlled venting of the containment using systems that were not, by design, intended for venting. The SAMG strategies also deal with the possible negative consequences of a hydrogen leak outside the containment.

11.3.5 Prevention of containment overpressure

Measures designed for coping with accidents threatening the integrity of the containment by high pressure are described in the SAMG strategies, which employ all available means to reduce pressure in the containment. The corresponding SAMG strategies provide instructions for preventive measures leading to reduced pressure in the containment in case its integrity is threatened by overpressure:

- Sprinkler systems in the containment (standard sprinkler system in the containment or fire water);
- Injecting containment using fire pumps is an alternative method (assuming there is a power supply). The locations of the fire pumps and their tanks ensure sufficient diversification when compared with containment sprinkler systems.
- Ventilation and venting units in the containment (with coolers);
- Various (non-design) routes for filtered or unfiltered venting.

11.4 References

[CZE11a] "National Report on „Stress Tests“ NPP Dukovany and NPP Temelín Czech Republic, Evaluation of Safety and Safety Margins in the light of the accident of the NPP Fukushima", State Office for Nuclear Safety, Czech Republic, December 2011.

[CZE11b] Dukovany licensee stress test review report, linked on the ENSREG stress test site. http://www.cez.cz/edee/content/file/pro-media-2012/02-unor/final-report-st-edu.pdf, accessed on 28.7.2014.

[CZE11c] Stress tests of nuclear power plants – ČEZ, a.s. Evaluation of Nuclear Safety and Safety Margins of Temelín NPP, linked on the ENSREG stress test site. http://www.cez.cz/edee/content/file/pro-media-2012/02-unor/final-report-st-ete.pdf, accessed on 28.7.2014.

12 Annex 4: Severe Accident Management in Finland

The information provided in this annex is taken from the reference [FIN11] and [QFI13]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [FIN11]. Therefore the strategies presented here reflect mainly the situation end of 2011.
A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of on-going R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Finish NPPs to provide guidance for further ASTEC development.
Currently there are two nuclear power plant (NPP) sites in Finland, one on Hästholmen island in Loviisa and the other one on Olkiluoto island in Eurajoki. Loviisa NPP is owned and operated by Fortum Power and Heat Oy (Fortum) and the Olkiluoto NPP by Teollisuuden voima Oyj (TVO).
Loviisa site has two VVER-440 reactors of Soviet design equipped with specific safety features including leak tight steel shell containment to meet Western nuclear safety requirements.
In Eurajoki, there are two BWRs of AB Asea Atom design. The first unit Olkiluoto 1 (OL1) achieved first criticality in July 1978 and started commercial operation in October 1979, and unit 2 (OL2) in October 1979 and in July 1982, respectively. There is also on EPR under construction on the site (OL3).

12.1 Severe Accident Management in Loviisa 1&2 (VVER-440)

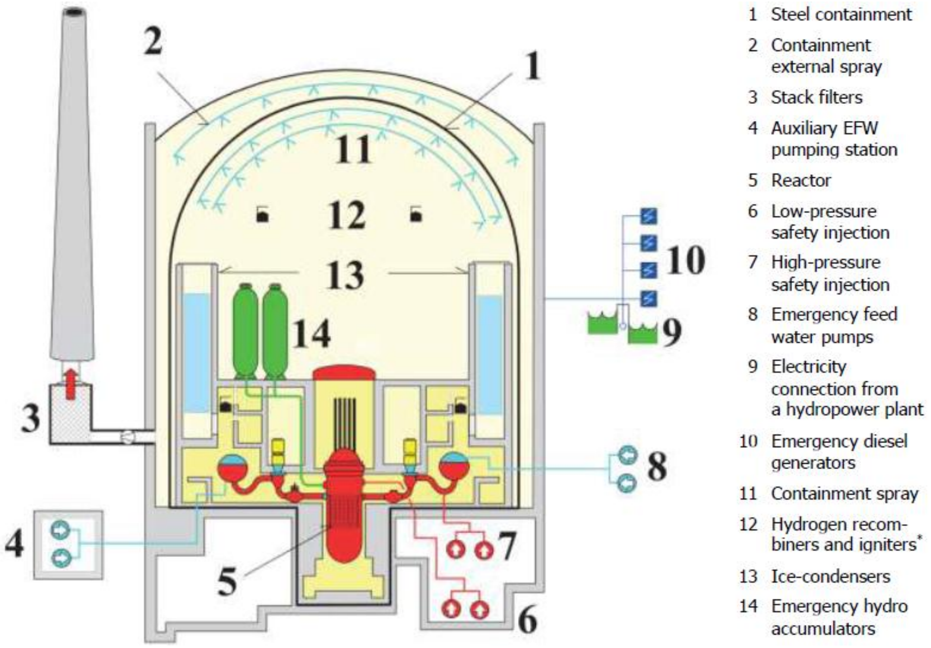

Figure 4. Main safety systems of Loviisa 1&2 NPP.

SAM Guidelines are based on the SAM safety functions. Immediate SAM measures are carried out within the Emergency Operation Procedures (EOP). After carrying out immediate actions successfully, the operators concentrate on monitoring the SAM safety functions. The SAM guidance focuses on monitoring the leak tightness of the containment barrier, and on the long-term issues. The transition to SAM Guidelines takes place when the reactor core is damaged or is close to doing so, and there is no re-turn to EOPs thereafter.

The main safety systems illustrated in Figure 4.

12.2 Accident Management measures in place at the various stages of loss of the core cooling function (Loviisa)

Before occurrence of fuel damage in the reactor pressure vessel: As an ultimate measure, RCS feed & bleed is utilized to depressurise the RCS and to provide possibility for consequent ECC injection to enable core cooling as the last resort. Within the EOPs RCS depressurisation and other immediate actions supporting the SAM are carried out, when core exit temperatures permanently exceed 450°C thus showing that the core is uncovering. These are lowering the thermal shield of the RPV lower head and forcing open the ice condenser doors, to provide adequate flow paths around the RPV outer wall and ensure efficient mixing of the containment atmosphere, respectively.

After occurrence of fuel damage in the reactor pressure vessel: The main goal is to protect the containment function with different systems and measures.

After failure of the reactor pressure vessel: Loviisa NPP SAM strategy relies on retaining corium inside the pressure vessel.

12.2.1 Maintaining the containment integrity after occurrence of significant fuel damage in the reactor core

The SAMG strategy is based on SAM safety functions whose purpose is to ensure containment integrity and isolation.

The SAM safety functions are:

- containment isolation,
- RCS depressurisation,
- in-vessel retention (IVR) of corium by reactor pressure vessel external cooling,
- hydrogen management, and
- management of containment pressure by containment external spray.

In addition to the above SAM safety functions, sub-criticality and fuel pool cooling have to be ensured during a severe accident. All these issues are part of the SAM Guidelines and SAM handbook.

The design basis for all SAM safety functions is that the actions can be done, when the other supplies have been lost, with dedicated independent SAM electrical systems and dedicated independent SAM automation from SAM control room or main control room. The SAM strategy has led to a number of hardware changes at the plant. Also the SAM Guidelines and SAM handbook were prepared.

In most cases, the containment is isolated automatically through a plant protection signal much before the transition into the severe accident regime. However, isolation of the containment and/or confirming isolation status is also part of the SAM Guidelines, due to the crucial importance of successful containment isolation before releasing radioactive substances from the core.

Containment isolation signals may be actuated manually by the operator through a key switch. The containment isolation is in most SBO cases ensured by fail-safe valves or check valves. In those lines where two motor operated valves are needed to be closed, the electrical feed is ensured by batteries, emergency diesels and SAM-diesels. Isolation in SBO is not needed for some valves, but all these valves have hand-wheels for manual closure.

12.2.2 Elimination of fuel damage/melt-down at high pressure

If the pressure in the RCS exceeds significantly the normal operating pressure of 123 bar, the pressurizer safety valves will open at 137 bar. When the core exit temperatures permanently exceed 450°C, the RCS depressurisation is carried out, i.e. all four depressurisation valves will be opened.
The RCS depressurisation is an interface action between the preventive and mitigative parts of the severe accident management. If the RCS feed function is operable, the depressurisation may prevent the core damage. If not, the mitigative actions and measures to protect the containment integrity and mitigate large releases are started. New manually actuated depressurisation capability has been designed and implemented for severe accident management through motor-operated relief valves (two parallel lines with two similar valves in each line).

12.2.3 Management of Hydrogen risks inside the containment

To ensure adequate flow paths for efficient mixing of hydrogen to containment upper compartment atmosphere, a dedicated system for opening the ice condenser doors is implemented. Passive autocatalytic recombiners: 154 in the containment, 84 in the steam generator space, 66 in the dome and 4 in some dead-end spaces. Glow plug system arranged based on DDT (Deflagration-to-Detonation Transition) criteria to prevent flame acceleration.
Loviisa NPP containments are equipped with ice-condenser containments, which are relatively large in size but have a low design pressure of 1.7 bar (abs). The ultimate failure pressure has been estimated to be above 3 bar (abs). An intermediate deck divides the containment in the upper and lower compartments. All the nuclear steam supply system components are located in the lower compartment and, therefore, any release of hydrogen into the containment will be directed into the lower compartment. The main route for hydrogen and steam to reach the upper compartment, which is significantly larger in volume, is through the ice-condensers.
The two redundant system to force open the ice condenser doors was installed in 2001 (LO1) and in 2002 (LO2). The doors of the two ice condensers at each unit can be forced open with pressurized nitrogen operated pneumatic cylinders. After opening, the doors are locked into place requiring no nitrogen pressure. The doors can be opened by operator action or without electricity and automation by manual operations of nitrogen valves, which are accessible during a severe accident.
The PARs installed in 2003 are fully passive components and do not require any operator or local actions apart from shutdown state recovery actions. The arrangement of the PARs ensures hydrogen concentrations low enough not to cause a threat to the containment. The new glow plug system was installed in 2003. The glow plugs are powered by SAM diesel generators, and they are arranged based on DDT criteria in an optimal way to prevent flame acceleration. Opening of ice-condenser doors and activation of glow plug system are included in SAM Guidelines and EOPs.

12.2.4 Prevention of containment overpressure

Ice condensers ensure flooded cavity. The lower neutron and thermal shield can be lowered down. Lowering mechanism is based on water-hydraulic cylinder that lowers the neutron and thermal shield as hydraulic pressure is relieved. Containment internal spray starts automatically if the pressure exceeds 1.17 bar (abs). Also containment external spray system, which is started manually when the containment pressure reaches the design pressure of 1.7 bar (abs). There is a possibility to use the external spray system without dedicated cooling system by water injection from external sources, e.g. a fire truck or other external pump.

The concrete used in the reactor cavity of Loviisa NPP does not contain any CO2, the amount of non-condensable gases (except for hydrogen) generated during core-concrete interaction would be negligible. Therefore, the overpressure protection of containment could be limited to condensing the steam produced. An obvious way of doing this is to spray the exterior of the containment steel shell. Later on, the concept of in-vessel retention was introduced to Loviisa NPP, which excludes core-concrete interactions altogether and thus finally ensures that no non-condensable gases apart from hydrogen need to be considered.
Other possibilities to remove decay heat from the containment are by the LPSI and containment internal spray system heat exchangers through the intermediate cooling system.

12.2.5 Prevention of re-criticality

Controlling core sub-criticality is one of the main accident management objectives. When the original core geometry is lost and a molten pool has been formed in the vessel, re-criticality should not be of concern, even if unborated water is used. However, diagnosing the state of the core geometry in a severe accident may be a challenging task. The risk of misdiagnose is therefore significant. Use of unborated water outside of the RPV, in the reactor cavity, may be problematic as well, even if it has no impact on the sub-criticality of corium inside the reactor pressure vessel. If the reactor pressure vessel failed, the sub-criticality of the debris in the cavity could not be ensured. The debris may become critical if it forms a rubble bed and this bed contains right amount of water acting as a moderator. Thus, only borated water should be used in severe accident, and this is considered in the SAM Guidelines.
Boron injection system:

- chemical/boron supply system provide boron control for make-up system by two redundant high-capacity pumps. The system has also low-capacity pumps (diverse from high-capacity pumps) capable of injecting high concentration boron solution directly into RCS at design pressure of 137 bar.
- The make-up system's two high-capacity pumps inject boron solution from the boron/chemical supply system into the RCS up to pressures of 138 bar. The system has also two redundant low-capacity pumps to provide boron injection into the RCS at elevated pressures up to 160 bar.
- Both the HPSI (High Pressure Safety Injection) and LPSI (Low Pressure Safety Injenction) increase boron concentration in primary coolant when injecting water from the Emergency Core Cooling System (ECCS) tank.
- Borax in the ice condensers ensures high enough boron concentration in the sump water also during the ECC recirculation mode.

12.2.6 Prevention of basemat melt-through

In the beginning HPSI and LPSI systems inject water from ECCS tank. In the case of LOCA leaked coolant is collected to LPSI system and containment spray system sumps. It is also possible to inject water to reactor shaft from fire-fighting system.
Some of the design features of Loviisa NPP make it most amenable for using the concept of In-Vessel Retention of corium by external cooling of the RPV as the principal means of arresting the progress of a core melt accident. Such features include the low power density of the core, large water volumes both in the primary and in the secondary side (long time delays, water for cavity flooding), no penetrations in the lower head of the RPV, and finally, ice condensers ensure a flooded cavity in most severe accident scenarios.
In-vessel retention is mostly ensured by passive means, such as flap valves at inlet and outlet of reactor cavity and strainers. Active operations are required only to lower neutron and thermal shield. After the initial lowering no electricity is needed. Cylinder lowering can also be

executed without electricity by manual operations of hydraulic valves. Loviisa NPP SAM strategy strongly relies on retaining corium inside the pressure vessel, as described above. However, if all means to cool corium inside the pressure vessel are supposed to fail it is possible to enter in a situation where bottom part of the reactor pressure vessel is damaged and molten corium falls to reactor cavity. Primary circuit de-pressurisation prevents high pressure scenarios and vessel failure itself should not jeopardize the containment integrity in case the reactor cavity is dry. If there is water in the reactor cavity in this case the reactor cavity is pressurized by interaction between molten corium and water.

12.3 Severe Accident Management in Olkiluoto 1&2 (BWR)

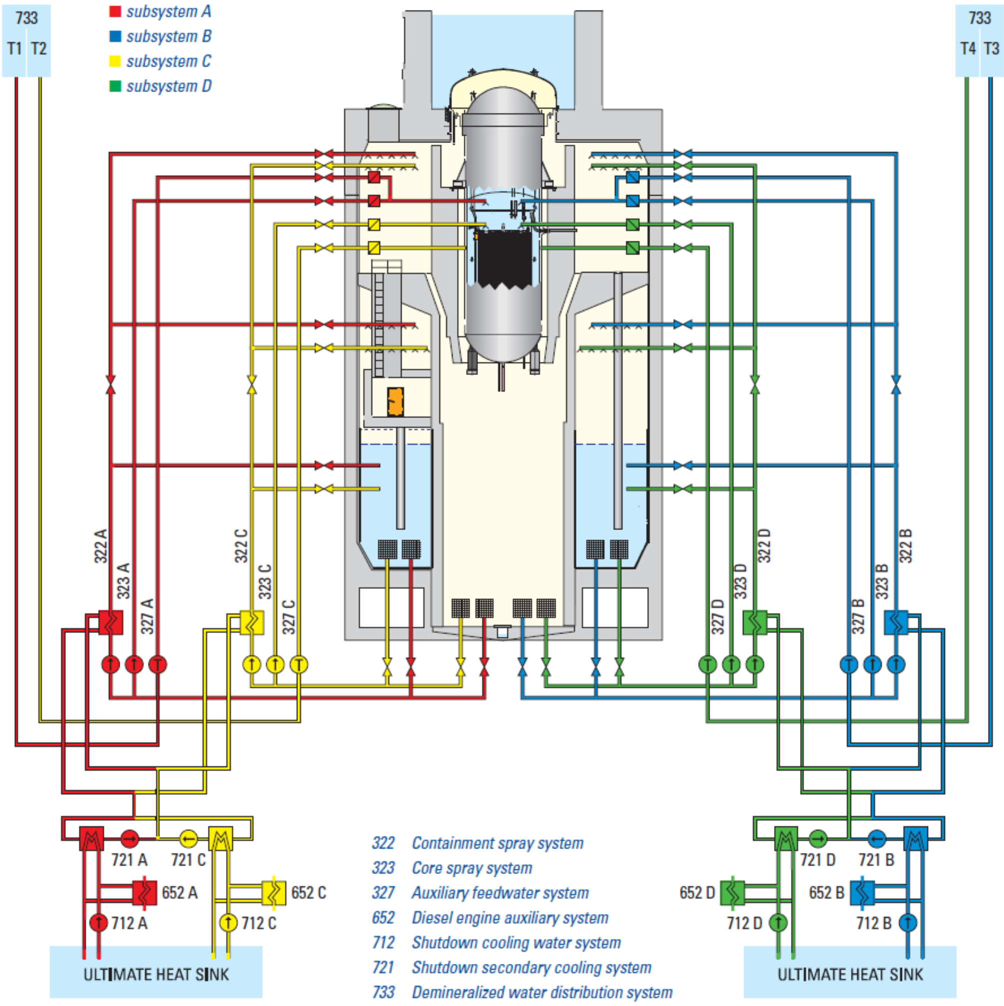

Figure 5. Emergency cooling systems in Olkiluoto 1&2.

There are event-oriented operating procedures for events within design, ranging from insignificant disturbances to postulated design basis accidents. The reactor protection system and the engineered safety systems are automated to function so that no operator actions are needed during the first 30 minutes after an incident or an accident. It is assumed that the allowed operator response time of 30 minutes is enough for finding out what has happened and which procedure to follow. If the reactor protection systems or some of the safety systems fail, a disturbance may develop into an emergency condition beyond design. The same may happen, if the initiating event is more severe than what has been postulated within the design basis. To cope with emergency conditions beyond design, a set of SBEOPs is available. The actions included in the EOPs aim at restoring the operation of the normal safety systems.

The emergency cooling systems are illustrated in Figure 5.

12.4 Accident Management measures in place at the various stages of loss of the core cooling function (Olkiluoto)

Before occurrence of fuel damage in the reactor pressure vessel: The accident management measures included in the symptom based EOPs aim at restoring the operability of the normal safety systems in order to preserve the integrity of the fuel and the primary circuit.
After occurrence of fuel damage in the reactor pressure vessel: If it becomes obvious that a severe reactor accident is imminent (if the reactor cannot be made sub-critical or if the reactor water level cannot be restored within a certain time) the operators are guided to start the most time critical severe accident management measures (depressurisation of the reactor, flooding of the reactor cavity). However, the efforts to start core cooling are still continued, until there is a clear indication of pressure vessel melt-through.
After failure of the reactor pressure vessel, it is clear that the only intact line of structural defence-in-depth is the reactor containment and that core coolability is no longer possible to restore. However, even at that stage, efforts to start the normal containment heat removal systems are still continued.

12.4.1 Maintaining the containment integrity after occurrence of significant fuel damage in the reactor core

The main goal for the accident mitigation is preservation of containment integrity and leak tightness, so that if releases from the containment become necessary, they can be performed in a controlled manner and the releases can be filtered. To this end, the following new systems were implemented:

- protection of the penetrations in the lower drywell against direct contact with the molten corium;
- containment filtered venting system;
- system for filling the containment with water from an external source (the fire water reservoir); and
- a dedicated instrumentation system for monitoring the conditions inside the reactor containment in connection with severe accidents.

Even though the management and mitigation of severe accidents was not included in the original design basis of OL1 and OL2, several original plant systems also play an important part in the severe accident management schemes, such as:

- reactor depressurisation system (to prevent pressure vessel melt-through under high pressure);
- devices for gravity driven flooding of the lower drywell (provision for core melt relocation into the compartment); and
- the fire-fighting water systems (provides water for filling the containment in order to reach a safe stable state).

12.4.2 Elimination of fuel damage/melt-down at high pressure

Automatic depressurisation of the reactor pressure vessel will be actuated, if the water level in the reactor has been below the level of 0.7 m above the top of active fuel for more than 15 minutes. The depressurisation is performed by opening 8 valves of the reactor relief system. The safety valves open also if the reactor pressure exceeds the design pressure (85 bar). If the battery capacity is depleted, the depressurisation valves will close again. For these sequences, the reactor relief system has two so-called fast opening valves which will open and stay open in case the battery backed power for their control valves is lost. The steam released through valves is led underwater in the condensation pool.

12.4.3 Management of Hydrogen risks inside the containment

To prevent hydrogen burns or detonations the containments of OL1 and OL2 plant units are normally nitrogen inerted during power operation. Inerted containment is the main feature for hydrogen management. During power operation only for a short time before shut-down to refuelling outage and after start-up from refuelling outage the oxygen content of the containment atmosphere may be higher than 2%.
The pressure control of hydrogen, and also the control of other non-condensable gases, is based on the containment over-pressurisation protection systems.
The limiting case for hydrogen generation assumes that that 100% oxidation of the core zirconium takes place in the lower drywell during 300 seconds after vessel breach. This will produce a total of 1800 kg of hydrogen. Due to inerted containment, no hydrogen is assumed to burn.

12.4.4 Prevention of containment overpressure

The containment pressure can be decreased by following systems:

- Containment vessel spray system: The containment vessel spray system condenses steam in the containment atmosphere and thus decreases the containment pressure. The spray also washes radioactive particles from containment atmosphere. The system is not dedicated to severe accident management, but if available, it can be used for mitigating the consequences of a severe accident.
- Containment over-pressurization protection system: Containment overpressure protection system is a pipeline with a diameter of 600 mm, and the relief valve is a rupture disc with a bursting pressure of 7 bar. It has been designed to blow from the upper drywell directly into the atmosphere. The filtered venting system consists of pressure relief lines from wetwell and drywell. The filter unit consists of a wet scrubber with venturi nozzles followed by a combined droplet separator and stainless steel fibre filter. The rupture disc is designed to burst at a pressure of 5.5 bar.
- Containment filtered venting system: The function of the filtered venting system is to enable release of steam and gases from the containment to the environment in a controlled way in cases where pressure rise may threaten the integrity of the containment. The system consists of pressure relief lines from wetwell and drywell, a two stage filter unit and an exhaust line to the environment. The system can be actuated by opening the isolation valves in one of the venting lines. The drywell venting line penetrates the containment wall close to the ceiling, which makes water filling of the containment possible. If a release becomes necessary before containment water filling can be started, the wetwell venting line has to be used. In this way, the release of activity can be decreased by using the scrubbing efficiency of the condensation pool. Another pipeline is connected in parallel with the drywell venting line, bypassing the normally closed isolation valves. This line contains a rupture disc and two normally open isolation valves. The rupture disc has been designed to burst at a lower pressure than the disc in the containment over-pressurization protection system to eliminate the risk of an unfiltered release in accident situations, when large amounts of radioactive fission products may be present in the containment atmosphere.

12.4.5 Prevention of re-criticality

OL1 and OL2 have two diverse systems for shutting down the reactor. Beside the control rods, boron injection using enriched Boron-10 is also an efficient means of rapidly reaching sub-criticality. The boron injection system will be automatically actuated in connection with ATWS event sequences, but not in connection with symptoms typical of severe accidents.

The SAM procedures guide the operators to start boron injection manually in case AC power supply is re-established after long interruption, e.g. if the depressurisation of the reactor has already been automatically actuated on low reactor water level due to loss of water injection. It will only take the boron system a few minutes to inject enough boron solution to assure sub-criticality of the core in all conceivable situations, as long as the pressure vessel is intact.

12.4.6 Prevention of base-mat melt-through

In-vessel retention of the corium has not been considered a practical severe accident management scheme for OL1 and OL2. This is due to the fact that it would take too long to inject the amount of water needed to submerge the bottom of the reactor vessel (about 1300 m3) into the containment.

To protect the basemat and the penetrations in the lower drywell, the compartment has to be flooded with water before the pressure vessel melt-through. This can be done using two pipelines that belong to the original containment spray system. There are two valves in each pipeline that must be opened to get the gravity driven flooding started. The valves can be opened either remotely from the main control room or locally through manual actions. Each pipeline has a flow capacity of about 0.15 m3/s.

The amount of water available for flooding of the lower drywell corresponds to a water pool depth of approximately 9 meters in that compartment. In order to make sure that all the penetrations in the lower drywell are submerged before the molten corium enters that compartment, the flooding has to be started at least half-an-hour before pressure vessel melt-through (here, a single failure is assumed that would make one of the two flooding lines unavailable). This means that in connection with worst case scenarios, the flooding has to be started within 30 minutes from the onset of the accident. Flooding of the lower drywell is the most time critical operator action in the accident management schemes.

12.5 References

[FIN11] "European Stress Tests for Nuclear Power Plants, National Report FINLAND" STUK, Radiation and Nuclear Safety Authority - December 30, 2011.

[QFI13] Replies to CESAM WP40 questionnaire by VTT.

13 Annex 5: Severe Accident Management in France

The information provided in this annex is taken from the reference [FRA11], [QFR13] and [RAI09]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [FRA11]. Therefore the strategies presented here reflect mainly the situation end of 2011.
A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of on-going R&D projects. But the target here is to compile the main Severe Accident Management strategies for the French NPPs to provide guidance for further ASTEC development.

Table 20. Nuclear Power Plant units in France.

Site	Number of reactors	Net power[1] (MWe)	Thermal power[2] (MWth)	Type of reactor	Date of first divergence
Belleville	2	1310	3817 (4117)	P4	Reactor 1 : 1987-9 Reactor 2 : 1988-5
Blayais	4	910	2785 (2905)	CPY (CP1)	Reactor 1 : 1981-5 Reactor 2 : 1982-6 Reactor 3 : 1983-7 Reactor 4 : 1983-5
Bugey	4	Reactor 2 : 910 Reactor 3 : 910 Reactor 4 : 880 Reactor 5 : 880	2785 (2905)	CP0	Reactor 2 : 1978-4 Reactor 3 : 1978-8 Reactor 4 : 1979-2 Reactor 5 : 1979-7
Cattenom	4	1300	3817 (4117)	P'4	Reactor 1 : 1986-10 Reactor 2 : 1987-8 Reactor 3 : 1990-2 Reactor 4 : 1991-5
Chinon	4	905	2785 (2905)	CPY (CP2)	Reactor 1 : 1982-10 Reactor 2 : 1983-7 Reactor 3 : 1986-9 Reactor 4 : 1987-10
Chooz	2	1500	4720	N4	Reactor 1 : 1996-7 Reactor 2 : 1997-3
Civaux	2	1495	4720	N4	Reactor 3 : 1997-11 Reactor 4 : 1999-11

Cruas	4	915	2785 (2905)	CPY (CP2)	Reactor 1 : 1983-4 Reactor 2 : 1984-8 Reactor 3 : 1984-4 Reactor 4 : 1984-10
Dampierre	4	890	2785 (2905)	CPY (CP1)	Reactor 1 : 1980-3 Reactor 2 : 1980-12 Reactor 3 : 1981-1 Reactor 4 : 1981-8
Fessenheim	2	880	2785 (2905)	CP0	Reactor 1 : 1977-3 Reactor 2 : 1977-6
Flamanville	2	1330	3817 (4117)	P4	Reactor 1 : 1985-9 Reactor 2 : 1986-6
Golfech	2	1310	3817 (4117)	P'4	Reactor 1 : 1990-4 Reactor 2 : 1993-5
Gravelines	6	910	2785 (2905)	CPY (CP1)	Reactor 1 : 1980-2 Reactor 2 : 1980-8 Reactor 3 : 1980-11 Reactor 4 : 1981-5 Reactor 5 : 1984-8 Reactor 6 : 1985-7
Nogent	2	1310	3817 (4117)	P'4	Reactor 1 : 1987-9 Reactor 2 : 1988-10
Paluel	4	1330	3817 (4117)	P4	Reactor 1 : 1984-5 Reactor 2 : 1984-8 Reactor 3 : 1985-8 Reactor 4 : 1986-3
Penly	2	1330	3817 (4117)	P'4	Reactor 1 : 1990-4
					Reactor 2 : 1992-1
Saint Alban	2	1335	3817 (4117)	P4	Reactor 1 : 1985-8 Reactor 2 : 1986-6
Saint Laurent	2	915	2785 (2905)	CPY (CP2)	Reactor 1 : 1981-1 Reactor 2 : 1981-5
Tricastin	4	915	2785 (2905)	CPY (CP1)	Reactor 1 : 1980-2 Reactor 2 : 1980-7 Reactor 3 : 1980-11 Reactor 4 : 1981-5

13.1 The French approach to the complementary safety assessments (CSAs)

These Complementary Safety Assessments (CSA) are part of a two-fold approach: on the one hand, performance of a nuclear safety audit on the French civil nuclear facilities in the light of the Fukushima event, which was requested from ASN on 23rd March 2011 by the Prime Minister, pursuant to article 8 of the TSN Act and, on the other, the organisation of "stress tests" requested by the European Council at its meeting of 24th and 25th March 2011.

In its CSA reports, EDF describes the site emergency organisation planned to respond to incident, accident or severe accident (SA) situations. This organisation is described in the site On-Site Emergency Plan (PUI), which is required by the regulations and devised to cover situations presenting a significant risk for the safety of the facilities, and which can lead to the release of radioactive, chemical or toxic substances into the environment. The PUI covers the management of SAs.

The procedures implemented in the management of SAs, the training and exercise drills are also detailed in the CSA reports. These three points form part of the GIAG (Severe Accident Intervention Guide) and the sites' PUI baseline. In practice the initial operator training syllabus presented by EDF already includes a part devoted to "Severe Accidents", and exercises simulating SA situations are held regularly.

13.2 Existing accident management measures further to loss of core cooling

In the CSA specifications, ASN asked EDF to describe the accident management measures currently in operation at the different stages of a severe accident, particularly further to loss of the core cooling function:

- before the fuel in the reactor vessel becomes damaged;
  - possible actions to prevent fuel damage;
  - elimination of the possibility of fuel damage at high pressure.
- after the fuel in the reactor vessel has been damaged;
- after failure of the reactor vessel (core meltdown in the reactor pit).

13.2.1 Before the fuel in the reactor vessel becomes damaged

In the CSA reports, EDF indicates that the safety procedures for the reactor fleet in service and the EPR rely on a strategy of defence in depth, which can be summarized as follows:

- measures are taken to avoid incidents;
- if an incident occurs, the protection systems bring the reactor to a safe condition;
- safeguard systems prevent a more severe accident from leading to core meltdown.

The existing measures to prevent entry into a severe accident situation (therefore before the fuel in the reactor vessel becomes damaged), particularly further to situations of flooding, earthquake, loss of electrical power or of the heat sink, come under the incident/accident operation (CIA) procedure.

The measures that can be taken on the reactor fleet to prevent fuel damage aim at restoring a means of core cooling, mainly by using the steam generators or/and by injecting water into the reactor vessel. The possible measures consist in:

- if necessary, restoring an electrical panel that can energise the backup systems;
- deploying an ultimate alignment for injecting water into the vessel of the impacted reactor.

13.2.2 Severe Accident Management Guidelines

Severe accident management guidelines (SAMG) have been developed by EDF since many years, with the objective to define actions based on the containment protection (in the emergency operating procedures (EOP), before SAMG application, the main objective is to assure the short and long terms core cooling).

Regarding the international practice, the severe accident guidelines for the French PWRs may appear singular because it gives a very high importance on the prevention of early containment failure and conducts to limit the possibility of core cooling when the water injection is prohibited.

The latest versions of SAMG include some specific recommendations regarding in-vessel water injection to limit the risks on the reactor containment, for example:

- water injection should be avoided at the beginning of core degradation if the flow rate is not sufficient to compensate both residual power and oxidation power (the idea is to avoid hydrogen production with high kinetics regarding PARs (passive autocatalytic recombiners) capabilities); from a practical point of view, the safety injection system is the only mean able to cope with this recommendation;
- water injection should be avoided after few hours of core degradation if a sufficient break does not exist on the reactor cooling system (RCS); this condition has been drafted to avoid RCS pressurization by injected water vaporization and then DCH;

On the other hand, before or after fuel damage, some other strategies are fully similar to international practices like the one to decrease the pressure in the primary system by opening the pressuriser safety valves when entering SAMG (when Core Exit Thermocouple reaches 1100°C or when high dose rate is measured in the containment).

13.2.3 After the fuel in the reactor vessel has been damaged

Beyond this point, a severe accidents management procedure aims at limiting the consequences in the event of core meltdown. If it has been impossible to avoid a severe accident, the operating priorities are turned towards controlling containment and reducing releases.
In the CSA report, EDF indicates the existing measures in response to the identified risk in a severe accident situation. They are indicated below.

<u>Prevention of re-criticality</u>
The fuel assembly geometry, the presence and arrangement of the control rods and neutron absorbers, the boron content of the water in the primary system and the PTR tank (IRWST for the EPR reactor) were studied at the design stage to exclude the risk of re-criticality in the case of design-basis accidents.
However, in the event of a severe accident, following the loss of the primary coolant as a result of the unavailability of all the safeguard systems, the core heats up and can start to melt. If the primary coolant is not recovered rapidly, the fuel and the core structure suffer damage, the core loses its shape, gradually forming a bed of debris and/or a corium pool which subsequently becomes relocated in the reactor vessel coolant inlet plenum or perforates the bottom of the vessel to reach the reactor pit. In this case the initial margins against re-criticality could be significantly reduced.
In the CSA reports, EDF indicates that it has carried out reactivity studies to analyse the risk of return to criticality for different corium configurations - compact or fragmented - in the reactor vessel or the reactor pit, on the basis of realistic assumptions (conservative in some cases).
These studies conclude:

- the criticality risk is nil when the corium is not fragmented in the water;
- the criticality risk is excluded when the borated water is injected at the minimum boron concentration of the PTR tank.

<u>Elimination of the risk of high-pressure fuel damage or core meltdown leading to direct heating of the containment</u>
In the CSA reports, EDF indicates for the reactors in operation that the prevention of pressure meltdown sequences is based on voluntary opening of the pressuriser SEBIM valve tandems. The opening of the three valve tandems causes rapid depressurisation of the primary system which eliminates the risk of having a highly pressurised reactor vessel when melt-through occurs and the risk of loss of containment through its direct heating. Opening of the valve tandems is required in the majority of situations well before entry into a severe accident on a primary system overheat criterion. In a situation of total loss of the electrical power supplies, valve tandem opening is required in the event of loss of the steam generator supply from the turbine driven auxiliary feedwater pump (TPS ASG). Confirmation of valve opening is required by the severe accident operating documents.

<u>Management of hydrogen risks</u>
The hydrogen released in the containment (through the primary system breach, the pressuriser relief tank, or the corium pool) where it is then mixed by the convection

movements. In the CSA reports, EDF indicates that Passive Autocatalytic Recombiners (PAR) have been installed on all the reactors in operation in order to reduce the hydrogen concentration in the reactor building (BR) in the event of a severe accident. This installation has been effective since the end of 2007.

Risk of slow pressurisation of the containment

The risk of slow containment pressurisation is first managed by Containment spray system. If the usage of the containment spray is not successful or if the containment spray is not available, On the reactor fleet in service, this risk of slow containment pressurisation is dealt with by the existence of the venting-filtration system called "U5" and an associated operating procedure allowing decompression and filtration of the reactor containment in order to maintain its long-term integrity.

In the CSA report, EDF specifies that to exclude any risk of hydrogen combustion in the U5 system that could be induced by condensation of the vapour in the piping, there is a preheating system (venting line conditioning). This conditioning is lost in the event of total loss of the electrical power supplies. Although measures are taken to limit the risk of hydrogen combustion in the U5 venting line (pressure reduction upstream of the line limiting the risk of condensation, recombiners substantially limiting the hydrogen concentration), EDF has undertaken to re-examine the hydrogen risk and its possible impacts on the U5 system.

Risk of reactor containment leaktightness fault

On the reactors in service, confirmation of the isolation of the containment penetrations is required as part of the immediate actions on entry into a severe accident situation. The activity is monitored so that restoration measures can be implemented if necessary. The U2 operating procedure (continuous monitoring of containment integrity) which is part of incident/accident operating procedure (CIA) is applicable in a SA situation. Its aim is to monitor the containment integrity under accident conditions and if necessary restore the reactor containment (by isolating the areas concerned, reinjection of highly radioactive effluents, etc.).

13.2.4 After reactor vessel melt-through

Prevention of basement melt-through

In the CSA reports, EDF states that maintaining the corium in the vessel avoids the ex-vessel corium-concrete interaction phase and thus contributes to the goal of maintaining the integrity of the containment. Stabilization of the situation in the vessel entails restoring a means of injecting borated water into the reactor coolant system within a sufficiently short period of time to avoid vessel rupture.

Assuming failure of the vessel, the corium pours into the reactor pit. In the CSA reports, EDF states the strategy currently in place on the reactors in operation, which is to inject water:

- by an input of water subsequent to vessel failure, using reactor cooling system makeup through the breach at the bottom of the vessel, in accordance with severe accident operations. Furthermore, when the reactor pit is initially dry or containing a low water level, the risk of a steam explosion is considered to be low. According to EDF, the conclusions of the MCCI (Molten core concrete interaction) programme run under the aegis of the OECD confirm this ex-vessel reflooding strategy. This international scientific programme dedicated to the ability to cool the corium-concrete mixture, demonstrated on an experimental scale that a corium pool can be stabilised by the injection of water;
- by flooding of the reactor pit prior to vessel failure, linked to operation of the reactor building containment spray system (EAS) if available before entering the severe

accident phase. If the reactor pit is flooded up to the level of the vessel bottom head, this significantly reduces the risk of basement meltthrough, as the retention of a part of the cooled corium in the vessel and corium contact with the water in the reactor pit reduces the quantity of corium that will contribute to the corium-concrete interaction (CCI)

13.3 Examples of key systems installed on French NPPs for Severe Accident Management

Instrumentation for hydrogen
Following a requirement of the French Safety Authority, EDF has developed some specific instrumentation that should help the operators and emergency teams in understanding the situation regarding hydrogen release during a severe accident.
This instrumentation is based on thermocouples installed on PARs and uses the high temperature of the catalyser plates during the hydrogen recombination with oxygen.

Instrumentation for the vessel failure detection
Following a requirement of the French Safety Authority, EDF has developed a specific instrumentation able to inform the operators and emergency teams on the occurrence of a vessel rupture.
This instrumentation is based on thermocouple located in the reactor cavity.

13.4 Perspectives

Severe accident management should be improved in the near future for the French Gen II PWRs in the context of the post-Fukushima action plans (SAM will be robust to external events) but also in the context of LTO (the EPR safety features are seen as an objective). In particular, in the context of LTO, the French Safety Authority has requested EDF to examine to possibility to reduce the iodine impact in case of use of the containment venting system, to examine solutions able remove heat from the containment without venting and to assure corium stabilization in case of vessel failure.

13.5 References

[FRA11] Complementary Safety Assessments of the French Nuclear Power Plants (European "Stress Test", report by the French Nuclear Safety Authority, December 2011

[QFR13] Replies to CESAM WP40 questionnaire by IRSN

[RAI09] Progress in the Implementation of Severe Accident Measures on the operated French PWRs – Some IRSN views and activities, E. Raimond et al., Slide presentation, OECD/NEA Workshop on Implementation of Severe Accident Management Measures, Switzerland, Oct 2009.

14 Annex 6: Severe Accident Management in Germany

The information provided in this annex is taken from the reference [GER11] and [QGE13a-d]. Much more information regarding systems to cope with severe accidents can be found especially in this reference document [GER11]. Therefore the strategies presented here reflect mainly the situation end of 2011.
Some of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of on-going R&D projects. This is related especially to the development of a complete set of SAMG which did not exist before. But the target here is to compile the main Severe Accident Management strategies for the German NPPs to provide guidance for further ASTEC development.

14.1 General data about German nuclear power plants

In Germany there are plants with pressurised water reactors (PWR) and boiling water reactors (BWR) of different construction lines in operation. All operated plants have been built by Siemens Kraftwerk Union (KWU). Similar Plants are under operation in Spain, Netherlands and Switzerland. According to the time of their construction, the nuclear power plants with pressurised water reactors can be classified according to four construction lines, whereas those with boiling water reactors belong to two different construction lines. The construction line is given for each plant in the second column of Table 21, the expected operating period is listed in the last column.

Table 21. Main characteristics of German NPP.

Operator	NPP	Power [MW brutto]	Type	Operating Periode (13. AtG-Novelle)
EnBW	Obrigheim*	357	PWR 1	01.04.1969 – 11.05.2005
RWE	Biblis A	1.225	PWR 2	26.02.1975 – 06.08.2011
EnBW	Neckarwestheim 1	840	PWR 2	01.12.1977 – 06.08.2011
RWE	Biblis B	1.300	PWR 2	31.01.1977 – 06.08.2011
VATTENFALL	Brunsbüttel	806	BWR 69	09.02.1977 – 06.08.2011
e-on	Isar 1	912	BWR 69	21.03.1979 – 06.08.2011
e-on	Unterweser	1.410	PWR 2	06.09.1979 – 06.08.2011
EnBW	Philippsburg 1	926	BWR 69	26.03.1980 – 06.08.2011
e-on	Grafenrheinfeld	1.345	PWR 3	17.06.1982 – 31.12.2015
VATTENFALL	Krümmel	1.402	BWR 69	28.03.1984 – 06.08.2011
RWE	Gundremmingen B	1.344	BWR 72	19.07.1984 – 31.12.2017
RWE	Gundremmingen C	1.344	BWR 72	18.01.1985 – 31.12.2021
e-on	Grohnde	1.430	PWR 3	01.02.1985 – 31.12.2021
EnBW	Philippsburg 2	1.468	PWR 3	18.04.1985 – 21.12.2019
e-on	Brokdorf	1.480	PWR 3	22.12.1986 – 31.12.2021
e-on	Isar 2	1.485	PWR 4	09.04.1988 – 31.12.2022
RWE	Emsland	1.400	PWR 4	20.06.1988 – 31.12.2022
EnBW	Neckarwestheim 2	1.400	PWR 4	15.04.1989 – 31.12.2022

As a result of a political decision in the aftermath of the Fukushima event some older plants are in permanent shutdown since this moratorium. This decision is based on an amendment of the

Atomic Energy Act which entered into force on 6th August 2011. The eight plants which are in permanent shutdown (beside of the Obrigheim plant) are highlighted in Table 21.

14.2 Severe Accident Management

In response to the severe accidents at Three Mile Island and especially after the Cher-nobyl accident in 1986, the German Reactor Safety Commission (RSK) was asked to check whether any measures to enhance the NPPs safety and to cope with severe ac-cidents are possible and if so, what these measures could be [RSK87]. The results of the German Risk Study „Deutsche Risikostudie Kernkraftwerke - Phase B“ (1981-1989) [GRS89], the first large comprehensive study including deterministic and probabilistic results of severe accidents based on a PWR reference plant, significantly influenced the development w.r.t. severe accident management in Germany.

First requirements for a Severe Accident Management (SAM) program regarding be-yond-design-basis events starting from power operation only were published in autumn 1988 after intensive discussions within the RSK [RSK87]. The concept was called "Anla-geninterner Notfallschutz", and the primary intention was the prevention of severe acci-dents starting at power operation. Some selected mitigative measures for dominating phenomena were proposed as well. For both necessary hardware modifications have been considered. The filtered containment venting system was one of the systems which was recommended and installed very early, in the late 1980s [RSK88], [RSK91] the hydrogen counter measures followed.

The final RSK recommendation regarding a Severe Accident Management Program was published in 1992 [RSK92] and provided all details for SAM concepts to be developed and implemented by the licensees to deal with severe accidents starting from full power operation. With respect to accident management and its organisation in NPPs a distinct line is drawn in Germany between the design basis area and the beyond-design-basis area. Accidents within the design basis area are dealt with by so-called 'event-oriented pro-cedures' if the event is clearly identifiable by use of a decision tree. If this is not the case, a set of "symptom-oriented procedures" is additionally in place. Both sets of procedures are comprised in the Operating Manual (Betriebshandbuch, BHB) [KTA11a]. BDBAs are dealt with by using the so-called "Notfallhandbuch (NHB)" or (beyond-design-basis) Accident Management Manual [KTA11b]. The NHB is structured along the same lines as the symptom-oriented part of the operating manual, i.e. it is based on the fundamental safety function concept. The NHB includes preventive (core intact) as well as a few mitigative severe accident management procedures (core damaged). The emphasis is, however, on the prevention side and limited "guidance" is available up to now besides these procedures for the core damage situation. The mitigative procedures describe e.g. how to operate the filtered containment venting system installed as part of the Severe Accident Management Program.

In 2009/2010, the German RSK started a renewed discussion on the implemented SAM measures in Germany. This resulted in the publication of new and extended recommendations: "Basic recommendations for the planning of emergency control measures by the licensees of nuclear power plants" [RSK10]. Special focus there is on: emergency response organisation, internal and external alert procedure, communication in case of an emergency, technical and organisational matter of emergency organisation, emergency documentation.

A short overview for the implementation of accident management measures (4/2011) is given in Table 22 and Table 23. Afterwards, a more detailed description is provided.

Table 22. Implementation of accident management measures in BWRs (4/2011).

Measure	BWR type 69				BWR type 72	
	KKB	KKI 1	KKP 1	KKK	KRB II B	KRB II C
Emergency manual	●	●	●	●	●	●
Independent injection system (steam driven turbo-pump)	●	●	●	●	□	□
Additional injection and refilling of the reactor pressure vessel	●	●	●	●	●	●
Assured containment isolation	●	●	●	●	✓	✓
Diverse pressure limitation for the reactor pressure vessel	●	●	●	●	●	●
Filtered containment venting	●	●	●	●	●	●
Containment inertization by N_2 and implementation of PARs	●	●	●	●	●*	●*
Supply-air filtering for the control room	●	●	●	●	●	●
Emergency power supply from neighbouring plant	□	□	●	□	●	●
Increased capacity of batteries (2 hours)	●	✓	●	●	✓	✓
Restoration of off-site power supply	●	●	●	●	●	●
Additional off-site power supply (underground cable)	●	●	●	●	●	●
Sampling system in the containment	❍	●	●	❍	●	●

* wetwell N2 inerted, drywell and wetwell equipped with passive autocatalytic recombiners
✓ design ● realised through backfitting measures ❍ applied for □ not applicable

Table 23. Implementation of accident management measures in PWRs (4/2011).

Measure	Generation 2				pre-KONVOI				KONVOI		
	KWB A	GKN 1	KWB B	KKU	KKG	KWG	KKP 2	KBR	KKI 2	KKE	GKN 2
Emergency manual	●	●	●	●	●	●	●	●	●	●	●
Secondary-side bleed	●	●	●	●	●	●	●	●	●	✓	✓
Secondary-side feed	●	●	●	●	●	●	●	●	●	●	●
Primary-side bleed	●	●	●	●	●	●	●	●	●	●	●
Primary-side feed	●	●	●	●	●	●	✓	●	●	✓	✓
Assured containment isolation	●	●	●	●	●	✓	●	●	●	✓	✓
Filtered containment venting	●	●	●	●	●	●	●	●	●	●	●
Passive autocatalytic recombiners to limit hydrogen formation	●	●	●	●	●	●	●	●	●	●	●
Supply-air filtering for the control room	●	●	●	●	●	●	●	●	●	✓	●

Measure	Generation 2				pre-KONVOI				KONVOI		
	KWB A	GKN 1	KWB B	KKU	KKG	KWG	KKP 2	KBR	KKI 2	KKE	GKN 2
Emergency power supply from neighbouring plant	●	●	●	□	□	□	●	□	□	□	●
Sufficient capacity of the batteries (2 hours)	●	●	●	✓	●	✓	●	●	●	●	●
Restoration of off-site power supply	●	●	●	●	●	●	●	●	●	●	✓
Additional off-site power supply (underground cable)	●	●	●	●	●	●	●	●	●	●	●
Sampling system in the containment	❍	●	●	●	●	●	●	●	●	●	●

✓ design ● realised through backfitting measures ❍ applied for □ not applicable

14.3 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

14.3.1 Before occurrence of fuel damage

Almost all of the PWRs use the following AM measures:

- Use of operational margins and hardened systems (like volume control system, emergency borating system, use of LP ECC pump as "booster pump" for HP ECC pump to inject water from containment sump etc. for injection at high pressure)
- Secondary bleed and feed by feed water system and tank or by a mobile pump
- Primary bleed and feed by installed ECCS systems
- Emergency injection into the demineralized-water reservoirs and spent fuel pool
- Restoration of AC power supply (e.g. third grid connection)
- Restoration of damaged/failed safety systems

In BWRs, where automatic depressurization of the RPV is initiated automatically at given criteria, typically the following measures are implemented:

- High-pressure injection into the RPV
- Medium-pressure injection into the RPV
- Low-pressure injection into the RPV using diverse water reservoirs and mobile systems
- Containment venting

Steam driven injection systems are only available for the BWR type 69 plants, which are no longer operated. BWR type 72 uses an additional independent injection and cool down system with a diverse reactor protection system.

14.3.2 Measures after the occurrence of fuel damage

Preventive measures based on water injection into the damaged core with the objective to cool the core and to achieve a coolable state are foreseen. The measures will be described in the SAMG documents, which will be implemented until the end of 2013.
PWRs: Active flooding of the reactor pit before the RPV failure is not possible. Late water injection on top of the melt in the cavity might be a strategy, if the ECCS systems will be

available. Passive autocatalytic re-combiners (PAR) are installed in the containment for severe accident conditions. A filtered containment venting system is installed and will be put into operation at a given containment pressure. German PWRs have no spray systems.
BWRs: Start-up of a boron injection system TW is possible to prevent re-criticality. For preventing a melt-through of the RPV, there exists the possibility to cool the reactor pressure vessel from outside by flooding the drywell of the containment. The success probability was analysed within the plant specific PSAs and is assumed to be quite different. The amount of water needed is quite high for BWR type 72 Plants. A system for filtered containment venting is installed in all Plants connected to the wetwell only and will be put into operation if a given pressure criterion is reached. Accident-proof instrumentation could be used to some extent to determine the current plant status. All containments are inerted with nitrogen during operation, except the drywell of BWR type 72 Plants. There in addition passive autocatalytic recombiners are installed in the containment (drywell and wetwell). The implementation of passive autocatalytic recombiners in the reactor building above the spent fuel pool in operating BWRs is under discussion.

14.3.3 Measures after the failure of the RPV/a number of pressure tubes

PWRs: Studies of PSA level 2 showed that first a dry phase MCCI in the reactor pit occurs. Due to the erosion of the biological shield respectively the basemat until air ventilation channels below the cavity floor are reached, a water ingression into the reactor pit after several hours is probable. Further studies of the coolability of a melt exiting from the barriers will be carried out within the framework of the SAMG.
BWR type 72: Severe Accident Management Measures for flooding the drywell of the containment are provided and described in the Accident Management Manual. The success probability was analysed within the plant specific PSA. Penetrations inside the containment wall nearby the cavity have been protected by brig walls to enhance the protection against melt attack and a early containment failure.

14.4 Maintaining containment integrity after an occurrence of significant fuel damage (up to core meltdown) in the reactor core

The strategy is based on the

1. Elimination of fuel damage/meltdown at high-pressure (primary depressurization)
2. Management of hydrogen risks inside the containment (PARs/inertisation)
3. Prevention of containment overpressure (filtered venting system)
4. Prevention of base mat melt-through (water injection)
5. Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting containment integrity – use of permanently installed and mobile diesel systems
6. Measuring and control instrumentation needed for protecting containment integrity – use of permanently installed and mobile diesel systems

14.5 Accident management measures to restrict radioactive releases

- Radioactive releases after a loss of containment integrity: Due to the robust and conservative design of the containment as well as the measures established for containment protections e.g. filtered venting, no Severe Accident Management . Measures for restricting activity releases into the environment after the containment integrity is lost are foreseen in the Accident Management Manual.

- Accident management after uncovering of the top of fuel in the spent fuel pool: Currently, there are no specific Severe Accident Management Measures described in the Accident Management Manual for the conditions after uncovering of the top of fuel in the spent fuel pool. In future the prevention of such a case will be further supported by additional measures to inject water into the SFP. The implementation of passive autocatalytic recombiners in the reactor building above the spent fuel pool in operating BWRs is under discussion.

14.6 References

[GER11] EU Stress test – National Report of Germany – Implementation of the EU Stress Tests in Germany, October 31, 2011

[GRS89] Gesellschaft für Reaktorsicherheit, (GRS) mbH, German Risk Study on Nuclear Power Plants, Phase B, GRS-A-1600, 1989

[KTA11a] KTA 1201, Requirements for the Operating Manual, Safety Standards of the Nuclear Safety Standards Commission (KTA), issued 2009-11

[KTA11b] KTA 1203, Requirements for the Accident Management Manual, Safety Standards of the Nuclear Safety Standards Commission (KTA), issued 2009-11

[QGE13a] Replies to CESAM WP40 questionnaire by GRS

[QGE13b] Replies to CESAM WP40 questionnaire by KIT

[QGE13c] Replies to CESAM WP40 questionnaire by RUB

[QGE13d] Replies to CESAM WP40 questionnaire by USTUT

[RSK10] Basic recommendations for the planning of emergency control measures by the licensees of nuclear power plants; Recommendation of SSK and RSK, issued in the 242th Meeting of SSK on 01./02. July 2010, authorized in the 244th Meeting of SSK on 03. November 2010, issued in the 429th Meeting of RSK on 14. Oktober 2010

[RSK87] Überprüfung der Sicherheit der Kernkraftwerke mit Leichtwasserreaktor in der Bundesrepublik Deutschland, Ergebnisprotokoll der 218. RSK-Sitzung am 17.12.1986 und der 222. RSK-Sitzung vom 24.06.1987

[RSK88] Abschlussbericht über die Ergebnisse der Sicherheitsüberprüfung der Kernkraftwerke in der Bundesrepublik Deutschland durch die RSK, Ergebnisprotokoll der 238. RSK-Sitzung am 23.11.1988

[RSK91] Spezifikationen für Filtersysteme in den Druckentlastungsstrecken des Sicherheitsbehälters von Druckwasserreaktoren und Siedewasser-reaktoren, Stellungnahme der RSK, 263. Sitzung am 24.06.1991

[RSK92] Behandlung auslegungsüberschreitender Ereignisabläufe für die in der Bundesrepublik Deutschland betriebenen Kernkraftwerke mit Druckwasserreaktoren, Positionspapier der RSK zum anlageninternen Notfallschutz im Verhältnis zum anlagenexternen Katastrophenschutz, Ergebnisprotokoll der 273. RSK-Sitzung am 09.12.1992

15 Annex 7: Severe Accident Management in Hungary

The information provided in this annex is taken from the references [HUN11], [QHU13] and [LAJ09a+b]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [HUN11]. Therefore the strategies presented here reflect mainly the situation end of 2011.
A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Hungarian NPPs to provide guidance for further ASTEC development.

Table 24. Nuclear Power Plant units in Hungary.

NPP	Start of operation	Power [MWth]	Reactor Type
Paks Unit 1	1982	1485 MW	VVER440/213
Paks Unit 2	1984	1485 MW	VVER440/213
Paks Unit 3	1986	1485 MW	VVER440/213
Paks Unit 4	1987	1485 MW	VVER440/213

15.1 Status of Severe Accident Management Modifications and SAMG strategies

In Paks NPP, an integrated analysis and then a modification programme was launched as far back as the year 2008 to prepare for mitigation of consequences of low probability beyond design basis accidents that might lead to severe damage to the reactor, called severe accidents. The work resulted in several technological modifications required for the introduction of the severe accident management guidelines, most of which are already implemented at Paks NPP; however the completion level of them is different in the various units. The status of the separate modifications in the units is shown in Table 25.
The development of the Severe Accident Management Guidelines (SAMG) was launched in collaboration with Westinghouse Electric Belgium Co. Therefore the SAMGs in Paks follow the generic Westinghouse SAMG approach with special adaptations for Paks VVER 440/2013 units.
The principal elements of severe accident management strategies are as follows:

- Depressurization of the primary system
- Water injection into the primary system
- In-vessel melt retention
- Preventing excessive vacuum in the containment
- Preventing containment overpressure
- Decreasing fission product release

This leads especially to the following measures depending of the stage of the accident (the list is not exhaustive).

Table 25. Status of SAM modifications or scheduled implementation in Hungary.

Table 1-2: Status of severe accident management modifications or scheduled date of implementation

Modification	Unit 1	Unit 2	Unit 3	Unit 4
Construction of reactor cavity flooding system	Implemented	2012 main outage	2013 main outage	2014 main outage
Construction of autonomous power supply to designated consumers	Implemented	Implemented	Implemented	Implemented
Installation of passive hydrogen recombiners	Implemented	Implemented	Implemented	Implemented
Reinforcement of cooling circuit of spent fuel pool against loss of coolant	Implemented	Nov-Dec, 2012	Feb-Mar, 2013	Jan-Feb, 2012
Installation of severe accident measurement system	Implemented	Jun-Aug, 2012	Sept-Oct, 2013	May-June, 2013
Introduction of severe accident management guidelines	Dec 31, 2011	Dec 31, 2012	Dec 31, 2013	Dec 31, 2014

The modifications implemented at Paks NPP with regard to severe accident management are aimed at stopping any assumed severe accident event sequence and to bring the unit to safe cold sate. Two key elements of the practical execution of severe accident management are the execution of the technical modification belonging to SAM and the introduction of SAMG.

15.1.1 Measures before fuel damage in the RPV

In the case of total blackout and/or the loss of the ultimate heat sink, primary pressure is high in the early stage of the process; therefore the most important function is the reduction of the pressure. Prior to the evolution of extensive core damage, the pressure reduction is made according to the Symptom-oriented Operating Procedures.
This can be achieved by depressurising the Steam Generators when the temperature measured by the Core Exit Thermocouple (CET) exceed 370 C or by depressurising directly the RCS through Pressuriser safety valves or the letdown system valves when the temperature measured by the Core Exit Thermocouple (CET) exceed 550 C.

15.1.2 Measures after fuel damage in the RPV

At first the external cooling of the reactor pressure vessel requires water discharge from the localization tower (from the bubble trays) to the floor of the containment, and then the water can be discharged to the reactor cavity from there by the force of gravity. The electrical power of the discharge valves can be provided from the normal, safety and severe accident power supplies. The discharge of water from the localization tower has to be started before the evolution of extended core damage, when the core outlet temperature reaches 550 °C.
Other way can be used to inject water in the containment if needed, for example through the Steam Generators.
Throughout the accident, injection into the SGs can be achieved through the Auxiliary Feedwater system but also though alternate source like fire water injection.
Several types severe accident diesel generators (fixed, mobiles) for supplying electrical power to SAM instruments and systems are also available.

15.1.3 Measure to eliminate fuel damage/melt-down on high pressure

In the case of total blackout and/or loss of the ultimate heat sink, the primary circuit pressure is high in the initial phase of the process. The reduction of the pressure is important because: certain elements of the emergency core cooling system can start to operate only on lower pressure level, and the consequences of any damage to the reactor pressure vessel on high pressure have to be avoided by any means. The Symptom-based Operating Procedures give instruction on unconditional pressure reduction above 550°C core outlet temperature with the assumption that all attempts to restore the cooling of the core were unsuccessful. If the core outlet temperature further increases during the application of the Symptom-based Operating Procedures and then exceeds the value of 800 °C in the case of total blackout, or the value of 1100 °C in any other case, then the SAMGs have to be applied. The SAMGs include new instructions to reduce the primary pressure with all available means.

15.1.4 Measures to manage Hydrogen risk inside the containment

The 60 (30 pairs) NIS type passive autocatalytic severe accident recombiners installed for hydrogen management in the containment significantly reduce the quantity of hydrogen generated during the analysed severe accident processes.

15.1.5 Measures to prevent underpressure in the containment

Excessive depression may be established in the containment of a VVER- 440/213 due to the following physical phenomena:

- Relocation of air from the main building into the air-traps and concurrent steam condensation
- Release of non-condensing gases through the containment leakage
- Decreasing the fraction of hydrogen and oxygen as a result of the operation of catalytic recombiners or hydrogen burn

The Containment depression may be intensified by the operation of the containment spray system. This excessive vacuum can develop in case of 3 operating trains of the spray system. The remedy is quite simple: stopping one or two trains of the spray system when containment pressure reaches the atmospheric level.

15.1.6 Measures to prevent overpressure of the containment

After the flooding of the reactor cavity, the residual heat of the molten core in the reactor pressure vessel, by heat transfer trough the wall of the vessel, warms up the coolant in the cavity. The evaporation of the coolant increases the quantity of steam in the containment; if the sprinkler system will be used to reduce pressure. If not successful, the only possibility is to discharge air through the venting system of the containment; thus damage to the containment can be avoided.

15.1.7 Measures to prevent containment bypass in case of SG tube rupture

In case of Steam Generator tube or collector rupture, blowdown of the SG inventory into the containment can be put in place.

15.2 References

[HUN11] "National report of Hungary on the targeted Safety Re-assessment of Paks Nuclear Power Plant" Hungarian Atomic Energy Authority – 29 December 2011

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[LAJ09a] "Verification of the SAMG for Paks NPP with MAAP code calculations", Gábor Lajtha, Zsolt Téchy (NUBIKI, Hungary); József Elter, Éva Tóth (Paks NPP, Hungary) - OECD/NEA Workshop on Implementation of Severe Accident Management Measures PSI, Villigen, Switzerland, ISAMM 2009-October 26-28, 2009

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16 Annex 8: Severe Accident Management in Romania

The information provided in this annex is taken from the reference [ROM11]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [ROM11]. Therefore the strategies presented here reflect mainly the situation end of 2011.

A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Romanian NPPs to provide guidance for further ASTEC development.

16.1 Main characteristics of the Cernavoda NPP units

All the units are pressurised heavy water reactors (PHWR), of CANDU 6 type, designed by AECL (Atomic Energy of Canada Ltd.). Each unit is provided with a dedicated Spent Fuel Bay (SFB) for the spent fuel temporary storage. The SFB is designed to accommodate the fuel discharged during 8 years. After 6-7 years of operation, the spent fuel bundles are transferred to the on-site, naturally air cooled dry storage facility (IDSFS) for the spent fuel long term storage.

Table 26. General data on Cernavoda NPP units.

Reactor	Type	Gross Capacity MW(e)	First Criticality	Operating Status
Cernavoda-1	CANDU-6	706.5	16th of April 1996	in operation
Cernavoda-2	CANDU-6	706.5	6th of May 2007	in operation
Cernavoda-3	CANDU-6	720	-	under preservation, plans for resuming construction
Cernavoda-4	CANDU-6	720	-	under preservation, plans for resuming construction
Cernavoda-5	CANDU-6	-	-	under preservation

The Cernavoda NPP has implemented a set of Severe Accident Management Guidelines (SAMGs). The objectives of the SAMGs are:

- To terminate core damage progression;

- To maintain the capability of containment as long as possible;
- To minimize on-site and off-site releases.

The SAMGs for the Cernavoda NPP have been developed based on the generic CANDU Owners Group (COG) SAMGs for a CANDU-6 type of plant. In developing the generic SAMGs, COG adopted the Westinghouse Owners Group (WOG) approach, with the necessary technical modifications suitable for implementation in CANDU plants, based on extensive CANDU specific severe accident analysis and research.

Table 27. SAMGs for Cernavoda NPP.

SAMG	Priority	Scope of application
Severe Accident Guidelines (SAG)	SAG-1	Inject into Heat Transport System
	SAG-2	Control Moderator Conditions
	SAG-3	Control Calandria Vault Conditions
	SAG-4	Reduce Fission Product Release
	SAG-5	Control Containment Conditions
	SAG-6	Reduce Containment Hydrogen
	SAG-7	Inject into Containment
Severe Challenge Guideline (SCG)	SCG-1	Mitigate Fission Product Release
	SCG-2	Reduce Containment Pressure
	SCG-3	Control Containment Atmosphere Flammability
	SCG-4	Control Containment Vacuum

16.2 Accident management measures to restrict the radioactive release

Fission product releases are monitored as part of the Diagnostic Flow Chart (DFC). Severe Accident Guideline 4, Reduce Fission Product Releases, is called upon if the radiation dose measurements exceed the SAG-4 set point (0.2mSv/h at the station boundary). The set point covers both airborne releases and direct shine from containment (the conditions should be prolonged and not triggered by a short term transient release).

Strategies to reduce fission product release from the containment are grouped in four categories:

- Stop the release (box-up the containment);
- Slow down the release rate (reduce containment pressure);
- Remove airborne component of fission products (time, dousing, chemicals, water, local air coolers);

16.2.1 Accident Management after uncovering of the top of fuel in the fuel pool

Based on WANO SOER 2011-2 recommendations, an emergency operating procedure called "APOP G04 - Spent Fuel Bay cooling abnormal conditions" was developed, validated and issued in order to address prolonged/ extended loss of Spent Fuel Bay cooling capability

To support APOP G04 execution, the following design changes and operational measures have been implemented or are in progress:

- tubing was installed above Spent Fuel Bay for H2 sampling;
- hard level gauge were installed in the Spent Fuel Bay and reception bay;
- a new pipe seismically qualified that has connection outside Spent Fuel Bay is installed in order to be used to add water in the bay using connections from the fire truck or from a mobile pump (in progress);
- ventilation of the area above the bay can be provided by opening SFLA doors or by designing a hatch in the roof and ventilation openings in the walls (in progress).

During the validation exercise for APOP G04 the worst case was considered, with the normal make-up water and the back-up fire water being considered unavailable. The procedure was followed and a source of make-up water was established 30 minutes after initiation of the event, using fire truck and hoses.

16.3 References

[ROM11] "ROMANIA – National Commission for Nuclear Activities Control – National Report on the Implementation of the Stress Tests"– December 2011

17 Annex 9: Severe Accident Management in the Slovak Republic

The information provided in this annex is taken from the reference [SR11], [QSR13a+b], [ENS12] and [SR12]. Much more information regarding systems to cope with severe accident can be found especially in this reference document [SR11]. Therefore the strategies presented here reflect mainly the situation at the end of 2011.
A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Slovak Republic NPPs to provide guidance for further ASTEC development.

Table 28. Nuclear Power Plant units in the Slovak Republic.

NPP	First connection to the grid	Power [MWe]	SAMG
Bohunice 3	1984	505	Prepared from 2002 to 2004
Bohunice 4	1985	505	
Mochovce 1	1998	470	Update 2015
Mochovce 2	1999	470	
Mochovce 3	Under construction	470	
Mochovce 4	Under construction	470	

17.1 Severe Accident Management

Development and implementation of the accident management programme, including mitigation of severe accidents has been an on-going process in all nuclear units in Slovakia. In 2004-2005, an overall study defining technical specification of modifications and extensions of the VVER-440/V213 basic design needed for implementation of SAMG was prepared. The project of implementation of modifications to support the severe accident management on the basis of SAMG was proposed in compliance with all the requirements and recommendations in Slovak legislation in 2006 - 2007. The SAM implementation project was initiated in 2009 as the common NPP Bohunice 3&4 (EBO 3&4) and Mochovoce 1&2 (EMO1&2) project with deadline in 2013 in EBO and the follow-up implementation in EMO1&2 (implementation accelerated after the Fukushima, with the new deadline 2015).
SAM project being currently implemented in both NPP Bohunice 3&4 and EMO1&2 is based on originally defined scope with assumptions for occurrence of a severe accident on only one of two units. In view of the lessons learned from the Fukushima accident the project completion will be followed by evaluation of a possible extension to management of a severe accident on both units at the same time and to all plant operational states.

17.2 Main SAM strategies used

The SAM strategies are applied after SAMG entry. Criterion for SAMG entry is core exit temperature:
> 800 °C - in case of no AC power (station blackout scenario),

> 1100 °C - all scenarios with AC power available

17.2.1 Management of AC/DC power

System for Emergency Power Supply for SAM
There are 3 diesel generators of the safety divisions available in a case of loss of external power sources. Added is 1 diesel generator per unit dedicated to SAM (DG-SAM).
The SAM systems (electro valves, spray, external …) are supplied from batteries of the system SZN S, capacity about 2 h (depends on number of consumers that are operated).

17.2.2 Reaching and maintaining reactor shutdown/sub-criticality

Efficient control rods are available.
All available water sources for providing water into the primary circuit in case of accident and successful reactor shutdown contain "sufficiently borated water" for maintaining sub-criticality. Boron injection with HP injection (40 g H_3BO_3/kg)
Four accumulators, HP and LP tanks of the ECCS and CONT spray system;
For SAM: external source of water.

17.2.3 Management and recovery of the ultimate heat sink

For VVER-440/V213 operated in Slovakia there is adopted IVR strategy. There are the following means for decay heat removal within separate phases of the severe accident control: quenching of the core, transition to low pressure injection in recirculation mode, spray system in recirculation mode for long term heat removal from containment atmosphere possibly also dedicated fan cooler system for long term heat removal are intended to be operated (see "containment heat removal strategy").
Three independent systems of essential service water are available.
Feeding of SGs from external FW tanks with capacity that is sufficient for 72 h is available.
Special nozzles for SG feeding were installed and mobile firefighting cars equipped with high-pressure pumps are available.

17.2.4 Injection into the RPV/RCS/ RCP seals

There are safety systems (HP+LP+ confinement spray), with automatic transition from direct mode to recirculation mode after depletion of injection tanks to the confinement sump. Four accumulators are available.
Additionally: An external source of borated water (system dedicated to SAM) for injection to primary circuit, CONT spray system, open reactor and spent fuel pool. The system consist of tanks, special pump and connection pipelines to the associated systems (primary circuit, containment spray collector, open reactor and spent fuel pool). Start-up of the system is performed manually after SAMG entry and according to decision of the specialists presented in Technical Support Centre the injection of water is directed to selected system (e.g. either to primary circuit or to containment spray collector).
Open reactor case
Operators can start above mentioned system and inject water into the reactor vessel. Based on available water level measurements, the water delivery can be adjusted to ensure the flooding of core for very long time (~3 days). Moreover, the water available in the bubble condenser trays and in the 4 accumulators can be used. Then, the time margin up to start of fuel heat-up is increased for several other days.
Spent fuel pool
The same as for open reactor case.

17.2.5 Depressurization of the RCS

The technical solution dedicated to SAM consists in addition of one depressurization pipeline to the existing system of pressurizer relieve and safety valves. This new pipeline is equipped with a special valves which are opened manually (i.e. signal is initiated manually) just after entry to SAMG.

17.2.6 External cooling of the RPV

A system for flooding the reactor cavity and external cooling of RPV dedicated to SAM has been implemented. The function is as follows:

- after entry to SAMG the drainage of bubble tower trays is initiated, with the aim of getting to the confinement floor sufficient volume of coolant for reactor cavity flooding and long-term operation of the IVR system. Dedicated and qualified drainage valves that are used in normal operation for the water level maintenance in bubbler trays are being opened and the content of upper trays is being poured through lower ones. Water is being discharged onto SG box floor;
- reactor cavity flooding valves protected by inlet construction and isolation siphon are installed on the floor of both corridors to allow cavity flooding and to prevent coolant losses by outflow to the ventilation system room. The overflow edge of the cavity-flooding system is located sufficiently high above the elevation of confinement sumps in order to allow normal recirculation function of safety systems (ECCS, spray system)
- steam being produced around RPV wall is released through dedicated back flaps to SG box at the upper part of reactor cavity.

17.2.7 Molten Corium stabilization

The IVR strategy is used. The molten corium is stabilized in the RPV lower plenum providing that external cooling of RPV is successful. Technical measures were implemented in order to enable timely reactor cavity flooding and efficient cooling of the RPV outer surface via natural circulation.

17.2.8 Injection into the containment

An external source of borated coolant dedicated to severe accident allows coolant delivery to spray system, primary circuit and spent fuel pool, but without the possibility of coolant recirculation.

17.2.9 Containment heat removal (spray, fan coolers, etc.)

Heat accumulation capacity of the containment can be temporary increased by spraying from external coolant sources (special SAM system).
Concerning long-term heat removal from containment atmosphere there are two dedicated systems on field of severe accident:

- dedicated and qualified recirculation spray system for long term heat removal from containment atmosphere (i.e. 1 of 3 containment spray system trains is charged during SA and qualified for SAM conditions),
- one of the existing circulation ventilation system (i.e. fan with cooler) is partly qualified for SAM conditions and can be used as backup system in the case when containment spray is not available.

17.2.10 Containment overpressure protection (venting, spraying, etc.)

See previous point concerning spraying and long term heat removal form the containment. The special feature of VVER-440/V213 unit are 4 air-traps (4 volumes located on 4 floors in bubble condenser tower) where air and non-condensable gases are transported within the blow down phase of LOCA scenarios and consequently being stored here. Thus, a significant fraction of oxygen is separated from SG boxes and adjacent volumes (this decreases the risk of dangerous forms of hydrogen burning but supports the potential for oxygen starvation conditions in the later phase of hydrogen recombination). Air-traps are normally connected with the volume of bubble tower upstream the bubbler trays via special check-valves (note that shaft of this tower is connected with SGs boxes with 2 corridors).

There is a non-negligible risk of under-pressure occurrence within DBA scenario. To suppress such risk the Vacuum Breaker system has been installed. However, there are other important functions of the vacuum breaker system that can be used in managing severe accidents:

- to return of air from pressurised air traps into confinement volume upstream bubbler trays with the aim to enable recombination of residual hydrogen that has gathered there (prevention of oxygen starvation conditions that would decrease the efficiency of PARs);
- to prevent creation of pressure difference between adjacent air-traps, which could destroy their common wall (floor - ceiling). Such event could lead to damage of bubble tower and to loss of containment integrity.

17.2.11 Monitoring and management of combustible gases (mainly Hydrogen) in the containment

- Hydrogen concentration in the selected containment volumes is monitored through measurements.
- Hydrogen management in the containment consists of large capacity passive autocatalytic recombiners. These recombiners are of AREVA's design (28 x FR-1500T and 4 x FR-1500T at EBO, 27 x FR-1500T and 6 x FR-1500T at EMO1&2) and are located mainly in SG boxes and in selected associated compartments, as well as in the bubble tower shaft including connecting corridors.

17.3 SAM strategies, summarizing the measures being implemented in the Slovak NNPs include [SR12]:

- dedicated means for the primary circuit depressurization,
- hydrogen management using passive autocatalytic recombiners,
- containment underpressure protection,
- in-vessel corium retention via external reactor vessel cooling ,
- dedicated large external tanks with the boric acid solution with dedicated power source and pump aimed at possible fuel flooding and serving as a supplementary source of coolant for the reactor cavity flooding and for washing out the fission products from the containment atmosphere,
- spent fuel pool and external source tanks using mobile source connected to the external connection point on walls of the reactor building and auxiliary building,
- associated I&C needed for severe accident management.

17.4 References

[ENS12] "Summary report National Action ENSREG, Post Fukushima accident, Technical summary on the implementation of comprehensive risk and safety assessments of nuclear power plants in the European Unions" COM (2012) 571 Final.

[QSR13a] Replies to CESAM WP40 questionnaire by VUJE.

[QSR13b] Replies to CESAM WP40 questionnaire by IVS.

[SR11] "The Stress Tests for Nuclear Power Plants in Slovakia", Nuclear Regulatory Authority of the Slovak Republic, Slovak Republic, 30 December 2011.

[SR12] "National Action Plan of the Slovak Republic, Regarding actions to comply with the conclusions from the stress tests performed on nuclear power plants" Marta ZIAKOVA, NRASR (Nuclear Regulatory Authorities Slovak Republic) December 2012.

18 Annex 10: Severe Accident Management in Slovenia

The information provided in this annex is taken from the reference [SLO11], [SLO12] and [IAEA04]. Much more information can be found in this reference document [SLO11] which described in an extensive manner the Westinghouse Owner's Group (WOG) Accident Management Procedures/Guidelines. The strategies presented here reflect the situation end of 2011.
A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Slovenian NPP to provide guidance for further ASTEC development.

Table 29. Nuclear Power Plant in Slowenia.

NPP	First criticality	Power [MWth]	Reactor Type
Krsko	1981	1994 MW	Westinghouse 2-loop PWR

Slovenian nuclear requirements and safety assessment are based on WENRA, IAEA and applicable USNRC technical standards, regulations and guides. The SAMG have been adopted based on the generic guidelines provided by the vendor (Westinghouse). Slovenian SAM is in line with IAEA TECDOC-953. The SAMG were initially validated with the installation of the full scope simulator in 2000. After every modification, SAMG are updated and validated [SLO11].

18.1 Main types of Accident Management Procedures

Accident management and corrective measures, individual emergency response actions and the activities for maintaining emergency preparedness are dealt with in detail in different types of Krško NPP procedures.
The main sets of procedures dealing with accident and of interest in the frame of ASTEC development are:

- abnormal operating procedures (AOPs),
- emergency operating procedures (EOPs),
- severe accident management guidelines (SAMGs),

AOPs and EOPs are used by operators in the MCR to carry out operations actions on plant components and systems in case of abnormal or emergency operational conditions of the plant corresponding to design basis accidents (DBA) and beyond design basis accidents (BDBA) not involving core damage.
Engineered safety features (ESF) is the designation given to systems provided to protect the public and plant personnel by minimizing both the extent and the effects of any accidental release of radioactive fission products from the reactor coolant system, particularly those following a LOCA. These safety features function to localize, control, mitigate, and terminate such accidents and to hold the offsite environmental exposure levels within the limits.
The plant status evaluation team in TSC evaluates overall operational and safety status of the plant during an accident and supports the MCR crew as regards particular operations measures. In case of a severe accident when the EOP's are no more effective in preventing core damage the transition from EOP's to SAMG's is performed. Shift supervisor in the MCR makes a decision on the transition from EOPs to SAMG's based on transition criteria. The overall

objective of the SAMGs is to terminate the severe accident condition so that three primary goals associated with SAMG's are achieved:

- to return the core to a controlled stable status;
- to maintain or return the containment to a controlled stable status;
- to terminate any fission product releases from the plant.

The plant status evaluation team in the TSC evaluates SAMG's and recommends severe accident management strategies to the emergency director. The emergency director makes final decisions on the implementation of particular severe accident management strategies.

18.2 Emergency Operating Procedures (EOP)

Given the existence of the automatic ESFs, the EOPs, and well-trained licensed operators, the probability that any initiating event will lead to core damage is low. However Krško NPP does not consider it negligible, therefore a severe accident management program has been implemented in addition to general, abnormal and emergency operating procedures development. It includes both the development of plant-specific severe-accident management guidelines and training of personnel who would be tasked with managing a severe accident, should one ever occur.

According to the reactor plant design and operation NPP has developed procedures which are responding to any abnormalities of the particular system. MCR operating staff has been trained to respond by the plant condition by appropriate sets of procedures: Alarm Response Procedure (ARP), Abnormal Operating Procedures (AOP), Emergency Operating Procedures (EOP), Function Restoration Guidelines (FRG) and Fire Response Procedures (FRP).

Figure 6 shows operators response to abnormal situation by using different level of procedures.

Emergency Operating Procedures (EOP) are written so that trained operational shift crew will be able to identify an emergency from the symptoms available, take immediate actions on the expected course of the event, mitigate the consequences, and place the plant in a stable and safe condition. These procedures are entered always when reactor trip occurs or when it should occur. In the EOPs, the emphasis is on preventing core damage. The EOPs contain two types of procedure, whose use depends on whether the event can be diagnosed or not (Figure 6).

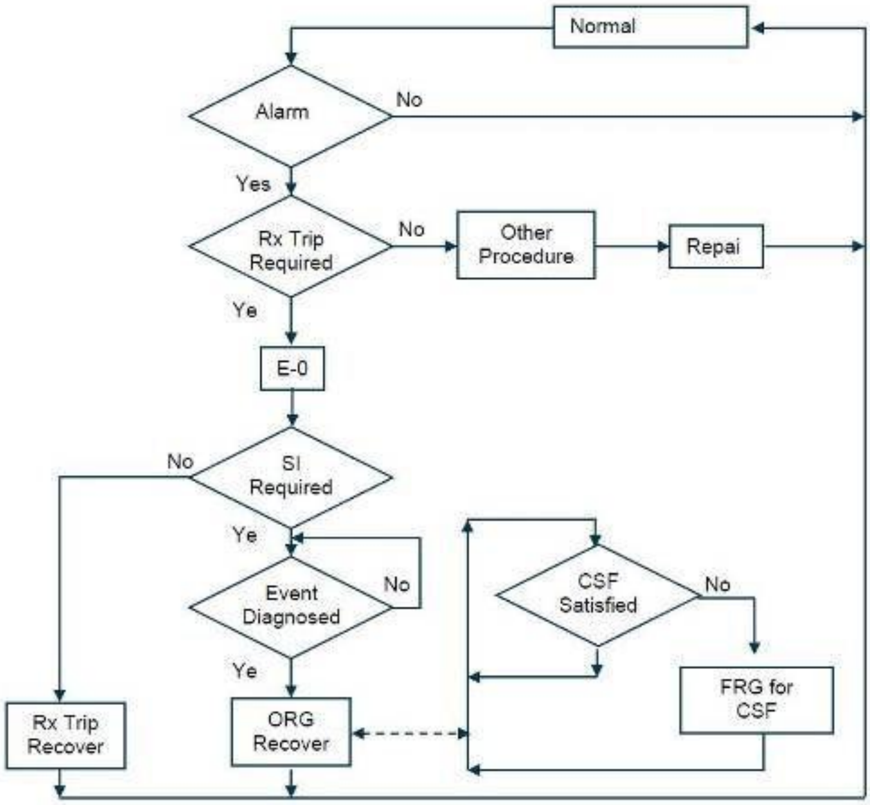

Figure 6. Operators response to abnormal situations (Slovenia).

Optimal Recovery Procedures deal with situations where diagnosis is possible, and they cover both design basis situations such as:

- Reactor trip with or without a safety injection,
- Loss of coolant (primary or secondary),
- Steam line break,
- Steam generator tube rupture

and also certain beyond design basis situations such as for example:

- Loss of all AC power
- Loss of primary coolant recirculation
- Uncontrolled depressurization of all SGs
- and numerous others.

For situations where diagnosis is not possible, Function Restoration Guidelines (FRG's) are provided. FRGs provide an explicit, systematic mechanism for evaluation and restoration of the plant safety state in terms of Critical Safety Functions (CSF) status As long as the fuel matrix/cladding, RCS pressure boundary and containment barrier are intact, the plant poses no threat to the health and safety of the public. CSFs, which are continuously monitored after entry to EOPs, if satisfied, are sufficient to maintain the fuel matrix/cladding, RCS pressure boundary and containment vessel barrier. CSFs in order of priority are as follows:

1. Subcriticality (minimizing energy production in the fuel),
2. Core cooling (providing adequate reactor coolant for heat removal from the fuel),
3. Heat sink (providing adequate secondary coolant for heat removal from the fuel),
4. RCS integrity (preventing failure of RCS),
5. Containment integrity (preventing failure of containment vessel),
6. Reactor coolant inventory (providing adequate inventory).

Relation between CSFs and barriers is shown on Figure 7.
It is important to note that the EOP package deals with preventive measures for all types of event: those within the design basis and also those beyond design basis.

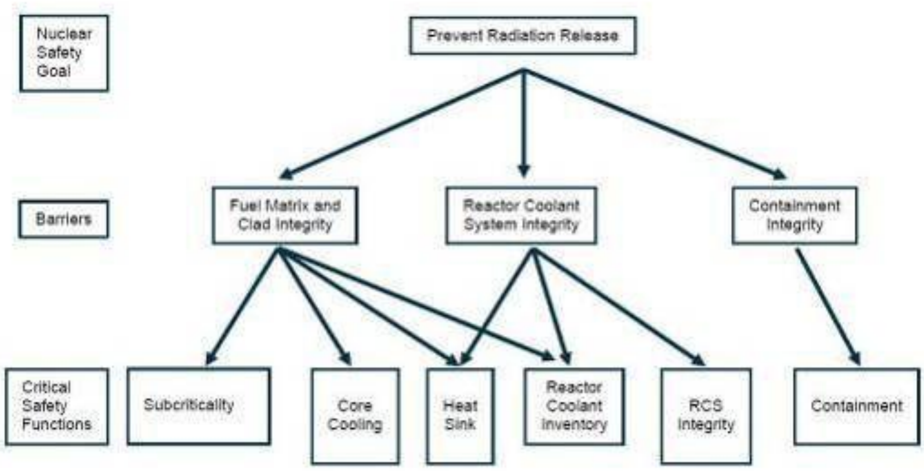

Figure 7. Relation between CSFs and barriers.

18.3 Severe Accident Management Guidelines (SAMG)

A combination of substantial equipment failure and consequently a lack of operators' actions is the most likely scenario that leads to the core damage event. For core damage to occur the core must be uncovered and remain uncovered long enough to overheat. Initially, core heat-up is driven by the decay heat. But once the cladding becomes hot enough, the heat released by the zirconium-steam reaction dominates and accelerates the core heat-up.
The various types of operating procedure and guideline and their interfaces, are shown in Figure 8.

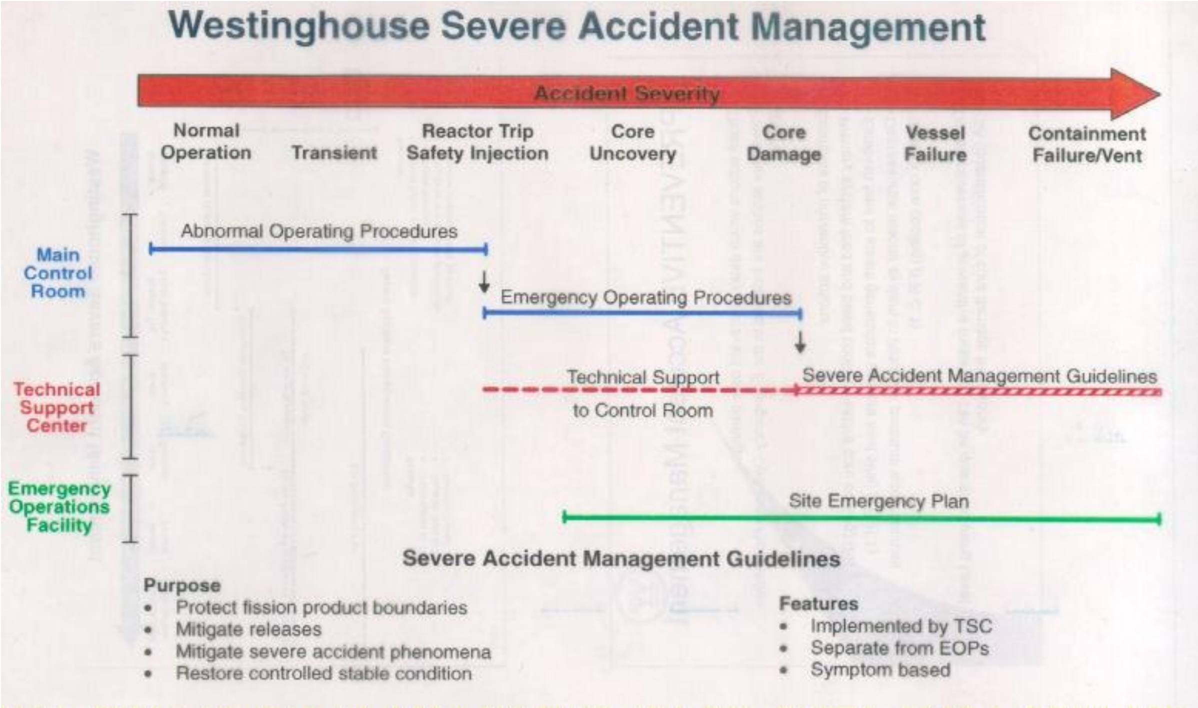

Figure 8. Operating procedures/guidelines and their interfaces.

The SAMG provide guidance for managing the in-plant aspects of an accident that progresses to core damage and in which the design bases of the plant are grossly violated. Because EOPs prioritize core damage prevention, they may not be appropriate after core damage. Thus, the EOPs are terminated when the transition from EOP to SAMG is made. Refer to Figure 9.
EOPs are rule-based procedures. A rule-based procedure requires that a very specific action would be taken for a given plant condition. It requires little or no evaluation of unintended consequences or negative impacts that might arise from taking the action. Most of the SAMG could not be developed as rule-based procedures. The SAMG include variables in the applicability and magnitude of positive and negative impacts associated with a given action. The SAMG require evaluation and decision-making processes to select proper actions for implementation. Therefore most of the SAMG consist of knowledge-based guidelines. A knowledge-based guideline does not mandate that a particular action is taken for a given condition. Rather, it identifies potential strategies for a given condition and leaves it up to the user to decide the best course of action under the circumstances. The consensus within the SAMG developer was that the plant evaluation team is best-suited for using knowledge-based guidance, which requires an essentially engineering approach.
In the EOPs, the emphasis is on preventing core damage. In the SAMG, the presumption is that core damage has already occurred. Therefore, when the transition from the EOPs to the SAMG is made, priorities shift – from preventing core damage to preserving the containment fission product barrier and arresting the progression of core damage.

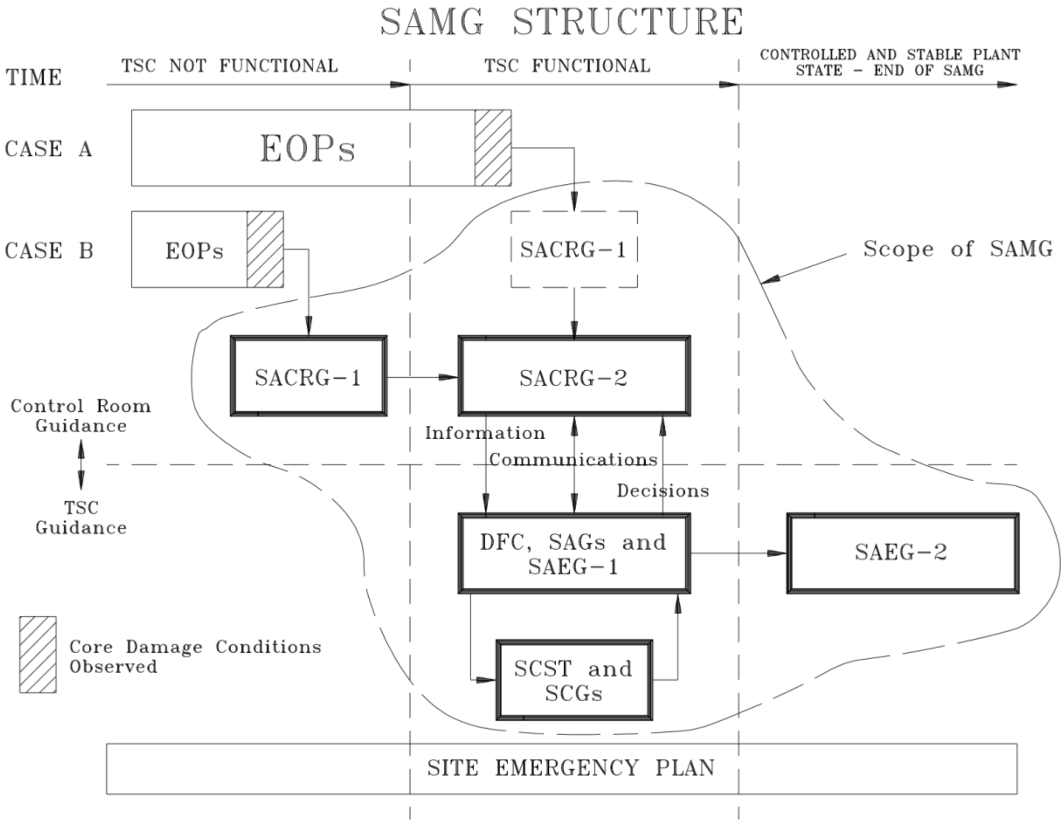

Figure 9. Graphic presentation of transition from EOPs to SAMGs (before and after TSC is functional) and SAMG structure.

There are three possible transfers from EOPs into SAMG:

- FR-C.1 Loss of cooling Reactor Core
- FR-S.1 No shutdown of reactor (ATWS)
- ECA-0.0 Loss of electricity supply – Blackout

These three transfers for emergency statuses into SAMG are sufficient and cover all possible severe accident scenarios.

The transition from these EOP procedures into SAMG being effective only when:

- the thermocouple reading of the core exit temperature exceeds a given value (650°C) and
- measures to restore cooling have failed,

In the event of core damage accident there are three major types of response actions:

- Control and termination of fission product releases
- Prevention of severe challenges to the containment fission product boundaries
- Recover core cooling

Secondary actions are to:

- Minimize fission product releases while achieving primary goals;
- Maximize equipment and monitoring capabilities while achieving the primary goals.

The Westinghouse Owners Group (WOG) SAMGs were developed to enhance the capability of the plant emergency response staff to:

- Diagnose the plant status during an accident which progresses to core damage,
- Perform a systematic and logical evaluation of possible severe accident strategies to choose the optimal strategy at any point in a severe accident and
- Evaluate the effectiveness of a severe accident strategy once it is implemented.

SAMG are primarily used for evaluators, not for implementers (operators). An evaluator is a member of the Plant Evaluation Team (PET) tasked with any of the following activities:

1. Diagnosing conditions that require entry into specific guidelines;
2. Assessing availability of equipment to perform the required strategy;
3. Evaluating the positive and negative impacts of strategies presented in certain guidelines;
4. Providing the recommended actions;
5. Interpreting the response of plant parameters following strategy implementation;
6. Assessing the effectiveness of implemented strategies and determining whether additional mitigation is needed;

18.3.1 SACRG-1

SACRG-1, »SAMG Control Room Guide – Initial Response« is the initial SAMG used by the control room staff. The control room staff enters SACRG-1 from the EOPs when conditions indicate that significant core damage is occurring. The control room staff is using SACRG-1 until the TSC is operational and the PET is ready to use SAMG.

Actions in SACRG-1 are:

1. Taking manual control of equipment to prevent automatic actuation of inactive equipment
2. Controlling hydrogen equipment .
3. Providing sufficient containment water inventory to allow ECCS recirculation capabilities and to mitigate possible consequences of vessel failure which could impact containment integrity in relatively short period (less than a day).
4. Controlling (depressurizing) RCS pressure to prevent high pressure vessel failure or steam generator tube failure by opening pressurizer PORVs.
5. Continuing attempts to restore core cooling. Following RCS depressurization, additional methods for injecting water may become available.
6. Controlling containment pressure to avoid hydrogen severe challenge conditions and addressing possible ignition sources. Containment should be maintained enough steam inert, but not over-pressurized.
7. Controlling steam generator water inventory; these actions are already implemented in EOPs.
8. Controlling and establishing effective containment and secondary system pressure boundaries.

The last step of SACRG-1 returns the Control Room to the point where the status of the TSC is checked again.

18.3.2 SACRG-2

A second control room guideline is used when the TSC is staffed and functional, and monitoring the plant status. It provides actions to respond to a severe accident in which the core may be damaged.

18.3.3 Diagnostic Flowchart (DFC)

On entry to the SAMG, the TSC begins immediately to monitor the DFC.

If a setpoint is exceeded in DFC, the TSC implements the corresponding Severe Accident Guideline (SAG), taking into account orders of priority.

A list of all Krško NPP SAGs is in Table 30. Prioritization of SAMG strategies is shown on Figure 10.

Table 30. List of Krško NPP TSC Severe Accident Guidelines (SAG).

TSC guidelines and associated DFC parameter		
Guideline	**Description**	**DFC parameter**
SAG-1	Inject into steam generators	Water level in all SGs
SAG-2	Depressurize RCS	RCS pressure
SAG-3	Inject into RCS	Core temperature
SAG-4	Inject into containment	Containment water level
SAG-5	Reduce fission product releases	Site releases and SFP temperature
SAG-6	Control containment conditions	Containment pressure
SAG-7	Reduce containment hydrogen	Containment hydrogen concentration
SAG-8	Flood containment	Containment water level

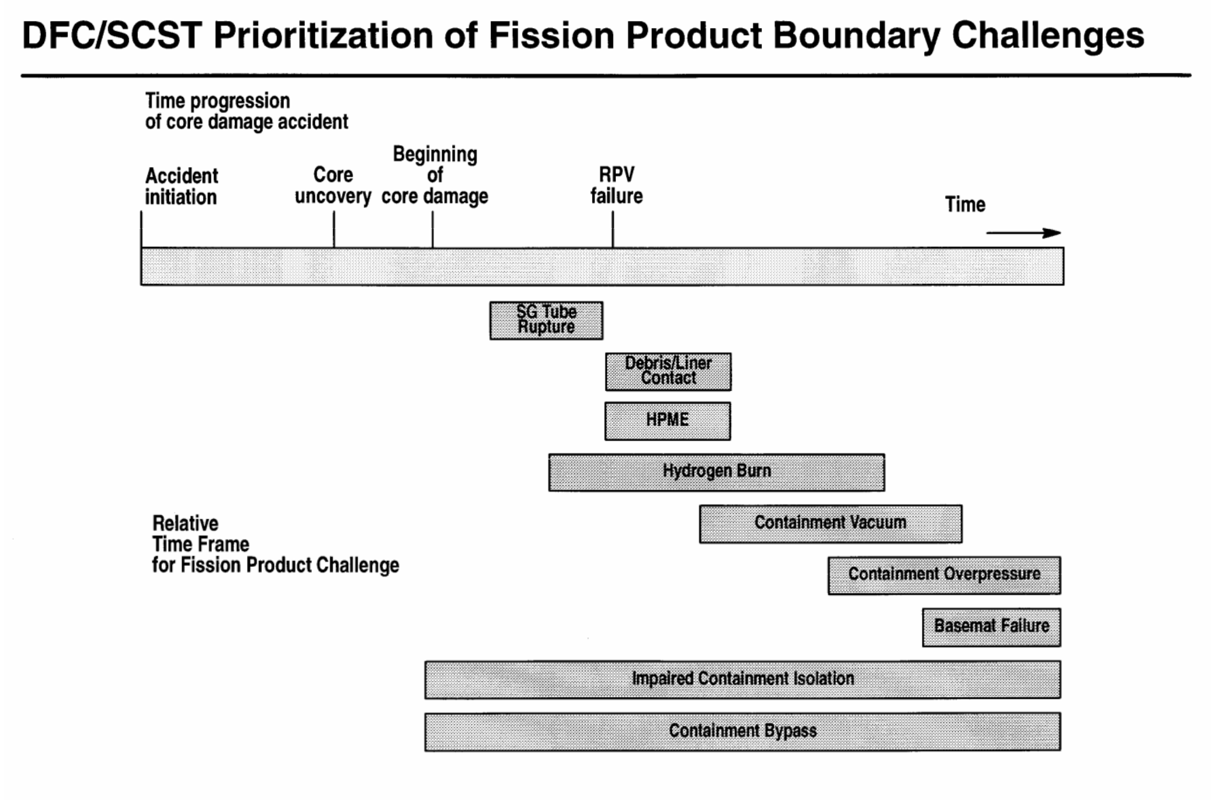

Figure 4: Diagnostic prioritization of SAMG strat gies based on prioritization

Figure 10. Diagnostic prioritization of SAMG strategies based on prioritization.

18.3.4 Inject into Steam Generators (SAG-1)

The purposes of injecting into the SGs are to protect the SG tubes from creep rupture (preventing breaching containment fission product barrier), to scrub fission products that enter the SGs via tube leakage and to provide a heat sink for the RCS.
There are many pumps that can inject water into the SGs, including two AC motor driven and one steam driven auxiliary feed pumps, main feed pumps, condensate pumps, service water pumps, AC firewater pump and one diesel driven firewater pump.
Normal suction sources from condensate storage tanks and service water are extended with additional sources as fire protection tank, water treatment tanks, pretreated water tanks, condenser hotwell, city water, circulating water tunnel, and the Sava river water.

18.3.5 Depressurize the RCS (SAG-2)

Lowering of RCS pressure during a severe accident is one of the top priorities of severe accident management to prevent a high pressure melt ejection, to prevent creep rupture of the SG tubes when the SGs are dry, to allow RCS makeup from low pressure injection sources and to maximize RCS makeup from any injection source.
There are several ways of depressurizing the RCS, including pressurizer relief valves, pressurizer spray, SG depressurization, and other RCS vent paths.

18.3.6 Inject into the RCS (SAG-3)

The purposes of injection into the RCS are: to remove stored energy from the core when it has been uncovered, to provide an ongoing decay heat removal mechanism, to prevent or delay vessel failure and to provide a water cover to scrub fission products released from the core debris.
There are at least three types of pumps that can be used to inject into the RCS during a severe accident; charging pumps, high head safety injection SI pumps, and low head (residual heat removal) SI pumps with suction from borated RWST and containment sump, boric acid tanks, reactor makeup water storage tanks, boron recycle holdup tanks, spent fuel pool and reactor makeup water storage tank.

18.3.7 Inject water into the containment (SAG-4)

The purposes of injection into the containment are: to prevent or mitigate the consequences associated with MCCI, to scrub fission products released from ex-vessel core debris and to allow ECCS recirculation.
Possible means of injecting water into the containment are: containment spray pumps, gravity feed from RWST to containment recirculation sump, ECCS and reactor coolant pressure boundary break as it is addressed in SAG-3 (Inject into the RCS) and portable fire protection pumps.
After vessel failure, basement melt-through can be prevented by ensuring that the core debris ex-vessel is water covered and cooled.
Krško NPP has wet reactor cavity, i.e. cavity is connected to containment sump with a 4-inch pipe. Consequently, flooding the containment will also flood reactor cavity. There are several methods available to inject water into containment:

- containment spray,
- RWST gravity drain,
- fire protection pipes for reactor coolant pumps,
- vacuum relief pipes

18.3.8 Reduce fission product releases (SAG-5)

The purpose of reducing fission product releases is to protect the health and safety of the public. Following the onset of core damage, fission products will be released from the cladding gap and possibly from the fuel matrix, and the fission products will be released either to the containment (through a break in the RCS or pressurizer relief or safety valves), to the SGs (through a tube leak or rupture), to the auxiliary building (through ECCS break located outside containment), or to the containment annulus.

Using the containment spray (CI) pumps or the reactor containment recirculation fan coolers (RCFC) can reduce fission product releases due to fission product scrubbing and containment pressure reduction.

Using the containment annulus negative pressure fans can reduce fission product release in the case that containment leaks to containment annulus (intermediate space between steel containment and concrete reactor building outside structure). They are effective as long as filters are active.

18.3.9 Control containment conditions (SAG-6)

The purposes of controlling containment conditions are to prevent a challenge to containment integrity due to high containment pressure, to prevent a challenge to containment penetration seals due to high containment temperature, to minimize the challenge on containment equipment and instrumentation due to a harsh containment environment, to reduce the airborne fission product concentrations, and to mitigate fission product leakage from containment. There are two generic containment heat sinks identified that are capable of depressurizing the containment to near ambient conditions following a severe accident: the containment spray system and containment recirculation fan cooler units.

18.3.10 Reduce containment Hydrogen (SAG-7)

Following core uncovery, the core would heat up and the fuel cladding would oxidize in the presence of steam. One of the products of the cladding oxidation reaction is hydrogen, which can accumulate in the RCS or in the containment if a venting pathway exists from the RCS. Following significant core damage, it is very likely that hydrogen concentration would reach 4% to 6% in the containment. Consequentially, the hydrogen can ignite and cause a spike in the containment pressure and temperature. The purpose of reducing containment hydrogen is to prevent hydrogen from accumulating to the point where the containment may be severely challenged. This is achieved by using one of two methods: intentionally igniting the hydrogen in containment, and using hydrogen recombiners.

18.3.11 Flood containment (SAG-8)

The purpose of flooding the containment is to establish cooling of the core material as a long-term strategy when other strategies have been ineffective. Specifically, if the vessel has failed, then the containment may need to be flooded to a level that ensures all core material remaining inside the vessel is covered with water. Three major benefits can be released by flooding containment to submerge the cover material remaining in the reactor vessel: any core material which remains in the vessel after reactor vessel failure would be cooled; water on the containment floor may quench the core debris following vessel failure preventing basement melt-through; fission products released from core debris on the containment floor would be scrubbed.

The amount of water necessary to flood the containment to the elevation of the top of active fuel, which would ensure in-vessel core debris cooling, is approximately 3 RWST volumes. The major impact that a flooded containment can have on the accident progression is described in severe accident guideline SAG-4.

18.4 Severe Challenge Guidelines (SCGs)

If SCG setpoint is exceeded in Severe Challenge Status Tree (SCST), the TSC stops monitoring the DFC (i.e., evaluating SAG) and refers to appropriate SCG. The difference between SCGs and SAGs is that immediate actions for mitigating severe consequences are implemented without any evaluation of negative impacts due to implementation of suggested strategies. List of SCGs is in Table 31.

Table 31. List of Krško NPP TSC Severe Challenge Guidelines (SCGs).

TSC SCGs and associated SCST parameters		
Guideline	**Description**	**SCST parameter**
SCG-1	Mitigate fission product releases	Site releases and SFP level
SCG-2	Depressurize containment	Containment pressure
SCG-3	Control hydrogen flammability	Containment hydrogen below severe challenge
SCG-4	Control containment vacuum	Containment pressure

18.4.1 Mitigate fission product releases (SCG-1)

The strategy for mitigating the fission product releases is called from the SCST. The purpose of reducing fission product releases is to protect the health and safety of the public. Actions in SCG-1 are identical to SAG-5, but they are implemented without any delay.

18.4.2 Depressurize containment (SCG-2)

The purpose of depressurizing the containment is to mitigate a severe challenge to the containment integrity due to high containment pressure. The consequence of not taking actions per this guideline will be containment failure leading to the uncontrolled release of high levels of fission products to the atmosphere. The only objective in SCG-2 is to reduce containment pressure far enough to mitigate a severe challenge. Guidance for reducing containment pressure to ambient conditions is then provided in SAG-6.

18.4.3 Control Hydrogen flammability (SCG-3)

The purpose of controlling hydrogen flammability is to mitigate a severe challenge to the containment integrity due to a hydrogen burn. The consequence of not taking actions per this guideline will be containment failure leading to the uncontrolled release of high levels of fission products to the atmosphere.

18.4.4 Control containment vacuum (SCG-4)

The purpose of controlling containment vacuum is to mitigate a severe challenge to the containment integrity due to the strong vacuum in containment. Containment pressure less than the lower containment pressure design basis can result in a severe challenge to the containment structure via buckling of the containment liner. A vacuum in containment may occur when noncondensable gases are released from the containment either via an unisolated leak that is subsequently isolated or from containment venting.

18.5 TSC long-term monitoring (SAEG-1)

This guideline provides information to the TSC to monitor the long-term concerns associated with the implementation of strategies contained in the SAGs and the SCGs. Specifically, the information contained in this guideline relates to:

- Actions which must be taken after a strategy is implemented to ensure that the strategy can be continued in the long term,
- Actions which must be taken to ensure that a function can be continued in the long term, for systems functioning prior to entry into the SAMG,
- The potential for primary recovery methods to become available after an alternative recovery method has been implemented.

18.6 SAMG termination (SAEG-2)

This guideline is used after the plant is declared to be in a controlled and stable state using TSC diagnostic flowchart (i.e., core temperatures, reactor vessel level, site releases, containment pressure, containment hydrogen, SFP temperature are below limits and in acceptable state). It provides information for the TSC that is important to supplement recovery actions after the use of the SAMG is discontinued. Specifically, the information contained in this guideline relates to:

- Plant conditions that may prohibit recovery actions,
- Special conditions for long-term monitoring as a result of strategies that have been implemented and are continuing after the time the SAMG is terminated,
- High radiation concerns that should be taken into account in the recovery actions.

18.7 Computational Aids (CA)

Computational aids are part of SAMG. They are typically charts that illustrate and simplify more complex relationships of different parameters, with the intention of allowing a quick assessment of the situation and thus proper action to be taken.
A useful example is a generic Computational Aid (CA) provided to assess injection into the reactor cooling system, to recover the core. Operational characteristics of charging/safety injection pumps, and PRZ relief valves, are combined into a diagram of injection flow rate over RCS pressure. Moreover, regions in the diagram are marked according to likely success of a given flowrate. The TSC will identify easily if the existing feed and bleed can be expected to be successful, and, if this is not the case, which pumps/valves would need to be operated for a successful refilling of the core.

18.8 References

[IAEA04] Implementation of Accident Management Programmes in Nuclear Power Plants, IAEA Safety Report Series No. 32, Vienna, 2004.

[SLO11] National Report for Slovenia on EU Nuclear Stress Tests, December 2011;

[SLO12] Slovenian Post-Fukushima National Action Plan, December 2012

19 Annex 11: Severe Accident Management in Spain

The information provided in this annex is taken from the reference [SPA11], [QSP13], [ENS12] and [SPA12]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [SPA11]. Therefore the strategies presented here reflect mainly the situation end of 2011.

A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Spanish NPPs to provide guidance for further ASTEC development.

Table 32. Nuclear Power Plant units in Spain.

NPP	Start of operation	Power [MWe]	Reactor Type
Garoña[1]	1971	466	BWR, General Electric
Cofrentes	1985	1092	BWR, General Electric
Almaraz 1	1981	1035	PWR, Westinghouse
Almaraz 2	1983	980	PWR, Westinghouse
Ascó 1	1984	1032.5	PWR, Westinghouse
Ascó 2	1986	1092	PWR, Westinghouse
Trillo 1	1988	1066	PWR, Siemens-KWU
Vandellós 2	1988	1087	PWR, Westinghouse

The implementation of the SAMGs at the Spanish plants of American design, both Pressurised Water Reactor (PWR) plants and Boiling Water Reactor (BWR) plants, followed a process that was parallel in time, such that all these plants have had SAMGs since the year 2001:

- PWR Westinghouse, Severe accident management guidelines (SAMGs) have been incorporated at this plant on the basis of the standards of the Westinghouse owners group (PWROG).
- BWR GE, Severe Accident Management Guidelines (SAMGs) have been incorporated at this plant on the basis of the standards of the BWROG.

This implementation was performed in accordance with the practices of the country of origin (USA), and then applying the criterion of using only equipment already available at the plants. The only Spanish plant of German design (Trillo NPP), the implementation of these procedures (Operating Manual and Severe Accidents Manual) was also performed in accordance with the practices of the country of origin of the technology. The scope of these manuals focuses more on reinforcing strategies to prevent core damage than on the mitigation of its consequences. Spanish NPPs have not developed severe accident management guidelines for all power conditions. In Spain the development of a more comprehensive and systematic set of SAMGs is

[1] Cessation of the production. Stopped Dec 2012, BOE 10th July 2013.

still on going for some plants. Trillo NPP: is developing the symptom-based SAMG for mitigation of the consequences of severe accidents and maintenance of containment integrity.

19.1 Main SAM strategies used

19.1.1 Management of AC/DC power

USNRC SBO rule implemented in Spanish NPPs.
PWRs (Westinghouse& Siemens) & BWR (GE): EOP for SBO including shedding of non-critical loads.
Procedure for recovery of AC/DC from new portable diesel generators to specific charges critical for accident management.

19.1.2 Reaching and maintaining reactor shutdown/sub-criticality

USNRC ATWS rule implemented in Spanish NPPs.
PWRs (Westinghouse): Sub-criticality is one of the Critical Safety Functions. FRG-S.1 (EOP for ATWS). Injection of borated water to the RCS included in the procedures.
PWR (Siemens): Sub-criticality is one of the Objectives of Protection. A guideline for the case of ATWS exists in the Operation Manual. Injection of borated water to the RCS included in the procedures.
BWR (GE): Procedures for power control in EOP. Injection of borated water to the RCS included in the procedures.

19.1.3 Management and recovery of the ultimate heat sink

PWRs (Westinghouse): EOP/SAMG packages. EDMG for: AFW Turbine-driven pump manual/local actuation (SG as a UHS); provide alternative access to the UHS by means of portable equipment and hoses.
PWR (Siemens): EOP/SAMG packages. EDMG for: Diesel-driven pump manual/local actuation (SG as a UHS); provide alternative access to the UHS by means of portable equipment.
BWR (GE): EOP/SAG packages (including Dedicated Containment Venting). EDMG for: RCIC Turbine-driven pump manual/local actuation (SG as a UHS); provide alternative access to the UHS by means of portable equipment.

19.1.4 Depressurization of the Steam Generators (SGs)

PWRs (Westinghouse): EOP/SAMG packages. EDMG for: Depressurization using relief valves by manual/local actuation
PWR (Siemens): EOP and SAMG packages. EDMG for: Depressurization using relief valves by manual/local actuation

19.1.5 Injection into (feed) the SGs

As it is defined in "Management and recovery of the ultimate heat sink".

19.1.6 Core cooling

New procedures for the core cooling using portable equipment and for recovering of the electrical power have been considered after Fukushima accident. (see "Management of AC/DC power")
PWRs (Westinghouse & Siemens): EOP/SAMG packages

BWR (GE): EOP/SAG packages. New procedure to avoid the use of containment venting for residual heat removal from the containment (taking into account that the core is being cooled by RCIC and safety-relief valves are open to the suppression pool).

19.1.7 Injection into the PRV /RCS /RCP seals

New procedures for the core cooling using portable equipment have been considered after Fukushima accident.
PWRs (Westinghouse & Siemens): EOP/SAMG packages
BWR (GE): EOP/SAG packages

19.1.8 Depressurization of the RCS

Enhancements of the use of safety/valves in total SBO conditions have been considered after Fukushima accident.
PWRs (Westinghouse): EOP/SAMG packages
PWR (Siemens): EOP and SAMG packages. RCS Bleed (through PRV) and Feed strategy will be implemented.
BWR (GE): EOP/SAG packages

19.1.9 Spraying within the RPV

BWR (GE): EOP/SAG packages

19.1.10 Molten Corium stabilisation

There is not specific guidance for this strategy. Capability to inject/spray the containment can provide water for molten corium in the ex-vessel phase.

19.1.11 Spraying into the containment

New procedures to implement strategies of containment spray using portable equipment (mobile pumps, hoses and fast connections) during total SBO conditions have been considered after Fukushima accident.
PWRs (Westinghouse): EOP/SAMG packages. EDMG for: Spray into the containment
PWR (Siemens): Not applicable (there is no Containment Spray System)
BWR (GE): EOP/SAG packages. EDMG for: Spray into the containment

19.1.12 Injection into the containment

New procedures to implement strategies of containment injection using portable equipment (mobile pumps, hoses and fast connections) during total SBO conditions have been considered after Fukushima accident.
PWRs (Westinghouse): EOP/SAMG packages. EDMG for: injection into the containment
PWR (Siemens): Containment injection not covered in current EOP. EDMG for: injection into the containment
BWR (GE): EOP/SAG packages. EDMG for: Injection into the containment

19.1.13 Containment heat removal (spray, fan coolers, etc.)

By applying the SAM strategies above:

- "Management and recovery of the ultimate heat sink"
- "Spraying into the containment"
- "Injection into the containment"

19.1.14 Containment overpressure protection (venting, spraying, etc.)

By applying the SAM strategies above:

- "Spraying into the containment"

Procedures for containment venting in EOP for pressure and hydrogen control. The NPPs are going to install a filtered vent as an additional improvement to protect the containment. The BWR plants already have a "hard" venting system.

19.1.15 Monitoring and management of combustible gases (mainly H2) in the containment

According to SAMG.
PWRs (Westinghouse): Passive Autocatalytic Recombiners (PAR's) will be installed in the near future in those areas of the containment that might present a risk of hydrogen accumulation.
PWR (Siemens) already has PARs.

19.1.16 Other SAMG strategies

<u>Reduction of source term (filtered venting, spray, pH control, etc.)</u>
Containment Venting: Spanish PWRs will implement a new Filtered Containment Venting System. Procedures will be issued.
New procedures for external spray of buildings (containment an SFP building), where releases may be produced, using new mobile equipment.

<u>Containment isolation</u>
New procedures (or enhancement of existing procedures) for guaranteeing the isolation containment during total SBO conditions have been considered after Fukushima accident.
PWRs (Westinghouse & Siemens): EOP/SAMG packages
BWR (GE): EOP/SAG packages

<u>Connection of mobile equipment</u>
Connection of mobile equipment is being done with the Extensive Damage Mitigation Guidelines.

<u>Spent fuel pool make-up</u>
Current strategies for SFP make-up are included in EOP and Abnormal Operating Procedures. Additional strategies for SFP make-up included in new procedures (EDMG) during total SBO conditions.

<u>Assessment of accuracy of the information provided by the instrumentation in severe accident conditions</u>
According to SAMG that include, in some cases, provisions to confirm instrumentation

19.2 References

[ENS12] "Summary report National Action ENSREG, Post Fukushima accident. Technical summary on the implementation of comprehensive risk and safety assessments of nuclear power plants in the European Unions" COM (2012) 571 Final.

[QSP13] Replies to CESAM WP40 questionnaire by CIEMAT.

[SPA11] "Stress tests carried out by the Spanish nuclear power plant, Final report" Consejo de Seguridad Nuclear, Spain, 31 December 2011.

[SPA12] "Post-Fukushima European Action Plan, Spain National Plan" Consejo de Seguridad Nuclear, Spain, December 2012.

20 Annex 12: Severe Accident Management in Sweden

The information provided in this annex is taken from the reference [SWE11]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [SWE11]. Therefore the strategies presented here reflect mainly the situation end of 2011.

A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Swedish NPPs to provide guidance for further ASTEC development.

There are 10 nuclear power reactors in operation in Sweden; seven BWR and three PWRs. All the BWRs were designed by the domestic vendor ASEA-ATOM (later ABB Atom, now Westinghouse Electric Sweden AB) and all the PWRs by Westinghouse (USA). The three oldest BWRs have external main recirculation loops while the other four units have internal recirculation pumps with no large pipes connected to the reactor pressure vessel below core level. The BWR containments are all of the PS-type and various layouts of the vent pipe configuration and pressure suppression pools. All PWRs are 3-loop standard Westinghouse design reactors.

Table 33. Swedish nuclear power plants.

Reactor	Operator	Type	MWe net	Commercial operation	Intended decommissioning
Oskarshamn 1	OKG	BWR	473	1972	2022?
Oskarshamn 2	OKG	BWR	638	1974	2034
Oskarshamn 3	OKG	BWR	1400	1985	2035
Ringhals 1	Vattenfall	BWR	859	1976	2026
Ringhals 2	Vattenfall	PWR	866	1975	2025
Ringhals 3	Vattenfall	PWR	1045	1981	2041
Ringhals 4	Vattenfall	PWR	950	1983	2043
Forsmark 1	Vattenfall	BWR	987	1980	2040
Forsmark 2	Vattenfall	BWR	1000	1981	2041
Forsmark 3	Vattenfall	BWR	1170	1985	2045
Total (10)			9388		

Besides the operating NPPs, the Barsebäck nuclear power plant is situated in the south of Sweden 30 km from Malmö and 20 km from Copenhagen whose two 600 MWe BWR units commenced operations in 1975 and 1977, and were shut down in November 1999 and May 2005.

20.1 Generic Severe Accident Management strategies

All of the currently operating plants in operation have chosen the Multi-Venturi Scrubber System (MVSS) concept to fulfill the requirements of filtered venting, and a conceptual illustration of the overall severe accident mitigation concept for the BWRs and PWRs is presented in Figure 11 and Figure 12, respectively. The major component is the scrubber system comprising a large number of small venturi scrubbers submerged in a pool of water. The water contains chemicals for adequate retention of iodine. A venture scrubber is a gas cleaning device that relies on the passage of the gas through a fine mist of water droplets. The design of the venturis is based upon the suppliers' broad experience in this area, gained when designing venturis for cleaning polluted gases from various industrial plants.

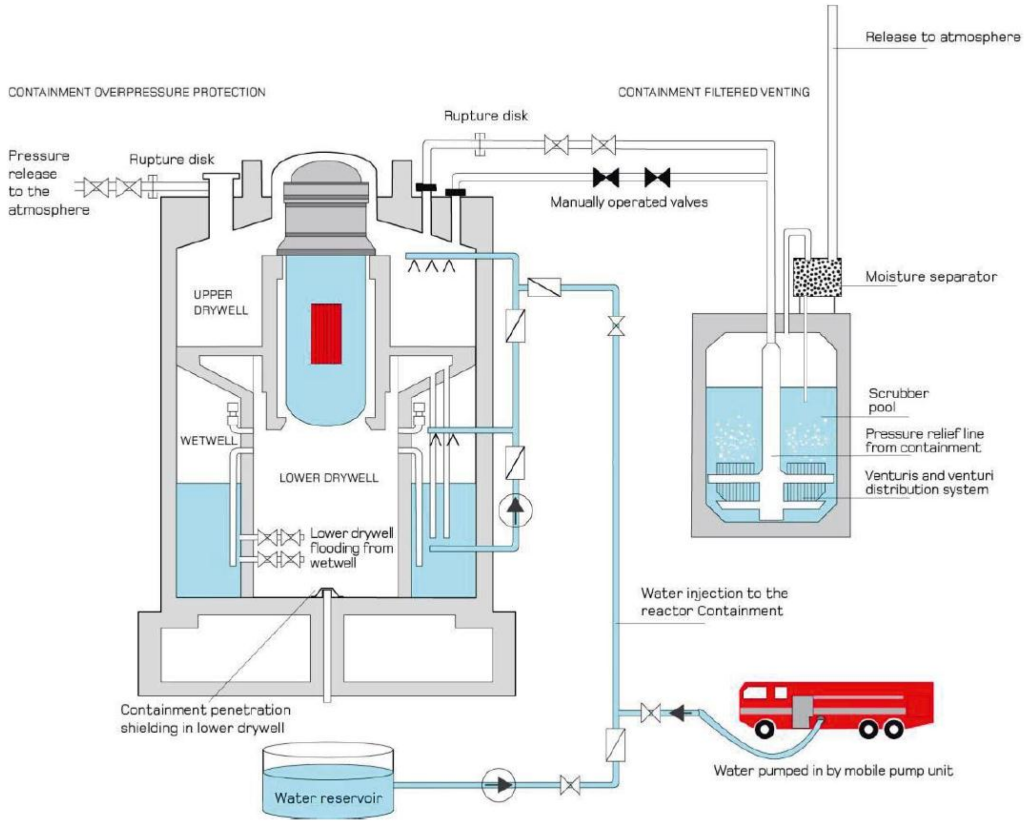

Figure 11. Overall severe accident mitigation concept for Swedish BWRs.

The MVSS can be activated automatically, via a rupture disc, or manually. There are two separate venting lines from the containment for these two modes of operations. The venting line with the rupture disc is always open so that no operator actions are needed to vent this way. The design principle of the system is the same for BWRs and PWRs. The system is made inert to avoid hydrogen combustion.

The Swedish strategy for dealing with a core melt in BWRs is to let the core debris fall into a large volume of water in the lower regions of the containment. This is a quite uncommon approach and only a few reactors in the world apply this strategy.

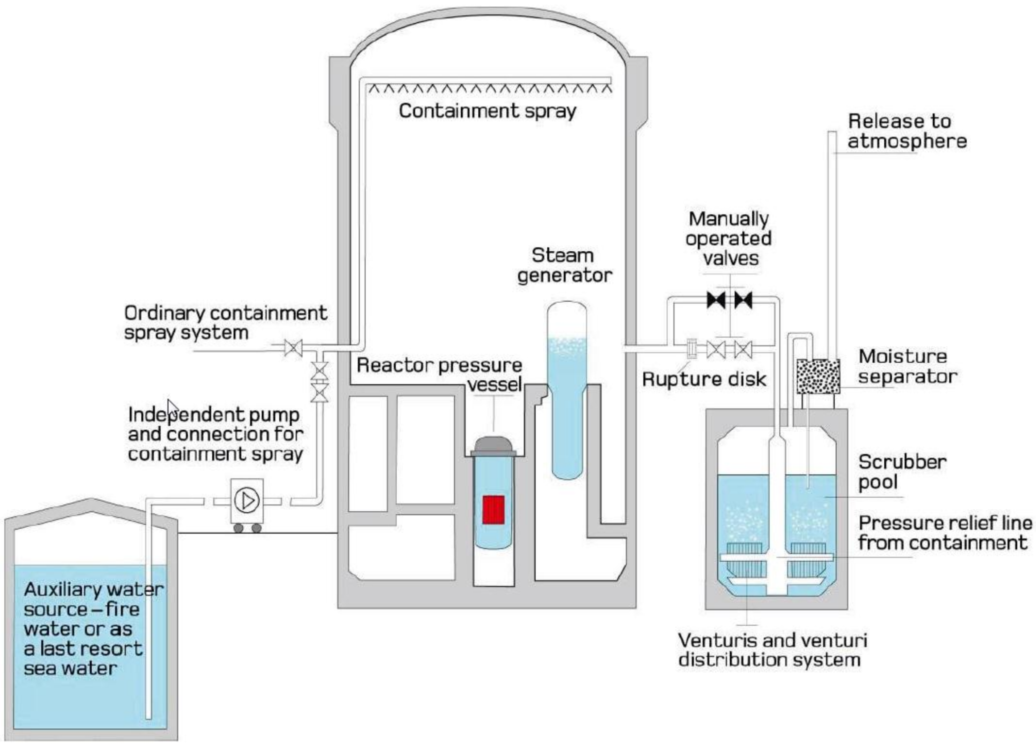

Figure 12. Overall severe accident mitigation concept for Swedish PWRs.

20.2 Forsmark NPP

In a scenario with loss of the core cooling function it should be noted that some of the strategies described can only be carried out if electrical power (AC) is available, while other measures are entirely passive or can be carried out when a power system with battery back-up (DC) or compressed air supply is available.

20.2.1 Before occurrence of fuel damage

In the event of station blackout without power recovery there is no possibility to restore the core cooling function. The strategy is then to mitigate the consequences of an accident resulting in fuel damage and reactor vessel melt-through, by protecting the containment and minimizing the release of radioactive substances to the surroundings

20.2.2 After occurrence of fuel damage

In the case power supply is available, it is always prioritized to try to start different systems for injecting water to the reactor.

20.2.3 After failure of the reactor pressure vessel

The priority is maintaining the integrity of the containment in order to minimize the consequences in the form of release of radioactivity to the surroundings. The chemistry in the containment is important for minimizing corrosion and retaining iodine in the sump in the bottom of the containment. It is recommended to achieve a pH of approx. 10 within a week of the start of the accident.

20.2.4 Fuel damage in high pressure scenario

In order to prevent high-pressure melt-through of the reactor vessel in case of fuel damage in high pressure scenario, the reactor water discharge valves are used, opening when the level in the reactor vessel reaches extremely low level (L4). At level L4, the core is still covered with water. The depressurization function is reliant on a battery back-up system but not on any other electrical power supply.

20.2.5 Management of hydrogen risks inside the containment

This is easy in the short term since the containment is filled with nitrogen gas and no hydrogen combustion can occur.
The containment pressure can be decreased by following systems:

- Containment filtered venting system
- Containment spray system
- Containment over-pressurization protection system and

The filtered pressure relief system consists of a water-filled scrubber connected to the upper drywell of the containment. If the pressure in the containment exceeds 5.5/6.0 bar (Forsmark 1 and 2/Forsmark 3), a rupture disc will burst and part of the atmosphere in the containment will be released via the scrubber.

20.2.6 Basemat melt through

The strategy for preventing basemat melt through is to fill the space below the reactor vessel with water. The space below the reactor vessel is automatically filled with water 30 minutes after isolation of the containment.

20.2.7 Lost containment integrity

After loss of containment integrity there are no accident management measures.

20.2.8 Spent fuel pools

The storage facilities for Forsmark 1-3 consist of three spent fuel pools filled with water: two fuel pools and a cask pool. All the pools are water filled during normal operation. The temperature of the fuel pool is monitored by temperature gauges. Strategies for cases with uncovered fuel are not available at present.

20.3 Oskarshamn NPP

In a scenario with loss of the core cooling function, the accident strategy focuses on maintaining control of the reactor's reactivity and on restoring the electrical power supply in order to be able to utilize the plant's various water and cooling systems.

20.3.1 Before occurrence of fuel damage

Systems that are not normally used for water injection to the reactor pressure vessel can be used. However, even these systems require electrical power supply to be available. This also includes depressurization of the reactor pressure vessel to achieve injection with low pressure core cooling system, as well as measures to prevent high pressure core melt in a later stage. Both depressurizations can be performed with only battery power available. The batteries have a capacity of two hours at nominal output.

20.3.2 After occurrence of fuel damage

The time available before the reactor pressure vessel fails is, in the worst case, estimated to be about four hours. Filling of the lower drywell by opening of the valves from the condensation pool is part of the preparations for the melting through of the RPV. This measure means that the area under the RPV is filled with water and that the molten metal ends up in the water at a later stage. This action is performed automatically when the level is less than 0.5 m above the core for 10 min and if this action is not performed automatically it will be performed manually. Spraying of the drywell has the goal of keeping the pressure and temperature at low levels, and to wash out any radioactive particles and aerosols.

20.3.3 After failure of the reactor pressure vessel

The accident strategy primarily focuses on the mitigation of consequences on the environment. All isolation valves are closed and the containment is filled with water up to a level corresponding to the top of the core; this is expected to start eight hours after the initiating event. Filling of the containment is accomplished by spraying into drywell using water injection that is not dependent on the availability of electrical power. Residual heat removal through the scrubber is established and the water level in the scrubber is adjusted. This is performed using mobile generator as electrical power supply.

20.3.4 Fuel damage in high pressure scenario

The pressure is reduced inside the reactor pressure vessel by motor-operated pressure relief valves that open automatically at low level by using battery-supplied AC power. The batteries have a capacity of two hours at nominal output, but may probably be utilised after this period of time. These motor-operated pressure relief valves are dedicated to this task and will remain open after they have opened. Alternatively, the safety relief valves and automatic depressurization valves can be opened, the governing valves of which require battery supplied DC power.

20.3.5 Management of hydrogen risks inside the containment

Is normally achieved by filling the containment with nitrogen. Anyway, should a severe accident occur the permanent recombiners will start acting 90 minutes after the containment isolation is activated. Should a situation arise where the permanently installed recombiners are not able to keep down the oxygen level, a portable recombiner is available. It is connected to the containment atmosphere through connections outside the containment.
The containment pressure can be decreased by following systems:

- Containment filtered venting system.
- Containment over-pressurization protection system
- Containment spray system

At the beginning of an accident when the containment pressure increase is due to the escaping steam-water mixture at high pressure and temperature the containment venting can be direct to the environment. The rupture disc to the environment bursts at 0.6 MPa for Oskarshamn 1 and at 0.65 MPa for Oskarshamn 2-3. After depressurization directly to the atmosphere is completed, valves (battery backed power supplied) automatically will close 20 minutes after containment isolation is activated. Spraying with water can then start in the containment in order to wash out the containment atmosphere, which causes a simultaneous pressure reduction by condensation of steam.

20.3.6 Basemat melt through

Due to different configuration of the containments in Oskarshamn 1-3, the strategies for prevention of basemat melt through may differ in the accident sequence:

- For Oskarshamn 1, the space under the reactor pressure vessel is equipped with a penetration which melts when it is hit by the melt from the reactor pressure vessel. This entails that the mixture of metal and fuel material which continued down into the water pool beneath and split into minor parts. A heat sink is created for the reactor pressure vessel by establishing cooling of the condensation pool through the activation of an electrical power supply independent system, which sprays and cools the containment and thereby also the condensation pool.
- For Oskarshamn 2, the condensation pool has a volume of 2000 m3 and with an area of 322 m2 entails that the melt is cooled and can remain cooled at the bottom of the reactor containment. When melting through the reactor pressure vessel, the melt will dwell for a short time in the control rod drive pit before it finally flows downwards and is split into minor parts in wetwell. No preparations are required to secure this course of accident sequence. Only continued surveillance of the level and temperature of the heat sink is performed during the course of the sequence.
- For Oskarshamn 3, in an accident situation where isolation of the containment has occurred and the level in the reactor pressure vessel is very low, an automatically initiated filling occurs in the space below the reactor pressure vessel (lower drywell). The filling occurs through the opening of valves from the condensation pool. This entails that the space beneath the RPV becomes water-filled and that the melt at a later stage will end up in water and there be cooled by thermal circulation. The filling is initiated when both low levels L3 and L4 are present for greater than 10 minutes. Initiation of this function requires battery-backed power.

20.3.7 Lost containment integrity

In the event that the containment integrity can no longer be maintained, the reactor building is the next barrier to the environment. A continued sprinkling of the containment atmosphere by the power supply independent system holds down the amount of airborne activity that could leak out. Depending on how much leakage occurs from the reactor containment, an isolation and emergency filter ventilation of the reactor building through carbon filters will occur to prevent or limit the release of activity into the environment.

20.3.8 Spent fuel pool

A calculation of the available time in the worst scenario, i.e., shortly after the start of an outage shows that there are about 18 hours for Oskarshamn 1, 40 hours for Oskarshamn 2 and 21 hours for Oskarshamn 3 available before the pool temperature reaches 100°C where action should be taken. Measures may also be taken after 100°C, but then the working environment has deteriorated due to the boiling in the pools and breathing protection, which is kept near the main control room, may be required. If also the duration with adequate radiation protection is taken into account (2 m water above top of fuel), about 91 hours for Oskarshamn 1, 6 days for Oskarshamn 2 and 74 hours for Oskarshamn 3 are available to take measures.

20.4 Ringhals NPP

For Ringhals 1, the first priority in a scenario with loss of the core cooling function is to prevent fuel damage. If fuel damage has occurred the strategies are shifted towards mitigating the consequences to the public which means protecting containment integrity.
For Ringhals 2- 4, the initial accident management measures in a scenario with loss of the core cooling function are focused on restoring cooling function guided by event based Westinghouse ORG, the so called "E-procedures". In case critical safety functions are challenged a transition will be made to the F-procedures - Function restorating - which are more symptom-oriented. In case the temperature measured by CET (Core Exit Thermocouple) exceeds a certain value a transition will be made to the SAMG (Severe Accident Management Guidance) where priorities are changed from restoring core cooling to protect the fission product boundaries.

20.4.1 Before occurrence of fuel damage

For Ringhals 1 the actions before occurrence of fuel damage are as follows:

1. prevent criticality with control rods or boron injection,
2. containment isolation,
3. Reactor Pressure Vessel (RPV) pressure relief,
4. RPV injection
5. maintain/restore heat sink (condensation pool).

Last resort to prevent fuel damage will be to inject water into the RPV using the plants existing safety systems. The possibility for fuel damage in high pressure (which might lead to RPV failure at high pressure) is avoided by pressure relief of the RPV.
For Ringhals 2-4 the actions before occurrence of fuel damage are as follows:

1. maintaining level in steam-generators,
2. pressure relief in the primary system and
3. restoration of injection into the primary system.

20.4.2 After occurrence of fuel damage

For Ringhals 1 the actions after occurrence of fuel damage are as follows:

1. RPV pressure relief,
2. RPV injection and
3. maintain/restore heat sink (condensation pool).

For Ringhals 2-4 the actions after occurrence of fuel damage are as follows:

1. maintaining level in steam-generators,
2. pressure relief in the primary system,
3. restoration of injection into the primary system and
4. filling containment with water up to lower part of RPV.

20.4.3 After failure of the reactor pressure vessel

For Ringhals 1 the actions after failure of the reactor pressure vessel are as follows:

1. ensure that the ex-vessel corium is covered by water,
2. prevent over-pressurization of containment and
3. fill the containment up to lower part of RPV.

For Ringhals 2-4 the actions after failure of the reactor pressure vessel are as follows:

1. ensure that the ex-vessel core is covered by water,
2. prevent over-pressurization of containment and c) fill the containment with water up to lower part of RPV.

20.4.4 Fuel damage in high pressure scenario

In Ringhals 1 the possibility for fuel damage in high pressure scenario which might lead to RPV failure at high pressure is avoided by pressure relief of the RPV. The actions after occurrence of fuel damage are

1. RPV pressure relief,
2. RPV injection and
3. maintain/restore heat sink (condensation pool).

In Ringhals 2-4 the actions after occurrence of fuel damage in high pressure scenario are:

1. maintaining level in steam-generators,
2. pressure relief in the primary system,
3. restoration of injection into the primary system and
4. filling containment with water up to lower part of RPV.

20.4.5 Management of hydrogen risks inside the containment

For Ringhals 1, the SAM measures which will be performed for the management of hydrogen risks inside the containment after occurrence of fuel damage are as follows:

1. H2-deflagration/detonation is prevented by keeping the containment atmosphere inerted with nitrogen,
2. O2-generation from a large LOCA may be recombined with “Mobile Recombiner” and
3. the containment atmosphere may be adjusted through further injection of nitrogen from “Nitrogen storage for ventilation of containment”.

During outage as well as approximately 12 hours during start-up or shutdown the containment atmosphere is not inerted. There are no specific measures to handle hydrogen risks during these operations. There is no prepared system or strategy fully qualified for recombining O2-generation from severe accidents. A recently performed evaluation has confirmed that no actions are required within at least 7 days.
For Ringhals 2-4 the SAM measures which will be performed for protecting the containment function after occurrence of fuel damage is that H2-deflagration/detonation is prevented through PAR (Passive Autocatalytic hydrogen Recombiners) which even for very large core degradations will within some hours reduce the hydrogen concentration to safe levels through recombination of hydrogen and oxygen.

The containment pressure can be decreased by following system:

- Containment filtered venting system (BWR and PWR)
- Containment over-pressurization protection system (only BWR)
- Containment spray system (BWR and PWR).

20.4.6 Over-pressurization of the containment

Over-pressurization of the containment in Ringhals 1 (BWR) is normally prevented through the containment spray system. If this is not possible, containment overpressure may be reduced through filtered venting (CFV) which also will reduce pressure from non-condensable gasses, e.g. hydrogen. In addition, pressure can be reduced by activation of independent spray. CFV is inerted and can relieve a mixture of steam, hydrogen and nitrogen without dangerous hydrogen deflagrations.
Over-pressurization of the containments in Ringhals 2-4 (PWR) is normally prevented through spray. If this is not possible, containment over-pressure may be reduced through filtered vent (CFV). In addition, pressure will be reduced by activation of independent spray.

20.4.7 Basemat melt through

For Ringhals 1, the basemat melt through is prevented through maintaining a sufficiently high water level in wet-well. Since the wet-well extends over the whole area below the RPV and is initially filled with several meters of water it is unlikely that wet-well will be depleted. The combination of an initial big wet-well water volume and the possibility to maintain the wetwell inventory through independent spray is supposed to give a low probability for basemat failure. For Ringhals 2-4, the basemat melt through is prevented through maintaining a sufficiently high water level in the containment.

20.4.8 Spent fuel pool

For spent fuel pools (SFP), "adequate shielding" is considered to be more than 2 m water above the top of fuel. Several strategies have been identified as possible "before losing adequate shielding", whereas very few strategies have been identified as possible "after losing adequate shielding". "Uncover of the top of fuel" is considered to be a further development of "losing adequate shielding" and the applicable SAM measures for cooling are the same. "Occurrence of fuel degradation (fast cladding oxidation with hydrogen production)" is considered to be an even further development of "losing adequate shielding" and the applicable SAM measures for cooling are the same. There are no prepared SAM measures in place to cope with the specific effects of cladding oxidation and hydrogen production in the SFP. The time to boiling, loss of shielding and uncovery are very much dependent on whether the plant is in normal operation with limited amounts of fuel in the storage pool or in outage for refueling. During outage, adequate shielding would be lost within 2-3 days, while the times are extended to one week or more for a plant in normal operation.

20.5 References

[SWE11] Swedish action plan for nuclear power plants. Response to ENSREG's request. December 2012.

21 Annex 13: Severe Accident Management in Switzerland

The information provided in this annex is taken from the reference [SUI11]. Many more information regarding systems to cope with severe accident can be found especially in this reference document [SUI11]. Therefore the strategies presented here reflect mainly the situation end of 2011.

A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Swiss NPPs to provide guidance for further ASTEC development.

Table 34. Swiss nuclear power plants.

	KKB 1	KKB 2	KKG	KKL	KKM
Thermal power [MW]	1130	1130	3002	3600	1097
Gross electrical output [MW]	380	380	1035	1220	390
Net electrical output [MW]	365	365	985	1165	373
Reactor type	PWR	PWR	PWR	BWR	BWR
Reactor supplier	W	W	KWU	GE	GE
Turbine supplier	BBC	BBC	KWU	BBC	BBC
Generator data [MVA]	2 x 228	2 x 228	1140	1318	2 x 214
Main heat sink	River water	River water	Cooling tower	Cooling tower	River water
Commercial operation started in	1969	1971	1979	1984	1972
Spent fuel pools (SFP)	2 SFPs in separate building	2 SFPs in separate building	1 SFP in P-containment, 1 SF loading pond in S-containment	1 SFP in P-containment, 1 SFP in separate building	1 SFP in S-containment
Interim waste storage facility	Internal interim storage facility (air-cooled)	Internal interim storage facility (air-cooled)	External wet storage facility	External interim storage facility	Internal interim storage facility
Holder of operating licence	Axpo AG	Axpo AG	Kernkraftwerk Gösgen-Däniken AG	Kernkraftwerk Leibstadt AG	BKW FMB Energie AG
Number of reactor cooling loops	2	2	3	-	-
Containment type	Full pressure containment	Full pressure containment	Full pressure containment	Mark III containment with venting system	Mark I containment with venting system

There are a total of five nuclear power plant units at four different sites in Switzerland. These are:

- Beznau nuclear power plant (KKB) with two units
- Gösgen (KKG) with one unit
- Leibstadt (KKL) with one unit
- Mühleberg (KKM) with one unit

Three of the Swiss nuclear reactors are pressurised water reactors (PWR); two of these are of American design and one is of German design. The other two Swiss nuclear reactors are American boiling water reactors (BWR) of different generations. The key technical data for the nuclear power plants are shown in Table 34.

To allow a clear presentation of the safety systems present in the Swiss nuclear power plants they have been subdivided into three "Safety trains" by which the plants can be brought into a safe shutdown state in case of accidents. The safety trains can be used in all plant operational states (full power operation as well as low power and shutdown states). The characteristics of the three safety trains are described below:

- Safety train 1: This consists of the conventional safety systems which are used to control accidents due to internal events (such as loss of coolant accidents (LOCAs), internal flooding) and, depending on the original design concept of the nuclear power plant, external events related to natural causes (such as earthquakes and external flooding).
- Safety train 2: The special emergency systems constitute another safety train which is primarily intended to control accidents due to external events, but which also provides further protection in addition to the conventional safety systems in the case of internal events.
- Safety train 3: The preventive accident management measures implemented in all nuclear power plants constitute the third safety train. This train consists exclusively of manual measures that are to be implemented locally by operating staff; they are stipulated in specific emergency procedures, are ordered by the emergency staff and are carried out with the deployment of either permanent built-in or mobile equipment.

Table 35. Safety functions implemented in Swiss NPPs.

Fundamental safety functions	Safety functions	Plants											
		KKB			KKG			KKL			KKM		
		1	2	3	1	2	3	1	2	3	1	2	3
Control of reactivity	Shutdown of reactor and ensured sub-criticality	X	X	X	X	X	X	X	X	X	X	X	X
Cooling of fuel assemblies	Removal of decay heat from the reactor and reduction of pressure	X	X	X	X	X	X	X	X	X	X	X	X
	Removal of decay heat from the SFP and wet storage facility	X	-	X	X	X	X	X	-	X	X	-	X
	Removal of decay heat from the primary containment	X	X	X	X	-	X	X	X	X	X	X	X
Containment of radioactive substances	Isolation of primary containment	X	-	X	X	-	X	X	-	-	X	-	-
	Protection of primary containment integrity by:												
	Venting of the primary containment	-	-	X	-	-	X	-	-	X	-	-	X
	Prevention of deflagration and explosion of hydrogen	-	-	X	-	-	X	-	-	X	-	-	X

21.1 Limiting radioactive releases

The maximum possible retention of radioactivity in the plant in the event of core damage is one of the central objectives of the SAMG. Elements of SAMG include, but are not limited to, the deployment strategy and the specific procedure if it becomes necessary to vent the containment with the Filtered Containment Venting System, FCVS. The FCVS in place in the nuclear power plants is conceived for accidents that exceed the design basis. It ensures that the primary containment does not fail due to excessive internal pressure, with the uncontrolled release of radioactive substances. A wet scrubber is integrated into the release train.

21.2 Accident management measures in place at the various stages of a scenario of loss of the core cooling function

Measures as part of Severe Accident Management (SAM) after occurrence of fuel damage are established and incorporated into the procedures (Severe Accident Management Guidance, SAMG) at all the nuclear power plants. Both the PWR plants (KKB, KKG) use SAMG entry criteria geared to the core exit temperature (CET). At KKG, additional provision is made for two entry paths based on an increased hydrogen concentration or dose rate in the containment.

The measures ordered in the SAMG include, but are not limited to, filtered venting of the containment before or after an RPV failure. In each Swiss plant, provision is also made for a supply to the containment – preferably before, but also after the RPV failure. In the case of the BWRs, this is specifically intended to ensure that the core melt falls into a pool of water in case of an RPV failure. At KKB, on the other hand, this measure is intended for external cooling of

the RPV, so as to prevent it from failing. If coolant is lost into the containment, the measures to supply the RPV would also contribute to the supply for the containment.
The accident management measures after failure of the RPV are largely identical to those prior to such a failure. The differences relate to the priorities and objectives. After an RPV failure, for example, the supply of water to the suppression pool (SP) of the containment at KKL would have a lower priority than the supply of water for cooling the residual melt remaining in the core. After an RPV failure and the beginning of damage to the containment foundation, flooding of the containment (via injection into the reactor coolant system) is used at KKG to decelerate the interaction between melt and concrete and to minimise discharges if the core melt reaches the containment sump area.

21.3 Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

21.3.1 Hydrogen control, containment venting and containment isolation

In order to prevent hydrogen deflagrations or detonations in the primary containment, all the nuclear power plants have systems such as igniters, thermal or passive autocatalytic recombiners or mixing systems. The KKM containment is inerted with nitrogen.
All the plants have a system for venting the containment which, according to the relevant safety analysis reports, is fitted with filters with decontamination factors of at least 100 for iodine and 1000 for aerosols. The containment venting systems all have (at least partly) a two-train contaminated gas piping, of which one train is sealed with a rupture disk. The shutoff valve upstream of the rupture disk is closed during normal operation at KKG but is open at the other plants.
At all the plants, the containment venting systems are used not only to prevent hydrogen deflagrations or detonations but also to prevent a containment overpressure failure.

21.3.2 Prevention of basemat melt-through

KKB, KKL and KKM each refer to their accident management measure of flooding the containment. For the case of failure of both core cooling and RPV depressurization (e.g. within the course of a total SBO), KKL emphasizes that the pressure build-up in the containment pushes a large amount of water from the suppression into the drywell to fill the drywell and pedestal to the top of weir wall level. KKG describes the supply of emergency cooling medium to form a melt crust on the corium surface, and the importance of a high sump water level.

21.4 Accumulations of hydrogen outside the containment

KKB postulates the integrity of the containment. KKB regards H2 accumulations outside the containment as unlikely. KKL puts forward one case involving H2 accumulations in the secondary containment. This would happen if a core meltdown occurred during plant shutdown and, at the same time, an open material gate linking the primary and secondary containments cannot be closed promptly. At KKM, H2 accumulations are not expected to reach a hazardous limit. Possible leaks from the primary containment are diluted in those parts of the building, which then absorb them to such an extent that explosive gas mixtures are not expected to develop. KKG does not examine this topic. However, ENSI has required a systematic re-evaluation of the issue in all plants.

21.5 Accident management measures to restrict the radioactive releases after loss of containment integrity

At all the nuclear power plants, some of the accident management measures (e.g. SG injection, RPV injection, containment venting, containment injection) requested before the loss of containment integrity are also used to limit radioactive releases after loss of containment integrity. In this context, the BWR operators (KKL, KKM) refer to measures specified in instructions on monitoring the secondary containment. KKL mentions the verification of containment isolation and RPV venting as two measures for this purpose, which are ordered in the quoted instructions. KKM emphasises that these instructions must also be worked through after the transition to SAMG. At both the PWR plants (KKB, KKG), special measures to limit radioactive releases after the loss of containment integrity are integrated into the SAMG.

21.6 Accident management measures in place at loss of the spent fuel pool cooling

In case of a failure of the systems used in operation for cooling the spent fuel pools (SFP), staggered (defence-in-depth) measures come into play. Initially, their aim is to use permanently installed alternative systems (e.g. the shutdown cooling system at KKM) to restore a cooling circuit. For this purpose, the relevant sections of the first safety train (at KKG, KKL and KKM) are available and, at KKG, those of the second safety train are also available for SFP cooling. In this case, some manual measures may have to be implemented by the plants' operating staff. Operation of the systems as such is handled from the main control rooms (KKG, KKL, KKM) or from the emergency control room (at KKG only). The respective measures are stipulated in accident procedures. After loss of the operationally utilised SFP cooling system at KKB, it is necessary to deploy the alternative pool cooling system that is assigned to the third safety train.

If it proves impossible to reconnect a cooling circuit, heat is removed by vaporisation and/or evaporation cooling. In this case, prepared accident management measures are implemented whereby the vaporisation and/or evaporation volume is compensated by re-injecting water into the SFP. These accident management measures (safety train 3) are implemented with the help of mobile operational equipment kept available on-demand on site, such as fire extinguishing pumps, fire water tender vehicles and fire brigade hoses. At the KKB, KKL and KKM plants, it is necessary to implement manual measures in the storage pool area, e.g. to establish hose connections as far as the SFP or to operate valves. By contrast, the injection into the SFP at the KKG plant is effected by means of a connection that is permanently installed in the annular space and then via pipes in the independent pool cooling system. Due to the dimensioning of the SFPs, sufficiently long time is available after the failure of SFP cooling at all the plants, in every operating condition (power operation or shutdown with full core discharge), in order to implement the prepared accident management measures. As the SFPs at the KKB and KKL plants are located in separate buildings, the accident management measures for SFP cooling can be implemented there regardless of conditions in the containment.

Incipient evaporation of the SFP inventory is quoted by all the operators as a cliff-edge effect. Depending on the reactor type and plant configuration, the time until this occurs (after a failure of SFP cooling) is of the order of several hours for the PWR plants (KKG: 6 hours; KKB: 13 hours) or several days for the BWR plants. Three operators (KKB, KKL, KKM) point out that incipient evaporation of the SFP inventory would substantially impair accident management measures (for restoration of SFP cooling), and they announce back-fitting measures to counteract this impairment (e.g. SFP filling level indicators in the main control room and in the emergency control room).

In those plants where the SFPs are located outside the primary containment (KKB, KKL, KKM), no specific measures have currently been prepared in order to counteract the release of H_2 following a Zr-water reaction in the SFPs. In case of a total failure of SFP cooling, KKL and KKM do not expect any uncovering of the fuel assemblies in the storage pools that could lead to major releases of activity, on account of the large water reserve for the SFPs and the prolonged periods thereby available to bring alternative water injections into operation. At KKB, no release of H_2 is expected as long as the fuel assemblies remain covered and the fuel does not heat up beyond 800°C.

In the event that a water injection into the SFP at the KKG plant is impossible even with accident management measures, isolation of the containment can be implemented in order to minimise the release of activity. A pressure failure of the containment within 72 hours of the occurrence of the event is excluded by KKG. An H_2 mixing and reduction system is installed within the containment. In addition to the SFP inside the containment, KKG also has an external wet storage facility for spent fuel assemblies. Heat removal here is ensured by inherently safe passive natural circulation via two parallel cooling circuits. In an emergency, this pool can also be supplied by prepared fire brigade resources.

Table 36. Accident management provisions implemented in Swiss NPPs.

<table>
<tr><th></th><th>KKB (PWR)</th><th>KKG (PWR)</th><th>KKL (BWR)</th><th>KKM (BWR)</th></tr>
<tr><td rowspan="2">Limiting radioactive release</td><td colspan="4">Filtered Containment Venting System with a wet scrubber integrated into the release train.
Decontamination factors of at least 100 for iodine and 1000 for aerosols.
The containment venting systems have (at least partly) a two-train contaminated gas piping, of which one train is sealed with a rupture disk.</td></tr>
<tr><td>The shutoff valve upstream of the rupture disk is open during normal operation</td><td>The shutoff valve upstream of the rupture disk is closed during normal operation</td><td>The shutoff valve upstream of the rupture disk is open during normal operation</td><td>The shutoff valve upstream of the rupture disk is open during normal operation</td></tr>
<tr><td rowspan="2">Accident management measures in place at the various stages of a scenario of loss of the core cooling function</td><td>Coolant supply to the containment to provide external cooling of the RPV so as to prevent it from failing</td><td></td><td colspan="2">Coolant supply to the containment to ensure that the core melt falls into a pool of water</td></tr>
<tr><td></td><td>After an RPV failure and the beginning of damage to the containment foundation, flooding of the containment is used to decelerate the interaction between melt and concrete and to minimise discharges if the core melt reaches the containment sump area</td><td>After an RPV failure the supply of water to the suppression pool (SP) of the containment would have a lower priority than the supply of water for cooling the residual melt remaining in the core</td><td></td></tr>
<tr><td rowspan="2">Hydrogen control, containment venting and containment isolation</td><td colspan="4">all the nuclear power plants have systems such as igniters, thermal or passive autocatalytic recombiners or mixing systems.</td></tr>
<tr><td></td><td></td><td></td><td>The KKM containment is inerted with nitrogen</td></tr>
</table>

Table 36 (continued)

<table>
<tr><th></th><th>KKB (PWR)</th><th>KKG (PWR)</th><th>KKL (BWR)</th><th>KKM (BWR)</th></tr>
<tr><td>Prevention of basemat melt through</td><td>flooding the containment</td><td>KKG describes the decelerating effect of filtered venting on the reaction speed of the concrete-melt interaction, the supply of emergency cooling medium to form a melt crust on the corium surface, and the importance of a high sump water level</td><td>Flooding the containment. For the case of failure of both core cooling and RPV depressurization (e.g. within the course of a total SBO), KKL emphasizes that the pressure build-up in the containment pushes a large amount of water from the suppression into the drywell to fill the drywell and pedestal to the top of weir wall level.</td><td>flooding the containment</td></tr>
<tr><td>Accumulations of hydrogen outside the containment</td><td>KKB postulates the integrity of the containment and regards H2 accumulations outside the containment as unlikely</td><td>KKG does not examine this topic</td><td>KKL puts forward one case involving H2 accumulations in the secondary containment. This would happen if a core meltdown occurred during plant shutdown and, at the same time, an open material gate linking the primary and secondary containments cannot be closed promptly.</td><td>At KKM, H2 accumulations are not expected to reach a hazardous limit. Possible leaks from the primary containment are diluted in those parts of the building, which then absorb them to such an extent that explosive gas mixtures are not expected to develop</td></tr>
<tr><td rowspan="2">Accident management measures to restrict the radioactive releases after loss of containment integrity</td><td colspan="4">some of the accident management measures (e.g. SG injection, RPV injection, containment venting, containment injection) requested before the loss of containment integrity are also used to limit radioactive releases after loss of containment integrity</td></tr>
<tr><td colspan="2">special measures to limit radioactive releases after the loss of containment integrity. The measures are specified in the SAMG. The ones mentioned in the NPP reports include containment spraying (KKB) and optimization of the operation of the annulus exhaust air handling system (KKG).</td><td>KKL mentions the verification of containment isolation and RPV venting as two measures for this purpose</td><td>KKM emphasises that the verification of containment isolation and RPV venting must also be worked through after the transition to SAMG</td></tr>
</table>

Table 37. Accident management provisions for the Spent Fuel Pool in Swiss NPPs.

	KKB (PWR)	KKG (PWR)	KKL (BWR)	KKM (BWR)
Alternative cooling circuit	Third safety train	First safety train Second safety train Third safety train	First safety train Third safety train	First safety train Third safety train
Location of the SFP	Separate building	Inside the reactor building [?]	Separate building	Separate building
Time before start of evaporation of the SFP inventory (limiting case with fresh core reloaded)	13 hours	6 hours	Several days	Several days
counteract the release of H2	No	H2 mixing and reduction system	No	No
isolation of the containment	No	Yes	No	No
external wet storage facility	No	Yes. Heat removal here is ensured by inherently safe passive natural circulation via two parallel cooling circuits	No	No

21.7 References

[SUI11] "EU Stress Test" Swiss National Report – ENSI review of the operators' reports – December 2011

22 Annex 14: Severe Accident Management in The Netherlands

The information provided in this annex is taken from the reference [NED11], [BOR11] and [VAY01]. Much more information regarding systems to cope with severe accident can be found especially in this reference document [NED11]. Therefore the strategies presented here reflect mainly the situation end of 2011.
A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the Dutch NPP to provide guidance for further ASTEC development.
One NPP is operated in the Netherlands. The main characteristics of this plant given in the table below.

Table 38. Characteristics of Borssele NPP (KCB).

Borssele	
Type	Two-loop PWR
Thermal power	1365 MWth
First criticality	1973
Containment building characteristics	spherical steel shell in concrete reactor building
Steam generator replacement	no
Fuel storage pool capacity	730 m^3
Designer	KWU

22.1 Characteristics of SAMGs

The national report describes the Emergency Organisation and stresses the importance of simulator-based exercises involving all groups implicated in the organisation.
The Borssele SAMGs cover all plant states from shutdown to full power. They are based on the generic Westinghouse Owners Group (WOG). However some adaptations were necessary as KCB is a Siemens KWU design and some strategies used in Westinghouse NPP were not directly applicable to Borsele KCB. KCB has also studied elements of other approaches in SAMG. An example is the system interdependency matrix, developed by ERIN Inc., USA, for the Boiling Water Reactor Owners Group (BWROG). Such a method supports decision making about which systems to restore to service. KCB has also studied whether the mechanism to estimate the plant damage states (a characteristic of the Combustion Engineering Owners Group approach) would be beneficial.
KCB SAMG consists of:

- Severe Accident Control Room Guidelines (SACRG);
- Severe Accident Guidelines (SAG): A SAG is activated when plant parameters exceed the level for controlled, stable operation as indicated in the Diagnostic Flow Chart (DFC);

- Severe Challenge Guideline (SCG) : A SCG is activated when an immediate and severe challenge to containment fission product boundaries occurs as indicated in the Severe Challenge Status Tree (SCST);
- Severe Accident Exit Guidelines (SAEG): SAEGs describe actions to ensure long-term operation after a controlled, stable operation has been received;
- Computational Aids (CA).

The Severe Accident Guidelines (SAGs), Severe Challenge Guidelines (SCGs) and Severe Accident Exit Guidelines for KCB are

- SAG-1 Inject into the steam generators
- SAG-2 Depressurise the reactor coolant system
- SAG-3 Inject into the reactor coolant system
- SAG-4 Inject into the containment (outside the Cavity)
- SAG-5 Reduce fission product releases
- SAG-6 Control containment conditions
- SCG-1 Mitigate fission product releases
- SCG-2 Reduce containment pressure
- SCG-3 Control hydrogen flammability
- SCG-4 Control containment vacuum
- SAEG-1 Long-term monitoring
- SAEG-2 SAMG termination

Details of concrete measures available within these procedures are given in Appendix 6.1 of [BOR11].

The strategies used to limit radioactive releases after core melt are provided by the SAMGs. In particular the guidelines SAG-5, SAG-6 and SCG-1 give strategies to mitigate radioactive releases.

Contaminated water produced during an accident can be stored in the controlled area in the storage and waste water tanks which are normally used for contaminated process water.

22.1.1 Accident management measures after occurrence of fuel damage

These measures are described in the guidelines

- Injection into the reactor coolant system, SAG-3;
- Depressurising the reactor coolant system: SAG-2.

More than one SAG may be evaluated at a time and the implementation of strategies should follow the priorities dictated. Other SAGs which might be important after the occurrence of fuel damage with respect to core cooling are:

- Injection into the steam generators: SAG-1;
- Injection into the containment: SAG-4.

22.1.2 Accident management measures after failure of the reactor vessel

After the failure of the reactor vessel core debris will leave the primary system. The accident management measures currently in place for cooling ex-vessel core debris outside the cavity and for scrubbing fission product releases of ex-vessel core debris outside the cavity are described in Severe Accident Management Guideline SAG-4:

- Injection of water from the TJ storage tanks by the containment spray pumps;
- Injection of water from the TJ storage tanks by gravity drain.

22.1.3 Maintaining the containment integrity after occurrence of significant fuel damage in the reactor core

The SAM goals supported by strategies under the different SAGs are:

- Elimination of fuel damage/meltdown in high pressure
- Management of hydrogen risk inside the containment
- Prevention of overpressure of the containment
- Prevention of re-criticality
- Prevention of basemat melt through

22.1.4 Accident management measures to restrict the radioactive releases

A loss of containment integrity during the severe accident implies radioactive releases and requires the strategies described in the following procedures:

- reducing the fission product releases: SAG-5 ;
- controlling the containment conditions: SAG-6;
- mitigating fission product releases: SCG-1 ;
- injection into the containment: SAG-4pool.

Accident management has also to address the dangers linked to the uncovering of the top of fuel in the fuel pool that are linked to hydrogen production, lost radiation shielding and fission product release.

SAM strategies proposed in [BOR11] are:

- active opening of relief hatches between the installation area and the operations area in the containment. This will improve/start the natural circulation between the installation area and the operations area in order to reduce the probability of high local hydrogen concentrations;
- controlling the containment conditions: SAG-6 ;
- controlling hydrogen flammability: SCG-3.
- for radiation shielding, work at refilling the pool from one of the available sources, see discussion in report [2]
- for restricting releases after the damage of spent fuel in the fuel pool, strategies are the same mentioned above for the containment, because the fuel pool is placed within the containment.

22.2 References

[BOR11] Complementary Safety margin Assessment NPP Borssele, Stress test report of the Licensee EPZ, http://www.rijksoverheid.nl/onderwerpen/kernenergie/documenten-en-publicaties/rapporten/2011/11/02/final-report-complementary-safety-margin-assessment.html , November 2011.

[NED11] "Netherlands' National Report On the post-Fukushima stress test for the Borssele Nuclear Power Plant, Ministerie van Economische zaken, Landbouw & Innovatie, December 2011.

[VAY01] Implementation of Severe Accident Management at the Borssele NPP, The Netherlands, George Vayssier, Mario van der Borst, Charles Wike, OECD Workshop on the Implementation of Severe Accident Management, Paul Scherrer Institute, Villigen, Switzerland, 10 - 13 September 2001

23 Annex 15: Severe Accident Management in the UK

The information provided in this annex is taken from the reference [UK11]. Much more information regarding systems to cope with severe accident can be found especially in this reference document [UK11]. Therefore the strategies presented here reflect mainly the situation end of 2011.

A limited number of these strategies could be modified in the future to take into account the lessons learned from the Fukushima accident, the results of the EU Stress Tests and the results of ongoing R&D projects. But the target here is to compile the main Severe Accident Management strategies for the UK NPPs to provide guidance for further ASTEC development.

As ASTEC V2.0 and V2.1 are developed for LWRs and not Gas Cooled Reactors, only Sizewell B is considered here.

Table 39. Nuclear Power Plant (LWR) in the UK.

NPP	First criticality	Power [MWth]	Reactor Type
Sizewell B	1988	3444	Westinghouse SNUPP PWR

SNUPP: Standardised nuclear power plant system

Sizewell B is a standard Westinghouse PWR. Based on the information provided in reference [UK11] and [SEH12] it can be assumed that the SAMG strategies in place at Sizewell B are comparable to the generic Westinghouse SAMG strategies. Some of these accident management measures taken from [UK11] are highlighted below, especially the one for maintaining containment integrity in case of Severe Accident.

23.1 Maintaining the Containment Integrity after Occurrence of Significant Fuel Damage (up to Core Meltdown) in the Reactor Core

23.1.1 Elimination of fuel damage/melt-down in high pressure

The UK PWR does not include specific depressurisation design provisions dedicated to the elimination of high ‐ pressure melt ejection (HPME). The design intent is that, should the operator fail to depressurise using normal systems and a high ‐ pressure failure of the primary circuit occurs, the corium would be confined to the lower compartment in containment and direct containment heating of this area would have only a modest effect on the internal pressure.

23.1.2 Management of Hydrogen risks inside the containment

At the time of the EU Stress tests, Sizewell B had limited hydrogen mitigation provisions in the form of a small number of recombiners within the containment building; these would be of limited use during a severe accident. Passive Autocatalytic Recombiners (PAR) to cope with Severe Accident situations are being installed now in Sizewell B.

The hydrogen purge system, which provides a diverse means of reducing hydrogen concentrations in accidents, provides a means of connecting to the emergency exhaust HVAC system and could potentially be used to vent hydrogen from the containment under severe accident conditions. However, this route would not be feasible if hydrogen concentrations were above 3%, which would be likely following a severe accident, therefore this would have to be assured with ample hydrogen recombination capacity, which Sizewell B does not have.

23.1.3 Prevention of overpressure of the containment

Sizewell B has some design features that would limit the occurrence of over‐pressurisation of the containment; namely, the large volume, provision of containment fan coolers and water spray system and, as a last resort, the reactor building fire suppression system could be used for additional cooling. ONR notes that EDF NGL (the licensee) is currently considering the feasibility of installing containment venting capability; this capability could be provided by installing a filtered containment vent. In addition, EDF NGL will consider whether there is a need to add further passive autocatalytic hydrogen recombiners and a flexible means of injecting water into the containment using portable external equipment.

23.2 Prevention of Basemat Melt Through

23.2.1 Potential design arrangements for retention of the Corium in the RPV

As there is no formal in‐vessel retention strategy at Sizewell B it has to be assumed that corium will eventually end up on the basemat where it can be cooled by cavity flooding using the reactor building spray system or fire suppression system. The design of the building should ensure that between one and three metres of water is in the cavity prior to vessel failure, even without the operation of engineered safeguards.

23.2.2 Potential arrangements to cool the Corium inside the containment after RPV rupture

At the PWR, water is required to cool the corium in the containment; it can be injected via the reactor building spray or containment fire suppression systems.

23.3 References

[SEH12] B. Sehgal (2012): “Nuclear safety in Light Water Reactors – Severe accident phenomenology”.

[UK11] "European Council "Stress Tests" for UK nuclear power plants – National Final Report"- Office of Nuclear Regulation – December 2011.

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