Nuclear Regulation
NEA/CNRA/R(2022)2
April 2025
www.oecd-nea.org

# BWR Severe Accident Mitigation Guidelines (SAMG) Information Compilation and Reactor Water Level Measurement

OECD
BETTER POLICIES FOR BETTER LIVES
NEA
NUCLEAR ENERGY AGENCY

Nuclear Energy Agency

**NEA/CNRA/R(2022)2**

**Unclassified** **English text only**

**7 April 2025**

**NUCLEAR ENERGY AGENCY**
**COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES**

# BWR Severe Accident Mitigation Guidelines (SAMG) Information Compilation and Reactor Water Level Measurement

This document is available in PDF format only.

**JT03563860**

## ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 38 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, Colombia, Costa Rica, Czechia, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Korea, Latvia, Lithuania, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Türkiye, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

## NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 34 countries: Argentina, Australia, Austria, Belgium, Bulgaria, Canada, Czechia, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Romania, Russia (suspended), the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Türkiye, the United Kingdom and the United States. The European Commission and the International Atomic Energy Agency also take part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes;
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management and decommissioning, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

# COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES (CNRA)

The Committee on Nuclear Regulatory Activities (CNRA) addresses NEA programmes and activities concerning the regulation, licensing and inspection of nuclear installations with regard to both technical and human aspects of nuclear safety. The Committee constitutes a forum for the effective exchange of safety-relevant information and experience among regulatory organisations. To the extent appropriate, the Committee reviews developments which could affect regulatory requirements with the objective of providing members with an understanding of the motivation for new regulatory requirements under consideration and an opportunity to offer suggestions that might improve them and assist in the development of a common understanding among member countries. In particular it reviews regulatory aspects of current safety management strategies and safety management practices and operating experiences at nuclear facilities including, as appropriate, consideration of the interface between safety and security with a view to disseminating lessons learnt. In accordance with *The Strategic Plan of the Nuclear Energy Agency: 2017-2022*, the committee promotes co-operation among member countries to use the feedback from experience to develop measures to ensure high standards of safety, to further enhance efficiency and effectiveness in the regulatory process and to maintain adequate infrastructure and competence in the nuclear safety field.

The committee promotes transparency of nuclear safety work and open public communication. In accordance with the NEA Strategic Plan, the committee oversees work to promote the development of effective and efficient regulation.

The committee focuses on safety issues and corresponding regulatory aspects for existing and new power reactors and other nuclear installations, and the regulatory implications of new designs and new technologies of power reactors and other types of nuclear installations consistent with the interests of the members. Furthermore, it examines any other matters referred to it by the NEA Steering Committee for Nuclear Energy. The work of the committee is collaborative with and supportive of, as appropriate, that of other international organisations for co-operation among regulators and consider, upon request, issues raised by these organisations. The Committee organises its own activities. It may sponsor specialist meetings, senior-level task groups and working groups to further its objectives.

In implementing its programme, the committee establishes co-operative mechanisms with the Committee on the Safety of Nuclear Installations (CSNI) in order to work with that committee on matters of common interest, avoiding unnecessary duplications. The committee also co-operates with the Committee on Radiological Protection and Public Health (CRPPH), the Radioactive Waste Management Committee (RWMC), and other NEA committees and activities on matters of common interest.

# *Table of contents*



**List of tables**

# *List of abbreviations and acronyms*

| | |
|---|---|
| ABWR | Advanced boiling water reactor |
| AEC | Atomic Energy Council of Chinese Taipei |
| AM | Accident Management |
| AMT | Accident management team |
| AOP | Abnormal operating procedures |
| AOT | Allowed outage time |
| APRI | Accident phenomena of risk importance (SA research project in Sweden) |
| ATWS | Anticipated transient without scram |
| BWR | Boiling water reactor |
| BWROG | Boiling Water Reactors Owners' Group |
| CAPS | CNRA activity proposal sheet |
| CCI | Core-concrete-interaction |
| CNRA | Committee on Nuclear Regulatory Activities (NEA) |
| CSN | Spanish Nuclear Safety Council |
| CV | Containment venting |
| DEC | Design extension condition |
| DW | Drywell |
| EDMG | Extensive damage management guidelines (guidelines developed in Spain to make use of the portable equipment for RPV-containment-fuel pool injection, and for other functions related to DEC) |
| ELAP | Extended loss of AC power |
| ENSI | Swiss Federal Nuclear Safety Inspectorate |
| EOP | Emergency operating procedures |
| EPG | Emergency procedure guidelines |
| FCV | Filtered containment venting |
| FLEX | Flexible use equipment[1] |
| GRS | Gesellschaft für Anlagen- und Reaktorsicherheit (Germany) |
| HPCS | High pressure core spray system |
| HVAC | Heat ventilation air conditioning |
| IAEA | International Atomic Energy Agency |
| KK6/7 | Kashiwazaki-Kariwa Unit 6 and 7 |
| LCO | Limiting condition for operation |
| LOCA | Loss-of-coolant accident |

1 Equipment that maintains long-term core and spent fuel cooling and containment integrity that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from off site.

| | |
|---|---|
| LPCS | Low pressure core spray system |
| MCCI | Molten core concrete interaction |
| MSCRWL | Minimum steam cooling reactor pressure vessel water level[2] |
| NEA | Nuclear Energy Agency |
| NEI | Nuclear Energy Institute |
| NERP | Nuclear emergency response plan (Chinese Taipei) |
| NHB | Emergency operating manual (Germany) |
| NNERC | National Nuclear Emergency Response Centre (Chinese Taipei) |
| NRA | Nuclear Regulation Authority in Japan |
| NRC | Nuclear Regulatory Commission (United States) |
| PAR | Passive autocatalytic recombiner |
| PC | Primary containment |
| PCT | Peak cladding temperature |
| PWR | Pressurised water reactor |
| PWROG | Pressurised Water Reactor Owners' Group |
| RC/L | Reactor pressure vessel level control |
| RC/P | Reactor pressure vessel pressure control |
| RC/Q | Power control |
| RCS | Reactor coolant system |
| RPV | Reactor pressure vessel |
| RR | Radioactive release |
| SA | Severe accident |
| SAG | Severe accident guidelines |
| SAM | Severe accident management |
| SAMG | Severe accident management guidelines |
| SAWA | Severe accident water addition |
| SAWM | Severe accident water management |
| SC | Secondary containment |
| SFP | Spent fuel pool |
| SRO | Senior reactor operator |
| SSM | Swedish Radiation Safety Authority |
| SSMFS | SSM's regulations |
| STUK | Radiation and nuclear safety authority in Finland |
| TAF | Top of active fuel |
| TPC | Taipower Company |
| TSC | Technical support centre |

[2] MSCRWL defined to be the lowest RPV water level at which the covered portion of the core is capable of generating sufficient steam to preclude PCT in the uncovered portion of the core from exceeding 1500°F

| | |
|---|---|
| TSG | Technical support guideline |
| URG | Ultimate response guideline |
| US NRC | US Nuclear Regulatory Commission |
| WPBWR | Working Party on BWR (NEA) |
| WW | Wetwell |
| YVL | Finnish authority's regulatory guides |

# *Executive summary*

This report presents nuclear plant operator and regulatory practices in the area of severe accident management (SAM) in general and more specifically the regulation of its severe accident mitigation guidelines (SAMGs). This is accomplished by describing the content and implementation of SAMGs in the participating countries and economies. The information was compiled via a questionnaire developed by and distributed among members of the Nuclear Energy Agency (NEA) Working Party on Boiling Water Reactors (WPBWR) within the Committee on Nuclear Regulatory Activities (CNRA).

The results from the questionnaire show that most participants use the reactor pressure vessel (RPV) water level as a criterion for transfer from the emergency operating procedures (EOPs) to the SAMGs. The exact RPV water level differs between countries and reactor design, but the goal is to identify when one can expect severe core damage if the accident progression continues.

The results from the questionnaire also show that there are many similarities in SAMG concepts among the participating countries and economies. However, the practice of implementing SAMG varies among licensees due to different reactor designs and regulatory requirements.

While no direct conclusions should be drawn from this report regarding the quality of various SAMGs, it does provide insights that may be useful as NEA member countries and economies consider enhancing the regulatory framework and implementation of SAMGs at nuclear power plants. This report was approved by the CNRA at its 47th meeting on 2-3 June 2022.

# 1. Background

The purpose of the questionnaire was to collect and compile information on practices in the area of SAMG. The basis for this was the Swedish SSM's input to the WPBWR's work. It was identified that the quality of the SAMG for Swedish BWRs was poor, and Sweden initiated a project to enhance the quality. To receive some input and support for the work, the SSM requested the assistance of WPBWR members to make a questionnaire regarding current practices in the countries and economies where BWRs are in operation. When this topic was proposed within the WPBWR, it was clear that more countries and economies were interested in this area, i.e. compiling information regarding industry and regulatory practices related to SAMG.

The participating countries and economies were:

- Finland
- Germany
- Japan
- Spain
- Sweden
- Switzerland
- Chinese Taipei
- United States

The questionnaire is presented in Annex A.

The main purpose of this activity is to benchmark the practices from all the regulators and to describe at a high level how the regulations are applied to SAMG in the participating countries and economies.

All countries and economies that participate in the work of WPBWR provided answers to the questionnaire.

# 2. Results

The following sections contain a discussion of the answers that each country and economy provided to the questionnaire.

## 2.1. General

### *2.1.1. Regulations*

**Finland**

Finland has two levels of regulation in the SAMG area; STUK Y/1/2018, Section 20 "Safety of operation", and YVL-guide A.6, "Conduct of operations at a nuclear power plant", paragraphs 710-711 and 713-719.

**Germany**

In Germany, the requirement for SAMG is presented in "Safety Requirements for Nuclear Power Plants", Section 3.1 (10).

**Japan**

While there are no explicit regulations for SAMG in Japan, there are Severe Accident Measures Requirements for light water reactors that provides regulations covering serious events, such as serious damage to the reactor core. The general criteria relevant for SAMG are included in "Review standard for the technical capability required for taking measures necessary for preventing the occurrence and expansion of a severe accident". These criteria show some items that each licensee's SAMG should include. For example, securing access routes, making spare parts available, preparing procedures, and having an organisational structure.

**Spain**

In Spain, the regulation relevant for SAMG is included in "Nuclear Safety Council Instruction IS-36" of 21 January 2015, on emergency operating procedures and the management of severe accidents at nuclear power plants.

**Sweden**

There are no explicit regulations for SAMG in Sweden. Instead there are some general requirements in the 5$^{th}$ chapter, § 2, SSMFS 2008:1, "The Swedish Radiation Safety Authority's Regulations and General Advice concerning Safety in Nuclear Facilities"[3] that are applied to SAMG. Below is a short description.

<u>Regulation</u>

Documented guidelines shall have been drawn up at the facility for measures which may be necessary to implement in order to control and mitigate the consequences of beyond-design-basis accidents.

<u>*General advice*</u>

Guidelines for management of accidents that have not been taken into account in the facility design should be developed to the extent that is possible and reasonable with respect to the need for the protection of the environment. The guidelines should be systematically developed and aim to assist in the management of accidents identified in

3. Since 1 March 2022 this regulation has been replaced by SSMFS 2021:6 "The Swedish Radiation Safety Authority's Regulations and General Advice concerning Operation of Nuclear Power Plants". With respect to SAMG there are no major changes in the regulation.

the plant-specific analyses of accidents that are not a part of the original design bases, including severe accidents. The guidelines should be well co-ordinated with the facility's EOP. The verification and validation of the facility's SAMG should be documented. Personnel involved in the management of severe accidents should be regularly trained in the use of the guidelines and the transition between instructions, while guidelines should be specifically highlighted.

**Switzerland**

In Switzerland, the ENSI guideline B12 regulates the SAMG requirements for the Swiss nuclear power plants.

**Chinese Taipei**

Chinese Taipei does not have a SAMG-specific regulation. The licensee voluntarily established its own SAMG based on BWROG severe accident guidelines. For maintaining the effectiveness of severe accident management at nuclear power plants, the regulator issues orders to require the licensee to enhance the SAMG in compliance with the latest version of BWROG severe accident guidelines.

**United States**

The United States does not have a regulation that requires SAMG for the operating fleet. However, the US nuclear power industry has made commitments to the NRC regarding SAMG using NEI 14-01, "Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents", published in February 2016 and the NRC confirms this commitment through inspections as noted below:

- Industry committed to adopting the latest revision of the applicable owner's group generic SAG within three years or two refuelling cycles from the issue date.
- The NRC updated the Reactor Oversight Process relative to industry's commitment. Inspection Procedure 71111.18, "Plant Modifications", requires a verification of inclusion of the latest SAMG by 2022. The PWROG issued new SAMG in February 2016 (Revision 0), while the BWROG issued a Revision 4 to the SAMG in June 2018.
- It should be noted that NRC inspectors will only verify that licensees have included the latest updates of SAMG into their procedures. The NRC will not inspect the effectiveness of the licensee's implementation of SAMG strategies.

### *2.1.2. Structure and content of EOP and SAMG per country/economy*

**Finland**

In Finland, the BWR units have symptom-based EOP for postulated accidents and design extension conditions. The SAMG is a stand-alone operator guideline that has a looping structure, with continuous tasks and once-performed actions. Only credited and separate systems designed and implemented to manage severe accidents are credited in the SAMG. The guide is based on the design basis path to manage severe reactor accidents and mitigate their consequences. The SAMG covers all safety functions that are relevant to severe accident management: depressurisation of the primary circuit, hydrogen management, ensuring ex-vessel melt retention, pH-control, detection and prevention of re-criticality, ensuring containment isolation and integrity, management of containment pressure and monitoring the accident progression. The SAMG has the needed system operations to control and manage the severe accident situation, including references to specific system guides. There are instructions to verify actions, monitor parameters, start or stop systems, and close or open specific valves. Besides instructions for the system operations, the SAMG includes the most important emergency preparedness actions.

**Japan**

The Japanese BWR SAMG consists of three parts, abnormal operating procedures (AOP), emergency operating procedures (EOP) and severe accident operating procedures (SOP).

AOP: Procedures for typical scenarios within the scope of design basis accidents.

EOP: Procedures for the corresponding operation according to the observed plant parameter trends.

SOP: Procedures for preventing the expansion of severe accidents and mitigating their impact.

The EOP are classified into “reactor control,” “primary containment control,” “secondary containment control,” “unexpected situation,” “AC/DC power supply restoration,” and “EOP/SOP interface” according to each purpose. They consist of a “flowchart”, which is a visualisation of each procedure or manual for the operator’s easy recognition, and a “corresponding operation” for each procedure. The SOP is composed of a “flow chart”, which is a visualisation of the contents of the corresponding operation (RPV control, PCV control, R/B control) to be performed after core damage for the operators’ easy recognition.

**Spain**

The IS-36 establishes the requirements to be met by the Spanish nuclear power plants in relation to the EOP and SAMG. The requirements for structure and content of the EOP and SAMG are described in the fourth chapter of this Instruction. IS-36 requires, e.g. that:

- The EOP shall be consistent with the rest of the plant operating procedures and with the SAMG.
- The SAMG shall be developed by means of a systematic process specific to each nuclear power plant.
- The procedures and guidelines shall be technically correct and accurate, complete, explicit, user-friendly and reliable (verified and validated), and shall be integrated and consistent with the rest of the man-machine interfaces of the facility.

Spain’s BWR EOP and SAMG were developed following the generic EOP/SAMG from the BWR Owners’ Group.

**Sweden**

Work is ongoing in Sweden in which the licensees are renewing the structure and content of their severe accident guidelines in line with established SAMG. Both licensees (from 2021 there will only be two sites with BWRs in Sweden) are co-operating in this work, but the new SAMG will be different between the sites.

**Switzerland**

The initial response to an abnormal event is guided by a symptom-based EOP, which contains checks of the automatic activation of safety systems as well as indications for manual activation of safety systems if they did not start automatically and transfers to other symptom-based or event-based procedures. The SAMG structure and content is plant specific. The common structure is a guidance for plant state diagnosis at the beginning and a related guidance for prioritisation of the measures to be taken. The goals of SAMG are to stop the process of core damage, to maintain the integrity of the containment, and to limit the radioactive releases as much as possible. The measures to achieve these goals contain the typical measure listed at the beginning of the questionnaire, see Annex A. The SAMG in Switzerland covers all relevant operational states.

**Chinese Taipei**

Chinese Taipei's BWR SAMG have two parts:

- Part 1: RPV and primary containment flooding in which water levels, RPV pressure and reactor power are monitored and controlled.
- Part 2: RPV, containment and radioactivity release control in which parameters for the containment are monitored and controlled. Included in the part is also the radioactivity release control.

The EOP contain parts to ensure that the core is cooled, the reactor is shut down, the containment's integrity is maintained, and the release of radioactive substances is limited. The EOP also include more detailed instructions of contingencies not only for controlling the parameters of the RPV under more degraded conditions but also when to exit the EOP and enter the SAMG.

**United States**

The BWR emergency procedure guidelines (EPG) and SAG are presented as a flow chart. Individual licensees develop their EOP using the guidance in the generic EPG, and their plant-specific SAMG from the SAG. Each EOP consists of a high-level guideline followed by specific guidance to address specific safety parameters (e.g. vessel level control, vessel pressure control, containment parameters). The specific guidance is provided in sub-tier flowcharts or "legs". The EOP should be entered whenever a defined entry condition occurs or an explicit direction to do so is encountered. Each EOP/SAG has specific "symptom-based" entry and exit criteria based on plant indications (e.g. vessel level is below Level 3). When plant parameters indicate that the core can no longer be adequately cooled (i.e. inability to maintain RPV water level above the minimum steam cooling RPV water level), operators are directed within the procedure to exit the EOP and enter the SAMG. When an EOP/SAMG is entered, the sub-tier procedures within that EOP/SAMG are all performed "concurrently" and prioritised by the senior reactor operator (SRO).

*EPG content*

The EPG contain four high-level guidelines:

- reactor pressure vessel (RPV) control;
- primary containment (PC) control;
- secondary containment (SC) control;
- radioactive release (RR) control.

*SAG content*

The SAGs consist of two high-level guidelines:

- RPV and primary containment flooding;
- containment radioactivity release control.

SAG strategies are much less prescriptive than the EOP strategies. They provide a big picture approach and priorities/goals for limiting core damage and its impact. Each strategy provides recommendations for RPV injection and venting, PC injection and venting, and the use of DW (drywell) and SC (secondary containment) sprays depending on plant conditions.

The BWROG also provides technical support guidelines (TSGs) that are used by the licensee's technical support centre (TSC) evaluators to assess plant status, safety function performance, and action strategy and priority. In addition, the BWROG provides calculation aids for the evaluators to make real-time plant-specific calculations to support their assessments, such as estimating the time to reach a specific PC water level given the current injection rate.

### *2.1.3. Criteria for transfer from EOP to SAMG*

As can be seen below most countries and economies use the RPV water level as a criterion for transfer from the EOP to the SAMG. The exact level differs between countries/economies and reactor designs, but the goal is to identify when one can expect severe core damage if the accident progression continues. In Germany, where the RPV level is not used, the core temperature is used instead; the aim, however, is the same.

Below is a table that summarises the criteria in a format that is not specific to countries or economies. Each country or economy's specific criteria are presented below the table.

**Table 2.1. Non-Country/economy Specific Criteria to transfer from EOP to SAMG**

| Type | Criteria for transfer from EOP to SAMG |
|---|---|
| Complete failure of reactor shutdown systems | Unsuccessful reactor trip with both control rod insertion and boration failing |
| Complete failure of core cooling systems | Time since core cooling stopped |
| Minimum cooling condition to prevent the core damage | • The RPV water level cannot be maintained above minimum steam cooling RPV water level (MSCRWL)<br>• A specific water level with time limit (30 min)<br>• Low water level in the RPV and no core cooling system in operation<br>• Core temperature > 336°C for more than 10 min |
| Core damage is occurring | • Local dose rate inside drywell >750 mGy/h<br>• High radiation in the containment<br>• High radioactivity in the containment<br>• High radioactivity in the upper drywell<br>• Hydrogen concentration in the containment<br>• High temperature in the water blowing valves in the system for pressure relief of the RPV |
| Shift manager's decision | When at the discretion of the shift manager, the need to switch to SAMG is determined |

**Finland**

Two transfer criteria are used:

- a specific water level with time limit (30 min); or
- an unsuccessful reactor trip with both control rod insertion and boration failing.

**Germany**

Two transfer criteria for each of the reactor and SFP are used:

- Reactor:
  - core temperature > 336°C for more than 10 min; or
  - local dose rate inside drywell >750 mGy/h.
- Spent fuel pool:
  - water temperature > 80°C; or
  - water level < 9.33 m and strongly decreasing; water level < 4.65 m issuing of the pre-alert.

**Japan**

BWR-specific severe accident countermeasure operations based on the new regulatory standards are currently under review. However, according to the example of Kashiwazaki-Kariwa Units 6 and 7, when the operator determines that the core is damaged it is decided to shift to countermeasures for severe accidents. This is stipulated in the safety regulations.

**Spain**

Five transfer criteria are used:

- The water level in the RPV cannot be restored and maintained above MSCRWL, having required "emergency depressurisation" in Contingency 1; or
- The water level in the RPV cannot be restored and maintained above MSCRWL, and it is not possible to inject the nominal flow rate with HPCS or LPCS in Contingency 1; or
- The water level in the RPV cannot be restored and maintained above the suction level of the jet pumps (2/3 of the core) in Contingency 1; or
- It has been determined that damage to the core is occurring during the flooding process established in Contingency 4; or
- When at the discretion of the shift manager, the need to switch to SAMG is determined.

**Sweden**

The criteria for transferring from EOP to SAMG are described in general terms in the bullet list below:

- ratio between time since scram and operation time for core cooling systems; or
- low water level in the RPV (some distance below TAF) and decreasing water level; or
- activity in the upper drywell; or
- hydrogen-concentration more than a specific value in wetwell; or
- high temperature in the water blowing valves in the system for pressure relief of the RPV.

**Switzerland**

The criteria for transferring from EOP to the SAMG are plant-specific. In general, a transfer takes place when core damage has happened or is soon to happen and cannot be prevented. The related parameters include core exit temperature (two plants), reactor level (two plants), hydrogen concentration in the containment (two plants), radiation level in the containment (one plant), state of violation of elementary protection goals (one plant) and adverse containment conditions in ATWS scenarios (one plant).

**Chinese Taipei**

Two transfer criteria are used:

- if the RPV water level cannot be maintained above MSCRWL; or
- if RPV water level cannot be determined and it is determined that core damage is occurring by performing the evaluations based on the TSG.

**United States**

In general, two criteria are used:

- determination that core damage is occurring;
- determination that the RPV level cannot be restored and maintained above the MSCRWL.

### *2.1.4. Spent fuel pool*

**Finland**

The spent fuel pool is not directly addressed in the SAMG of the BWRs of Finland.

**Germany**

Strategies and procedures for both shutdown modes of the plant and the SFP are available in the SAMG concept of the plant in order to mitigate severe accident progression in the event of a damaged RPV and SFP. Features regarding safety for design extension conditions related to the SFP have been installed and are briefly described as follows.

For the SFP, which is located outside the containment but inside the reactor building, an additional feeder line has been installed. The SFP can be fed through that line from outside the reactor building by using a mobile pump/fire truck. As a dedicated mitigative measure for the hydrogen management implemented in the aftermath of the Fukushima Daiichi severe accident, 12 passive autocatalytic recombiners (PARs) have been installed inside the reactor building above the SFP.

**Japan**

The SFP is included in the SAMG by giving directions on how to provide cooling and restoring the water level with mobile equipment. The purpose of this is to avoid boiling. Boiling in the SFP could impact the equipment in the reactor building that is used to manage severe accident scenarios.

**Spain**

The SFP is addressed in the SAMG. There are two sections, one for the control of temperature and another for the control of the water level in the SFP.

The temperature of the SFP is monitored in the same way as during normal operation of the plant. The temperature of the SFP is controlled to keep it below the high temperature alarm value (XX°C). When this temperature cannot be maintained, measures are taken to resume cooling capacity. If this is not successful and the SFP cannot be maintained below the design temperature (YY°C), it is ordered to align all available cooling systems to keep the temperature below this value. The cooling of the core and the fuel pool are simultaneous objectives of the EOP/SAG, and the available systems should be operated to satisfy both purposes as far as possible. However, although the available systems can alternate in their use of injection/cooling to the RPV and contribution to the SFP (provided that the continuous injection to the RPV is not required), the function of preventing the debris from breaking the RPV has preference over the functions of the SFP.

The level of the spent fuel pool is controlled according to the normal operating procedures and the response procedures in case of alarm to maintain the level between the high- and low-level alarm settings. Maintaining the level of the pool above the low-level alarm setting, the level is maintained above the overflows, allowing the water to flow to the skimmers and maintaining the NPSH of the spent fuel cooling system pumps. If the operator cannot restore and maintain the SFP water level above a specific level covering the fuel racks, then the water flow into the SFP can be increased using predefined alternative sources of water, except those that are used to mitigate a severe accident that may be occurring at the same time in the reactor. [4]

**Sweden**

The updated SAMG and parts of the EOP include measures to ensure that the water level in the SFP is above the TAF. One licensee has these measures in the EOP and the other licensee in the SAMG. The basis for remaining in the EOP is to prevent severe fuel damage in the SFP.

**Switzerland**

The SAMG covers accidents in the reactor and in the SFP. For accidents during shutdown, the state of the SFP is subject to the guidance on diagnosis and measures in the SAMG. For accidents during power operation, the EOP guides the monitoring of the SFP state and provides guidance for both measures to maintain or reestablish appropriate conditions or entry into the SAMG if necessary.

**Chinese Taipei**

The SFP is not addressed in the SAMG. Maintaining the spent fuel pool cooling and water level is, however, one of the main objectives of the URG, the Ultimate Response Guidelines, which can be viewed as a defense-in-depth supplement to the EOP and were developed for mitigating the reactor and spent fuel pool accident conditions caused by a large-scale severe compound external event like the Fukushima Daiichi accident. The management of the SFP in severe accident conditions includes the following measures: various SFP makeup strategies, installation of extra makeup and spray flow paths, and upgrades to the instruments for monitoring the water level and temperature in the SFP.

**United States**

Both the EOP and SAG address the SFP level and temperature by providing primary and alternate makeup sources and injection strategies based on specific level and temperature thresholds.

### *2.1.5. Licensees' organisation*

**Finland**

The control room personnel manage the severe accident situation in a nuclear power plant based on the SAMG. The emergency preparedness organisation can give advice

4. The purpose of these actions is to increase the level in the SFP using all the available systems: normal makeup systems, emergency systems or alternative systems (for instance, firefighting systems or alternate systems based on portable pumps using the connections implemented in the plant after the Fukushima Daiichi accident). The licensee will have a predefined list of the systems, and their delivery pressures, specified in plant specific guidelines as complementary information for the operators. Protecting the reactor coolant system has preference over protection of the spent fuel pool.

and strategies if accident progression deviates from SAMG, e.g. if some SAM-system fails to operate.

The head of the emergency preparedness organisation has the ultimate decision power for the site. The emergency preparedness organisation supports the control room and is responsible for all the emergency actions outside the control room.

**Germany**

In case of a beyond-design-basis accident, a crisis team will be installed to make decisions regarding severe accident management. The crisis team is subdivided into several teams (operation, monitoring, special services, radiological protection, etc.) and it is supported by the shift supervisor and the shift itself.

**Japan**

Basically, the initial response is performed by the operator for normal operation. Additional staff members for SAMG response after the accident occurrence are secured in the office. After that, additional personnel for long-term countermeasure are gathered as events progress.

As an example, at Kashiwazaki-Kariwa Units 6 and 7 during weekdays there are 231 persons as resident emergency on-site responders and 99 persons during nights and holidays. These persons in general terms come from the following organisations: The nuclear power plant, which is supported on-site by the plant manufacturer, subcontractors and the operator’s branch office. Upon request there is also a nuclear emergency assistance centre to support the accident management. Off-site there is also support from the operator’s head office and the plant manufacturer’s head office.

**Spain**

Onsite, the personnel in the main control room have the responsibility for carrying out EOP response, but the SAG are a responsibility of the technical support centre (TSC) personnel.

Onsite, the team for the management of severe accidents consists of three people: one co-ordinator and two evaluators. The transition from EOP to SAG cannot be made until the TSC and the severe accident management team are established and operational.

The staff of the main control room continues to operate the plant following the instructions given by the severe accident management team through the co-ordinator.

**Sweden**

In the updated instructions/guidelines, when the decision to transfer from the EOP to SAMG is taken, the operational decisions are also transferred from management Level 3 (shift manager) to management Level 2 or 1 depending on the radiological consequences of the decision. At the same time, the working processes shift so that the main control room and the technical support centre work in parallel with the support of identical symptom-based flow charts that describe each party’s responsibilities. The decision hierarchy changes so that the site manager (management Level 1) makes strategic decisions, plant manager (management Level 2) makes tactical decisions and the shift manager (management Level 3) makes the operational decisions in order to implement the strategic and tactical decisions.

**Switzerland**

For the evaluation of accident management measures a special team is designated to support (but not replace) the operators in the main control room. The team is composed of plant experts and safety engineers. They issue recommendations to the accident response co-ordinator.

**Chinese Taipei**

During emergency conditions, the onsite TSC will start to operate when the licensee escalates the emergency classification level from unusual event to alert. The TSC provides plant management and technical support to personnel located in the control room. It has technical data displays and plant records available to assist in the detailed analysis and diagnosis of abnormal plant conditions and any significant release of radioactivity to the environment. The TSC is the primary communications centre for the plant during an emergency.

When staff in the of TSC foresee the management of plant operations will transfer from the use of the EOP to the SAG, an AMT will start to operate and provide the TSC with appropriate suggestions for responding to the severe accidents according to the SAG and the TSGs. Once they enter a SAG, the reactor operators in the main control room will follow the instructions from the TSC.

The members of the AMT consist of the operation section manager, supporting shift manager, quality control section manager, and nuclear engineering section manager. An information sharing system has been developed to perform those actions required by the SAG. With this system, reading the flow charts becomes easier, the efficiency of group discussions improves, the information for decision-making becomes more transparent, and the contents of management guidelines become more complete.

The organisational structure for the AMT is as follows:

### Table 2.2. Chinese Taipei Accident Management Team Organisational Structure

| Division | Responsibility | Technical Support Manual |
|---|---|---|
| Plant operation | • Determine the priority with which systems should be returned to service<br>• Identify timing for actions directed by the SAG<br>• Evaluate containment flooding strategies | Plant-specific EOP/SAG action assessment guideline |
| Safety analysis and evaluation | • Evaluate the availability of plant systems<br>• Monitor the trend of control parameters<br>• Estimate the flow rate of water injection into the RPV | Plant-specific system status assessment guideline and plant status assessment guideline |
| Reactor engineering | • Confirm the reactor will remain shutdown<br>• Identify whether the RPV has been breached by core debris<br>• Estimate offsite Radioactivity Release rate | Plant-specific plant status assessment guideline |
| Safety parameter display system (SPDS) | • Evaluate the availability of instrumentation used to determine values of SAG control parameters<br>• Monitor the trend of control parameters | Plant-specific control parameter assessment guideline |

**United States**

NUREG-0654, *Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants* provides generic guidance to US nuclear power plants and state and local governments to develop radiological emergency plans. Ten CFR Part 50.47 contains regulatory requirements for emergency planning and preparedness. Ten CFR Part 50 Annex E contains additional details regarding emergency plan implementation and staff training requirements. Table B-1 in NUREG-0654 provides guidance for nuclear power plants when developing an emergency response organisation on-shift and augmented staffing levels. The table was recently updated, and US nuclear power plants can adopt the new format. NEI 12-01, *Guideline for Assessing Beyond Design Basis Accident Response and Communications*

*Capabilities* provides recommended criteria for use in performing an assessment to identify licensee personnel that should be available to respond to a beyond-design-basis external event affecting multiple units at a site.

In general, a nuclear power plant would be at a site area emergency or general emergency when entering the SAMG (most likely general emergency). Most nuclear power plants staff their emergency response facilities (technical support centre/operations support centre and emergency operations facility) at the alert level and above.

Since a plant would likely be at a general emergency when entering the SAMG, all of the staffing in Table B-1 would be applicable. A simplified version of Table B-1 is shown below (see Table B-1 in ADAMS ML18022A352 for a complete description). Individual nuclear power plant requirements are contained in the plant-specific emergency plan:

## Table 2.3. Simplified NUREG-0654 Table B-1

| Function | On-Shift Personnel | Technical Support Center (For Alert-level) | Emergency Operations Facility (for SAE/GE) |
|---|---|---|---|
| **Command and control** | Shift manager | Emergency co-ordinator | Emergency director |
| **Communications** | Communicator | Two communicators | Communicator |
| **Radiation protection** | Radiation protection personnel | Additional radiological protection personnel | No additional personnel |
| **Radiation protection supervision** | Shift manager | Site radiological protection co-ordinator | Radiation protection manager |
| **Dose assessments and projections** | Dose assessments and projection staff | Dose assessments and projection staff | Dose assessments and projection staff |
| **Emergency classification** | Emergency classification advisor | Emergency classification advisor | No additional personnel |
| **Engineering (for specific disciplines)** | Core/thermal hydraulics engineer | - Electrical/I&C<br>- Mechanical<br>- Core/Thermal<br>- Additional engineers as needed | No additional personnel |
| **Security** | Per security plan | Security liaison | No additional personnel |
| **Repair team** | None | (As Needed)<br>Electricians<br>Mechanics<br>I&C Technicians | No additional personnel |

In NEI 12-01, the Nuclear Energy Institute specifies recommended staffing considerations for expanded response functions that would be required for beyond-design-basis accidents. Table 3.1 of NEI 12-01 includes key roles and staffing considerations for the implementation of severe accident management strategies. The table suggests the following:

- Evaluation: A team for the evaluation of severe accident management strategies. This team would be located in the TSC or the emergency operations facility, and its composition would be governed by plant-specific procedures and guidelines. There would be one team for each affected unit, and members may include personnel responsible for performing other emergency plan functions for the same assigned unit.

- Implementation: A team for the implementation of severe accident management strategies. The number and composition of this team would ensure there are sufficient personnel capable of implementing any two severe accident management strategies at each unit. This team should not include personnel who are assigned to other functions by the emergency plan.

### *2.1.6. Regulators' organisation*

**Finland**

STUK has dedicated inspectors focusing mainly on severe accident management, analysis, strategies and the SAMG in the Finnish nuclear power plants. In addition, the SAMG is included in the oversight of the entire set of plant procedures by experts on this field.

STUK has its own emergency response unit and operation plan which includes a preparedness centre and trained personnel. All STUK employees participate in emergency preparedness.

**Germany**

There is no special organisational unit at the regulators for the supervision of the on-site emergency preparedness measures. Supervision is carried out by inspectors who deal in depth with the topic but are also responsible for other topics. If necessary (e.g. to assess changes to the installation or to carry out exercises with the licensee), expert organisations are involved.

**Japan**

In the event of a severe accident, the organisation shifts to an emergency system (response by the entire government) based on the Act on Nuclear Disaster. The trigger for this is when a sign of the occurrence of an event is seen; e.g. the occurrence of a huge earthquake, a large tsunami warning, or loss of a design basis safety function during reactor operation.

In such a case, an emergency notification is sent to emergency personnel (usually responsible for work such as examinations and inspections in NRA) who have been rented a dedicated mobile phone. These emergency personnel will gather at NRA's emergency response centre or at the affected nuclear power plants site to co-ordinate emergency response.

Response to a severe accident involves co-ordination between the NRA and other government entities to evaluate any potential threat to the civilian population and recommended protective actions (such as resident evacuations) when necessary.

Such a response is an action by the whole government, including the Nuclear Regulatory Agency, and instructs residents to evacuate according to the progress of the event.

**Spain**

See the answer presented in Section 2.1.7.

**Sweden**

The SSM conducts oversight based on the regulation described in Section 2.1.1. One dedicated group at the SSM has the responsibility to perform oversight based on this.

The now ongoing oversight of the SAMG in Sweden is based on an injunction from 2016 that states, in short, that the licensees shall improve the SAMG and provide an education and training plan for the developed SAMG. The updated SAMG shall contain descriptions of pros and cons for the proposed measures that deal with identified phenomena, criteria for the transition from the use of EOP to SAMG, evaluating necessary resources regarding specialists within the area of severe accident phenomena and severe accident management, and how these specialists can support the TSO in a long-term perspective.

**Switzerland**

The regulator's emergency organisation assembles when an accident takes place. It observes and conducts its own analysis regarding accident progression and potential releases. The SAMG does not trigger a different setting of the ENSI emergency organisation.

**Chinese Taipei**

The AEC reviews the licensee's SAMG and oversees the nuclear emergency response of nuclear power plants.

In case of a nuclear accident, the AEC activates the National Nuclear Emergency Response Centre (NNERC), which consists of the personnel for nuclear safety, radiological protection and emergency response from each department. The AEC would oversee the response measures, evaluate the accident, and notify the key stakeholders and the public of information associated with accident conditions and protective actions.

**United States**

The NRC does not maintain updated copies of plant-specific SAMG. In addition, the generic (BWROG and PWROG) versions of the SAMG are not widely distributed across the NRC. Certain staff (technical trainers, BWR/PWR technology experts) have provided feedback to the owners' groups on the generic SAMG. In the event of a severe accident, these individuals would be involved with the regulatory response to the accident and provide technical assistance to the NRC's response effort.

### *2.1.7. Regulators' inspection and oversight*

**Finland**

The SAMG must be based on the plant-specific analyses and system descriptions (in the Final Safety Analysis Report). The regulator reviews and approves these. When updated, the SAMG is submitted to the regulator and requirements for the SAMG can be given if needed.

The functionality of the SAMG is noted in the emergency drills and training, which the regulator oversees.

All procedures are discussed in the periodic inspections of related topics (in this case the inspection related to severe accident management).

All changes in safety classified systems are submitted to the regulator. The severe accident management systems are classified to safety class 3.

**Germany**

Supervision of the procedures and precautions of the accident management system by the licensee takes place through various methods. These include:

- on-site inspections;
- supervision of changes;
- event-related implementation of new findings, e.g. from accidents in other nuclear power plants;
- periodic safety reviews.

As part of the on-site inspections, the regulator randomly verifies that the licensee:

- ensures the functionality of emergency equipment through maintenance and periodic testing;
- applies and tests preventive and mitigative measures in exercises;
- reviews and updates the preventive and mitigative measures, taking into account results from emergency exercises;
- has prepared a suitable emergency organisation with adequate emergency facilities and equipment;
- can provide sufficiently qualified and trained staff.

In some cases, emergency exercises carried out by the licensee are evaluated by authorised experts commissioned by the regulator.

In principle, changes to the technical and organisational emergency protection arrangements are subject to official supervision. The scope and depth of regulatory supervision in the area of beyond-design-basis accidents are typically lower than in the area of design basis accidents.

**Japan**

Since the BWR inspection practice is currently under review by the NRA, the Japanese PWR inspection practice is described as an example. The maintenance status of SA components, surveillance status, SA sequence drills (e.g. loss of feedwater combined with failure of water injection systems) and their component drills (e.g. cable connection, pump car movement, startup of emergency response centre) is inspected as part of the oversight.

At present, these drills are performed in accordance with a predetermined accident sequence, and this includes the response to changes in plant conditions.

**Spain**

The SAMG development was required by the CSN through conditions or complementary technical instructions to the authorisations (renewal every ten years). SAMG were implemented in Spanish nuclear power plants between 1999 and 2004.

New developments after the Fukushima Daiichi accident are being considered (EDMG and their interaction with SAMG, containment flooding, implementation of filtered containment venting systems, implementation of PAR).

The CSN has performed several inspections of the process of initial development and updates of the SAMG:

- specific CSN inspections during the development phase;
- post-Fukushima Daiichi inspections to review the implementation of improvements for severe accident management;
- specific inspections to the process of updating of SAMG.

**Sweden**

The SSM oversees the SAMG in the same way as other areas. SAMG updates are reviewed by the SSM using the same standards that are used in other inspection activities. The area of SAMG became even more important after the Fukushima Daiichi accident and was highlighted during the stress tests performed in the EU. The Swedish regulations were updated based on the outcome from the stress tests and based on this, inspection activities were initiated.

**Switzerland**

The regulator performs inspections related to the availability of SAMG equipment (e.g. mobile pumps) and their storage, the exactness of the indication of the SAMG (e.g. location of keys), and the update process of the SAMG. In addition, SAMG specialists of the regulator perform inspections during emergency exercises to ensure that SAMG processes are used appropriately. The SAMG is a separate issue of the periodic safety review.

**Chinese Taipei**

The AEC requests that the licensee update its SAMG as the latest guidelines and research results become available. The SAMG are reviewed and the periodic training on these procedures are required.

The AEC oversees the performance of the annual Nuclear Emergency Response Plan (NERP) exercise, which includes drills of the accident management team. The regulatory inspections also focus on the usage of SAMG and the accident management team's decision-making process.

**United States**

There is no regulatory requirement for SAMG implementation. The NRC decided to leave the SAMG as a voluntary industry initiative. US nuclear power plants have committed to maintaining their plant-specific SAG (SAMG) current with Owners Group guidance and to ensure the impact on SAMG implementation is part of their design change implementation process when changes are made to a nuclear power plant.

- The NRC updated the Reactor Oversight Process relative to the commitment. Inspection Procedure 71111.18, "Plant Modifications", requires (as of 2022) a verification that the licensee has updated their SAMG to the latest version issued by their applicable owners' group. The PWROG issued new severe accident technical guidelines in February 2016 (Revision 0), while the BWROG issued Revision 4 to severe accident technical guidelines guidance in June 2018.
- It should be noted that NRC inspectors will only verify that licensees have included the latest updates of SAMG into their procedures. Inspectors can include observations of this process in an inspection report, but they will not inspect the effectiveness of the licensee's implementation of SAMG strategies, and the licensees will not be subject to NRC regulatory enforcement related to their implementation of SAMG strategies.
- Inspectors can cite a finding for the licensee's failure to update their SAMG. Most licensees have a local procedure that requires this. In the past, inspectors have cited licensees for failure to follow their local procedures and incorporate design changes into the SAMG.

## 2.2. Severe accident phenomena

### *2.2.1. Phenomena included in regulations*

As the questionnaire contained some specific severe accident phenomena, the results are presented in Table 2.4. This format was chosen to increase the readability and to facilitate comparison between answers.

## Table 2.4. Measures for severe accident phenomena in different countries/economies

| Phenomena | Finland | Germany | Japan | Sweden | Switzerland | Chinese Taipei | United States |
|---|---|---|---|---|---|---|---|
| Over-pressurisation of containment | Filtered containment venting (primary)<br><br>In case of a break in the primary circuit in combination with a failing pressure suppression function: an automatic over-pressurisation protection system (non-filtered). | Filtered venting system and wetwell spraying | Alternative core cooling system and filtered containment venting | Filtered containment venting and containment sprinkling | Filtered containment venting | Monitoring and control of primary containment pressure | US Mark I and Mark II BWR containments are equipped with hardened wetwell (WW) vents. The Mark I containments also have a drywell vent. The FLEX implementation strategies provide guidance on when to vent. Typically it is tied to a containment pressure (e.g. 10 psig) during an extended loss of AC power. US plants with isolation condensers do not credit containment venting. Only one US Mark III containment plant (Grand Gulf) uses venting. The other (Mark III plants) use some type of suppression pool cooling. If preemptive venting is not used, the EOP require venting prior to reaching primary containment pressure limit (around 60 psig for a Mark I containment). |
| Hydrogen burn: inside and outside containment | Inside containment: Inert containment atmosphere (nitrogen), containment integrity during accidents and filtered containment venting<br><br>Outside containment: airing of reactor building | Inside containment: PAR inside wetwell and drywell<br><br>Outside containment, inside reactor building: PARs above spent fuel pool | Inside containment: Inert containment atmosphere, filtered containment venting and hardened wetwell (WW) vents<br><br>Outside containment: PAR inside reactor building | Inert containment atmosphere, airing of reactor building and filtered containment venting | Inside containment: active and passive ignitors; PAR<br><br>Outside containment: PAR in annulus (secondary containment) | Monitor and control drywell and containment hydrogen concentrations. | Hydrogen burn is prevented in Mark I and II containments by using an inert atmosphere ($N_2$) and then controlling $H_2$ using feed and bleed (containment atmosphere dilution system) or in some cases hydrogen recombiners. Mark III containments use $H_2$ ignitors (glow plugs) to control $H_2$.<br>Rev 4 of the EPG(EOP)/SAG also added guidance for potential $H_2$ migration to other buildings (reactor building, turbine building, radwaste building, etc). If HVAC is lost, open a vent path as high as possible in the building and if $H_2$ might be released into the building, also open a low point vent to establish some natural circulation ventilation. |

## Table 2.4. Measures for severe accident phenomena in different countries/economies (Continued)

| Phenomena | Finland | Germany | Japan | Sweden | Switzerland | Chinese Taipei | United States |
|---|---|---|---|---|---|---|---|
| Steam explosion: in-vessel and ex-vessel | Ensuring containment integrity: analysing the capabilities of various parts of the containment to resist the tolerance from steam explosion.<br><br>The lower airlock as well as some pipe elbows are strengthened. | Containment design, no endangerment by steam explosion which has been shown by deterministic analyses and experiments. | In-vessel: Considered not relevant in-vessel<br><br>Ex-vessel: Limiting the amount of water injection into pedestal as MCCI countermeasure or cooling debris | Considered not relevant in-vessel, for ex-vessel the water injection to lower drywell is limited. | In-vessel: To mitigate the impact on containment integrity: reactor depressurisation, alternate power supply for SRV operation (accident management [AM] procedure), mechanical SRV blocking in position open (AM procedure)<br><br>Ex-vessel: guidance (SAMG) on optimal containment level control (concerning both function of the suppression chamber and sufficient water level in the drywell). | No answer | The SAG direct operators to restore and maintain containment pressure low (<15 psig) in anticipation of the core going ex-vessel. If the core goes ex-vessel, there is a specific caution that warns that adding water to hot core debris may result in rapid steam generation challenging primary containment limits. US plants were not required to strengthen containment. |
| Long-term cooling of core debris | Filling of the containment by fire pumping station pumps or by an external pump | Flooding of drywell/control rod drive compartment; spraying into drywell and wetwell | Water injection to pedestal and reactor | Water injection to lower drywell | Containment flooding, water injection to pedestal and reactor | Monitor and control RPV and primary containment water levels | The EOP and SAG are designed to maintain the WW vent viable. US plants would employ Severe accident water management (SAWM) strategies in order to do this. NEI 13-02 provides detailed guidance on this topic. |

## Table 2.4. Measures for severe accident phenomena in different countries/economies (Continued)

| Phenomena | Finland | Germany | Japan | Sweden | Switzerland | Chinese Taipei | United States |
|---|---|---|---|---|---|---|---|
| High pressure melt-through | Automatic depressurisation of the reactor circuit with the blowdown valves (before entering SAMG). Pressure control with the fast-opening valves, which can be kept open using nitrogen gas or fire water | Automatic depressurisation of RPV; 11 self-medium-controlled safety valves plus three additional diverse motor-driven safety valves | Depressurisation of RPV | Motor operated safety relief valves | Filtered containment venting, containment flooding, reactor stabilizezed on, alternate power supply for SRV operation (AM procedure), mechanical SRV blocking in position open (AM procedure) | No answer | This is a much greater concern for PWRs. All BWRs will have emergency stabilized prior to going to the SAG. As long as pressure suppression capability is maintained, this should not be a concern. |
| Core-concrete interaction | Flooding of lower drywell | Flooding of drywell/control rod drive compartment; spraying into drywell and wetwell | Water injection to pedestal and corium shield | Water injection to lower drywell | Containment Flooding, water injection to pedestal and corium shield | Monitor and control RPV and primary containment water levels | CCI is considered in the technical support guidelines. The mitigating strategies are the same as the ex-vessel strategies described above. Use severe accident water addition (SAWA) to stabilize the core debris (and arrest the CCI) and then maintaining the core debris cool using SAWM. (NEI 13-02). |
| Re-criticality: in-vessel and ex-vessel | Detection by SIRM-detectors | Boron poisoning system | Not considered specifically | Not considered specifically | In/Ex-vessel: SLCS (standby liquid control system), Alternate boron injection (AM procedure) | Monitor and control reactor power, i.e. only in-vessel re-criticality | The possibility of re-criticality is extremely low in a BWR. If it does occur, the strategy is to lower injection but maintain it above the minimum decay heat removal injection rate, and maintain the vessel level to as low as possible (but above top of active fuel). |

Below are the complete answers regarding the severe accident phenomena. These are also presented since they contain some additional information beyond what is in Table 2.4.

**Finland**

Severe reactor accident analyses must cover the phenomena associated with the severe accident management strategy. The SAMG are based on the severe accident management strategy and the related analyses.

Requirements for deterministic safety analysis are given in YVL B.3 and for probabilistic risk assessment in YVL A.7.

**Germany**

In the German nuclear regulatory framework (Interpretation I-7, Requirements for accident management), additional requirements for several phenomena to be expected during SA scenarios are available. The following dedicated phenomenon-specific requirements regarding SAMG can be found:

Requirements for the filtered containment venting system:

- Connection of the vent line at a containment region with expected low aerosol concentration during SA sequences.
- Considering endangerment by combustible gases for the design of the venting system.

Requirements for a containment sampling system:

- Equipment for sampling the containment atmosphere and containment sump should be available.

Requirements for analyses to assess the effectiveness of mitigative EOP:

- Scenarios with severe core damage or severe damage of fuel assemblies inside SFP must be considered for the analyses.
- Effectiveness of mitigative SAM measures (EOP and SAMG) must be assessed by demonstrating that the integrity of the containment is maintained.
- Objectives of the analysis of mitigative SAM measures are to demonstrate that:
    - Either the deflagration of gases ($H_2$ or CO) can be excluded or containment integrity is not endangered.
    - The failure of the containment due to continuously increasing pressure can be avoided.
    - Containment venting can be reliably initiated below the design pressure of the containment.
    - The specifications of the filters of the venting system are maintained.
    - A failure of the containment by sub-pressure can be excluded.
    - Filtered containment venting can be interrupted and restarted.
    - Deflagration inside the whole containment venting system can be excluded or the deflagration does not endanger the functioning of the system.

**Japan**

See Table 2.4.

**Spain**

The IS-36 establishes the requirements to be met by the Spanish nuclear power plants in relation to EOP and SAMG. Chapter 5, “Severe accident management aids”, details requirements for the management of different aspects related to the phenomenology of severe accidents.

Regarding the structure of the EOP-SAMG in Spanish BWRs, they were developed following the generic EPG-SAG from the BWR Owners’ Group.

**Sweden**

Since 1988, when the filtered containment venting systems were installed in Sweden, the licensees have been obliged to follow research activities both internationally and within Sweden. A research programme where both the licensees and SSM contribute with funding has therefore been ongoing since early 1980s, called the APRI-project (accident phenomena of risk importance). The focus of this research is on increasing the understanding of phenomena related to severe accidents. One group at the SSM has the responsibility to perform oversight based on this.

The injunction from the SSM does not include any requirements regarding which phenomena to include (or exclude); it is up to the licensees to identify relevant phenomena.

One licensee in Sweden has presented how the following phenomena are taken care of:

- hydrogen burn: Inert containment atmosphere, airing of reactor building and filtered containment venting;
- high pressure melt through/direct containment heating: RPV depressurisation;
- steam explosion: not applicable in-vessel; for ex-vessel, the water injection to the lower drywell is limited;
- over-pressurisation of containment: filtered containment venting;
- molten core concrete interaction (MCCI): water injection to lower drywell.

**Switzerland**

The licensees are required to follow the state of the art and therefore consider all relevant phenomena in the design and implementation of SAMG.

**Chinese Taipei**

The purpose of the SAG is to:

1. Remove heat from the RPV
2. Retain any core debris in the RPV
3. Maintain primary containment integrity
4. Scrub fission products from the containment atmosphere
5. Prevent or minimise core-concrete interaction
6. Submerge the core and core debris

**United States**

*Hydrogen Burn: Inside and outside containment*

Hydrogen recombiners are not passive and require power for the heaters. Mark III containments use hydrogen ignitors (glow plugs) to control hydrogen. A FLEX diesel is even provided for the ignitors during extended loss of AC power (ELAP).

The hydrogen burns at Fukushima outside containment were caused by containment leakage (DW head bolts stretching, etc.) and in the case of Unit 4, cross flow from Unit 3 venting. Part of the hardened vent design criteria is to prevent leakage to other systems and other units. With the containment vents, there should be no containment overpressure causing leakage, and the cross flow would be minimized, thereby minimising hydrogen burn outside containment.

### *2.2.2. Phenomena not included in regulations*

A common position for several regulators is that it is up to the licensee to provide evidence that a specific phenomenon is not relevant for a specific unit.

**Finland**

The licensee is required to justify the chosen strategy through analyses or experiments. Analyses must cover all relevant phenomena and the licensee has responsibility to identify them. Some phenomena are naturally not relevant to all nuclear power plant designs.

If the licensee demonstrates in the safety analyses that a phenomenon is not relevant or physically impossible to occur in the plant, it can be excluded from the management measures and SAMG.

**Germany**

No phenomena are known which are not covered by the safety concept of the German nuclear power plants.

**Japan**

The new regulatory standards specify accident sequences that must be considered when considering countermeasures for a severe accident, but if PRA finds a new sequence specific to each plant, it should also be included in the review.

In the current review, the core damage frequency (PRA) due to the tsunami became significant even when the seawall was installed, so it is assumed that the tsunami may exceed the seawall and these countermeasures were required to secure the ultimate heat sink (TOKAI Dai-2 BWR5. It could not expect mobile components).

**Spain**

See Section 2.2.1.

**Sweden**

The regulation does not point out any phenomena at all. Tt is up to the licensee to ensure that all relevant accident conditions are taken care of.

In-vessel steam explosions have been considered as a phenomenon that is not necessary to take measures against; this was a finding in the early APRI-projects. The basis for this is that the structures that will be impacted from these loads will withstand the loads.

Re-criticality is not addressed in any of the existing SAMG or the updated ones. Ex-vessel re-criticality was evaluated in the 1980s and considered as a residual risk.

However, SSM has put an injunction on the licensees to provide an analysis of the containment's capability to withstand steam explosions. If this cannot be demonstrated, the identified weak parts must be strengthened.

**Switzerland**

Containment low atmospheric pressure can be an issue with respect to containment integrity and relevant for the application of the venting system.

**Chinese Taipei**

The phenomena described in Section 2.2.1 are covered by the SAG strategies.

**United States**

The United States does not have a regulation for SAMG. Plants are required under 10 CFR 50.155 to have mitigating strategies in place to handle beyond-design-basis events. With numerous unique designs in existence in the United States, approval of mitigating strategies is done on a plant-by-plant basis.

## 2.3. Long-term management

### *2.3.1. Regulation and content of SAMG*

**Finland**

Regulation STUK Y/1/2018 requires that the nuclear power plant be designed so that it can be reliably brought into a safe state after a severe reactor accident. (Safe state following a severe reactor accident shall refer to a state where the conditions for the controlled state of a severe reactor accident are met and, in addition, the pressure inside the containment is low enough that leak from the containment is minor, even if the containment is not leak-tight.)

The SAMG focuses on the immediate actions and management of severe accidents to bring the nuclear power plant to a safe state.

**Germany**

The main goal of the SAMG concept is to mitigate the severe accident progression, to minimise the release of radionuclides, and to take the plant to a secured and safe state in the long term. In the German nuclear regulatory framework (Interpretation I-7 (Requirements for accident management) of the safety requirements for nuclear power plants), the main goals of SAMG strategies/procedures are:

- termination of the core melt sequence;
- protection of intact barriers for retention of radionuclides;
- limitation of the release of radionuclides; and
- achievement of a controllable plant state in the long term.

**Japan**

The SAMG focuses on the immediate actions and management of severe accidents to bring the nuclear power plant to a safe state. In Japan, there are special regulations separate from severe accident regulations for long-term recovery after severe accidents (for example, the regulation for Fukushima Daiichi Nuclear Power Plant).

**Spain**

There are no specific provisions in the regulation for long-term recovery after a severe accident (in the months-years range).

The SAMG deal with the accident itself.

**Sweden**

Regarding the long-term management of severe accidents there are no explicit regulations. § 8 of SSMFS 2008:17[5] states that “It shall be possible to cool a molten core over an extended period of time”, which indicates that long-term management should be in place.

**Switzerland**

The licensees are required to consider long-term management. Sufficient resources (human and material) must be available for extended time periods after the accident.

**Chinese Taipei**

The SAMG are symptom-based procedures to cope with severe accidents, depending on real-time operational parameters to mitigate the event consequence. The AEC ordered the licensee to build up, maintain and update the plant-specific SAMG. The content of plant-specific SAMG will be discussed through the reactor oversight programme, exercises and inspection processed by the AEC.

**The United States**

The SAMG are not a regulatory requirement in the United States. The 10 CFR 50.155 requires licensees to have mitigating strategies for beyond-design-basis events and extensive damage mitigating guidelines in place to ensure continued protection of the core, containment and spent fuel pool.

## 2.4. Training of personnel

### *2.4.1. Transition from EOP to SAMG*

**Finland**

It is YVL-guide A.4 that regulates training of personnel; see paragraphs 325 and 327-330 and also Annex E “Control room operators” to the guide.

The control room operators must have a STUK-approved licence, which is granted for one to four years. The requirements for the licence include a suitable degree, work experience, initial training, a written examination, demonstration of professional skills in a training simulator and an oral examination.

In accident simulation, the operator trainee identifies the disturbance, performs the first actions required by the procedures, detects the abnormally functioning component, corrects the situation, and determines any further action required.

Besides scheduled training, if there are changes in procedures there will also be additional simulator training.

If there are changes in procedures, there will be additional simulator/classroom training for the control room operators.

**Germany**

Measures of the NHB are trained on at least once every three years. That must be done by the shift supervisors, reactor operators, head of crisis team and the heads of the teams for emergency operations. Information regarding the training of personnel with respect to the use of SAMG measures is not available, but it can be assumed that the periods for training should be like the training of NHB measures.

5. Since 1 March 2022 this regulation has been replaced by SSMFS 2021:4 “The Swedish Radiation Safety Authority’s Regulations and General Advice concerning Design of Nuclear Power Plants”. With respect to SAMG there are no major changes in the regulation.

**Japan**

Drills must be conducted as described below:

- Since severe accident measures must be diverse enough to manage various plant situations, education and drill related to such measures shall enable trainees to improve their knowledge of the behaviour of a plant during severe accident.
- In addition to periodical education to improve the knowledge base according to the role of each member of personnel, practical drills shall be planned to check the effectiveness of the severe accident measures' implementation organisations and the support organisation.
- Plant personnel shall be sufficiently familiar with the plant and spare parts, etc. through daily maintenance by experiencing practical work such as replacing parts during day-to-day work.
- Personnel shall be trained in how to manage accidents under various conditions such as high radiation levels, and during nighttime and under bad weather conditions.
- Through normal maintenance work and inspection, personnel shall be trained and prepared to quickly use information and manuals related to equipment and equipment used during accidents.

**Spain**

IS-36, Chapter Eight "Training on EOP's and SAMG's" regulates the training of the personnel with respect to EOP and SAMG.

The nuclear power plant licensee shall provide a periodic training programme for all personnel responsible for performing manoeuvres included in the EOP and SAMG, in accordance with an analysis of the tasks and responsibilities assigned to each job post. The frequency of the training shall be in accordance with the safety significance and complexity of the manoeuvres to be covered in the training.

The operating shift personnel, of both the control room and auxiliary operators, shall be periodically trained on all the EOP and SAMG tasks and manoeuvres for which they are responsible.

The personnel of the main emergency response centre at the plant, or technical support centre, shall be trained periodically on strategies and use of the SAMG, within their responsibilities.

Training on the transition from the EOP to the SAMG shall be included in the training programme for the operating shift and plant technical support centre personnel.

Within the process of initial and ongoing training on the EOP, a full-scope replica simulator is considered by the CSN to be adequate for initial and ongoing training and for licence examinations it [a full-scope replica simulator] shall be used. Within the process of initial and ongoing training on the SAMG, severe accident calculation programmes and simulators shall be used wherever feasible.

**Sweden**

Chapter 5, § 2 of SSMFS 2008:1 states that "Personnel shall be familiar with the instructions, procedure and guidelines." There is no other specific requirement that regulates training on the use of EOP and SAMG.

**Switzerland**

The ENSI guideline B10 regulates the training of plant personnel. Beyond-design basis accidents are part of the training and requalification of plant operators.

**Chinese Taipei**

In Chinese Taipei, an on-site nuclear emergency response drill (or on-site nuclear exercise) is required for each nuclear power plant every year to test the plant's capability to respond to an emergency, and to help the personnel in the TSC, the AMT and the main control room to perfect their emergency response skills so that they can bring the severe accident into a controlled state and minimise the radioactivity release to the environment.

According to the Regulations on Nuclear Reactor Operators' Licenses, the reactor operator training programme should include emergency procedures for the nuclear power plant. Under the two-year requalification training programme, the licensed reactor operators are also required to have a course on the SAG and the recent development of severe accident management.

A new AMT member is required to learn the SAG and the technical support manuals related to their responsibilities in the team. The new member should pass an exam given by the leader of the AMT. The AMT requires training every two years for the members to review the SAG strategies through a lecture or an exercise.

**United States**

The mitigating strategies rule, 10 CFR 50.155, requires licensees to conduct training on: 1) Strategies and guidelines to mitigate beyond-design-basis external events from natural phenomena, and 2) strategies and guidelines to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant impacted by a [beyond-design-bases] event due to explosions or fire. Ten CFR Part 50.120 and 10 CFR Part 55 discuss the requirements for operator training and the concept of a systems approach to training for US nuclear power plants.

While there is no regulatory requirement for licensees to incorporate SAMG as the standard for mitigating beyond-design-basis events, to the extent a licensee implements SAMG as their choice to comply with the rule, they would be expected to train operators as part of their systems approach to training, which is required by regulations.

### *2.4.2. Frequency of training*

**Finland**

Basic training is done periodically according to a schedule. Modifications/lessons learnt/new information about severe accident are trained for as needed.

The control room personnel’s basic education includes a severe accident part with training on simulator and theoretical lessons based on the MELCOR analyses of the nuclear power plant.

The licensed control room operators have scheduled training programmes. Operators train on the EOP and SAMG regularly on the licensee’s plant simulator.

The licensee has education about severe accidents for all personnel.

The emergency preparedness personnel have a training matrix, which includes training about severe accidents.

**Germany**

Measures of the NHB must be trained at least once every three years. In addition, each year an emergency drill is performed which includes internal severe accident management measures.

**Japan**

The safety standards review criteria for “operational safety programmes” in the new regulatory standards require that training for SAMG-related personnel be regularly conducted at least once a year. The criteria also require that these activities be evaluated regularly and that necessary measures be taken.

**Spain**

See Section 2.4.1

**Sweden**

There are no regulations regarding frequency for training. However, licensees provide training for emergency preparedness personnel on a recurring schedule.

**Switzerland**

The ENSI guideline B11 describes the requirements for plant exercises. Beyond-design-basis accident exercises involving the emergency response team take place at regular intervals (every two years; a different plant every time).

**Chinese Taipei**

The training is periodically performed as described in Section 2.4.1

**The United States**

Ten CFR Part 55 describes training requirements for licensed plant operators. Most licensees will conduct periodic refresher training during licensed operator requalification and engineering support or emergency preparedness training. Requalification is required every 24 months.

Ten CFR Part 50 Annex E contains requirements for periodic emergency preparedness exercises on an eight-year exercise cycle.

### *2.4.3. How training is performed*

**Finland**

There is a full-scale training simulator, but after depressurisation the simulator is not capable of simulating the scenario. After simulator training, the training is done by classroom training. See also the answers in Sections 2.4.1 and 2.4.2.

**Germany**

Drills are conducted on-site, especially regarding the preparation and connection of mobile equipment. The periodic diagnosis of the plant state (state of the RPV and state of the containment) and the performance of the strategies and procedures of the SAMG concept are trained on a full-scale simulator, though the simulator is not able to simulate a whole SA sequence. Thus, only the organisational aspects of SAMG procedures are trained.

**Japan**

On the regulatory side, there are full-scope simulators that can also reproduce severe accidents (PWR, BWR, ABWR), and there are training programmes for startup and shutdown operations, transient, design basis accidents, severe accidents, and so on.

In the private sector, there are companies that conduct training for both BWRs and PWRs, and they have full-scope simulators. There are various programmes, from beginner to advanced, for operator training.

**Spain**

See Section 2.4.1

**Sweden**

In general, the training is performed via tabletop exercises. The full-scale simulator cannot be used for severe accident conditions.

**Switzerland**

The plant exercises take place in the full-scale simulator. The plant simulator can calculate up to the onset of core damage. To simulate the accident after core damage, a MELCOR simulation (usually performed in advance) takes over to display the relevant plant parameters.

**Chinese Taipei**

The TPC uses a full-scale simulator for training the reactor operators, the members of the TSC, and the AMT to mitigate the severe accident. When the reactor core damage begins, the lecturer will freeze the simulator and use a tabletop exercise to train the relevant personnel to respond to the emergency situations.

In addition, the TPC has developed MAAP4-based severe accident simulation models for Chinshan and Kuosheng Nuclear Power Plants, respectively. For the on-site nuclear emergency response drills, the TPC uses MAAP4 severe accident analysis code to help the members of the AMT make better decisions for the actions directed by the SAG.

**United States**

Classroom and/or “tabletop” settings are generally used for severe accident training. Very few utilities have simulators that model severe accidents. These licensees do run severe accident EP drills using the simulator but once the core melts, all indications are provided by drill proctors.

# References

Regulations, guidelines or standards relevant for SAMG and mentioned in this document:


| | | |
|---|---|---|
| Finland | - | STUK Y/1/2018, Section 20 “Safety of operation” |
| | - | YVL-guide A.4 “Organisation and personnel of a nuclear facility” |
| | - | YVL-guide A.6 “Conduct of operations at a nuclear power plant” |
| | - | YVL-guide A.7 “Probabilistic risk assessment and risk management of a nuclear power plant” |
| | - | YVL-guide B.3 “Deterministic safety analyses for a nuclear power plant” |
| Germany | - | “Safety Requirements for Nuclear Power Plants”, Section 3.1 (10) |
| | - | Nuclear Safety Standards Commission’s (KTA) safety standard “Requirement for the Emergency Manual” (KTA 1203) |
| Japan | - | No references provided. |
| Spain | - | Nuclear Safety Council Instruction IS-36, of 21 January 2015 |
| Sweden | - | SSMFS 2008:1 “The Swedish Radiation Safety Authority’s Regulations and General Advice concerning Safety in Nuclear Facilities” |
| Switzerland | - | ENSI guideline B10 “Basic Training, Recurrent Training and Continuing Education of Personnel in Nuclear Installations” |
| | - | ENSI guideline B12 “Notfallschutz in Kernanlagen” |
| Chinese Taipei | - | No references provided. |
| United States | - | NEI 14-01, “Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents” |
| | - | Inspection Procedure 71111.18, “Plant Modifications” |
| | - | Post-Fukushima Order 12-051, “Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation (Effective Immediately)” |
| | - | Mitigating Strategies Order 12-049, “Issuance of order to modify licenses with regard to requirements for mitigation strategies for beyond design basis external events” |
| | - | 10 CFR 50.155, “Mitigation of beyond-design-basis events” |
| | - | NEI 13-02, “Industry guidance for compliance with order EA-13-109, BWR Mark I & II Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions” |
| | - | NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants” |
| | - | 10 CFR 50.47 |
| | - | 10 CFR Part 50 Appendix E |
| | - | NEI 12-01, “Guideline for Assessing Beyond Design Basis Accident Response and Communications Capabilities” |
| | - | EA-13-109, NRC-2013-0128, “Order Modifying Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions (Effective Immediately)” |

# Annex A. Questionnaire – SAMG for BWR

# Introduction

The purpose of this questionnaire is to collect information regarding SAMG (severe accident management guidelines) for BWRs in different countries/economies and with different designs.

According to IAEA NS-G-2.15[6] SAMG are used in the mitigatory domain of accident management. In this phase of the accident progression uncertainties may exist both in plant status and in the outcome of actions. Consequently, the guidance for the mitigatory domain should not be prescriptive; instead, the guidance should propose a range of possible actions including pros and cons for these.

According to the same IAEA safety guide, EOP are used in the preventive domain of an accident's progression. Since plant status will be known from instrumentation, the guidance is normally presented as procedures and is prescriptive in nature. EOP are generally limited to actions taken before core damage occurs.

Typical severe accident phenomena that are discussed in this questionnaire include:

| Phenomena | Typical measures |
|---|---|
| Over-pressurisation of containment | Filtered containment venting (FCV) or only containment venting (CV) or spraying water inside containment |
| Hydrogen burn: inside and outside containment | Passive autocatalytic recombiner (PAR), FCV/CV, igniters… |
| Steam explosion: in-vessel and ex-vessel | Design measures aiming at strengthening the containment vessel, avoiding core melt to drop in water, etc. |
| Long-term cooling of core debris | Water filling of containment |
| High pressure melt through | Reactor pressure vessel venting/depressurisation |
| Core-concrete interaction | Avoiding high pressure melt through |
| Re-criticality: in-vessel and ex-vessel | Boron management |

6 IAEA (2009), "Severe Accident Management Programmes for Nuclear Power Plants: Safety Guide No. NS-G-2.15", International Atomic Energy Agency, Vienna, www-pub.iaea.org/MTCD/Publications/PDF/Pub1376_web.pdf.

# Questionnaire

| **General** |
|---|
| Describe the regulation relevant for SAMG in your country.<br>It would be appreciated if the answer contains a reference to the regulation. |
| |
| Describe in general terms the licensees' EOP and SAMG with respect to structure and content. The EOP can be described only briefly. |
| |
| Describe the criteria for transfer from the use of EOP to the use of SAMG. |
| |
| Describe if and how the spent fuel pool is addressed in the SAMG. |
| |
| Describe briefly the licensee's organisation during a severe accident with respect to the use of SAMG. Does the organisation change when one transfer from EOP to SAMG? |
| |
| Describe how the regulator conducts inspection and oversight of the licensee's SAMG. Consider aspects like plant changes and other changes (e.g. organisational) that might have an impact on SAMG. |
| |
| Describe in general term the regulator's organisation with respect to SAMG issues. |
| |
| **Severe accident phenomena**<br>These questions refer to the phenomena described above |
| Describe how the phenomena are included in the requirements that are set up for the SAMG. |
| |
| Describe if there are any phenomena that are not covered by the requirements.<br>If yes, what is the basis for this? (Could be plant-specific design, analyses that demonstrate that a specific phenomenon is not relevant, etc.) |
| |
| **Long-term management** |
| Describe the long-term management with respect to SAMG, both the regulation and the content of the licensees' SAMG. |
| |
| **Training of personnel** |
| Describe how the training of personnel with respect to use of SAMG and the transition from EOP to SAMG is regulated. |
| |
| Describe how often training is performed. Recurring or on a need basis? |
| |
| Describe how the training is performed? Full-scale simulator? Other type of simulator? |
| |