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VVER Working Group

Regulatory approaches and criteria used in severe accident analyses and severe accident management

Regulators involved in the VVER working group discussions:	AERB, HAEA, NNSA, Rostechnadzor, STUK and TAEK
Regulators which support the present report:	AERB, HAEA, NNSA, Rostechnadzor, STUK and TAEK
Compatible with existing IAEA related documents:	Yes

CONTENT

I. Introduction

In June 2014, during the 2nd VVER Working Group (VVERWG) meeting [1], the representative from Rostechnadzor expressed the interest of the VVER's family to understand differences in regulatory approaches and oversight practices used in different countries related to severe accident assessment and severe accident management.

It was suggested to establish the technical experts subgroup on severe accidents (TESG on SA) to have further discussions among regulators to better understand differences in regulatory approaches and oversight practices as well as to identify commendable practices in this area.

In December 2014 [2], it was agreed that the TESG on SA would conduct a discussion and prepare a technical report devoted to the following topics:

- Methodology for severe accident analyses;
- Technical provisions for safety systems;
- Severe accident management operating strategies (SAMG);
- Radiological impact assessment.

This report has been prepared on the basis of answers given by VVERWG members to the questionnaire elaborated by SEC NRS (Russia) with proposals received from other subgroup members. The regulatory bodies from following countries have participated in the preparation of this Technical report: AERB (India), HAEA (Hungary), NNSA (China), Rostechnadzor (Russian Federation), STUK (Finland) and TAEK (Turkey).

II. Scope and Objectives

This document is intended to summarize key aspects of the regulatory requirements and existing practices in the field of severe accident assessment and severe accident management. It is supposed that the current activity will be a preliminary stage before national safety review of their VVER application and will allows to share the regulatory experience, to highlight the items where approaches of regulators in member countries are similar and also to identify and discuss the differences.

The focus of the information presented in this technical report, as well as the activity of the VVER TESG on SA is on the events which lead to reactor core or fuel damage.

III. Comparative summary of main findings

In the following text the expression «all member countries» means the regulatory authorities of the following countries - Finland, Hungary, India, People's Republic of China, Russian Federation and Turkey. Similarities in the regulatory approaches are presented in the text; the differences are listed with bullets.

3.1. General and legal items

The common approach of representatives of member states assumes that the issues relating to severe accidents have to be under control of national regulators and conform to international agreements, domestic legislation, regulations and guidelines. The requirements of IAEA and other international organizations (WENRA, etc.) should also be taken into consideration. The volunteer initiative of the licensee is possible and encouraged, but the measures taken by the licensee on this basis have to be reviewed and agreed by the national regulator.

3.2. Procedures and Guidelines

The development of SAMG covering prevention and mitigation stages of severe accident are mandatory for licensee in all member states.

SAMGs should be symptom-based and corresponding to administrative and technical requirements. Entry and exit criteria have to be clearly defined on the basis of measured parameters. Exit criteria are the set of conditions which define the stable and safe NPP state.

In all member countries it is required that decision making person has to be identified unambiguously. Usually this responsibility is relying on the operator and assigned to the emergency director. Technical support center has a supporting role (to elaborate the recommendations on management strategy).

In all member countries the review of SAMG and corresponding technical basis are a part of licensing, and verification and validation of SAMG are required. Plant simulator and results of SA analyses could be a base of SAMG validation. The periodic emergency training and drills should be used to verify SAMG.

The SAMG compliance with current NPP state is required and to be confirmed.

3.3. Equipment

In all member countries, the equipment for the severe accident management shall be foreseen in the design. The equipment dedicated for SAM and its capacity shall be enough to provide the fulfilment of the main safety functions (subcriticality, cooling of the damaged fuel and confinement of the radioactivity). The list of equipment dedicated for SAM and its technical characteristics is defined by the designer and reviewed by the regulator within licensing.

Existence of the special instrumentation and control (I&C) qualified for SA conditions are required in all member states. Their set and characteristics shall provide the monitoring of NPP state and implementation of SAM strategies in the course of SA.

In all member countries the integrity and leak tightness of containment should be confirmed for SA conditions based on the results of SA analyses. The integrity of the containment should be confirmed based on the criteria defined for SA conditions (pressure and temperature). Hydrogen release and distribution in the containment shall be considered.

Cooling of the containment during SA should be provided taking into account of all heat sources. Containment filtered venting system (CFVS) can be considered as an ultimate option to decrease the containment pressure.

In all member countries it is required to eliminate the detonation of hydrogen in the containment. The technical provisions to monitor and to decrease the hydrogen concentration in the containment shall be foreseen in the design. Elimination of sustained deflagration and hydrogen detonation shall be demonstrated on the basis of SA analyses taking into account the functioning of technical provision foreseen in the design.

In all member countries the requirements are already established or near to be established to consider multiunit accidents, all locations of the fuel (reactor, spent fuel pool), and consequences of the extreme external events.

The requirements to the type of technical means that are different in member countries:

Type of equipment:

- Only permanently installed systems are credited for SAM in Finland;
- Permanently installed and mobile equipment are credited for SAM in China, India, Hungary, Russian Federation and Turkey.

Independency and single failure criteria:

- In Finland an active equipment dedicated for SAM must be independent and single failure tolerant;
- In Hungary the equipment dedicated for SAM must be independent, single failure tolerance is prescribed only for: severe accident I&C systems; systems for cleaning the containment; and the ventilation and air conditioning systems of the unit main and backup control rooms; the technical support center and the emergency response center;
- In India the independence is required and single failure tolerance is desirable;
- In Russian Federation and Turkey the independence and single failure tolerance are desirable;
- In China, the equipment dedicated for SAM should be independent as far as practicable according to new revision of HAF 102, 2016 (based on IAEA SSR-2/1, 2016), and single failure tolerance is desirable.

Safety classification:

- In Finland, Hungary, India and Turkey the technical means dedicated for SAM shall be safety classified;
- In China, the technical means dedicated for SAM may not necessarily be safety classified in requirement; however, some specific requirements are needed, such as seismic class, availability in severe accident condition etc;.
- In Russia the equipment dedicated for SAM shall be safety classified if calling time for this equipment is demanded within 72 hours.

Requirements on I&C:

- In Finland I&C dedicated for SA must be independent from all other I&C systems, safety classified and single failure tolerant;
- In Hungary the I&C dedicated for SA must be independent;
- In China and India, independency of I&C and single failure criteria with respect to I&C are not explicitly covered in the regulatory documents, but it is desirable. Instrumentation that is credited to operate during design extension conditions and during and after severe accident scenario shall be shown with reasonable confidence, to be capable of achieving intended function under the expected environmental conditions;
- In Russian Federation the requirements in respect of independency and single failure criteria for SA I&C are not established. There shall be I&C qualified for the SA conditions and sufficient for the SAM. If calling time for I&C is within 72 hours it shall be safety classified;

- In Turkey the requirements on I&C for SA are based on IAEA guides (SSR-2/1, NS-G 2.15 and also on NS-G 1.10) where the key plant parameters needed for both preventive and mitigatory accident management measures should be identified and it should be checked that all these parameters are available from the instrumentation in the plant. Alternative instrumentation should be identified where the primary instrumentation is not available or not reliable.

Leak tightness and integrity of the containment:

- In Finland the leak tightness of the containment must demonstrated up to pressure obtained from the SA analyses and increased by:
    - 50% (to account for the uncertainty);
    - pressure increase due to hydrogen burn calculated by Adiabatic Isochoric Complete Combustion (AICC) principle.
- In China, according to HAF102 2016, design provision shall be made to prevent the loss of the structural integrity of the containment in all plant states. The use of this provision shall not lead to an early radioactive release or a large radioactive release.

  There are no special requirements related to the leak tightness of the containment in Hungary, India, Russian Federation and Turkey.

Heat removal from the damaged fuel:

- In Finland, a nuclear power plant shall be equipped with systems to ensure the stabilisation and cooling of molten core material generated during a severe accident (see Section “Core Catcher”). Systems to remove heat from the containment to the ultimate heat sink during accidents are also required. The safety function to be performed by these systems is to reduce containment pressure and temperature, and to keep them at a sufficiently low level. The systems shall be independent of the systems designed for normal operation, anticipated operational occurrences and postulated accidents, as well as single-failure tolerant. In addition, the debris of a damaged reactor shall be cooled in such a way that release of radioactive materials to the containment atmosphere can be effectively reduced, and that the heat radiated from the debris will not endanger containment integrity.
- In Hungary sufficient and diverse heat removal solutions, which are also independent in respect of power supply, shall be provided for the removal of the residual heat from the reactor and the spent fuel pool. At least one design solution shall perform its function also during DEC events caused by external natural hazard factors. An additional independent heat removal solution shall be provided for the SA conditions.
- In China, according to HAF102 2016,
    - The capability to remove heat from the containment shall be ensured, in order to reduce the pressure and temperature in the containment, and to maintain them at acceptably low levels after any accidental release of high energy fluids. The systems performing the function of removal of heat from the containment shall have sufficient reliability and redundancy to ensure that this function can be fulfilled.
    - The design shall also include features to enable the safe use of non-permanent equipment for restoring the capability to remove heat from the containment

There are no special requirements related to the heat removal from the damaged fuel in India, Russian Federation and Turkey.

Devices for primary pressure decrease:

- In Finland severe accident pressure reduction system shall be independent of systems designed for the plant's operational conditions and postulated accidents and single failure tolerant (2X100%);
- In India although it is not explicitly covered in the regulations, it is desirable that severe accident pressure reduction system should be independent of systems designed for the plant's operational conditions and postulated accidents;
- In China, according to HAF 102, 2016, severe accident pressure reduction system shall be independent as far as practicable of systems designed for the plant's operational conditions and postulated accidents;
- In Russian Federation and Hungary there are no special requirements on SA primary pressure reduction system regarding its independency and single failure tolerance;
- In Turkey, the requirements on devices for primary pressure decrease are based on IAEA guides (SSR-2/1, NS-G 1.10 and NS-G 2.15) where mitigation strategies should be taken to depressurize the reactor circuit and containment to prevent the failure.

Core catcher:

- In Finland there is a requirement that nuclear power plant shall be equipped with systems to ensure the stabilisation and cooling of molten core material generated during a severe accident. Direct interaction of molten core material with the load bearing containment structure shall be reliably prevented;
- In China, adequate consideration shall be given to cool molten core debris, and to mitigate the effects of its interaction with concrete, a means of flooding the reactor cavity with water to prevent breach of the vessel, or a reinforced sump or cavity to catch and retain molten core debris should be adopted;
- In Hungary there is a requirement that the functions mitigating the consequences of accidents and, if necessary, the systems providing these functions shall be specified to such an extent that in severe accidents the molten fuel can be retained in a cooled condition within the containment. The destructive effect of the fuel melt on the structural integrity of the containment shall be prevented or shall be limited to the extent reasonably achievable;
- In India, provisions shall be made for transfer of residual heat from damaged / molten core to an ultimate heat sink to ensure that acceptable temperatures can be maintained in structures, systems and components important to the safety function of confinement of radioactive materials in the event of a severe accident. Use alternate paths and means to supply water to cool the molten corium/debris in the core catcher.

Severe accident safe state is a state which shall be achieved subsequent to a design extension condition with significant core damage or core melt phenomena. Severe accident safe state shall be reached at the earliest after an accident initiation. It should be possible to maintain this state indefinitely.

During this state there is:

- No possibility of re-criticality;
- Fuel or debris are continuously cooled;

  - Uncontrolled release of radioactivity to environment is arrested;
  - Means to maintain above conditions are available for long term, including critical parameter monitoring;
  - Monitoring of radiological releases and containment conditions.

  As the plant state is in design extension condition with core melt (severe accident), the severe accident safe state should be preferably reached within about one week from accident initiation. Severe accident monitoring instrumentation and control: it shall be possible to assess the information condition of core or debris.

- In Russian Federation, there are no special requirements on core catcher. However, there is special requirement to provide sub-criticality of the damaged (melted) core in the case of severe accident;
- Turkey - there is no special requirements on core catcher.

<u>Treatment of liquid radioactive effluents arisen during SAM:</u>

There are no special requirements related to treatment of liquid radioactive effluents arisen during SAM in China, Finland, India, Russian Federation, and Turkey.

- In Hungary, it shall be estimated with appropriate margins that what type and what quantities of radioactive wastes are expected to be produced during DEC events as well as their management and response to them. In their knowledge, solutions suitable for the interim storage and management of wastes shall be designed and their location shall be designated on-site.

3.4. Severe accident analyses

The review of severe accident analyses is a part of safety assessment.

<u>List of SA scenario:</u>

- In Finland the set of severe accident scenarios shall cover all actions required for the plant SA strategy and the phenomena associated with it. The analyses must justify the SAMG's strategies.
- In Hungary a minimum set of SA scenario are listed in the regulatory requirements, and it is also required to develop and justify the list of SA scenario for particular case.
- In China, India, Russian Federation and Turkey the set of SA scenario should be based on engineering judgment, deterministic and probabilistic safety assessments and its completeness and representativeness should be justified in SAR.

<u>SA acceptance criteria:</u>

The results of SA analyses should confirm the successful recovery of the main safety functions (subcriticality, cooling of the damaged fuel, confinement).

The integrity and leak tightness of the containment under the severe accident conditions must be proved on the base of SA analyses. SA analysis should consider all the phenomena that can aggravate the impacts on the containment and on the systems are within containment. Elimination of hydrogen detonation and sustained deflagration must be confirmed.

In all member countries criteria for cumulative CDF is equal 10^{-5} 1/year. Large and early release should be practically eliminated.

- In India additional criterion on the cumulative frequency from the internal events is less than 10^{-6}/year is incorporated.

Large radioactive release:

- In Finland the mean value of the frequency of a release of radioactive substances from the plant during an accident involving Cs-137 release into the atmosphere in excess of 100 TBq is less than $5 \cdot 10^{-7}$/year;
- In Hungary the frequency of large radioactive release should not exceed 10^{-6}/year;
- In India, Russian Federation, and Turkey the frequency of large radioactive release should not exceed 10^{-7}/year;
- In China, during the 12th Five-year Plan period (before 2015), the new NPPs should have comprehensive measures to prevent and to mitigate severe accidents, the LRF should be lower than 10^{-6}/year; after the 13th Five-year Plan period (after 2015), the new NPPs design should practically eliminate the possibility of large scale radioactive release;
- In Finland, Hungary, and India the release of radioactive materials arising from a severe accident shall not lead to the need for large scale protective measures and long term restrictions on the use of extensive areas of land;
- In Russian Federation and in Turkey the restriction on cumulative integral release is not implemented, but the consequences of radioactive release should not lead to the need for protective measures beyond 25 km;
- In Hungary it shall be demonstrated that in case of SA no urgent protective measures are required beyond a distance of 800 m from the NPP unit;
- In Hungary it shall be demonstrated that there is no need for any kind of temporary action, beyond a distance of 3 km from the NPP unit and there is no need for any kind of late phase protective measure beyond a distance of 800 m from the NPP unit;
- In Turkey the criteria will be based on IAEA guide SSG-2 to demonstrate that the consequences would be acceptably low.

Severe accident codes:

In all member countries codes used for safety analyses must be verified and validated. The approach for code validation is practically similar. Code validation should be provided on the basis of analytical tests, separate effect tests, integral test facilities, data of actual accidents that have happened, benchmarks (comparison with results produced by validated codes).

Administrative rules:

- In Finland, Hungary, and in India there is no requirements for code certification;
- In Russian Federation and in Turkey all codes, applied for safety analyses, shall be certified;
- In China and Turkey requirements will be based on IAEA guides NS-G 2.15 and NS-G 1.10.

Methodology for SA analysis in general should be based on realistic approach, however in case lack of knowledge the conservative assumptions are credited (the conservativeness of the assumptions should be proved). Realistic approach on SA analysis means to be based on best estimate codes plus sensitivity study if applicable.

The results of SA analyses should confirm the successful recovery of the main safety functions (subcriticality of the damaged fuel, cooling, localization).

IV. Conclusions and Recommendations

The current report presents the overview of the existing national regulatory practices in the field of SA management in condensed manner which are used by the regulatory authorities in support of licensing of the NPP with VVER designs. It is recommended by the VVER technical experts subgroup on severe accidents that this technical report be issued as a MDEP product.

It is possible to note similarity of approaches regarding the general requirements and recommendations. In all member states it is required to prove the integrity of the containment under SA conditions and practically exclude the radioactive release resulting in long-term protective measures and actions.

The distinctions in requirements to the technical means providing the fulfillment of the general requirements can result in separate changes in the national versions of the VVER type power plant projects, but the general safety concept regarding SA remains the same.

The information presented in the report will assist in appreciation of national practices used in severe accident analyses and could be used as a basis for the further activity of the VVER TESG on SA for elaboration of the common position regarding the items of common interest for regulators of the MDEP VVER member countries.

V. References

1. Summary Record of the 2nd Meeting of the VVER Working Group NEA MDEP, June 2014, Paris, France.

2. Summary Record of the 1st Meeting of the VVER Technical Expert Subgroup on Severe Accidents NEA MDEP, December 2014, Moscow, Russia.

Appendix A – Regulatory approaches used in Finland

QUESTIONNAIRE ON SEVERE ACCIDENT REQUIREMENTS AND PRACTICES

I. General

1. What types of regulatory controls are installed in respect of severe accident (i.e. licensed verses voluntary initiatives)?

Regulatory requirements concerning severe accidents have been implemented since 1988.

2. What is the legal basis this matter (lows, requirements, regulations?)

High level requirements are given in the Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant Y/1/2016.

Detailed requirements are given in the YVL Guides issued by STUK:

YVL A.7, PRA

YVL B.1, Safety design

YVL B.2, Safety classification

YVL B.3, Deterministic safety analyses

YVL B.5, Reactor coolant circuit

YVL B.6, Containment

YVL C.3, Limitation and control of radioactive releases.

Nuclear Energy Decree 1988/161 chapter 3, section 22b: (Translation from Government Decree on Safety of Nuclear Power Plants 717/2013).

The release of radioactive substances arising from a severe accident shall not necessitate large scale protective measures for the public nor any long-term restrictions on the use of extensive areas of land and water.

In order to restrict long-term effects the limit for the atmospheric release of cesium-137 is 100 terabecquerel (TBq). The possibility of exceeding the set limit shall be extremely small.

The possibility of a release in the early stages of the accident requiring measures to protect the public shall be extremely small.

Regulation STUK Y/1/2016 section 10 Engineered barriers for preventing the dispersion of radioactive substances:

3 c) In order to ensure containment building integrity,

i. the containment shall be designed to maintain its integrity during anticipated operational occurrences and, with a high degree of certainty, during all accident conditions;

ii. pressure, radiation and temperature loads, radiation levels on plant premises, combustible gases, impacts of missiles and short-term high energy phenomena resulting from an accident shall be considered in the design of the containment; and

iii. the possibility of containment leaktightness becoming endangered as a result of reactor pressure vessel fracturing shall be extremely low.

4. A nuclear power plant shall be equipped with systems to ensure the stabilisation and cooling of molten core material generated during a severe accident. Direct interaction of molten core material with the load bearing containment structure shall be reliably prevented.

Regulation STUK Y/1/2016 section 11 Safety functions and provisions for ensuring them.

4. The most important safety functions necessary to bring the plant to a controlled state and to maintain it must be ensured even if any individual component of a system providing the safety function is inoperable and even if any other component of a system providing the same safety function or of a supporting or auxiliary system necessary for its operation is simultaneously inoperable due to the necessity for its repair or maintenance.

8. The management of severe reactor accidents and the monitoring of the plant's status during severe accidents shall be implemented by means of systems that are independent of the systems designed for normal operation, anticipated operational occurrences and postulated accidents. The leaktightness of the containment during a severe reactor accident shall be reliably ensured.

9. The plant shall be designed so that it can be brought into a safe state after a severe accident.

II. Procedures and instructions

1. Should guidelines and/or procedures be a part of licensing?

YVL A.6, Operation of a Nuclear Power Plant,

“710. A nuclear power plant shall be provided with severe accident management guidelines. The guidelines shall describe the measures for mitigating the consequences of severe accidents.”

711. The procedures drafted for postulated accidents and design extension conditions shall be symptom-based or a combination of symptom-based and event-based procedures. If safety functions cannot be maintained with the procedures employed, symptom-based procedures shall be put to use. The severe accident management guidelines shall be symptom-based.

715. The severe accident management guidelines shall be based on a severe accident management strategy and related analyses.

719. Instructions shall be drawn up for field action to be taken defined in the operating procedures for emergencies and transients and severe accident management guidelines.

802. Any procedures closely relevant to operations shall be submitted to STUK for information following the licensee's approval procedure. Any amendments to the procedures will be reviewed by STUK.

2. Should licensing cover as prevention and mitigation consequences of severe accident?

Systems and procedures designed for postulated accidents and design extension conditions act to prevent severe accidents. The requirements listed for General Part questions 1 &2 cover mitigation of severe accidents.

3. Should structure and content of the procedure and instructions be a part of regulatory control?

See answer 1 on this section. The severe accident management guidelines and amendments to the guidelines are sent to STUK for information. If necessary, STUK can require changes to guidelines.

4. How deep (in detail) should be monitored structure and content of procedures and/or instructions? Please consider the following items: entry criteria; exit criteria; criteria for the transition from prevention to mitigation domain of SA procedures and/or instructions; harmonization procedure and/or instructions dedicated for SAM with other emergency procedures and instructions; coverage of such SAM aspects as malty-unit accidents, different locations of the fuel (reactor' spent fuel pool, on-site fuel storage); distribution of the responsibility during the application of emergency procedure and/or instructions in the course of SA (who make a decision?);

1) YVL A.6,

716. The operating procedures drafted for emergencies and transients and severe accident management guidelines shall be verified and validated to ensure that they are administratively and technically correct for each nuclear power plant unit concerned, and that they are compatible with the environment in which they will be used.

717. The validation and verification of the procedures and guidelines shall be systematic. The extent to which human factors are accounted for in the procedures shall be judged upon validation. The validation of the procedures and guidelines shall be based on simulations or other suitable methods, primarily using a training simulator.

718. The operating procedures drafted for emergencies and transients and severe accident management guidelines shall be kept up-to-date at all times and fit for purpose.

SAMG in Finland is based on the validation of the severe accident management systems. Validation (experimental validation, if necessary) is required of the system performance and system components.

2) Multi unit accidents should be considered. Radiation and Nuclear Safety Authority Regulation on the Emergency Arrangements of a Nuclear Power Plant Y/2/2016:

1. Emergency arrangements shall be planned to ensure that emergency situations are quickly brought under control, the safety of the individuals in the site area is assured, and timely action is taken to prevent or limit radiation exposure to the public in the emergency planning zone.

2. Planning shall take account of a simultaneous threat to nuclear safety occurring in all nuclear facilities in the site area and their potential consequences, especially the radiation situation on the site and in the surrounding area and the opportunities to access the area.

3. Planning shall take account of the fact that the emergency situation could continue for a prolonged period.

4. Planning shall be based on analyses of the time behaviour progress of severe accident scenarios resulting in a potential release. In such a case, variations in the state of the plant, the development of events as a function of time, the radiation situation at the plant, radioactive releases, radioactive release routes and weather conditions shall be taken into account.

5. Planning shall take account of events deteriorating safety, their controllability and the severity of consequences, and threats related to unlawful action and the potential consequences thereof.

3) In Finland, decisions are also in severe accident made by the control room and by plant emergency manager. Technical support center has only a supporting role.

5. How the operator must confirm the correctness and sufficiency of the procedures and instructions? Should analyses and/or experiments be an obligatory part of validation and verification process for procedures and instructions?

YVL A.6, 716. The operating procedures drafted for emergencies and transients and severe accident management guidelines shall be verified and validated to ensure that they are administratively and technically correct for each nuclear power plant unit concerned, and that they are compatible with the environment in which they will be used.

III. Equipment

1. Should a minimal list of technical means dedicated for SAM be determined by regulator?

The containment must be designed to withstand the severe accidents (STUK Regulation Y/1/2016) and be equipped with systems mitigating severe accidents. The means (what kind of systems) is not specified by the regulation but left to the designer.

2. Should severe accident management provisions (i.e. using mobile equipment verses hardening on site equipment) be performance based or prescribed by regulation?

Permanently installed systems are required in Finland. SAM-systems must be independent, single failure tolerant and safety classified.

3. Should be installed special requirements (regulations) in respect of mission times and minimum capacity for equipment dedicated for SAM?

Definition of the mission time and equipment capacity are part of the severe accident systems design process. This done by the designer. STUK reviews and accepts the system design based on the requirements given in the STUK Regulations and YVL Guides.

4. Should be special requirements (regulations) be installed for using mobile equipment versus stationary installed equipment?

Permanently installed systems are required in Finland. In some cases, the system back-up may be provided by mobile equipment.

5. Should be a special requirements (regulations) related to I&C for SA (I&C for SAM and indicate the status of SA progress)?

Severe accident I&C is required to perform SAM-actions and to monitor severe accident progression and containment status. SA I&C must be independent from all other I&C systems, single failure tolerant and safety classified (YVL B1 §5240, point 8).

YVL B.1 §5218 and §5241

6. Should be special requirements (regulations) related hydrogen management in the containment under the SA conditions?

Hydrogen detonations must be analyzed and practically eliminated. (YVL B1, 424).

YVL B.6,311:

The leak tightness of the containment in severe accidents shall be demonstrated

- using the containment temperature and pressure obtained from the severe accident analyses performed in compliance with Guide YVL B.3
- by increasing the maximum pressure (pressure difference) by a 50% safety margin and
- <u>by pressure increase due to hydrogen burn calculated according to the AICC principle.</u>
  - The 50% safety margin compensates for the uncertainties associated with the calculation methods and selection of calculation cases in severe accidents.
  - PWR: add 50 % to overpressure load of maximal pressure / temperature load from analyses + add pressure increase due to AICC hydrogen burn
  - BWR: add 50 % to overpressure load of maximal pressure / temperature load from analyses. Pressure increase due to hydrogen burn = 0 for inerted containment
  - Best estimate methods can be used in analyses and in acceptance criteria (margin taken into the loads).

7. Should be special requirements (regulations) for primary pressure decrease?

There must be an extremely small probability that the reactor pressure vessel failure during a severe accident endangers the containment integrity -> primary system depressurisation must be ensured in severe accidents. Primary system depressurization must be independent, single failure tolerant, safety classified system (2x100%).

YVL B5:

422 The primary circuit shall be provided with a pressure reduction system to prevent the failure of the RPV in the event of a severe accident to the extent that it could endanger containment integrity.

423 The severe accident pressure reduction system shall be independent of systems designed for the plant's operational conditions and postulated accidents.

424 The severe accident pressure reduction system shall be capable of performing its safety function even in the event of a single failure.

425 Valves intended for pressure reduction shall be so designed that once opened they stay open reliably.

8. Should be special requirements (regulations) related to auxiliary equipment to provide water injection into the reactor vessel, steam generators and containment?

There are no special requirements related to the use of auxiliary equipment in severe accident. It must be possible to fulfil the needed water injection functions (in containment and core catcher) in SA even if any single component designed for the function fails (single failure tolerance on SAM-systems).

9. Should be special requirements (regulations) cover the special technical provisions dedicated for recovery of off-site power, multi-units considerations, treatment of liquid radioactive effluents?

Loss of off-site power supply: YVL B1 5402: The plant shall be provided with systems permitting power supply from the main generator to the plant systems important to safety in case the connection to the off-site grid is lost. When this power supply is designed, due consideration shall be given to the requirements presented in Section 5.5 of Guide YVL E.7.

Multi-unit accidents should be considered (see answer 4 of this section). The plants are not required to prepare for treatment of liquid radioactive effluents.

IV. Methodology of severe accident analyses

1. Should the list of SA analyses be a subject of licensing?

Guide YVL B.3, Deterministic safety analyses: “309. Severe accident analyses shall cover all actions required for the plant severe accident strategy and the phenomena associated with the strategy”.

2. How the completeness and representativeness of SA analyses could be confirmed (bounding criteria)?

The analyses must justify the plant severe accident management strategy. The strategy must be as independent as possible of the initiating event causing the severe accident

3. What acceptance criteria are applicable for severe accident?

Criteria for releases are given in the Nuclear Energy Decree 1988/161.

Acceptance criteria for the PRA limits are given in Guide YVL A.7

305. The design of a nuclear power plant unit shall be such that the mean value of the frequency of reactor core damage is less than 10–5/year.

306. A nuclear power plant unit shall be designed in compliance with the principles set forth in a way that

a. the mean value of the frequency of a release of radioactive substances from the plant during an accident involving a Cs-137 release into the atmosphere in excess of 100 TBq is less than 5·10-7/year;

b. the accident sequences, in which the containment function fails or is lost in the early phase of a severe accident, have only a small contribution to the reactor core damage frequency

Acceptance criteria for the containment are given in the Guide YVL B.6, Containment of a NPP.

311. The leaktightness of the containment in severe accidents shall be demonstrated.

339. In a severe accident, it must be possible to decrease the pressure difference across the containment pressure boundary to a level consistent with the safe state following a severe accident.

340. With design solutions where containment pressure decrease in compliance with requirement 339 is done by venting gas from the containment into the environment, the venting system must be provided with an efficient filter. After filtering, the released gases shall be routed to the plant ventilation stack.

341. The containment structure and systems used for managing accidents shall prevent such gas burns, gas explosions or other energetic phenomena that may jeopardise containment leaktightness or the operability of the components needed for accident management.

343. The debris of a damaged reactor shall be cooled in such a way that release of radioactive materials to the containment atmosphere can be effectively reduced, and that the heat radiated from the debris will not endanger containment integrity.

4. Should the codes dedicated for SA analyses (neutronic, thermohydraulic, containment, source term) be under licensing control? If yes, what are requirements for validation and verification of these codes?

Validation and verification are required.

The STUK Regulation Y/1/2016, section 3 requires that 2. Nuclear power plant safety and the technical solutions of its safety systems shall be assessed and substantiated analytically and, if necessary, experimentally.

3. The analyses shall be maintained and revised as necessary, taking into account operating experience from the plant itself and from other nuclear power plants, the results of safety research, plant modifications, and the advancement of calculation methods.

4. The analytical methods employed to demonstrate compliance with the safety requirements shall be reliable, verified and qualified for the purpose. The analyses shall demonstrate the conformity with the safety requirements with high certainty. Any uncertainty in the results shall be considered when assessing the meeting of the safety requirements.

YVL B.3 405. The physical models and computer programs used for the analyses shall be validated by comparing the calculation results obtained by them to separate effects tests or integral tests or nuclear power plant incidents. Comparison with already validated models may also be utilised.

In Finland, the regulator does not review and approve the codes used for safety analyses.

Appendix B – Regulatory approaches used in India

QUESTIONNAIRE ON SEVERE ACCIDENT REQUIREMENTS AND PRACTICES

I. General

1. What types of regulatory controls are installed in respect of severe accident (i.e. licensed verses voluntary initiatives)?

Both the licensed and voluntary initiative approaches are followed in India. Regulatory requirements exist with respect to severe accidents in the Design codes - AERB/NPP-PHWR/SC/D(R-1) & AERB/NPP-LWR/SC/D and codes and guides on operation. In addition initiatives have been taken by the utilities voluntarily especially after TMI, Chernobyl and the recent Fukushima accidents to address severe accidents and SAMG. Multitier regulatory review is followed for the review of severe accident analysis, provisions for severe accident prevention and mitigation and SAMGs

2. What is the legal basis this matter (laws, requirements, regulations?)

The stipulations regarding the safety in design of NPPs as stated in the AERB/NPP-LWR/SC/D, 2015 covers the severe accident related requirements also. Following are some of the AERB regulatory documents covering this aspect:

A. AERB/NPP-LWR/SC/D, 2015: Design of Light Water Reactor Based Nuclear Power Plants.
B. AERB/NF/SC/S (Rev.1), 2014: Site Evaluation of Nuclear Facilities.
C. AERB/NPP-PHWR/SC/D (Rev. 1), 2009: Design of Pressurised Heavy Water Reactor Based Nuclear Power Plants.
D. AERB/NPP-PHWR/SM/D-2, 2004: Hydrogen Release and Mitigation Measures under Accident Conditions in Pressurised Heavy Water Reactors.
E. AERB/SM/O-2: PSA
F. AERB/SG/G-10 - Regulatory review of PSA
G. Safety Classification
H. Emergency preparedness plans
I. AERB/SG/D-26, Safety Guide on SAMG is under preparation

AERB/NPP-LWR/SC/D

- If the plant state is within design extension condition without core melt, it shall be brought to and maintained under safe state within 24 hours (desirable) or within 72 hours (mandatory). Thereafter safe shutdown state should be maintained.
- In the design extension condition with core melt, the containment system and its safety features shall be able to perform in extreme scenarios that include, among other things, melting of the reactor core.
- Containment shall maintain its role as a leak-tight barrier for a period that allows sufficient time for the implementation of off-site emergency procedures following the onset of core damage. Containment shall also prevent uncontrolled releases of radioactivity after this period.
- Severe accident management guidelines shall be prepared, taking into account the plant design features and the understanding of accident progression and associated phenomena.

- Equipment that is credited to operate (e.g. certain instrumentation) during design extension conditions and during and after severe accidents scenario shall be shown, with reasonable confidence, to be capable of achieving the intended function under the expected environmental conditions. Severe accident management guidelines should address uncertainties arising from any shortfalls in such qualification of specific equipment/instrument.
- Severe accident management guidelines shall be prepared, taking into account the plant design features and the understanding of accident progression and associated phenomena.
- Level-1 and Level-2 PSA shall be carried out

General Considerations for Instrumentation and Control System

Instrumentation shall be provided for determining the values of all the plant variables that can affect the fission process, the integrity of reactor core, the reactor coolant system and containment at the nuclear power plant, for obtaining essential information on the plant that is necessary for its safe and reliable operation, for determining the status of the plant in accident conditions and for making decisions for the purpose of accident management.

Instrumentation and recording equipment shall be such that essential information is available to support plant procedures during and following accidents by:
(i) Indicating important plant parameters and radiological conditions.
(ii) Identifying the locations of radioactive material.
(iii) Facilitating decisions in accident management.

Provision of Instrumentation

Instrumentation shall be provided for obtaining essential information on the plant that is necessary for its safe and reliable operation, for determining the status of the plant in accident conditions and for making decisions for the purposes of accident management. It shall enable for determining the values of all the main variables that can affect the fission process, the integrity of the reactor core, the reactor coolant systems and the containment at the nuclear power plant.

Severe Accident Monitoring I&C

For the purpose of accident monitoring and management, appropriate means shall be considered for the plant by which the operating personnel obtain information for event assessment and for the planning and implementation of mitigating actions.
It shall be possible to assess the information about the following:
(i) Condition of core or debris
(ii) Condition of reactor pressure vessel
(iii) Condition of containment
(iv) Condition of spent fuel storage pool
(v) Radiological situation in the plant, site and its immediate surroundings
(vi) Status of implemented accident management measures.
The measurement systems/instrument shall be capable of measuring over the entire range within which the measured parameters are expected to vary during accident conditions.

Requirements for Additional Facilities

It shall include development of accident management techniques that improve the capability of a plant to survive an extended loss of all AC power, loss of normal heat sinks and loss of normal access to plant site, etc., as a result of extreme events. It shall include equipment to respond to such challenges; procedures and guidance; equipment readiness, storage, and transportation;

and training. The increased equipment capability will consist of installed equipment, portable equipment stored onsite, and portable equipment in nearby establishments and other national facilities.

The provisions described in the following sections address requirements considering the possibility of an event that could be more severe than known historical events.

The design should recognize the need to provide accident management capabilities when the onsite and the offsite infrastructure are severely damaged. The approach should be phased, to consider the immediate need to maintain core cooling, spent fuel cooling and containment integrity and the potential need to maintain these capabilities for an extended period.

II. Procedures and instructions

1. Should guidelines and/or procedures be a part of licensing?

EOPs and SAMGs are required by AERB. Generic technical basis documents on SAMGs are reviewed.

AERB/NPP-LWR/SC/D, DESIGN OF LIGHT WATER REACTOR BASED NUCLEAR POWER PLANTS

Severe accident management guidelines shall be prepared, taking into account the plant design features and the understanding of accident progression and associated phenomena.

Equipment that is credited to operate (e.g. certain instrumentation) during design extension conditions and during and after severe accidents scenario shall be shown, with reasonable confidence, to be capable of achieving the intended function under the expected environmental conditions. Severe accident management guidelines should address uncertainties arising from any shortfalls in such qualification of specific equipment/instrument.

AERB/NPP/SC/O (Rev. 1) NUCLEAR POWER PLANT OPERATION
Operating Instructions and Procedures

A comprehensive administrative procedure shall be established for document control including operating instructions and procedures as per code of practice on Quality Assurance for Safety in Nuclear Power Plants (AERB/SC/QA).

Operating instructions and procedures* are required to be established to operate, maintain and manage the NPP in a planned, systematic and safe manner in order to assure:

(i) that all activities affecting safe operation have appropriate instructions or procedures;
(ii) compliance with OLCs and other regulatory requirements;
(iii) consistency with the design intent; and
(iv) management of the plant under abnormal conditions.

(* The use of symptom oriented procedures for dealing with anticipated operational occurrences and for the management of accident conditions is recommended as far as practicable, taking into account all the capabilities of the plant which are available under the respective conditions. Procedures should be written so that each action can be readily performed in a proper sequence by the designated responsible person. There should be proper integration of design provisions, emergency operating procedures, accident management and emergency preparedness plans.)

The operating procedures shall be written and verified by competent persons before approval by designated authority. These procedures shall have detailed instructions for qualified personnel to perform the specified tasks without direct supervision. The procedures shall include sections dealing with plant under normal operation and anticipated operational occurrences as well as appropriate actions for accident conditions including design basis accidents. Emergency operating procedures or guidance shall be developed for managing severe accidents.

The PM shall ensure that detailed operating instructions and procedures are set out in writing and be available for training and qualification of operating personnel before the commencement of operation. Every instruction and procedure shall be carefully evaluated to assess its significance to safety.

They shall comply, with reasonable margins on the safe side; with the established OLCs. The above provisions also apply to any subsequent modification of such instructions and procedures. The PM shall ensure that these instructions and procedures are carefully followed in operating the plant. 7.2.5 Responsibilities and lines of communication shall clearly be set out in writing for those situations in which the operating personnel are forced to deviate from written procedures. No other person shall interfere in their decisions relevant to safety.

Special procedures/tests, which could result in the deviation from approved OLCs shall be carried out only with the prior approval of AERB or its designated agency. During execution of these procedure/test, the overriding authority of the persons who manipulate the reactor controls and who supervise such operation to terminate the procedure and to bring the plant to a safe state shall not be jeopardised.

Adequate arrangements shall be made for the periodic review and revision, if necessary, of all instructions and procedures within a specified period of time in accordance with written procedures.

2. Should licensing cover as prevention and mitigation consequences of severe accident?

Systems and procedures designed for postulated accidents and design extension conditions help in prevention and termination of severe accidents.

AERB/SG/D-26, Safety Guide on Accident Management covers both preventive and mitigation aspects of severe accident.

AERB/NPP-LWR/SC/D, DESIGN OF LIGHT WATER REACTOR BASED NUCLEAR POWER PLANTS

Design Approaches

For the design of safety systems necessary within design basis conditions rigorous safety criteria and conservative engineering practices shall be followed. This includes use of adequate margins, approach of single failure criteria, rigorous quality and qualification requirement for systems required to cater for addressing design basis events or accidents.

For design extension conditions without core melt, additional safety features/systems (other than those provided for DBA), if envisaged, should be diverse from the safety systems for design basis conditions.

In design of complementary systems which are used to prevent or mitigate the consequences of design extension conditions with core melt or severe accident situations that involve large early releases (e.g. early containment failure), the design approach should be to prevent such sequences by significant margins.

AERB/NPP/SG/G-9 (R-2 Draft), STANDARD FORMAT AND CONTENTS OF SAFETY ANALYSIS REPORT FOR NUCLEAR POWER PLANTS

Severe Accident/Design Extension Conditions Evaluations

This part of the SAR should provide a description in sufficient detail of the analysis performed to identify accidents that can lead to significant core damage and/or off-site releases of radioactive material (severe accidents). The challenges to the plant that such events represent and the extent to which the design may reasonably be expected to mitigate their consequences should be considered, justified and referenced here.

The severe accident analysis should generally be carried out using best estimate assumptions, data, methods and decision criteria. Nevertheless reasonably conservative assumptions should be made which take account of the uncertainties in the understanding of the physical processes being modelled and in interpretation of the results in terms of predicted timing and severity of phenomena.

Another issue is connected with assumptions regarding operability of plant systems in case of severe accidents. Consideration of operability of all plant systems even beyond their normal operating range is usually recommended and acceptable for development of severe accident management guidelines, but is very complicated to rely on survivability of systems in demonstrating acceptability of the plant design. In addition, majority of systems would not be available due to complete lack of normal and emergency power supply. It is therefore advisable to demonstrate acceptability of the design using only systems dedicated to severe accident mitigation.

In addition to demonstration of the acceptability of the design, the results of the relevant severe accident analyses used in the development of the accident management programmes and emergency preparedness planning for the plant should be specified and presented. The accident management measures that could be carried out to mitigate the accidents' effects, and also to provide input for emergency planning and preparedness, should have been identified and optimized in the severe accident analysis. Reference should be made to those relevant chapters of the SAR in which these results are used.

Identification of Severe Accident Scenarios

The following should be addressed in this section

a) Bases

b) List of scenarios

Severe Accident Prevention

Provide a deterministic evaluation to show how the plant's severe accident preventive features would cope with the events.

Severe Accident Mitigation

The following should be addressed in this section

a) Describe Severe Accident Progression, both In-and Ex-Vessel.

b) Describe Severe Accident Mitigation Features for external reactor vessel cooling, hydrogen generation and control, core debris coolability, high-pressure melt ejection(as applicable), fuel-coolant interactions, core concrete interaction, containment bypass (including steam generator tube rupture and intersystem LOCA), equipment survivability, and other severe accident mitigation features.

Containment Performance Capability

This section should provide an overview of the containment design and address the containment performance goals identified

Accident Management

Describe those actions taken (as per AMG) during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases.

This should also include provisions regarding on-site emergency support centre with adequate infrastructure and training of plant operating and technical staff.

Consideration of Potential Design Improvements

Any consideration for potential design improvements that are planned at this stage should be brought out in this section.

3. Should structure and content of the procedure and instructions be a part of regulatory control?

SAMG submissions include the technical basis document prepared by designers/utility which includes the severe accident analysis and the symptom based SAMG, this is a generic document for the given type of NPP. Based on these documents the SAM guidelines for the individual NPPs are prepared with plant specific inputs. Both these submissions for severe accident management guidelines are reviewed in multitier manner at AERB, following the submission by the utility. If necessary, AERB can require changes to guidelines based on the review recommendations.

The draft AERB/SG/D-26, Safety Guide on Accident Management provides guidance on the contents of the technical basis document.

4. How deep (in detail) should be monitored structure and content of procedures and/or instructions? Please consider the following items: entry criteria; exit criteria; criteria for the transition from prevention to mitigation domain of SA procedures and/or instructions; harmonization procedure and/or instructions dedicated for SAM with other emergency procedures and instructions; coverage of such SAM aspects as multi-unit accidents, different locations of the fuel (reactor' spent fuel pool, on-site fuel storage); distribution of the responsibility during the application of emergency procedure and/or instructions in the course of SA (who make a decision?);
i) Safety guide on accident management, AERB/SG/D-26 is under preparation. The development is in similar lines with the IAEA documents and in particular to the IAEA/NSG/2.15 and IAEA DS483.

The review covers all the aspects related to

- Supporting Analysis
- Entry criteria
- Exit criteria
- Criteria for the transition from prevention to mitigation domain of SA procedures and/or instructions
- Time available and Training of the operators
- Instrumentation and computational aids
- Harmonization procedure and/or instructions dedicated for SAM with other emergency procedures and instructions;
- Coverage of such SAM aspects as multi-unit accidents,
- Different locations of the fuel (reactor' spent fuel pool, on-site fuel storage);
- Responsibilities of the operating organisation during the implementation of emergency procedure and/or instructions in the course of SA;

ii) Multi-Unit Accident
AERB/NF/SC/S (Rev.1): SITE EVALUATION OF NUCLEAR FACILITIES
2.1 (iii) For a multi-unit/multi-facility site, consequences of external events shall be assessed/reassessed considering their impact on all units/ facilities in the site, including common cause failures. Consequential effects due to incidences in one facility/unit on other facilities/units shall also be considered.
5.(v) In a multi-unit/multi-facility site, considerations shall be given to emergencies arising out of common cause failures due to external events.
3) In India, functions attributed to the Technical Support Centre are carried out by the Plant Advisory Group. Assembly place of Plant Advisory Group will initially be the main control centre and later on, this group can shift to Onsite Emergency Support Centre (OESC). On-site actions are coordinated and guided by the Plant Advisory Group as well as inputs from the utility headquarters.
Decisions in severe accident are made by the control room and by plant emergency managers. Technical support centre has only a supporting role.

Further, the document *Safety guide on accident management, AERB/SG/D-26 which is under preparation would cover the guidance in detail for multi-unit sites.*

5. How the operator must confirm the correctness and sufficiency of the procedures and instructions? Should analyses and/or experiments be an obligatory part of validation and verification process for procedures and instructions?

The operating procedures/SAMGs prepared for AOOs, DBAs and severe accidents should be verified and validated to ensure that they are correct and time available/time required for the operator to take actions are verified/validated through the safety analysis .
The supporting analysis with and without considering the SAMG features are performed by the utility, which is reviewed and independently verified by AERB. Analytical inputs are considered where clear experimental inputs are not available. Well established and validated analytical studies are desired If the analytical studies show lot of uncertainties and undesired assumptions, then further research emphasizing experimental studies are also recommended by AERB. These aspects are part of the Safety guide on SAMG, AERB/SG/D-26 which is under preparation. The procedures (SAMG) are validated based on whether the operators are understood and perform the tasks in required manner and are able to perform the tasks in the available time mentioned in the SAMGs which might be obtained from the safety analysis.

III. Equipment

1. Should a minimal list of technical means dedicated for SAM be determined by regulator?

The technical means based on the design of the NPPs are described and highlighted by the utility in the submissions which are reviewed by AERB. The technical means can be design specific taking into account the plant design features and the understanding of accident progression and associated phenomena with the aim of mitigating severe accident and maintaining containment integrity. However, some major requirements like availability of alternate heat sink, seismically qualified water tanks, power supply means for prolonged SBO, air cooled DGs, systems for hydrogen management are brought out in AERB safety code (AERB/NPP-LWR/SC/D). The utility proposals are extensively reviewed and approved by AERB. The regulatory requirements are specified in the AERB documents to ensure safety through successful SAM actions to avoid and mitigate severe accidents.

2. Should severe accident management provisions (i.e. using mobile equipment verses hardening on site equipment) be performance based or prescribed by regulation?

The regulations in India require effective severe accident management provisions to ensure safety. For DEC-B, severe accident management provisions are not explicitly prescribed by the regulator but their performance shall be demonstrated through safety analysis which is reviewed by AERB. The performance of the SAM provisions have to meet the prescribed regulatory requirements to prevent and mitigate severe accidents. Mobile equipment (on site) are also credited. SAM-systems are desired to be independent, single failure tolerant and safety classified.

The following is reproduced from the AERB safety code for LWRs:

“In designing additional safety systems/features or complementary safety features for preventing the design extension conditions and to mitigate the consequence of such scenario the possibility of application of the single failure criterion shall be explored. However, emphasis shall be to provide diversified backup systems and consideration should be given to the repair and replacement potential should a failure occur”.

3. Should be installed special requirements (regulations) in respect of mission times and minimum capacity for equipment dedicated for SAM?

Mission time and minimum capacity are based on the NPP design specific safety systems that are part of SAM measures. These are reviewed based on the availability and the nature of ultimate heat sink along with requirements specified in various AERB safety documents (See the reply to Q2). The alternate heat sink should be available to remove decay heat for a period 7 days and beyond. Adequate quantity of onsite storage of water shall be available for decay heat removal from core and spent fuel under all plant states for at least 7 days. (AERB/NPP-LWR/SC/D).

4. Should be special requirements (regulations) be installed for using mobile equipment versus stationary installed equipment?

Mobile equipment (e.g. fire trucks and diesel driven pumps) can be used as alternative backup for permanent installations, which are required for handling severe accidents. Such mobile equipment should be qualified for usage under all possible adverse conditions. These equipment should be accessible and available during assumed external events. Generally covered by demonstrating availability under postulated scenario.

5. Should be a special requirements (regulations) related to I&C for SA (I&C for SAM and indicate the status of SA progress)?

I & C systems design enhancement shall ensure equipment functionality to remain capable of monitoring plant conditions (essential plant parameters) under extreme environmental conditions associated with severe accident. Provision of alternative power supplies for essential instrumentation (supplies and connection capability) should be ensured (AERB/NPP-LWR/SC/D).
The measurement systems/instrument shall be capable of measuring over the entire range within which the measured parameters are expected to vary during accident conditions.
Equipment that is credited to operate (e.g. certain instrumentation) during design extension conditions and during and after severe accidents scenario shall be shown, with reasonable confidence, to be capable of achieving the intended function under the expected environmental conditions. Severe accident management guidelines should address uncertainties arising from any shortfalls in such qualification of specific equipment/instrument.

6. Should be special requirements (regulations) related hydrogen management in the containment under the SA conditions?

Containment system shall include:

- Features for management and removal of fission products, hydrogen, oxygen, and other substances that may be released into the containment atmosphere.
- The design shall consider containment response for pressure and temperature build-up expected during postulated Design Extension Conditions with core melt. Consideration shall be given to: potential for generation and behaviour of inflammable gases like hydrogen. Hydrogen detonations need to be analyzed and eliminated by design.

7. Should be special requirements (regulations) for primary pressure decrease?

Provision shall be made to ensure that the operation of pressure relief devices will protect the pressure boundary of the reactor coolant systems against overpressure, and will not lead to the release of radioactive material from the nuclear power plant directly to the environment. There is no explicit requirement on pressure reduction under severe accident condition, but it is required that the large early releases are to be practically eliminated. It indirectly requires that the primary should be depressurised so as to avoid the early containment failure. The pressure relief system for severe accidents in general shall be independent and single failure criterion is desirable.

8. Should be special requirements (regulations) related to auxiliary equipment to provide water injection into the reactor vessel, steam generators and containment?

There are no special requirements related to the use of auxiliary equipment in severe accident. It must be possible to fulfil the needed water injection functions (in primary, containment and core catcher) in SA.

9. Should be special requirements (regulations) cover the special technical provisions dedicated for recovery of off-site power, multi-units considerations, treatment of liquid radioactive effluents.

The emergency power supply at the nuclear power plant shall be capable of supplying the necessary power in anticipated operational occurrences and accident conditions, in the event of the loss of off-site power.

Safety systems and additional safety systems, required for DBAs and design extension conditions without a core melt scenario, shall not be shared and interconnected between multiple units, unless this contributes to enhanced safety. Capability of complementary safety systems, their support systems and onsite resource requirements for mitigating design extension condition with core melt scenario, shall be such that simultaneous handling of such events at all the reactors at a multiunit site is possible.

Systems shall be provided at the nuclear power plant for treating liquid and gaseous radioactive effluents to keep their amounts below the authorised limits on discharges and as low as reasonably achievable.

IV. Methodology of severe accident analyses

1. Should the list of SA analyses be a subject of licensing?
The list of PIEs to be considered by the utility is approved by AERB during the review of PSAR/FSAR.

AERB Safety Guide, AERB/SG/D-19 on Deterministic Safety Analysis is in the advanced stage of completion. AERB Safety Guide AERB/SG/D-5 lists out the typical list of postulated initiating events for PHWRs including multiple failures. This guide is also being revised. These two guides provide guidance in selection and grouping of PIEs.

2. How the completeness and representativeness of SA analyses could be confirmed (bounding criteria)?

AERB/NPP-LWR/SC/D
A systematic approach shall be adopted during the design of the nuclear power plant to identify a comprehensive set of postulated initiating events such that all foreseeable events with a significant frequency of occurrence and all foreseeable events with the potential for significant radiological consequences are anticipated and are considered in the design basis or in the design extension condition.
The postulated initiating events shall be identified on the basis of engineering judgement and a combination of deterministic assessment and probabilistic assessment. A justification shall be provided, to show that all foreseeable events have been considered.

The postulated initiating events shall include all foreseeable failures of structures, systems and components of the plant, as well as operating errors and possible failures arising from internal and external hazards, whether at full power, low power, refuelling or shutdown states.
An analysis of the postulated initiating events for the plant shall be made to establish the preventive measures and protective measures that are necessary to ensure that the required safety functions will be performed.

3. What acceptance criteria are applicable for severe accident?

AERB/NPP-LWR/SC/D: In the design extension condition with core melt, the containment system and its safety features shall be able to perform in extreme scenarios that include, among other things, melting of the reactor core. Containment shall maintain its role as a leak-tight barrier for a period that allows sufficient time for the implementation of off-site emergency procedures following the onset of core damage. Containment shall also prevent uncontrolled releases of radioactivity after this period. Severe accident management guidelines shall be prepared, taking into account the plant design features and the understanding of accident progression and associated phenomena.
The design shall be such that design extension conditions that could lead to large or early releases of radioactivity are practically eliminated. For design extension conditions that cannot be practically eliminated, only protective measures that are of limited scope in terms of area and time shall be necessary for protection of the public, and sufficient time shall be made available to implement these measures.
Severe accident safe state is a state which shall be achieved subsequent to a design extension condition with significant core damage or core melt phenomena. Severe accident safe state shall be reached at the earliest after an accident initiation. It should be possible to maintain this state indefinitely. During this state there is: (a) No possibility of re-criticality. (b) Fuel or debris are continuously cooled. (c) Uncontrolled release of radioactivity to environment is arrested. (d) Means to maintain above conditions are available for long term, including critical parameter monitoring. (e) Monitoring of radiological releases and containment conditions. As the plant state is in design extension condition with core melt (severe accident), the severe accident safe state should be preferably reached within about one week from accident initiation.
Safety targets for different accidents conditions are as below: (a) Limits on core damage frequency (CDF) should be 1E-6/reactor-year due to internal events, for power and shutdown states. (b) The cumulative core damage frequency should be less than 1E-5/ reactor-year for all internal events and external hazards including seismic hazards. (c) Accident sequences with core melt which would lead to early or large release have to be practically eliminated. Quantitative target for the early or large release shall be less than 1E-7 per year.
The requirements related to dose criteria is spelt in the AERB Safety Codes on Design and Siting which is as follows: Design extension condition with core melt (severe accident) In case of severe accident e.g. accidents with core melt within design extension conditions, the release of radioactive materials should cause no permanent relocation of population. The need for off-site interventions should be limited in area and time.

4. Should the codes dedicated for SA analyses (neutronic, thermohydraulic, containment, source term) be under licensing control? If yes, what are requirements for validation and verification of these codes?

The verification and validation aspects are reviewed by AERB. AERB Safety Guide on Deterministic Safety Analysis, under revision has the following proposed guidelines:

Verification
It is recommended that utility should have mechanisms for verification of computer codes to ensure that the code correctly performs all the intended functions and does not perform any unintended function. In general, the verification of the code design should ensure that the numerical methods, the transformation of the numerical equations into a numerical scheme to

provide solutions, and user options and their restrictions are appropriately implemented in accordance with the design requirements. The verification of the code design should be performed by means of review, inspection and audit. The verification of the code design should include a review of the design concept, basic logic, flow diagrams, numerical methods, algorithms and computational environment. If the code is run on a hardware or software platform other than that on which the verification process was carried out, the continued validity of the code verification should be assessed. The code design may contain the integration or coupling of codes. In such cases, verification of the code design should ensure that the links and/or interfaces between the codes are correctly designed and implemented to meet the design requirements. Comparisons with independent calculations should be carried out where practicable to verify that the mathematical operations are performed correctly. The tracking of errors and reporting of their correction status should be a continuous process and should be a part of code maintenance. The impacts of such errors on the results of analyses that have been completed and used as part of the safety assessment for a plant should be assessed.

Validation

Computer code validation should be performed and documented for all computer codes that are used for the deterministic safety analysis of nuclear power plants. The purpose of validation is to provide confidence in the ability of a code to realistically predict the safety parameter(s) of interest. If code is upgraded by improving changing the models of the code, appropriate validation should be carried out. Adequate documentation should be maintained for change in the version of code.

For validation of computer codes, following approaches are acceptable

(i) computational checks: checking of individual model against analytical solutions or with exiting correlations derived from experimental data wherever possible.

(ii) separate effect test: Separate effect tests addresses specific phenomena that may occur on a nuclear power plant but the test does not address the other phenomena that may occur at the same time.

(iii) integral test: Integral test are directly related to a nuclear power plant. All or most of the relevant physical process are represented. However these tests are may be at reduced scale, use substitute material or be performed at low pressure.

(iv) operational transients: Operational transients occur either in an actual nuclear power plant or an experimental rig which represents the plant at full scale and in realistic conditions. Validation through operational transients together with NPP tests is crucial to qualify the plant model. Though it is noted that data from actual operational transients are subject to measurement as available at the time of incident.

(v) inter code comparisons

(vi) Solving the standard/benchmark problem.

(vii) Commissioning data and Operational data

Computer code validation should be properly documented and validation report should be referenced in utility submissions for licensing.

Appendix C – Regulatory approaches used in Russian Federation

QUESTIONNAIRE ON SEVERE ACCIDENT REQUIREMENTS AND PRACTICES

I. General

1. What types of regulatory controls are installed in respect of severe accident (i.e. licensed verses voluntary initiatives)?

There are regulatory requirements In respect of several accidents (requirements set by NP-001-15 address SAR and SAMG.

2. What is the legal basis this matter (laws, requirements, regulations?)

NP-001-15 sets the requirements for severe accidents probability, large radioactive release probability, general requirements to SAR, general requirements to SAMG.

NP-006-98 sets the requirements for SAR in detail.

NP-026-16 sets the requirements for I&C.

NP-040-02 sets the requirements for the hydrogen explosion safety.

RD-03-34-2000 sets the requirement to the verification report for safety analysis codes.

RB-102-15 provides recommendations on SAMG structure and contents.

Administrative Regulations cover the licensing process, defines the set of documents for the license application.

II. Procedures and instructions

1. Should guidelines and/or procedures be a part of licensing?

Yes. NP-001-15 requires that SAMG (as a part of BDBA MG) shall be developed and Administrative Regulations requires that this document shall be provided in a set of documents for license application.

2. Should licensing cover as prevention and mitigation consequences of severe accident?

Yes. NP-001-15 (p.1.2.4) requires both measures for prevention and consequences mitigation as a parts of Defence-in-Depth strategy.

3. Should structure and content of the procedure and instructions be a part of regulatory control?

There are general requirements on content of the SAMG set by p.1.2.16, 4.1.5 NP-001-15. Detailed structure and content recommendations provided in RB-102-15.

4. How deep (in detail) should be monitored structure and content of procedures and/or instructions? Please consider the following items: entry criteria; exit criteria; criteria for the transition from prevention to mitigation domain of SA procedures and/or instructions; harmonization procedure and/or instructions dedicated for SAM with other emergency procedures and instructions; coverage of such SAM aspects as malty-unit accidents, different locations of the fuel (reactor' spent fuel pool, on-site fuel storage); distribution of the responsibility during the application of emergency procedure and/or instructions in the course of SA (who make a decision?);

In SAMG there should be specified:

- Entry and exit criteria (p.4.3.1 RB-102-15),
- Transition criteria (p.4.3.2 RB-102-15),
- Procedures/instructions that shall no longer be used after entry to the SAMG (p.4.5.1 RB-102-15)
- SAMG should cover multi-units accidents (if applicable) (p.2.12 RB-102-15)
- SAMG should cover all fuel and radioactive materials locations (reactor, spent fuel pool, on-site spent fuel storage, transport containers, radioactive waste storages etc) (p.2.10 RB-102-15)
- Distribution of the responsibility SAMG (p.4.2.1, 4.2.2 RB-102-15).

5. How the operator must confirm the correctness and sufficiency of the procedures and instructions? Should analyses and/or experiments be an obligatory part of validation and verification process for procedures and instructions?

SAMG shall be based on SAR (p.1.2.16, 4.1.5 NP-001-15). SAR shall contain analysis of BDBA (including SA) management strategies (p.15.6.3 NP-006-98). Correctness and sufficiency of the procedures should be confirmed by the technical rationale (p.4.2 RB-102-15).

III. Equipment

1. Should a minimal list of technical means dedicated for SAM be determined by regulator?

There is a requirement that the technical means for the SA shall exist (p. 1.2.4, 1.2.11, 1.2.18 NP-001-15) and there is a special requirement that the dedicated technical means for station blackout and loss of ultimate heat sink accidents shall exist (p.3.1.4 NP-001-15). The exact means is not specified by the regulation but left to the designer.

2. Should severe accident management provisions (i.e. using mobile equipment verses hardening on site equipment) be performance based or prescribed by regulation?

SAM provisions are performance based: they have to provide sufficient level of safety and larger radioactive release probability (p. 1.2.4, 1.2.11, 1.2.18 NP-001-15).

3. Should be installed special requirements (regulations) in respect of mission times and minimum capacity for equipment dedicated for SAM?

The mission time and equipment capacity shall be enough to fulfil the requirements (p. 1.2.4, 1.2.11, 1.2.18 NP-001-15). There are no specific requirements.

4. Should be special requirements (regulations) be installed for using mobile equipment versus stationary installed equipment?

There are no special requirements for using mobile equipment versus stationary installed equipment, but there are requirements to the dedicated technical means for some types of accidents (p.3.1.4 NP-001-15) that at the current level of science and technology could be fulfilled only by mobile equipment.

5. Should be a special requirements (regulations) related to I&C for SA (I&C for SAM and indicate the status of SA progress)?

There are no special requirements related to the I&C for SA currently installed. General requirements set by NP-001-15 for I&C are applied, and p.3.1.5 also requires that there shall be I&C enough for the SA management. The proposed this year NP-026-16 requires that the I&C for the SA shall be sufficient to provide the state of the main safety functions and to allow SA management. The sufficiency shall be proved in the design. The information about the controlled parameters provided by I&C for the SA shall be available during the whole accident and beyond. The I&C for the SA shall be designed to withstand simultaneous accident on all units of the plant. The power supply for the I&C for the SA shall be available during the time which shall be justified in the design. The I&C for the SA shall be independent from the normal I&C and the safety I&C.

6. Should be special requirements (regulations) related hydrogen management in the containment under the SA conditions?

Hydrogen explosion safety analyses must be provided in SAR (p.3.3 NP-040-02).

Hydrogen detonation shall be eliminated (p.2.1 NP-040-02).

Hydrogen deflagration is allowed during SA if is proved that the containment safety functions are not affected by it (p.2.1 NP-040-02).

The prevention measures against hydrogen detonation and deflagration shall be provided (p.2.4 NP-040-02), as well as mitigation measures for the detonation and deflagrations (p.2.5 NP-040-02).

7. Should be special requirements (regulations) for primary pressure decrease?

There are no special requirements for primary pressure decrease.

8. Should be special requirements (regulations) related to auxiliary equipment to provide water injection into the reactor vessel, steam generators and containment?

There are no special requirements related to the use of auxiliary equipment in severe accident but there is a requirement that auxiliary equipment dedicated for the station blackout and loss of ultimate heat sink accident shall be provided (p.3.1.4 NP-001-15).

9. Should be special requirements (regulations) cover the special technical provisions dedicated for recovery of off-site power, multi-units considerations, treatment of liquid radioactive effluents.

There shall be technical means for the station blackout and loss of ultimate heat sink accidents (p.3.1.4 NP-001-15). The SAMG should take multi-unit accidents into consideration (p.2.12 RB-102-15). The radioactive waste storages should be taken into consideration in SAMG.

IV. Methodology of severe accident analyses

1. Should the list of SA analyses be a subject of licensing?

Yes, SAR shall include analyses of the severe accidents, the preliminary list of BDBA is provided in regulatory documents and final list of BDBA including SA shall be justified in SAR (p.1.2.16 NP-001-15, p.15.2 NP-006-98).

2. How the completeness and representativeness of SA analyses could be confirmed (bounding criteria)?

The completeness and representativeness of SA analyses shall be proven is SAR.

3. What acceptance criteria are applicable for severe accident?

The total probability of the SA in the reactor and unit SFP shall be less than 10^{-5} unit/year (p.1.2.17 NP-001-15)

The total probability of the large radioactive release shall be less than 10^{-7} unit/year (p.1.2.17 NP-001-15)

The total probability of the SA in fuel storages outside the plant units shall be less than 10^{-5} site/year (p.1.2.17 NP-001-15)

4. Should the codes dedicated for SA analyses (neutronic, thermohydraulic, containment, source term) be under licensing control? If yes, what are requirements for validation and verification of these codes?

The codes for the safety analyses (including SA analyses) shall be validated and verified (p.1.2.9 NP-001-15). Requirements for the structure and content of the verification report are specified by RD-03-34-2000.

Appendix D – Regulatory approaches used in Turkey

QUESTIONNAIRE ON SEVERE ACCIDENT REQUIREMENTS AND PRACTICES

I. General

1. What types of regulatory controls are installed in respect of severe accident (i.e. licensed verses voluntary initiatives)?

2. What is the legal basis this matter (laws, requirements, regulations)?

List of Licensing Basis for Akkuyu Nuclear Power Plant:

A. Requirements

1. Turkish National Regulations
    i. TAEK Regulations
    ii. Other Turkish Regulations
2. IAEA Fundamentals and Requirements
3. Russian Federation Regulations
4. Third Party Regulations

B. Standards

1. Turkish National Standards
2. APC Proposed Standards

C. Guides

1. Turkish National Guides
2. IAEA Guides
3. Russian Federation Guides
4. Third Party Guides

Turkish National Regulations:

General requirements are given in the “Decree on Licensing of Nuclear Installations, No: 83/7405, 18/11/1983”

There are not any detailed requirements issued by TAEK.

IAEA Fundamentals and requirements:

SF-1, Fundamental Safety Principles,

GSR Part 4 Safety Assessment for Facilities and Activities General Safety Requirements Part 4, Revision 1, Series No. (draft DS462-5, 06-11-2014)

SSG-2 Deterministic Safety Analysis for Nuclear Power Plants Specific Safety Guide

SSR-2/1 Safety of Nuclear Power Plants Design

NS-G-2.15 Severe Accident Management Programmes for Nuclear Power Plants Safety Guide

NS-G-1.9 Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety Guide

NS-G-1.10 Design of Reactor Containment Systems for Nuclear Power Plants Safety Guide

NS-G-1.11 Protection against Internal Hazards other than Fires and Explosions in the Design of Nuclear Power Plants Safety Guide

NS-G-1.12 Design of the Reactor Core for Nuclear Power Plants Safety Guide

NS-G-1.13 Radiation Protection Aspects of Design for Nuclear Power Plants Safety Guide

<u>Russian Federation Regulations</u>

OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

NP-006-98 To Contents of Safety Analysis Report Of NPP with VVVER Reactors

NP-040-02 Regulations for Ensuring Hydrogen Explosion Protection at a Nuclear Power Station

RD-03-34-2000 Requirements to Contents and Composition of Verification Report and Justification of SW Applied for Justification of Nuclear Power Utilization Facilities Safety

NP-032-01 Nuclear Power Plant Siting. Main Criteria and Safety Requirements

NP-082-07 Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants

II. Procedures and instructions

1. Should guidelines and/or procedures be a part of licensing?

IAEA NS-G 1.10, Design of Reactor Containment Systems for Nuclear Power Plants

6.29. During and following a severe accident, in order to follow the general conditions in the containment and to facilitate the use of guidelines for the management of severe accidents, essential parameters for the containment such as pressures, temperatures, hydrogen concentrations, water levels and radiation dose rates should be monitored.

6.34. Guidelines for the management of severe accidents (severe accident management guidelines (SAMGs)) should be aimed primarily at maintaining or restoring the performance of the containment. SAMGs should be developed for managing accident conditions in co-ordination with on-site and off-site emergency organizations. SAMGs should be established to supplement, but not to replace, provisions in the design to prevent the failure of containment systems during or following a severe accident or to mitigate the consequences of such an accident.

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

2.17. Severe accident management should cover all modes of plant operation and also appropriately selected external events, such as fires, floods, seismic events and extreme weather conditions (e.g. high winds, extremely high or low temperatures, droughts) that could damage large parts of the plant. In the severe accident management guidance, consideration should be given to specific challenges posed by external events, such as loss of the power supply, loss of the control room or switchgear room and reduced access to systems and components.

3.32. The strategies and measures discussed in the previous section should be converted to procedures for the preventive domain (EOPs) and guidelines for the mitigatory domain (SAMGs). The procedures contain a set of actions to prevent the escalation of an event into a severe accident. The guidelines contain a set of actions to mitigate the consequences of a severe accident according to the chosen strategies. Procedures and guidelines contain the necessary information and instructions for the responsible personnel, including the use of equipment, equipment limitations, and cautions and benefits. The guidelines also address the various positive and negative consequences of proposed actions and offer options.

3.33. The procedures and guidelines should contain the following elements:

— Objectives and strategies;
— Initiation criteria;
— The time window within which the actions are to be applied (if relevant);
— The possible duration of actions;
— The equipment and resources (e.g. AC and DC power, water) required;
— Actions to be carried out;
— Cautions;
— Throttling and termination criteria;
— Monitoring of plant response.

3.34. The set of procedures and guidelines should include a logic diagram that describes a sequence of relevant plant parameters that should be monitored and which are linked to the criteria for initiation, throttling or termination of the various procedures and guidelines. The sequence should be in line with the priority of associated strategies, procedures and guidelines, as described in paras 3.27 and 3.39

3.45. Procedures and guidelines should be based on directly measurable plant parameters. Where measurements are not available, parameters should be estimated by means of simple computations and/or precalculated graphs. Parameters that can be obtained only after carrying out complex calculations during the accident should not be used as the basis for decisions.27

3.46. Procedures and guidelines should be written in a user friendly way and such that they can be readily executed under high stress conditions, and should contain sufficient detail so as to ensure that the focus is on the necessary actions.28 The procedures and guidelines should be written in a predefined format. Instructions to operators should be clear and unambiguous.

3.48. Procedures and guidelines should contain guidance for situations where the accident management equipment may be unavailable (e.g. because of equipment failure or equipment lockout). Alternate methods should be explored and, if available, included in the guidance.

3.53. In the development of procedures and guidelines, account should be taken of the habitability of the control room and the accessibility of other relevant areas, such as the technical support centre or areas for local actions. It should be investigated whether expected dose rates and environmental conditions inside the control room and in other relevant areas may give rise to a need for restrictions for personnel. It should be determined what the impact of such situations will be on the execution of the accident management programme; the need for replacement of staff for reasons of dose should also be considered.

IAEA SSG-2 Deterministic Safety Analysis for Nuclear Power Plants

5.14. For severe accidents, the operator emergency procedures and severe accident management guidelines should be assessed in addition to the objective of showing compliance with the acceptance criteria. These analyses should include the use of all the systems or components that are available to mitigate the consequences of the accident, and they should be based on the best available knowledge. Some regulatory authorities require the licensee to demonstrate that a release criterion for a severe accident is met under the assumption that within a prescribed time period the operator does not take any action.

8.21. Deterministic safety analyses should also be performed to assist the development of the strategy that an operator should follow if the emergency operating procedures fail to prevent a severe accident from occurring. The analyses should be carried out by using one or more of the specialized computer codes that are available to model relevant physical phenomena. For light water reactors, these phenomena include thermohydraulic effects, heating and melting of the reactor core, retention of the molten core in the lower plenum, interactions between molten core and concrete, steam explosions, hydrogen generation and combustion, and fission product behaviour.

2. Should licensing cover as prevention and mitigation consequences of severe accident?

IAEA SSR-2/1 Safety of Nuclear Power Plants Design

Requirement 7: Application of defence in depth

The design of a nuclear power plant shall incorporate defence in depth. The levels of defence in depth shall be independent as far as is practicable.

4.9. The defence in depth concept shall be applied to provide several levels of defence that are aimed at preventing consequences of accidents that could lead to harmful effects on people and the environment, and ensuring that appropriate measures are taken for the protection of people and the environment and for the mitigation of consequences in the event that prevention fails.

4.13. The design shall be such as to ensure, as far as is practicable, that the first, or at most the second, level of defence is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the nuclear power plant.

IAEA NS-G 1.10, Design of Reactor Containment Systems for Nuclear Power Plants

6.4. For existing plants, the phenomena relating to possible severe accidents and their consequences should be carefully analysed to identify design margins and measures for accident management that can be carried out to prevent or mitigate the consequences of severe accidents. For these accident management measures, full use should be made of all available equipment, including alternative or diverse equipment, as well as of external equipment for the temporary replacement of design basis components. Furthermore, the introduction of complementary equipment should be considered in order to improve the capabilities of the containment systems for preventing or mitigating the consequences of severe accidents.

I–30. Mitigation of severe accidents is achieved mainly by:

— A primary depressurization device that prevents containment bypasses via the steam generator tubes and failure of the reactor pressure vessel at high pressure, and thereby minimizes the consequences of missiles in the reactor pressure vessel and direct containment heating;

— Passive autocatalytic recombiners which prevent global detonation of hydrogen as well as local fast deflagration and the deflagration– detonation transition in combination with steam inerting, and the possibility of passive global convection within the containment;

— A core catcher in the molten core spreading compartment which stabilizes the material after temporary retention within the reactor pit by passive flooding and cooling with water from the in-containment water storage tank;

— An active containment heat removal system that ensures long term cooling of the containment atmosphere and of molten core material;

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

2.19. Design features important for the prevention or mitigation of severe accidents should be evaluated. Accordingly, existing equipment and/or instrumentation should be upgraded or new equipment and/or instrumentation should be added, if necessary or considered useful 7 for the development of a meaningful severe accident management programme, i.e. a severe accident management programme that reduces risks in an appreciable way or to an acceptable level. The decision to add or upgrade equipment may depend on cost–benefit considerations.

IAEA SSG-2 Deterministic Safety Analysis for Nuclear Power Plants

2.10. Beyond design basis accidents, including severe accidents, are typically treated separately in deterministic safety analyses, although some initiating events may be the same as for design basis accidents. The results help to determine the necessary measures to prevent severe accidents and to mitigate their radiological consequences if they do occur.

8.24. The measures can be broadly divided into preventive measures and mitigatory actions. Both categories should be subject to analysis.

8.25. Preventive measures are recovery strategies to prevent core damage. They should be analysed to investigate what actions are possible to inhibit or delay the onset of core damage. Examples of such actions are: various manual restorations of systems; primary and secondary feed and bleed; depressurization of the primary or secondary system; and restarting of the reactor coolant pumps. Conditions for the initiation of the actions should be specified, as should criteria for when to stop the actions or to change to another action.

8.26. Mitigatory measures are strategies for managing severe accidents to mitigate the consequences of core melt. Such strategies include: coolant injection into the degraded core; depressurization of the primary circuit; operation of containment sprays; and use of the fan coolers, hydrogen recombiners and filtered venting that are available in the reactors of different types that are in operation or being constructed. Possible adverse effects that may occur as a consequence of taking mitigatory measures should be taken into account, such as pressure spikes, hydrogen generation, return to criticality, steam explosions, thermal shock or hydrogen deflagration or detonation.

RF NP-006-98 To Contents of Safety Analysis Report of NPP with VVVER Reactors

1.7/11. Information on beyond design basis accidents: a list beyond design basis accidents considered; measures mitigating beyond design basis accident consequences; severe accident management measures.

1.11/3. While assessing NPP impact to the environment a list of all technical and organizational measures to prevent or mitigate negative NPP impact to the biosphere shall be considered and presented.

7.2.1.5 Should the calculations input information and analysis be connected with personnel actions, analysis results regarding the impact of erroneous personnel actions to safety as well as information on I&C, equipment installed to prevent or mitigate the consequences of normal operation condition violations and accidents shall be incorporated.

15.6.3. Measures for beyond design basis accident management

15.6.4. Assessment of efficient measures proposed to manage beyond design basis accident Herein, demonstrate through computation that the implementation of corrective action strategy under beyond design basis accidents caused by any of vulnerable points at all accident severity levels ensures either termination of emergency process evolution or sound mitigation of accident consequences.

15.6.5. Conclusion

On the basis of the information provided for in Section 6 the conclusions shall be made on the possibility and efficiency of the measures developed to manage beyond design basis accidents.

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.4. During normal operation all physical barriers shall be operable, and measures to be undertaken for their protection shall be ready available. On detection of loss of functions by anyone of physical barriers envisaged in the design or unavailability of measures for its protection the RP shall be shutdown and measures shall be taken to bring the NPP to safe condition.

3. Should structure and content of the procedure and instructions be a part of regulatory control?

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.16 Tentative lists of initiating events preceding design basis accidents and list of beyond design basis accidents (BDBA) including initiating events, sequence paths and consequences shall be specified in regulatory documents for each type of reactor. They shall include representative scenarios with severe consequences for defining the plan of possible response.

The final lists of BDBA, their realistic (not conservative) analysis containing assessment of probabilities of BDBA accident sequence paths including accidents involving core meltdown, consequences of BDBA, functioning of safety systems shall be established in the NPP design and presented in the NPP Safety Analysis Report.

Should the analysis of BDBA consequences not confirm that i. 1.2.17 is met, it would be necessary to provide in the design additional technical approaches to accident management for the purpose of mitigating their consequences.

The analysis of BDBA consequences presented in the NPP design is a basis for drawing up emergency planning procedures on personnel and population protection in the event of an accident and elaborating the manual on accident management.

4. How deep (in detail) should be monitored structure and content of procedures and/or instructions? Please consider the following items: entry criteria; exit criteria; criteria for the transition from prevention to mitigation domain of SA procedures and/or instructions; harmonization procedure and/or instructions dedicated for SAM with other emergency procedures and instructions; coverage of such SAM aspects as malty-unit accidents, different locations of the fuel (reactor' spent fuel pool, on-site fuel storage); distribution of the responsibility during the application of emergency procedure and/or instructions in the course of SA (who make a decision?);

IAEA SSG-2 Deterministic Safety Analysis for Nuclear Power Plants

8.26. Mitigatory measures are strategies for managing severe accidents to mitigate the consequences of core melt. Such strategies include: coolant injection into the degraded core; depressurization of the primary circuit; operation of containment sprays; and use of the fan coolers, hydrogen recombiners and filtered venting that are available in the reactors of different types that are in operation or being constructed. Possible adverse effects that may occur as a consequence of taking mitigatory measures should be taken into account, such as pressure spikes, hydrogen generation, return to criticality, steam explosions, thermal shock or hydrogen deflagration or detonation.

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.3. The accident management guidance should address the full spectrum of credible challenges to fission product boundaries due to severe accidents, including those arising from multiple hardware failures, human errors and/or events from outside, and possible physical phenomena that may occur during the evolution of a severe accident (such as steam explosions, direct containment heating and hydrogen burns). In this process, issues should also be taken into account that are frequently not considered in analyses, such as additional highly improbable failures and abnormal functioning of equipment.

5. How the operator must confirm the correctness and sufficiency of the procedures and instructions? Should analyses and/or experiments be an obligatory part of validation and verification process for procedures and instructions?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.99. All procedures and guidelines should be verified. Verification should be carried out to confirm the correctness of a written procedure or guideline and to ensure that technical and human factors have been properly incorporated. The review of plant specific procedures and guidelines in the development phase, in accordance with the quality assurance regulations, forms part of this verification process. In addition, independent reviews should be considered, where appropriate, in order to enhance the verification process.

3.100. All procedures and guidelines should be validated. Validation should be carried out to confirm that the actions specified in the procedures and guidelines can be followed by trained staff to manage emergency events.

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.16 Tentative lists of initiating events preceding design basis accidents and list of beyond design basis accidents (BDBA) including initiating events, sequence paths and consequences shall be specified in regulatory documents for each type of reactor. They shall include representative scenarios with severe consequences for defining the plan of possible response.

The final lists of BDBA, their realistic (not conservative) analysis containing assessment of probabilities of BDBA accident sequence paths including accidents involving core meltdown, consequences of BDBA, functioning of safety systems shall be established in the NPP design and presented in the NPP Safety Analysis Report.

Should the analysis of BDBA consequences not confirm that i. 1.2.17 is met, it would be necessary to provide in the design additional technical approaches to accident management for the purpose of mitigating their consequences.

The analysis of BDBA consequences presented in the NPP design is a basis for drawing up emergency planning procedures on personnel and population protection in the event of an accident and elaborating the manual on accident management.

RF NP-006-98 To Contents of Safety Analysis Report of NPP with VVVER Reactors

15.6.3. Measures for beyond design basis accident management

III. Equipment

1. Should a minimal list of technical means dedicated for SAM be determined by regulator?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.69. For dedicated or upgraded equipment, there should be sufficient confidence in the equipment and, where possible, demonstration of its capability to perform the required actions in beyond design basis and severe accident conditions should be provided. Demonstration of the capability of equipment should be provided where other assessment methods cannot provide sufficient confidence. However, the level of qualification applied to such equipment need not necessarily be the same as that typically required for components and systems that cope with design basis conditions. Similarly, requirements on the redundancy of such systems may also be relaxed compared to the requirements applied in the design basis domain.

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.4. During normal operation all physical barriers shall be operable, and measures to be undertaken for their protection shall be ready available. On detection of loss of functions by anyone of physical barriers envisaged in the design or unavailability of measures for its protection the RP shall be shutdown and measures shall be taken to bring the NPP to safe condition.

1.2.18. The system of technical and administrative measures on ensuring NPP safety shall be presented in the NPP Safety report which shall be elaborated by the operating organisation of the NPP or organisation which announced its intention to build and operate a NPP (applicant) with participation of NPP and RP designers. Any differences between information contained in the NPP Safety Analysis Report and information in the NPP design as well as differences between NPP design and its implementation are not allowable. The compliance of the NPP Safety Analysis Report to the real conditions shall be maintained by the operating organisation of the NPP during the whole NPP service life.

RF NP-006-98 To Contents of Safety Analysis Report of NPP with VVVER Reactors

15.6.3.3. Systems and equipment, which may be employed to arrive at safety objectives and to confine accident consequences

All NPP engineering systems (including non-safety related systems) shall be identified, which may be possibly employed apart from the design purposes or design operational modes to arrive at short-term safety objectives and to confine accident consequences at each level of its severity. Consider redundancy of the systems implementing the same function. Describe the possibility to use materials and equipment located at adjacent power units as well as off-site, and select transportation means.

15.6.3.5. Analysis of facility state data, which are available to operating personnel during accident evolution

Herein, determine the sufficient level of information required to monitor facility state features, to identify accident severity level, to control relevant engineering systems, to evaluate the effectiveness of management actions of beyond design basis accidents, technical means and methods allowing to obtain such data under forecasted conditions. Should an indirect assessment of the required parameters be implemented, provide for the methodology for such evaluation.

2. Should severe accident management provisions (i.e. using mobile equipment verses hardening on site equipment) be performance based or prescribed by regulation?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.68. If equipment and systems used to cope with design basis conditions are supplemented by additional equipment to mitigate severe accidents, the latter equipment should preferably be independent.

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.4. During normal operation all physical barriers shall be operable, and measures to be undertaken for their protection shall be ready available. On detection of loss of functions by anyone of physical barriers envisaged in the design or unavailability of measures for its protection the RP shall be shutdown and measures shall be taken to bring the NPP to safe condition.

1.2.18. The system of technical and administrative measures on ensuring NPP safety shall be presented in the NPP Safety report which shall be elaborated by the operating organisation of the NPP or organisation which announced its intention to build and operate a NPP (applicant) with participation of NPP and RP designers. Any differences between information contained in the NPP Safety Analysis Report and information in the NPP design as well as differences between NPP design and its implementation are not allowable. The compliance of the NPP Safety Analysis Report to the real conditions shall be maintained by the operating organisation of the NPP during the whole NPP service life.

3. Should be installed special requirements (regulations) in respect of mission times and minimum capacity for equipment dedicated for SAM?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.33. The procedures and guidelines should contain the following elements:

— Objectives and strategies;
— Initiation criteria;
— The time window within which the actions are to be applied (if relevant);

— The possible duration of actions;
— The equipment and resources (e.g. AC and DC power, water) required;
— Actions to be carried out;
— Cautions;
— Throttling and termination criteria;
— Monitoring of plant response.

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.4. During normal operation all physical barriers shall be operable, and measures to be undertaken for their protection shall be ready available. On detection of loss of functions by anyone of physical barriers envisaged in the design or unavailability of measures for its protection the RP shall be shutdown and measures shall be taken to bring the NPP to safe condition.

1.2.18. The system of technical and administrative measures on ensuring NPP safety shall be presented in the NPP Safety report which shall be elaborated by the operating organisation of the NPP or organisation which announced its intention to build and operate a NPP (applicant) with participation of NPP and RP designers. Any differences between information contained in the NPP Safety Analysis Report and information in the NPP design as well as differences between NPP design and its implementation are not allowable. The compliance of the NPP Safety Analysis Report to the real conditions shall be maintained by the operating organisation of the NPP during the whole NPP service life.

4. Should be special requirements (regulations) be installed for using mobile equipment versus stationary installed equipment?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.68. If equipment and systems used to cope with design basis conditions are supplemented by additional equipment to mitigate severe accidents, the latter equipment should preferably be independent.

5. Should be special requirements (regulations) related to I&C for SA (I&C for SAM and indicate the status of SA progress)*?*

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.71. Since the SAMGs depend on the ability to estimate the magnitude of several key plant parameters, the plant parameters needed for both preventive accident management measures and mitigatory accident management measures should be identified. It should be checked that all these parameters are available from the instrumentation in the plant. Where instruments can give information on the accident progression in a non-dedicated way, such possibilities should be investigated and included in the guidance.

3.72. The existing qualification for relevant instruments should be taken into account, and it should be recognized that such equipment may continue to operate well beyond its qualified range. Alternative instrumentation should be identified where the primary instrumentation is not available or not reliable. Where such instrumentation is not available, alternative means should be developed, for example, computational aids.

3.73. Use of instrumentation that is qualified for the expected environmental conditions is the preferred method to obtain the necessary information.

3.74. The effect of environmental conditions on the instrument reading should be estimated and included in the guidance. It should be taken into consideration that a local environmental condition can deviate from global conditions and, hence, instrumentation that is qualified under global conditions may not function properly under local conditions. The expected failure mode and resultant instrument indication (e.g. off-scale high, off-scale low, floating) for instrumentation failures in severe accident conditions beyond the design basis should be identified.

3.75. Severe accidents may present challenges to instrumentation beyond its design basis where such instruments may operate outside their design operating range. As the indication from instruments then may be in error, all indications used to diagnose plant conditions for severe accident management should be benchmarked against other direct or derived indications in order to reduce the risks associated with faulty readings. In practice, every key instrumentation reading from a non-qualified dedicated instrument that is used for diagnosis or verification should have an alternate method to verify that the primary reading (i.e. the reading from the dedicated instrument) is reasonable. When an alternative means of obtaining a key parameter value cannot be identified, consideration should be given to upgrading or replacing the instruments in order to provide that alternative indication.

3.76. In the development of the SAMGs, the potential failure of important non- qualified instrumentation during the evolution of the accident should be included and, where possible, alternative strategies that do not use this instrumentation should be developed. The ability to infer important plant parameters from local instrumentation or from unconventional means should also be considered. For example, the steam generator level can be inferred from local pressure measurements on the steam line and steam generator blowdown lines.

3.77. The need for development of computational aids to obtain information where parameters are missing or their measurements are unreliable should be identified and appropriate computational aids should be developed accordingly.

IAEA NS-G 1.10, Design of Reactor Containment Systems for Nuclear Power Plants [6.28-33]

6.28. For the management of severe accidents, appropriate instrumentation and procedures should be available to guide operator actions to initiate preventive or mitigatory measures. The instrumentation necessary for the management of severe accidents falls into four categories:

(1) Instrumentation for monitoring the general conditions in the containment;

(2) Instrumentation for monitoring the progression in the values of parameters of interest, specifically in relation to severe accidents;

(3) Instrumentation necessary for operators to execute emergency procedures;

(4) Instrumentation for assessing radiological consequences.

IAEA SSR-2/1 Safety of Nuclear Power Plants Design

Requirement 59: Provision of instrumentation Instrumentation shall be provided for determining the values of all the main variables that can affect the fission process, the integrity of the reactor core, the reactor coolant systems and the containment at the nuclear power plant, for obtaining essential information on the plant that is necessary for its safe and reliable operation, for determining the status of the plant in accident conditions and for making decisions for the purposes of accident management.

6.31. Instrumentation and recording equipment shall be provided to ensure that essential information is available for monitoring the status of essential equipment and the course of accidents, for predicting the locations of release and the amount of radioactive material that could be released from the locations that are so intended in the design, and for post-accident analysis.

6. Should be special requirements (regulations) related hydrogen management in the containment under the SA conditions?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.22. In the mitigatory domain, strategies should be developed to enable:

— 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.

Strategies may be derived from ‘candidate high level actions’, examples of which are given in Appendix II of Ref. [12] (Implementation of Accident Management Programmes in Nuclear Power Plants, Safety Reports Series No. 32, IAEA). Examples of mitigatory strategies are: filling the secondary side of the steam generator to prevent creep rupture of the steam generator tubes; depressurizing the reactor circuit to prevent high pressure reactor vessel failure and direct containment heating; flooding the reactor cavity to prevent or delay vessel failure and subsequent basemat failure; <u>mitigating the hydrogen concentration;</u> and depressurizing the containment to prevent its failure by excess pressure or to prevent basemat failure under elevated containment pressure.

IAEA NS-G 1.10, Design of Reactor Containment Systems for Nuclear Power Plants

6.5. For new plants, possible severe accidents should be considered at the design stage of the containment systems. The consideration of severe accidents should be aimed at practically eliminating the following conditions:

— Severe accident conditions that could damage the containment in an early phase as a result of direct containment heating, steam explosion or hydrogen detonation;

— Severe accident conditions that could damage the containment in a late phase as a result of basemat melt-through or containment overpressurization;

— Severe accident conditions with an open containment — notably in shutdown states;

— Severe accident conditions with containment bypass, such as conditions relating to the rupture of a steam generator tube or an interfacing system LOCA.

6.10. For new plants, the integrity and leaktightness of the containment structure should be ensured for those severe accidents that cannot be practically eliminated (para. 6.5). The long term pressurization of the containment should be limited to a pressure below the value corresponding to Level II for structural integrity.

6.22. In a severe accident, a large amount of hydrogen might be released to the atmosphere of the containment, possibly exceeding the ignition limit and jeopardizing the integrity of the containment. In the event of interactions between molten core material and concrete, carbon monoxide might also be released, contributing to the hazard. To assess the need to install special features to control combustible gases, an assessment of the threats to the containment posed by such gases should be made for selected severe accident sequences. The assessment should cover the generation, transport and mixing of combustible gases in the containment, combustion phenomena (diffusion flames, deflagrations and detonations) and the consequent thermal and mechanical loads, and the efficiency of systems for the prevention of accidents and the mitigation of their consequences.

6.23. Uncertainties remain concerning the production of hydrogen during severe accident sequences; these uncertainties are essentially linked to such phenomena as flooding of a partially damaged core at high temperatures, the late phase of core degradation, the slumping of molten core material into residual water in the lower head of the reactor pressure vessel, and the long term interactions between molten core material and concrete. For new plants, these uncertainties should be taken into account in the design and layout of the means of mitigation of the consequences of the combustion or deflagration of hydrogen, and in the design of the containment.

6.24. The efficiency of the means of mitigation of the consequences of combustion or deflagration should be such that the concentrations of hydrogen in the compartments of the containment would at all times be low enough to preclude fast global deflagration or detonation. Possible provisions in the design for achieving this goal are, for example, an enhanced natural mixing capability of the containment atmosphere coupled with a sufficiently large free volume, passive autocatalytic recombiners and/or igniters suitably distributed in the containment, and inerting. For new plants the amount of hydrogen expected to be generated should be estimated on the basis of the assumption of total oxidation of the fuel cladding.

6.25. The leaktightness of the containment for the most representative accident sequences should be ensured with sufficient margins to accommodate severe dynamic phenomena such as a fast local deflagration, if these phenomena cannot be excluded.

6.26. Even in an inerted containment, the concentrations of hydrogen and oxygen generated over a long period of time by water radiolysis may eventually exceed the ignition limit. If this is a possible threat, a hydrogen control system, passive hydrogen recombiners or other appropriate systems for mitigation and monitoring (e.g. systems for oxygen control and measurement) should be installed.

6.27. Provision should be made for hydrogen monitoring or sampling. The concentrations of other combustible gases and oxygen should also be monitored.

6.29. During and following a severe accident, in order to follow the general conditions in the containment and to facilitate the use of guidelines for the management of severe accidents, essential parameters for the containment such as pressures, temperatures, hydrogen concentrations, water levels and radiation dose rates should be monitored.

III–12. The generation and combustion of large volumes of hydrogen and carbon monoxide are severe accident phenomena that can threaten the integrity of the containment. The major cause of the generation of hydrogen is the oxidation of zirconium metal and, to a lesser extent, the interaction of steel or any other metallic component with steam when the metal reaches temperatures well above normal operating temperatures.

III–13. In addition, ex-vessel hydrogen generation needs to be considered. Such hydrogen is produced mainly as a result of the reactions of ex-vessel metallic core debris with steam, and in the long term by molten core–concrete interactions (para. III–17) and by the extended radiolysis of sump water.

III–15. Under severe accident conditions, significant hydrogen concentrations could be reached locally in a short time (of the order of some minutes to an hour, depending on the containment design, the scenario and the location) and globally in a longer period of time.

III–16. When the ignition limit is exceeded, combustion of hydrogen is possible and can take different forms, depending on the concentrations, the atmospheric conditions in the containment and the geometry: diffusion flames (which are mainly responsible for thermal loads), slow deflagrations (which are mainly responsible for quasi-static pressure loads), fast deflagrations (for which dynamic effects become important) and detonations (for which the velocity of the flame front exceeds the speed of sound in the unburnt gas, giving rise to extremely severe dynamic effects). Depending on the mode of combustion, the integrity of the containment may be threatened by stresses beyond the structural design limits.

NS-G-1.9 Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety Guide

3.47. Hydrogen and oxygen generated by the decomposition of H_2O (or D_2O) in the core can dissolve in the water and steam and be carried to any part of the RCS and connected systems. Gases dissolved in steam piping can easily accumulate when steam in a closed off section of piping cools down and condenses into water. A local accumulation of hydrogen gas in the RCS could give rise to the potential for an explosion that could result in severe damage. The design should be such that the possibility of combustible gas accumulation can be excluded.

IAEA SSR-2/1 Safety of Nuclear Power Plants Design

6.29. Design features to control fission products, hydrogen, oxygen and other substances that might be released into the containment shall be provided as necessary:

(a) To reduce the amounts of fission products that could be released to the environment in accident conditions;

(b) To control the concentrations of hydrogen, oxygen and other substances in the containment atmosphere in accident conditions so as to prevent deflagration or detonation loads that could challenge the integrity of the containment.

RF NP-040-02 Regulations for Ensuring Hydrogen Explosion Protection at a Nuclear Power Station

p. 2.1, p. 2.4, p. 2.5, p. 3.3

7. Should be special requirements (regulations) for primary pressure decrease?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.22. In the mitigatory domain, strategies should be developed to enable:

— 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.

Strategies may be derived from ‘candidate high level actions’, examples of which are given in Appendix II of Ref. [12] (Implementation of Accident Management Programmes in Nuclear Power Plants, Safety Reports Series No. 32, IAEA). Examples of mitigatory strategies are: filling the secondary side of the steam generator to prevent creep rupture of the steam generator tubes; <u>depressurizing the reactor circuit to prevent high pressure reactor vessel failure</u> and direct containment heating; flooding the reactor cavity to prevent or delay vessel failure and subsequent basemat failure; mitigating the hydrogen concentration; and <u>depressurizing the containment to prevent its failure by excess pressure or to prevent basemat failure under elevated containment pressure.</u>

3.64. For the mitigatory domain, in upgrading equipment the focus should be placed on preservation of the containment function and, in particular, the following functions should be taken account of:

— Containment isolation in a severe accident, including bypass prevention;

— Monitoring parameters in the containment, allowing an early diagnosis of the unit status including the concentration of fission products and hydrogen;

— Ensuring the leaktightness of the containment, including preservation of the functionality of isolation devices, penetrations and personnel locks, for a reasonable time after a severe accident;

— Management of pressure and temperature in the containment by means of a containment heat removal system;

— Control of the concentration of combustible gases, fission products and other materials released during severe accidents;

— Containment overpressure and underpressure protection;

— Prevention of high pressure core-melt scenarios;

— Prevention of vessel melt through;

— Prevention and mitigation of containment basemat melt through by the molten core;

— Monitoring and control of containment leakages.

IAEA NS-G 1.10, Design of Reactor Containment Systems for Nuclear Power Plants

6.13. Highly energetic severe accident conditions with the potential for damaging the containment should be virtually eliminated for new plants. Reliable depressurization of the reactor coolant system to prevent the ejection of molten core material and core debris and direct containment heating should be ensured as an accident management measure for existing and new plants.

6.29. During and following a severe accident, in order to follow the general conditions in the containment and to facilitate the use of guidelines for the management of severe accidents, essential parameters for the containment such as pressures, temperatures, hydrogen concentrations, water levels and radiation dose rates should be monitored.

III–19. The long term pressurization of the containment may also be affected by the availability or unavailability of containment sprays (or heat exchangers) and air coolers.

IAEA SSR-2/1 Safety of Nuclear Power Plants Design

Requirement 48: Overpressure protection of the reactor coolant pressure boundary

Provision shall be made to ensure that the operation of pressure relief devices will protect the pressure boundary of the reactor coolant systems against overpressure and will not lead to the release of radioactive material from the nuclear power plant directly to the environment.

Requirement 58: Control of containment conditions

Provision shall be made to control the pressure and temperature in the containment at a nuclear power plant and to control any buildup of fission products or other gaseous, liquid or solid substances that might be released inside the containment and that could affect the operation of systems important to safety.

6.28. The capability to remove heat from the containment shall be ensured, in order to reduce the pressure and temperature in the containment, and to maintain them at acceptably low levels after any accidental release of high energy fluids. The systems performing the function of removal of heat from the containment shall have sufficient reliability and redundancy to ensure that this function can be fulfilled.

8. Should be special requirements (regulations) related to auxiliary equipment to provide water injection into the reactor vessel, steam generators and containment?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.27. Priorities should be set between strategies, because possible strategies can have a different weight and/or effect on safety, and because not all strategies can be carried out at the same time. In the preventive domain, the priority of the strategies should be reflected in the priority established for the critical safety functions. In the mitigatory domain, priority should be given to measures that mitigate large ongoing releases or challenges to important fission product barriers (where 'large' means releases with levels of radioactivity that are above the general emergency levels, as defined in the plant emergency plan). The basis for the selection of priorities should be recorded in the background documentation. An example is a set of priorities that follows the evolution of many severe accidents; that is, the first priority is to the first fission product barrier to fail if no mitigatory measures are taken. <u>The setting of priorities should include the consideration of support functions (vital auxiliaries such as AC and DC power and cooling water).</u>

9. Should be special requirements (regulations) cover the special technical provisions dedicated for recovery of off-site power, multi-units considerations, treatment of liquid radioactive effluents.

IAEA SSR-2/1 Safety of Nuclear Power Plants Design

Requirement 79: Systems for treatment and control of effluents

Systems shall be provided at the nuclear power plant for treating liquid and gaseous radioactive effluents to keep their amounts below the authorized limits on discharges and as low as reasonably achievable.

6.61. Liquid and gaseous radioactive effluents shall be treated at the plant so that exposure of members of the public due to discharges to the environment is as low as reasonably achievable.

6.62. The design of the plant shall incorporate suitable means to keep the release of radioactive liquids to the environment as low as reasonably achievable and to ensure that radioactive releases remain below the authorized limits on discharges.

6.63. The cleanup equipment for the gaseous radioactive substances shall provide the necessary retention factor to keep radioactive releases below the authorized limits on discharges. Filter systems shall be designed so that their efficiency can be tested, their performance and function can be regularly monitored over their service life, and filter cartridges can be replaced while maintaining the throughput of air.

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

2.1. Reference [5] (Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. NS-R-1, IAEA) establishes the following requirements on addressing severe accidents and accident management in the design of nuclear power plants:

(5) For multiunit plants, consideration shall be given to the use of available means and/or support from other units, provided that the safe operation of the other units is not compromised.

IV. Methodology of severe accident analyses

1. Should the list of SA analyses be a subject of licensing?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

2.17. Severe accident management should cover all modes of plant operation and also appropriately selected external events, such as fires, floods, seismic events and extreme weather conditions (e.g. high winds, extremely high or low temperatures, droughts) that could damage large parts of the plant. In the severe accident management guidance, consideration should be given to specific challenges posed by external events, such as loss of the power supply, loss of the control room or switchgear room and reduced access to systems and components.

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.16 Tentative lists of initiating events preceding design basis accidents and list of beyond design basis accidents (BDBA) including initiating events, sequence paths and consequences shall be specified in regulatory documents for each type of reactor. They shall include representative scenarios with severe consequences for defining the plan of possible response.

The final lists of BDBA, their realistic (not conservative) analysis containing assessment of probabilities of BDBA accident sequence paths including accidents involving core meltdown, consequences of BDBA, functioning of safety systems shall be established in the NPP design and presented in the NPP Safety Analysis Report.

Should the analysis of BDBA consequences not confirm that i. 1.2.17 is met, it would be necessary to provide in the design additional technical approaches to accident management for the purpose of mitigating their consequences.

The analysis of BDBA consequences presented in the NPP design is a basis for drawing up emergency planning procedures on personnel and population protection in the event of an accident and elaborating the manual on accident management.

RF NP-006-98 To Contents of Safety Analysis Report of NPP with VVVER Reactors

15.2. List of beyond design basis accidents

15.2.1. Scenarios of beyond design basis accidents which result in increased radionuclide releases to the environment. NPP vulnerable points.

15.2.2. Typical groups of beyond design basis accident scenarios

15.2.3. Representative scenarios of beyond design basis accidents

2. How the completeness and representativeness of SA analyses could be confirmed (bounding criteria)?

3. What acceptance criteria are applicable for severe accident?

IAEA SSG-2 Deterministic Safety Analysis for Nuclear Power Plants

3.22. Acceptance criteria for design basis accidents may be supplemented by criteria that relate to severe accidents. These are typically core damage frequency, prevention of consequential damage to the containment, large early release frequency, probability of scenarios requiring emergency measures off the site, limitation of the release of specific radionuclides such as 137 Cs, dose limits and/or risks to the most exposed individual.

7.7. Where there are acceptance criteria for beyond design basis accidents, including severe accidents, it should be demonstrated that the consequences would be acceptably low.

RF NP-082-07 Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants

2.1.9. During RI design process one shall pursue that the cumulative frequency of severe beyond design basis accidents estimated on the basis of the probabilistic safety assessment would not exceed 10^{-5} per reactor-year.

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.17. To avoid the necessity of evacuating population beyond the area covered by planning protective measures established according to regulatory requirements to NPP siting efforts should be made to ensure that, the estimated probability rate of limiting emergency release did not exceed 10^{-7} per reactor year.

RF NP-032-01 Nuclear Power Plant Siting. Main Criteria and Safety Requirements

5.7. The boundary of the area of planned protective actions for nuclear power plants and nuclear energotechnological stations should not be moved away by farther than 25 km and for nuclear heating plants - by 5 km from the site boundary.

4. Should the codes dedicated for SA analyses (neutronic, thermohydraulic, containment, source term) be under licensing control? If yes, what are requirements for validation and verification of these codes?

IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants

3.126. Computer codes used for analysis should be validated to the extent possible. However, it should be noted that many codes used in the beyond design basis accident and severe accident cannot be subjected to the same level of validation as the codes used in the design basis domain, due to uncertainty in the understanding of the phenomena. Usually, no single code can cope with the entire range of phenomena, and special purpose codes may also need to be used. The operating organization of the plant should specify the proper codes and models for the various applications, and should justify their use. Where relevant, the operating organization of the plant should carry out a sensitivity analysis in addition to the uncertainty analysis, to find the relative weight of certain phenomena compared to others.

IAEA NS-G 1.10, Design of Reactor Containment Systems for Nuclear Power Plants

6.3. The validation domain of the computer codes used for evaluating all pertinent parameters should be verified to cover their expected range of variation adequately. Computer codes should not be used beyond their validation domain. As an exception, the use of computer codes beyond their range of validation might possibly be acceptable in areas for which it is widely recognized that there is a lack of coherent data. Such exceptions should be allowed only on the following conditions:

— The exception is clearly specified.

— A comprehensive sensitivity analysis is carried out to evaluate the effects of variations in the assumptions and in the modelling.

— An independent assessment is made of the credibility of the results.

— Appropriate margins are introduced if knowledge is limited.

RF NP-006-98 To Contents of Safety Analysis Report Of NPP with VVVER Reactors

15.3.5. Information on computation software verification

Emergency mode mathematical models used for safety analysis, development of accident management programs and simulator software shall be verified, i.e. compared with experimental data. The verification matrix shall cover all experimental installations employed to justify the software. There shall be at least one test bench structurally similar to NPP, i.e. having physical models of NPP major equipment which reflects key features of each prototype: reactor core, SG, drum separators, MCP, leaktight containment, passive heat removal system, etc.

RF NP-082-07 Nuclear Safety Rules for Reactor Installations of Nuclear Power Plants

2.1.15. The RI and NPP designs shall list methodologies and codes used for safety justification, and those used in SIS, along with the scope of their application. The codes and methodologies in use shall be verified and certified in accordance with the established procedures.

RF OPB-88/97, NP-001-97 General Regulations on Ensuring Safety of Nuclear Power Plants

1.2.9. The operating organisation of the NPP ensures NPP safety including measures on preventing accidents and minimising their consequences, accounting and control, physical protection of nuclear materials, radioactive substances and waste, radiation control over the environmental conditions in the safe area and surveyed area as well as provides the use of the NPP for those purposes for which it was designed and constructed.

The operating organisation of the NPP is fully responsible for NPP safety.

The operating organisation shall not be relieved of responsibilities of the NPP in connection with the independent activity and commitments of organisations performing works or rendering services to the NPP and by state safety regulation bodies.

Appendix E – Regulatory approaches used in China

QUESTIONNAIRE ON SEVERE ACCIDENT REQUIREMENTS AND PRACTICES

I. General

1. What types of regulatory controls are installed in respect of severe accident (i.e. licensed verses voluntary initiatives)?

Licensed with based requirements, more effective strategies are voluntary initiatives.

2. What is the legal basis this matter (laws, requirements, regulations)?

NNSA issued a series of regulations, guidelines and policy statements for the severe accident analysis and management:

NNSA Department Rules:

- HAF 102 “Safety of Nuclear Power Plants: Design” (Based on IAEA No. NS-R-1, 2000), revision of HAF102 was issued in Oct. 2016, mainly based on IAEA SSR-2/1, 2016.
- HAF 103 “Safety of Nuclear Power Plants: Operation”, (Based on IAEA No. NS-R-2, 2000).

Guidelines:

- HAD 102/06 “Design of Reactor Containment Systems” (Based on IAEA No.NS-G-1.10).
- HAD 102/17 “Safety Assessment and Verification for Nuclear Power Plants” (Based on IAEA No.NS-G-1.2).
- “Severe Accident management programme for NPP” (DRAFT, Based on IAEA No.NS-G-2.15).

post-Fukushima Policy Statement:

- General Technical Requirements on post-Fukushima Nuclear Accident Improvement Measures for NPPs (Tentative).
- “The 12th Five-year Plan and 2020 long-term goal for nuclear safety and radioactive pollution prevention and control”.

II. Procedures and instructions

1. Should guidelines and/or procedures be a part of licensing?

Yes, revision of HAF 102 2016 “Safety of nuclear power plants: design”

- 5.2.12 important event sequences that may lead to a severe accident shall be identified using a combination of probabilistic methods, deterministic methods and sound engineering judgement. Accident management procedures shall be established, taking into account representative and dominant severe accident scenarios.

HAF 103 “Safety of nuclear power plants: operation”

- 3.12 Plant staff shall receive instructions in the management of accidents beyond the design basis. The training of operating personnel shall ensure their familiarity with the symptoms of accidents beyond the design basis and with the procedures for accident management.

- 5.2.3 Either event based or symptom based procedures shall be developed for abnormal conditions and design basis accidents. Emergency operating procedures or guidance for managing severe accidents (beyond the design basis) shall be developed.

HAD 102/06 "Design of Reactor Containment Systems"

- 6.34. Guidelines for the management of severe accidents (severe accident management guidelines (SAMGs)) should be aimed primarily at maintaining or restoring the performance of the containment. SAMGs should be developed for managing accident conditions in co-ordination with on-site and off-site emergency organizations. SAMGs should be established to supplement, but not to replace, provisions in the design to prevent the failure of containment systems during or following a severe accident or to mitigate the consequences of such an accident.

Twelfth Five-year Plan and 2020 long-term goal for nuclear safety and radioactive pollution prevention and control

- "During the 12th Five-year Plan period (before 2015), the new NPPs should have comprehensive measures to prevent and to mitigate severe accidents, the CDF should be lower than 10^{-5}/year and the LRF should be lower than 10^{-6}/year."
- "For NPPs built in/after the 13th Five-year Plan period (after 2015), the design should practically eliminate the possibility of large scale radioactive release".
- Enhance the safety of NPPs in operation: The SAMG should be completed and implemented before the end of 2013. The availability of equipment used for severe accident mitigation and hydrogen control should be assessed, corresponding improvements should be implemented.
- Enhance the safety of NPPs under construction: The SAMG should be completed and implemented before first fuel loading, covering all operation states and multiple units at a site, the availability and accessibility of important equipment and instruments should be assessed under severe accidents.

And there are some specific requirements in "IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants".

2. Should licensing cover as prevention and mitigation consequences of severe accident?

Yes, revision of HAF 102 2016 "Safety of nuclear power plants: design"

- 4.4: The design of a nuclear power plant shall incorporate defense in depth. The levels of defense in depth shall be independent as far as is practicable. The levels of defense in depth shall be independent as far as practicable to avoid the failure of one level reducing the effectiveness of other levels. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems.
- 5.1.9 A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional

accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.

3. Should structure and content of the procedure and instructions be a part of regulatory control?

For SAMG, requirements in "IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants" are referenced, however, there is no fixed structure and content for SAMG.

4. How deep (in detail) should be monitored structure and content of procedures and/or instructions? Please consider the following items: entry criteria; exit criteria; criteria for the transition from prevention to mitigation domain of SA procedures and/or instructions; harmonization procedure and/or instructions dedicated for SAM with other emergency procedures and instructions; coverage of such SAM aspects as malty-unit accidents, different locations of the fuel (reactor' spent fuel pool, on-site fuel storage); distribution of the responsibility during the application of emergency procedure and/or instructions in the course of SA (who make a decision?);

Requirements in "IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants" are referenced

3.3. The accident management guidance should address the full spectrum of credible challenges to fission product boundaries due to severe accidents, including those arising from multiple hardware failures, human errors and/or events from outside, and possible physical phenomena that may occur during the evolution of a severe accident (such as steam explosions, direct containment heating and hydrogen burns). In this process, issues should also be taken into account that are frequently not considered in analyses, such as additional highly improbable failures and abnormal functioning of equipment.

5. How the operator must confirm the correctness and sufficiency of the procedures and instructions? Should analyses and/or experiments be an obligatory part of validation and verification process for procedures and instructions?

Requirements in "IAEA NS-G 2.15 Severe Accident Management Programmes for Nuclear Power Plants" are referenced.

3.99. All procedures and guidelines should be verified. Verification should be carried out to confirm the correctness of a written procedure or guideline and to ensure that technical and human factors have been properly incorporated. The review of plant specific procedures and guidelines in the development phase, in accordance with the quality assurance regulations, forms part of thisverification process. In addition, independent reviews should be considered,where appropriate, in order to enhance the verification process.

3.100. All procedures and guidelines should be validated. Validation should be carried out to confirm that the actions specified in the procedures and guidelines can be followed by trained staff to manage emergency events.

III. Equipment

1. Should a minimal list of technical means dedicated for SAM be determined by regulator?

Yes, HAD 102/06 "Design of Reactor Containment Systems"

- 6.1.5 For new plants, possible severe accidents should be considered at the design stage of the containment systems. The consideration of severe accidents should be aimed at practically eliminating the following conditions:

– Severe accident conditions that could damage the containment in an early phase as a result of direct containment heating, steam explosion or hydrogen detonation;

– Severe accident conditions that could damage the containment in a late phase as a result of basemat melt-through or containment overpressurization;

– Severe accident conditions with an open containment — notably in shutdown states;

– Severe accident conditions with containment bypass, such as conditions relating to the rupture of a steam generator tube or an interfacing system LOCA.

- 6.1.6. For severe accidents that cannot be practically eliminated, the containment systems should be capable of contributing to the reduction of the radioactive releases to such a level that the extent of any necessary off-site emergency measures needed is minimal.

Structural behavior of the containment, energy management, management of radionuclides, management of combustible gases, instrumentation should be considered.

The exact means is not specified by the regulation but left to the designer.

2. Should severe accident management provisions (i.e. using mobile equipment verses hardening on site equipment) be performance based or prescribed by regulation?

İn revision of HAF 102 2016 “Safety of nuclear power plants: design”:

- 5.1.9.4 The analysis undertaken shall include identification of the features that are designed for use in, or that are capable of preventing or mitigating, events considered in the design extension conditions. These features:

– Shall be independent, to the extent practicable, of those used in more frequent accidents;

– Shall be capable of performing in the environmental conditions pertaining to these design extension conditions, including design extension conditions in severe accidents, where appropriate;

– Shall have reliability commensurate with the function that they are required to fulfill.

- 6.3.5.5 The design shall also include features to enable the safe use of mobile equipment for restoring the capability to remove heat from the containment

So installed equipment should be considered in prior, and mobile equipment is required and can be used in extreme condition which installed equipment can not work.

3. Should be installed special requirements (regulations) in respect of mission times and minimum capacity for equipment dedicated for SAM?

In China, generally, installed equipment is needed within 8 hours, on-site mobile light equipment can be used within 72 hours, and off-site mobile equipment can be used after 72 hours.

İn revision of HAF 102 2016 “Safety of nuclear power plants: design”:

- 6.3.5.4 Design provision shall be made to prevent the loss of the structural integrity of the containment in all plant states. The use of this provision shall not lead to an early radioactive release or a large radioactive release.

HAD 102/06 “Design of Reactor Containment Systems”

- 6.2.1 For existing plants, the ultimate load bearing capacity (structural integrity Level III) and retention capacity (leak tightness Level II) of the containment structure should not be exceeded in severe accidents, to the extent that this can be achieved by practicable means. Furthermore, the molten core material and core debris should be stabilized within the containment.

- 6.2.3 For new plants, the integrity and leak tightness of the containment structure should be ensured for those severe accidents that cannot be practically eliminated. The long term pressurization of the containment should be limited to a pressure below the value corresponding to Level II for structural integrity

General Technical Requirements on post-Fukushima Nuclear Accident Improvement Measures for NPPs (Tentative).

For emergency water-injection

- 3.1.1 It should be able to remove the residual heat of core by the "feed-bleed" of the secondary circuit for a long time, and the flow of emergency water injection should be able to meet the need of removing the core residual heat in 6 hours after shutdown.
- 3.1.2 The equipment should be able in operation for at least 72 hours after the accident.
- 3.1.3 All of the preparations should be completed in 6 hours after the shutdown, and the equipment should be in an available state.
- 3.2.1 The flow rate of water injection should meet the need of removing the core residual heat after shutdown 6 hours through emergency water injection to the primary circuit by mobile pumps and related pipeline.
- 3.2.3 Equipment should ensure operation for 72 hours after the accident.
- 3.2.5 All of the preparations should be completed in 6 hours after the shutdown, and the equipment should be in an available state.

4. Should be special requirements (regulations) be installed for using mobile equipment versus stationary installed equipment?

Provisons of severe accident are basically installed equipment, and there are some requirements for mobile equipment:

Revision of HAF 102 2016 "Safety of nuclear power plants: design":

- 5.1.9.4 The analysis undertaken shall include identification of the features that are designed for use in, or that are capable of preventing or mitigating, events considered in the design extension conditions. These features:
  - Shall be independent, to the extent practicable, of those used in more frequent accidents;
  - Shall be capable of performing in the environmental conditions pertaining to these design extension conditions, including design extension conditions in severe accidents, where appropriate;
  - Shall have reliability commensurate with the function that they are required to fulfill.
- 6.3.5.5 The design shall also include features to enable the safe use of mobile equipment for restoring the capability to remove heat from the containment.
- 6.6.1.9 The design shall also include features to enable the safe use of mobile equipment to restore the necessary electrical power supply.
- 6.10.6 The design shall also include features to enable the safe use of mobile equipment to ensure sufficient water inventory for the long term cooling of spent fuel and for providing shielding against radiation.

General Technical Requirements on post-Fukushima Nuclear Accident Improvement Measures for NPPs (Tentative).

- Emergency mobile Water-Injection and Related Equipment should be prepared for secondary, primary circuit, or spent fuel pool.
- When all of the AC power supply (including DG) lost，the nuclear power plant should provide temporarily power with the emergency methods by using mobile emergency power source, in order to mitigate the accident results and provide interval for recovering the offsite or onsite AC source.
- Setting of the emergency water injection interface and isolating devices should be the same safety class with the connective systems, and the equipment behind the isolating device should have the same seismic requirements with the connective systems. Setting of interfaces should be convenient for operation and connection.

5. Should be special requirements (regulations) related to I&C for SA (I&C for SAM and indicate the status of SA progress)*?*

Yes, Revision of HAF 102 2016 "Safety of nuclear power plants: design":

- 6.4.1 Instrumentation shall be provided for: determining the values of all the main variables that can affect the fission process, the integrity of the reactor core, the reactor coolant systems and the containment at the nuclear power plant; for obtaining essential information on the plant that is necessary for its safe and reliable operation; for determining the status of the plant in accident conditions; and for making decisions for the purposes of accident management.

HAD 102/06 "Design of Reactor Containment Systems"

- 6.6 For the management of severe accidents, appropriate instrumentation and procedures should be available to guide operator actions to initiate preventive or mitigatory measures. The instrumentation necessary for the management of severe accidents falls into four categories:
  - Instrumentation for monitoring the general conditions in the containment;
  - Instrumentation for monitoring the progression in the values of parameters of interest, specifically in relation to severe accidents;
  - Instrumentation necessary for operators to execute emergency procedures;
  - Instrumentation for assessing radiological consequences.

6. Should be special requirements (regulations) related hydrogen management in the containment under the SA conditions?

Yes, Revision of HAF 102 2016 "Safety of nuclear power plants: design"

- 6.3.5.6. Design features to control fission products, hydrogen, oxygen and other substances that might be released into the containment shall be provided as necessary:
  - To reduce the amounts of fission products that could be released to the environment in accident conditions;
  - To control the concentrations of hydrogen, oxygen and other substances in the containment atmosphere in accident conditions so as to prevent deflagration or detonation loads that could challenge the integrity of the containment.

HAD 102/06 "Design of Reactor Containment Systems"

- 6.5.1 In a severe accident, a large amount of hydrogen might be released to the atmosphere of the containment, possibly exceeding the ignition limit and jeopardizing the integrity of the containment. In the event of interactions between molten core material and concrete, carbon monoxide might also be released, contributing to the hazard. To assess the need to install special features to control combustible gases, an assessment of the threats to the containment posed by such gases should be made for selected severe accident sequences. The assessment should cover the generation, transport and mixing of combustible gases in the containment, combustion phenomena (diffusion flames, deflagrations and detonations) and the consequent thermal and mechanical loads, and the efficiency of systems for the prevention of accidents and the mitigation of their consequences.
- 6.5.2 Uncertainties remain concerning the production of hydrogen during severe accident sequences; these uncertainties are essentially linked to such phenomena as flooding of a partially damaged core at high temperatures, the late phase of core degradation, the slumping of molten core material into residual water in the lower head of the reactor pressure vessel, and the long term interactions between molten core material and concrete. For new plants, these uncertainties should be taken into account in the design and layout of the means of mitigation of the consequences of the combustion or deflagration of hydrogen, and in the design of the containment.
- 6.5.3 The efficiency of the means of mitigation of the consequences of combustion or deflagration should be such that the concentrations of hydrogen in the compartments of the containment would at all times be low enough to preclude fast global deflagration or detonation. Possible provisions in the design for achieving this goal are, for example, an enhanced natural mixing capability of the containment atmosphere coupled with a sufficiently large free volume, passive autocatalytic recombiners and/or igniters suitably distributed in the containment, and inerting. For new plants the amount of hydrogen expected to be generated should be estimated on the basis of the assumption of total oxidation of the fuel cladding.
- 6.5.4 The leaktightness of the containment for the most representative accident sequences should be ensured with sufficient margins to accommodate severe dynamic phenomena such as a fast local deflagration, if these phenomena cannot be excluded.
- 6.5.5 Even in an inerted containment, the concentrations of hydrogen and oxygen generated over a long period of time by water radiolysis may eventually exceed the ignition limit. If this is a possible threat, a hydrogen control system, passive hydrogen recombiners or other appropriate systems for mitigation and monitoring (e.g. systems for oxygen control and measurement) should be installed.
- 6.5.6 Provision should be made for hydrogen monitoring or sampling. The concentrations of other combustible gases and oxygen should also be monitored.

7. Should be special requirements (regulations) for primary pressure decrease?

Yes, Revision of HAF 102 2016 “Safety of nuclear power plants: design”

- 4. 4. 7 The levels of defense in depth shall be independent as far as practicable to avoid the failure of one level reducing the effectiveness of other levels. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems
- 6.2.2 Overpressure protection of the reactor coolant pressure boundary: Provision shall be made to ensure that the operation of pressure relief devices will protect the pressure boundary of the reactor coolant systems against overpressure and will not

lead to the release of radioactive material from the nuclear power plant directly to the environment.

HAD 102/06 “Design of Reactor Containment Systems”

- 6.1.5 For new plants, possible severe accidents should be considered at the design stage of the containment systems. The consideration of severe accidents should be aimed at practically eliminating the following conditions:
  - Severe accident conditions that could damage the containment in an early phase as a result of direct containment heating, steam explosion or hydrogen detonation;
- 6.3.1 Highly energetic severe accident conditions with the potential for damaging the containment should be virtually eliminated for new plants. Reliable depressurization of the reactor coolant system to prevent the ejection of molten core material and core debris and direct containment heating should be ensured as an accident management measure for existing and new plants.

8. Should be special requirements (regulations) related to auxiliary equipment to provide water injection into the reactor vessel, steam generators and containment?

Yes, Revision of HAF 102 2016 “Safety of nuclear power plants: design”:

- 6.3.5.5 The design shall also include features to enable the safe use of mobile equipment for restoring the capability to remove heat from the containment.
- 6.6.1.9 The design shall also include features to enable the safe use of mobile equipment to restore the necessary electrical power supply.
- 6.10.6 The design shall also include features to enable the safe use of mobile equipment to ensure sufficient water inventory for the long term cooling of spent fuel and for providing shielding against radiation.

General Technical Requirements on post-Fukushima Nuclear Accident Improvement Measures for NPPs (Tentative).

- Emergency mobile Water-Injection and Related Equipment should be prepared for secondary, primary circuit, or spent fuel pool.
- Setting of the emergency water injection interface and isolating devices should be the same safety class with the connective systems, and the equipment behind the isolating device should have the same seismic requirements with the connective systems. Setting of interfaces should be convenient for operation and connection.

9. Should be special requirements (regulations) cover the special technical provisions dedicated for recovery of off-site power, multi-units considerations, treatment of liquid radioactive effluents.

Yes, Revision of HAF 102 2016 “Safety of nuclear power plants: design”:

- 6.9.1 Systems shall be provided at the nuclear power plant for treating liquid and gaseous radioactive effluents to keep their amounts below the authorized limits on discharges and as low as reasonably achievable.

Twelfth Five-year Plan and 2020 long-term goal for nuclear safety and radioactive pollution prevention and control

- Enhance the safety of NPPs under construction: The SAMG should be completed and implemented before first fuel loading, covering all operation states and multiple units at a site, the availability and accessibility of important equipment and instruments should be assessed under severe accidents.

IV. Methodology of severe accident analyses

1. Should the list of SA analyses be a subject of licensing?

Yes, revision of HAF 102 2016 "Safety of nuclear power plants: design"

- 5.1.9 A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.

2. How the completeness and representativeness of SA analyses could be confirmed (bounding criteria)?

revision of HAF 102 2016 "Safety of nuclear power plants: design"

- 5.1.9 <u>A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments</u> for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.

Twelfth Five-year Plan and 2020 long-term goal for nuclear safety and radioactive pollution prevention and control

- "During the 12th Five-year Plan period (before 2015), the new NPPs should have comprehensive measures to prevent and to mitigate severe accidents, the CDF should be lower than 10^{-5}/year and the LRF should be lower than 10^{-6}/year."
- "For NPPs built in/after the 13th Five-year Plan period (after 2015), the design should practically eliminate the possibility of large scale radioactive release".

3. What acceptance criteria are applicable for severe accident?

The general acceptance criteria of severe accident is the integrity of the containment.

Twelfth Five-year Plan and 2020 long-term goal for nuclear safety and radioactive pollution prevention and control

- "During the 12th Five-year Plan period (before 2015), the new NPPs should have comprehensive measures to prevent and to mitigate severe accidents, the CDF should be lower than 10^{-5}/year and the LRF should be lower than 10^{-6}/year."
- "For NPPs built in/after the 13th Five-year Plan period (after 2015), the design should practically eliminate the possibility of large scale radioactive release".

İn revision of HAF 102 2016 "Safety of nuclear power plants: design":

- 6.3.5.4 Design provision shall be made to prevent the loss of the structural integrity of the containment in all plant states. The use of this provision shall not lead to an early radioactive release or a large radioactive release.

And there are also some acceptance criteria for main severe accident phenomena.

General Technical Requirements on post-Fukushima Nuclear Accident Improvement Measures for NPPs (Tentative).

- The hydrogen concentration should be less than 10%(V/V), assuming the hydrogen generated from the metal-water reaction involving 100% of the fuel cladding metal in the active fuel region and distributed uniformly in the containment.

The pressure of RPV should not exceed 2MPa when RPV failure in severe accident to prevent HPME and DCH.

4. Should the codes dedicated for SA analyses (neutronic, thermohydraulic, containment, source term) be under licensing control? If yes, what are requirements for validation and verification of these codes?

HAD 102/06 "Design of Reactor Containment Systems"

- 6.1.2 The validation domain of the computer codes used for evaluating all pertinent parameters should be verified to cover their expected range of variation adequately. Computer codes should not be used beyond their validation domain. As an exception, the use of computer codes beyond their range of validation might possibly be acceptable in areas for which it is widely recognized that there is a lack of coherent data. Such exceptions should be allowed only on the following conditions:
  - — The exception is clearly specified.
  - — A comprehensive sensitivity analysis is carried out to evaluate the effects of variations in the assumptions and in the modelling.
  - — An independent assessment is made of the credibility of the results.
  - — Appropriate margins are introduced if knowledge is limited.

Some typical codes used in China for severe accident analysis are MAAP, MELCOR, GASFLOW, MC3D, and COM3D etc.

Appendix F – Regulatory approaches used in Hungary

QUESTIONNAIRE ON SEVERE ACCIDENT REQUIREMENTS AND PRACTICES

I. General

1. What types of regulatory controls are installed in respect of severe accident (i.e. licensed verses voluntary initiatives)?

In Hungary the regulatory control is installed in the regulations.

2. What is the legal basis this matter (laws, requirements, regulations)?

- Act CXVI of 1996 on Atomic Energy
- Governmental Decree 118/2011 (VII. 11.) Korm. on the nuclear safety requirements of nuclear facilities and on related regulatory activities
- Nuclear Safety Code: Annexes 1 to 10 of the Governmental Decree 118/2011 (VII. 11.) Korm.

II. Procedures and instructions

1. Should guidelines and/or procedures be a part of licensing?

Yes, they are part of the licensing:

1.2.5.0700. The following shall be **attached to the commissioning license application**, as having been reviewed based on the experience of the activities performed on the basis of the commissioning license:

a) the updated Final Safety Analysis Report which shall verify – taking into consideration the results of commissioning tests – that

aa) the nuclear facility operates in compliance with the valid design basis,

ab) the inspection, management, emergency operating and accident management provisions necessary for the safe operation are suitable for the attainment of the formulated objectives, and

ac) the safe operation is ensured under the operational limits and conditions set out in the Final Safety Analysis Report,

b) Operational Limits and Conditions document,

c) the document describing the procedures which ensure the maintenance of the condition of systems and system components important to nuclear safety as specified in the designs and the Final Safety Analysis Report,

d) emergency operating procedures,

e) accident management procedures*, and*

f) Nuclear Emergency Preparedness and Response Plan of the nuclear facility.

4.5.3.0430. The emergency operating instructions and **accident management guidelines** applicable to DEC operating conditions shall be **primarily based on appropriately qualified system components and measurements**.

2. Should licensing cover as prevention and mitigation consequences of severe accident?

4.5.3.0300. **Accident management guidelines shall be provided in order to mitigate the consequences of severe accidents** for those instances when measures applied to restore or substitute lost safety functions could not effectively prevent core damage.

3a.2.2.7700. The required means of accident management shall be designed and accident management guidelines shall be developed for the efficient mitigation of the consequences of beyond design basis conditions analysed in detail, including severe accident processes resulting in a full fuel melting, in such a way that the hazard to the environment and the population remains below a predefined, manageable level if the processes and means of accident management operate successfully.

3. Should structure and content of the procedure and instructions be a part of regulatory control?

Yes, they are part of the licensing:

1.2.5.0700. The following shall be **attached to the commissioning license application**, as having been reviewed based on the experience of the activities performed on the basis of the commissioning license:

a) the updated Final Safety Analysis Report which shall verify – taking into consideration the results of commissioning tests – that

aa) the nuclear facility operates in compliance with the valid design basis,

ab) the inspection, management, emergency operating and accident management provisions necessary for the safe operation are suitable for the attainment of the formulated objectives, and

ac) the safe operation is ensured under the operational limits and conditions set out in the Final Safety Analysis Report,

b) Operational Limits and Conditions document,

c) the document describing the procedures which ensure the maintenance of the condition of systems and system components important to nuclear safety as specified in the designs and the Final Safety Analysis Report,

d) emergency operating procedures,

e) accident management procedures*, and*

f) Nuclear Emergency Preparedness and Response Plan of the nuclear facility.

If the procedure changes, the licensee needs to apply for a modification license.

4. How deep (in detail) should be monitored structure and content of procedures and/or instructions? Please consider the following items: entry criteria; exit criteria; criteria for the transition from prevention to mitigation domain of SA procedures and/or instructions; harmonization procedure and/or instructions dedicated for SAM with other emergency procedures and instructions; coverage of such SAM aspects as malty-unit accidents, different locations of the fuel (reactor' spent fuel pool, on-site fuel storage); distribution of the responsibility during the application of emergency procedure and/or instructions in the course of SA (who make a decision?);

4.5.3.0110. In the case of a nuclear power plant having more than one unit, the emergency operating instructions and accident management guidelines shall also take into account the simultaneous accident or severe accident condition of more than one reactor and spent fuel pool, and the resources required for their implementation shall be determined also by considering thereof, including external assistance to be used. Special attention shall be paid to potential interactions between the reactor and the spent fuel pool during such accident situations.

4.5.3.0200. Emergency operating procedures shall be provided for the management of beyond design basis accidents up to the point of fuel melting, in order to restore or substitute lost safety functions and prevent fuel melting.

4.5.3.0300. Accident management guidelines shall be provided in order to mitigate the consequences of severe accidents for those instances when measures applied to restore or substitute lost safety functions could not effectively prevent core damage.

4.5.3.0400. Instructions for DEC1 operating conditions may be only symptom-oriented.

4.5.3.0600. The emergency operating procedures and accident management guidelines shall be verified and validated in the form in which they will be used in order to ensure administrative and technical correctness for the benefit of the nuclear power plant and the compatibility with the available human resources and the environment in which they will be applied

4.5.3.0700. The approach used for the nuclear power plant specific validation and verification shall be documented. During the validation of procedures and guidelines it shall be examined how effectively were the technical aspects of human factors taken into consideration. The validation of emergency operating procedures shall be based on representative simulation, with the use of a simulator where possible.

4.5.3.0410. Those measures shall be laid down in the emergency operating instructions and accident management guidelines by which it can be ensured to the extent reasonably achievable, in the case of a nuclear power plant having more than one unit, that one unit can support the other, in order to minimise the consequences.

4.5.3.0430. The emergency operating instructions and accident management guidelines applicable to DEC operating conditions shall be primarily based on appropriately qualified system components and measurements.

4.5.3.1000. During the preparation for the management of severe accidents, the transition from emergency operating instructions to accident management guidelines shall be practised.

3a.2.2.7600. Reaching a safe condition after a severe accident shall be ensured within the reasonably shortest time, but not later than within 168 hours, by restoring the damaged systems or by operating the accident management systems providing for the management of DEC operating conditions.

3a.2.2.8900. It shall be ensured by the appropriate design that:

a) regarding operator interventions:

aa) in the case of events resulting in DB2-4 or DEC operating conditions, there shall be no need for operator interventions for 30 minutes in the control room, and for one hour outside of the control room for complying with the release limits defined in the design,

ab) there shall be no need for on-site mobile light equipment to prevent a fuel melting within six hours in the case of an event resulting in DEC operating conditions, and to preserve the containment function within 24 hours in the case of an event resulting in DEC operating conditions and for 72 hours in the case of an event resulting in DB2-4 operating conditions,

ac) in the case of an event resulting in DB2-4 and DEC operating conditions, there shall be no need for on- or off-site mobile heavy equipment for 72 hours, and

ad) in the case of events resulting in DEC operating conditions, the containment system shall withstand the hazards for at least 12 hours, but possibly for 24 hours, without operator intervention;

b) regarding the heat sink:

ba) an appropriate heat sink shall be available on the long term in the case of events resulting in DB2-4 and DEC operating conditions,

bb) the emergency feedwater reserve shall be sufficient for at least 24 hours, and

bc) at the site of the nuclear power plant, water sufficient at least for 72 hours shall be available for the cooling of the steam generators;

c) in electric power supply:

ca) independence from external supply shall be ensured for at least 72 hours in the case of DB1-4 and DEC operating conditions,

cb) the batteries fulfilling a F1 safety function shall fulfil the safety function for at least six hours without recharging after shifting from DB2-4 operating conditions to a DEC1 or DEC2 operating condition due to a total blackout, and

cc) batteries supplying power to systems used for the management of DEC2 operating conditions shall perform their safety function for at least 24 hours without recharging and shall be independent of the batteries performing the F1 safety function.

In the Hungarian legislation there are no requirements on the distribution of the responsibility.

5. How the operator must confirm the correctness and sufficiency of the procedures and instructions? Should analyses and/or experiments be an obligatory part of validation and verification process for procedures and instructions?

4.5.3.0500. The accident management guidelines shall be developed systematically using a nuclear power plant specific approach. The accident management guidelines shall contain such strategies which allow for the management of series of events identified in the analyses of severe accidents. The analyses shall aim at the identification of nuclear power plant vulnerability

4.5.3.0600. The emergency operating procedures and accident management guidelines shall be verified and validated in the form in which they will be used in order to ensure administrative and technical correctness for the benefit of the nuclear power plant and the compatibility with the available human resources and the environment in which they will be applied.

4.5.3.0700. The approach used for the nuclear power plant specific validation and verification shall be documented. During the validation of procedures and guidelines it shall be examined how effectively were the technical aspects of human factors taken into consideration. The validation of emergency operating procedures shall be based on representative simulation, with the use of a simulator where possible.

III. Equipment

1. Should a minimal list of technical means dedicated for SAM be determined by regulator?

3a.2.2.6600. The DEC analyses shall identify all reasonably achievable measures by which severe accidents can be avoided. Irrespective of the success of the identified measures, preparations shall also be made for severe accidents. As part of the analyses, all reasonably achievable solutions by which the consequences of severe accidents can be mitigated shall also be identified.

3a.2.2.7400. During design, the accident management functions and the pressure reduction and hydrogen removal systems performing such functions during accidents shall be determined to such an extent that high-pressure processes in events causing fuel melting and early damage to the containment can be avoided

3a.2.2.7500. The functions mitigating the consequences of accidents and, if necessary, the systems providing these functions shall be specified to such an extent that in severe accidents the molten fuel can be retained in a cooled condition within the containment.

2. Should severe accident management provisions (i.e. using mobile equipment verses hardening on site equipment) be performance based or prescribed by regulation?

The Nuclear Safety Codes are rather performance based than prescriptive.

3. Should be installed special requirements (regulations) in respect of mission times and minimum capacity for equipment dedicated for SAM?

3a.2.2.8900. It shall be ensured by the appropriate design that:

a) regarding operator interventions:

ab) there shall be no need for on-site mobile light equipment to prevent a fuel melting within six hours in the case of an event resulting in DEC operating conditions, and to preserve the containment function within 24 hours in the case of an event resulting in DEC operating conditions and for 72 hours in the case of an event resulting in DB2-4 operating conditions,

ac) in the case of an event resulting in DB2-4 and DEC operating conditions, there shall be no need for on- or off-site mobile heavy equipment for 72 hours, and

3a.2.2.6900. Following DEC1 operating conditions considered in the extended design basis, the achievement of a controlled condition shall be ensured within 24 hours, and the achievement of a safe shutdown condition within 72 hours, at the latest.

3a.2.2.7600. Reaching a safe condition after a severe accident shall be ensured within the reasonably shortest time, but with regard to Section 146 a) to c) of Annex 10, not later than within 168 hours, by restoring the damaged systems or by operating the accident management systems providing for the management of DEC operating conditions.

4. Should be special requirements (regulations) be installed for using mobile equipment versus stationary installed equipment?

There are no special requirements regarding using mobile equipment versus stationary installed equipment. The only regulations mentioning mobile equipment are the following:

3a.2.2.8900. It shall be ensured by the appropriate design that:

a) regarding operator interventions

ab) there shall be no need for on-site mobile light equipment to prevent a fuel melting within six hours in the case of an event resulting in TAK operating conditions, and to preserve the containment function within 24 hours in the case of an event resulting in TAK operating conditions and for 72 hours in the case of an event resulting in TA2-4 operating conditions,

ac) in the case of an event resulting in TA2-4 and TAK operating conditions, there shall be no need for on- or off-site mobile heavy equipment for 72 hours

3a.3.7.0400. Permanently installed or mobile, automatic or manual fire extinguishing systems shall be installed, which shall be so designed and placed that their failure or incorrect or inadvertent operation does not significantly affect the ability to perform the safety functions of the systems, structures and components important to nuclear safety

3a.7.1.1000. If the use of mobile equipment forms part of accident response, such fixed connection points shall be established for it that can also be used in TAK1 and TAK2 operating conditions from a physical and radiological point of view.

5. Should be special requirements (regulations) related to I&C for SA (I&C for SAM and indicate the status of SA progress)?

3a.2.2.6800. An alternative power supply facility shall be provided for avoiding a total blackout.

3a.4.5.0500. An appropriate power supply shall be provided for the cases of DEC operating conditions in accordance with the required interventions and timeframe established by DEC analyses, taking into account hazard factors of natural origin.

3a.4.5.0800. The characteristics of uninterruptible power supply and the duration of allowable loss of essential energy supply shall be determined by safety substantiation. Batteries performing a function under DEC1 and DEC2 operating conditions shall have an appropriate capacity until they can be recharged or until another power supply solution can be provided.

3a.4.5.1300. An electricity source shall be designed, which:

a) is independent physically and in terms of system technology of the safety power source designed for the handling of DB2-4 operating conditions, and

b) is able to provide an appropriate power supply to prevent a DEC2 operating condition and to mitigate its consequences in the case of a total blackout.

3a.4.5.1400. Instrumentation suitable for the measurement of parameters necessary for monitoring fundamental safety functions shall be provided, thus ensuring the availability of information necessary for the reliable and safe operation of the nuclear power plant unit and the management of events resulting in DB2-4 and DEC1 and DEC2 operating conditions.

3a.4.5.2700. The instrumentation and control systems shall ensure:

c) actuating and monitoring devices,

cd) for manual interventions necessary for accident management.

3a.4.5.5200. The instrumentation shall provide information on the condition of critical safety functions and the process systems required for the management of the operating condition even under the circumstances of DB1-4, DEC1 and DEC2 operating conditions.

6. Should be special requirements (regulations) related hydrogen management in the containment under the SA conditions?

3a.2.2.7400. During design, the accident management functions and the pressure reduction and hydrogen removal systems performing such functions during accidents shall be determined to such an extent that high-pressure processes in events causing fuel melting and early damage to the containment can be avoided.

3a.3.2.1700. It shall be ensured that the physicochemical properties of the materials used in the containment can avoid hydrogen production during events resulting in DB2-4 and DEC1 operating conditions.

3a.4.6.1500. The containment as a system shall comprise:

a) all important parts of the primary circuit,

b) systems capable of controlling pressures and temperatures,

c) isolation elements, and

d) instruments serving for the management and removal of fission products, hydrogen, oxygen and other materials released into the containment atmosphere.

3a.4.6.2200. The cleaning of the containment atmosphere shall be so performed that the systems used for the management and monitoring of fission products, hydrogen, oxygen and other materials potentially released into the containment atmosphere ensure the reduction of the quantity and concentration of the fission products released into the environment, assuming a single failure, as well as the control of concentration of hydrogen or oxygen in the containment atmosphere in order to ensure the integrity of the containment.

7. Should be special requirements (regulations) for primary pressure decrease?

3a.3.3.1200. If a higher than allowable pressure can develop in them, the pressure retaining equipment and pipelines shall be equipped with an appropriate pressure limiting device. The pressure limiting devices shall be so designed that if they operate, the quantity of radioactive materials released into the environment has the lowest reasonably achievable level.

3a.3.3.1300. If a system or component important to nuclear safety is in contact with a system or component the operating pressure of which is higher than that of the former one, the system or component shall be designed for the pressure values of the latter system or component, or it shall be ensured by design solutions that even in the case of a single failure, the pressure in the system or component designed for the lower pressure does not exceed the design value.

8. Should be special requirements (regulations) related to auxiliary equipment to provide water injection into the reactor vessel, steam generators and containment?

3a.4.3.1100. If the capability of transferring the residual heat to the ultimate heat sink cannot be demonstrated for every operating condition with high reliability, a secondary ultimate heat sink and systems required for its operation shall be provided, which ensure through their location and design solutions that the heat removal safety function is not lost as a result of external hazard factors.

3a.4.6.1600. The heat removal system of the containment shall ensure the quick reduction of the pressure and temperature in the containment following a loss of coolant event, then it shall ensure their maintenance at a reasonably achievable low level, assuming single failure.

3a.7.1.1000. If the use of mobile equipment forms part of accident response, such fixed connection points shall be established for it that can also be used in DEC1 and DEC2 operating conditions from a physical and radiological point of view.

9. Should be special requirements (regulations) cover the special technical provisions dedicated for recovery of off-site power, multi-units considerations, treatment of liquid radioactive effluents.

3a.2.2.7900. During the analysis of external hazard factors resulting in DEC operating conditions, at least the following shall be performed for the identification of reasonably achievable safety enhancement measures:

a) the severity of the given event beyond which the fundamental safety functions cannot be ensured shall be determined,

b) it shall be demonstrated that sufficient margins are available for avoiding the cliff edge effect,

c) the most efficient ways of providing the fundamental safety functions shall be identified and evaluated,

d) events that simultaneously affect more than one unit or a redundant system, structure or component or have an effect on the on-site and regional infrastructure, off-site services and protective measures shall also be taken into account,

e) it shall be demonstrated that in the case of a nuclear power plant with more than one unit, common use resources are available in sufficient quantities, the compliance with which shall also be confirmed by on-site inspections.

3a.3.5.1600. In accident situations, it is allowed to apply connected support systems between the individual units if it can be demonstrated that they help the restoration of a given safety function during accident management. No interconnection is allowed between the units that would increase the probability or severity of consequences in the case of any unit.

3a.2.1.3100. In the case of a nuclear power plant with more than one unit, the independence of the units shall be ensured to the reasonable extent

3a.3.5.1500. In the case of a nuclear power plant with more than one unit, every unit shall have its own safety systems, structures and components for management of DB1-4 and DEC1 and DEC2 operating conditions to the extent reasonably achievable. The safety systems may be shared between the units only if warranted from a safety point of view

3a.4.5.1200. In the case of a nuclear power plant with more than one unit, direct electrical connection between the units shall be provided to the extent reasonably achievable in such a way that the propagation of possible failures from one unit to another can be practically excluded.

3a.6.3.0800. It shall be ensured by an appropriate technical solution that the liquid radioactive wastes collected in the accessory buildings containing systems, structures and components coming into contact with radioactive media or, if the nuclear power plant has it, in the secondary containment can be returned to the containment also in DB and DEC operating conditions if their quantity exceeds the capacity of the liquid waste processing system

3a.6.3.0900. It shall be estimated with appropriate margins that what type and what quantities of radioactive wastes are expected to be produced during DB3-4 and DEC1 and DEC2 events as well as their management and response to them. In their knowledge, solutions suitable for the interim storage and management of wastes shall be designed and their location shall be designated on-site.

3a.4.3.1000. In the case of systems containing irradiated fuel assemblies, such as the shutdown nuclear reactor or spent fuel pool, capabilities shall be provided for passive heat removal.

4.5.3.0110. In the case of a nuclear power plant having more than one unit, the emergency operating instructions and accident management guidelines shall also take into account the simultaneous accident or severe accident condition of more than one reactor and spent fuel pool, and the resources required for their implementation shall be determined also by considering thereof, including external assistance to be used. Special attention shall be paid to potential interactions between the reactor and the spent fuel pool during such accident situations.

IV. Methodology of severe accident analyses

1. Should the list of SA analyses be a subject of licensing?

3a.2.2.6300. For the extension of the design basis, the following shall at least be taken into account if they are not included in the design basis and if they are applicable for the given power plant type:

a) total blackout,

b) loss of systems performing reactor shutdown functions required under DB2-4 operating conditions,

c) rupture of a steam pipeline with additional damage to the heat exchange surface of the steam generator,

d) events that bypass the containment and result in direct releases to the environment,

e) total loss of feedwater,

f) loss of coolant with the total loss of an emergency core cooling system type,

g) uncontrolled level decrease during an operating condition involving natural circulation with partially filled loop or refuelling,

h) total loss of one or more auxiliary systems of equipment providing fundamental safety functions,

i) loss of cooling of the active core during the removal of residual heat,

j) loss of cooling of the spent fuel pool,

k) uncontrolled dilution of boric acid,

l) simultaneous rupture of more than one heat exchanger tube of the steam generator,

m) loss of safety systems required for the long-term management of an assumed initiating event,

n) loss of the containment pressure reduction function in operating conditions when it would be needed,

o) other events resulting in fuel melting,

p) crash of a military or civilian aircraft,

q) events resulting in multiple failures.

2. How the completeness and representativeness of SA analyses could be confirmed (bounding criteria)?

If the SA analyses is carried out based on the guideline N3a.33 (Deterministic safety analyses of severe accidents), than HAEA accepts its completeness and representativeness.

3. What acceptance criteria are applicable for severe accident?

3a.2.4.0700. For the fulfilment of the criterion of limited environmental impacts, for events resulting in DEC1 operating conditions and for events resulting in DEC2 operating conditions with consideration to the specifications of Section 3a.2.2.7000, it shall be demonstrated that

a) no urgent protective measures are required beyond a distance of 800 m from the nuclear reactor;

b) there is no need for any kind of temporary action, i.e. the temporary evacuation of the population, beyond a distance of 3 km from the nuclear reactor;

c) there is no need for any kind of subsequent protective measure, i.e. the final re-settlement of the population, beyond a distance of 800 m from the nuclear reactor

d) there is no need for any long-term restriction on food consumption.

4. Should the codes dedicated for SA analyses (neutronic, thermohydraulic, containment, source term) be under licensing control? If yes, what are requirements for validation and verification of these codes?

No requirements for code certification. The used codes shall be validated and verified.