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Assessment of Equipment Capability to Perform Reliably under Severe Accident Conditions

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ASSESSMENT OF EQUIPMENT CAPABILITY TO PERFORM RELIABLY UNDER SEVERE ACCIDENT CONDITIONS

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FOREWORD

The experience from the last 40 years has shown that severe accidents can subject electrical and instrumentation and control (I&C) equipment to environmental conditions exceeding the equipment's original design basis assumptions. Severe accident conditions can then cause rapid degradation or damage to various degrees up to complete failure of electrical and I&C equipment.

Information from accident monitoring instrumentation is needed not just during the accident itself but also for a long period after the onset of the event. Due to harsh environmental conditions in the containment or adjacent rooms, it may be impossible to access equipment for replacement or maintenance. This is especially true of equipment for containment isolation, heat removal and venting, as well as instrumentation for measuring the effects of degradation of fission product barriers.

Electrical and I&C equipment required to function during a severe accident has to be protected against harsh environments. This equipment can be physically separated, installed at a safer location or shielded against the effects of such an event. In case adequate protection cannot be accomplished or is not feasible, the equipment has to be assessed for its capability to perform reliably under severe accident conditions.

This publication provides the technical basis to consider when assessing the capability of electrical and I&C equipment to perform reliably during a severe accident. It provides examples of calculation tools to determine the environmental parameters as well as examples and methods that Member States can apply to assess equipment reliability.

This publication is intended for all personnel involved in the design, manufacture, licensing, operation and maintenance of electrical and I&C equipment required to function in severe accident conditions. The IAEA wishes to thank all participants in the consultants' meetings and the Technical Meeting and their Member States for their valuable contributions. The IAEA officer responsible for this publication was A. Duchac of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The implementation of any severe accident mitigation measures in accordance with severe accident management guidelines (SAMG) assumes that electrical and I&C equipment used for monitoring accident conditions remain functional during the accident and the post-accident phase.

As defined in Ref. [1], the plant states considered in the design of nuclear power plant includes operational states and accident conditions. The accident conditions comprise design basis accidents (DBA) and design extension conditions (DEC). The design extension conditions further include conditions without significant core degradation and conditions with core melting (severe accident).

Requirement 30 of Ref. [1] states: “A qualification programme for items important to safety shall be implemented to verify that items important to safety at a nuclear power plant are capable of performing their intended functions when necessary, and in the prevailing environmental conditions, throughout their design life, with due account taken of plant conditions during maintenance and testing”. Paragraph 5.48. of Ref. [1] also states that “the environmental conditions considered in the qualification programme for items important to safety at a nuclear power plant shall include the variations in ambient environmental conditions that are anticipated in the design basis for the plant”. Additionally, Refs [2] and [3] provide recommendations on qualification of electrical and I&C equipment as well as methods to preserve the qualification status for their intended safety function during the time in service.

The recommended practices for qualifying equipment important to safety for postulated design basis accidents have been established in Ref. [4] entitled “Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing”. However, Ref. [4] does not address assessment requirements for severe accident conditions. Requirement 20 of Ref. [1] states that “a set of design extension conditions shall be derived on the basis of engineering judgment, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant’s capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.”

Paragraph 5.29 of Ref. [1] requires that “the analysis undertaken shall include identification of the features that are designed for use in, or that are capable of preventing or mitigating, events considered in the design extension conditions. These features:

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

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

(c) Shall have reliability commensurate with the function that they are required to fulfil.”

The environmental parameters anticipated during accident conditions depend on the plant design, the type of the initiating event and the resultant level of core degradation. When accident conditions proceed to a severe accident, the equipment located in the containment may be exposed to conditions that significantly exceed the values for extended time periods for which the equipment has been qualified.

Environmental conditions to which the electrical and I&C equipment may be exposed can be derived based on data collected from severe accidents that have occurred and severe accident condition simulations. These values can be significantly higher than those anticipated during design basis accidents.

The use of traditional environmental qualification methods for design basis accident conditions (e.g. loss of coolant accidents, high energy line breaks) is acceptable. However, the qualification may need to be extended to address conditions valid for severe accidents.

1.2. OBJECTIVE

The objective of this publication is to provide:

- An international technical basis to be considered when assessing the electrical and I&C equipment reliable performance under severe accident conditions needed for implementation of mitigative measures during severe accidents;
- An overview of specific issues related to electrical and I&C equipment capability to perform reliably under severe accident conditions;
- Examples of calculation tools for determining the environmental parameters for severe accidents;
- Examples of methods that may be applied in Member States to assess reliable performance of electrical and I&C equipment under severe accident conditions;
- Examples and methods that Member States may apply to enhance the capability of equipment dedicated for severe accident conditions.

This publication makes reference to existing nuclear power plants and their related documentation as well as to those being planned or under construction.

1.3. SCOPE

This publication covers relevant aspects of assessing the capability of the accident mitigation and monitoring equipment of nuclear power plants that would be exposed to environmental conditions that significantly exceed environmental qualification values for extended time periods.

Typical examples of electrical and I&C equipment that are needed for severe ac accident mitigation and monitoring include:

- Sensors/transducers;
- Transmitters;
- Actuators (motors and solenoid drives);
- Cables and connection interfaces (splices, terminations, connection interfaces, etc.);
- Junction boxes;
- Containment penetrations (electrical and sealing function);
- Limit switches/position indicators.

With respect to environmental conditions electrical and I&C equipment can be categorized as follows:

- Installed in the containment and exposed directly to environmental conditions of the severe accident;
- Installed outside the containment and exposed to special environmental conditions as a result of the severe accident. This equipment is installed outside the containment and located either in the areas of or within pipes, vessels or ducts containing contaminated fluid or atmosphere.

The values of environmental parameters appropriate for these two categories include containment pressure and temperature, radiation levels and combustible gas concentration and may be obtained on the basis of calculations, experimental results, and (if necessary) on basis of engineering judgment. Type tests, survivability assessment or a combination of both can be used to determine whether the equipment can perform reliably during and after severe accidents.

1.4. STRUCTURE

This publication contains seven main Sections and seven Annexes.

Section 1 introduces the topic, the objective and scope of the publication. Section 2 discuses basic considerations needed for assessment of the equipment capabilities under severe accident conditions. Section 3 discusses the electrical and I&C equipment needed for severe accident mitigation and monitoring. Section 4 provides the methods for estimating environmental parameters to be used in the determination of assessment specifications. Section 5 discusses the electrical and I&C equipment design capability and the anticipated equipment performance under severe accident conditions. Section 6 discusses the methods applied in Member States for the assessment of electrical and I&C equipment to perform reliably under severe accident conditions. Section 7 contains summary and conclusions.

References in this publication provide links to important international documents, codes, standards and other guidance publications relevant to the design of electrical power systems and instrumentation and controls in NPPs.

Annexes to this publication provide examples of Member States practices to calculate severe accident environmental parameters as well as examples to enhance the capability of electrical and I&C equipment dedicated for severe accident mitigation to perform reliably.

2. EQUIPMENT CAPABILITY ASSESSMENT UNDER SEVERE ACCIDENTS

2.1. BACKGROUND

Equipment qualification for design basis accident conditions ensures that the credited equipment is capable of fulfilling the intended safety functions. The equipment qualification programme typically addresses:

- Suitability and correctness of equipment functions and performance;
- Capability to withstand the impact of environmental conditions during operational states and accident conditions considered in the design;
- Capability to withstand external hazards;
- Capability to withstand the impact of electromagnetic disturbances.

Most of the current operating nuclear power plants (NPP) were designed with equipment capable of coping with the design basis accidents, but their ability to cope with severe accidents need to be further evaluated. In contrast, new plants need to demonstrate capability of accident mitigation equipment under severe accident conditions.

While the anticipated environmental conditions and time duration for the design basis accidents are well defined in the plant safety analysis, the environmental conditions during severe accident depend on the type of the initiating event and the resultant level of core degradation. If the accident conditions proceed to a severe accident the environmental conditions in the containment might significantly exceed parameters for which the equipment has been already qualified.

The following inputs are needed for assessing the reliable performance of equipment under severe accident conditions:

- The environmental profiles,
- The mission times for which the reliable performance is to be assessed, and
- The intended safety function.

Severe accident environmental profiles can either be estimated or derived from the analysis/simulation of severe accidents, or from data obtained from actual severe accidents that have occurred.

An appropriate mission time for which the equipment is needed to function during a severe accident is an important input parameter for the assessment of reliable performance. This mission time may be on the order of weeks or even years due to limited accessibility for maintenance and replacement.

The qualification methods for qualifying electrical and I&C equipment important to safety for design basis accidents are well established in some national and international standards and regulations, such as:

- The IAEA Safety Reports Series No. 3 Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing;
- The United States Nuclear Regulatory Commission 10 CFR 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants;
- The Finish YVL Guides 5.2 (electrical equipment) and 5.5 (I&C equipment);
- The German KTA standard 3504 (electrical equipment) and 3505 (I&C field equipment);
- The IEC/IEEE Joint Logo Standard, Nuclear Facilities – Electrical Equipment Important to Safety: Qualification (IEC/IEEE 60780-323 standard (2016);
- The French RCC-E standard (in particular volume B);
- The Swedish KBA/TBA standard;
- Nuclear Power Plant Equipment Qualification Reference Manual, EPRI TR-100516, 1999, revision 2010 (document 1021067);
- Canadian Standards Association (CSA) N290.13 Environmental qualification of equipment for CANDU nuclear power plants.

These equipment qualification standards and regulations cover aspects of demonstrating the suitability of equipment for design basis accidents. Aspects of confirming the equipment performance under severe accident conditions are only partially addressed.

Severe accidents that have occurred during the last forty years have revealed that a greater effort is needed to assess reliable performance of the equipment needed for monitoring and mitigation of a severe accident. At present there are no international consensus standards that provide a technical basis on how to assess the reliable performance of the equipment.

A basic assumption is that equipment that has been already qualified to design basis accident conditions, has a higher probability of performing its intended safety function under severe accident conditions than equipment without qualification. This qualified equipment may have the capability to maintain its intended safety functions for a limited time under severe accident conditions.

The evaluation of reliable performance includes theoretical analysis, type testing, comparison with operating experience in other harsh environmental applications (aerospace, glass making, petrochemical industry, etc.), and material assessment. Methods of pre-ageing and seismic qualification of electrical and I&C equipment need to be considered, but are out of scope of this publication. Recommendations to protect the electrical and I&C equipment for withstanding the seismic hazards are provided in Ref. [5].

2.2. FISSION PRODUCT BARRIER INTEGRITY

During the progression of a severe accident, fission product barriers between the highly radioactive fuel inside the nuclear plant and the environment outside are challenged. Ref. [6] defines the main objectives of mitigation strategies to prevent failure(s) of the fission product barriers and to prevent subsequent radiological release to the environment. This includes actions to terminate core/fuel melt progression, maintain reactor pressure vessel integrity, maintain containment integrity and prevent containment bypass in order to reach a long term stable state of affected unit.

Plant instrumentation dedicated for monitoring fission product barrier integrity is necessary for the implementation of severe accident mitigation strategies, and is intended to provide adequate information for decision making.

2.3. SEVERE ACCIDENT ENVIRONMENTAL PROFILES

Environmental profiles (i.e. parameters versus time) resulting from the severe accident conditions to which electrical and I&C equipment may be exposed can be derived based on data collected from severe accidents that have occurred and severe accident condition simulations. The environmental profiles may also depend on the installation location of the equipment. The profiles can be derived using the following inputs:

- Estimation of profile durations (mission time dependent);
- Potential recurrence of specific phenomena (e.g. hydrogen combustion);
- Combination of chemical compounds that may have degradation effects;
- Radiation profiles (dose rate vs. time, ratio of β vs. γ radiation, energy of radiation);
- Other effects such as flooding, hydrogen combustion, etc.
- Interaction of the melted core with concrete.

The environmental parameters during a severe accident vary during different stages, due to ongoing physical processes and chemical reactions inside the reactor and the containment. Values of these parameters can significantly be higher than those anticipated during design basis accidents. Annex III provides an example of mapping of environmental parameters inside and outside the containment during a severe accident.

The onset of a severe accident is characterized by an increase in the physical quantities described in the following subsections.

2.3.1. Radiation, temperature, pressure and humidity

The loss of capability to cool the core will eventually lead to fission products release into the containment atmosphere causing an increase in pressure, humidity and temperature in the containment. Furthermore, fission products deposited on the surface of equipment may cause an additional heat up of the individual equipment surface.

An increase in the total radiation level is caused by the fission product release into the containment atmosphere. In contrast to the radiation dose occurring during normal operation, the radiation dose during a severe accident consists of γ-radiation and β-radiation, which influence the degradation effects on the equipment.

2.3.2. Flooding (submergence)

Flooding can occur due to the event or as a consequence of mitigation strategies. Flooding/ submergence may have an impact on the functionality of electrical and I&C equipment since the hydrostatic pressure has to be taken into account. Flooding of electrical and I&C equipment due to the increasing water level in the containment leads to a higher total integrated dose, because the equipment is in direct contact with the contaminated coolant. On the other hand, the impact of the temperature load may be reduced because the equipment is not directly exposed to combustion processes.

2.3.3. Explosive atmosphere

Hydrogen release and the generation of carbon monoxide are crucial phenomena to be considered when assessing severe accident phenomena. If uncontrolled combustion processes occur, the equipment and the containment structure may be exposed to extreme temperature and pressure peaks challenging their proper function and integrity. Maintaining the containment hydrogen concentration below the dangerous (explosive) limit is therefore essential.

2.3.4. Chemical processes

Severe accidents can lead to significant changes in the chemical composition of the containment atmosphere and the sump. These changes are a consequence of the release of aerosols, chemical compounds and degraded materials. The resultant harsh environment may impact the reliable performance of the equipment because of chemical degradation of insulating or sealant materials.

2.4. HARDWARE PROVISIONS FOR SEVERE ACCIDENT MANAGEMENT

Reference [6] provides recommendations to include hardware provisions for effective implementation of accident management strategies. Furthermore, it recommends that hardware characteristics and layout are assessed for their capability in meeting the accident management objectives. The assessment of reliable performance of electrical and I&C equipment requires a good understanding of the progression of a severe accident, and the correct order of response activities within the framework of previously defined mitigation strategies. Severe accident management is characterized by different stages as illustrated in Figure 1.

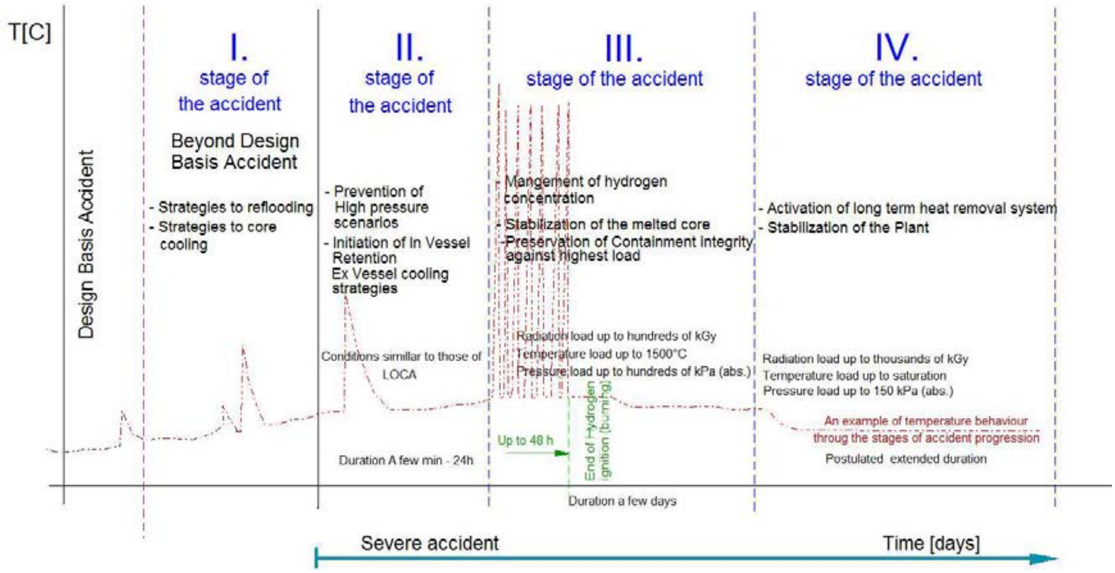

FIG. 1. Typical example of accident stages and associated environmental parameters (figure courtesy of VUJE, a.s.).

A set of specific mitigation strategies is used to limit the severe environmental conditions inside the containment. These strategies include, but are not limited to:

- Depressurization of the primary circuit;
- Cavity flooding (for in vessel retention only);

- Heat removal from containment;
- Quenching of the core;
- Cooling of corium;
- Possible containment filtered venting;
- Control of water level and temperature of the suppression pools for BWRs;
- Control of combustible and non-condensable gases;
- Recombination and/or ignition of hydrogen;
- Maintaining spent fuel pool water level.

Each stage of accident progression is associated with its own set of environmental parameters:

- Stage I: Parameters are associated with unsuccessful implementation of measures to cope with an initiating event. They are in the same range as expected for design basis accidents (DBA).
- Stage II: Parameters are associated with mitigating strategies for the prevention of high pressure scenarios. These scenarios involve the protection of the integrity of the reactor pressure vessel and the containment. Environmental conditions may exceed the limits anticipated for DBA.
- Stage III: Parameters are associated with the initial stages of the core melt. During this stage values of temperature, pressure, radiation and concentration of combustible gases reach their maximum.
- Stage IV: Parameters are associated with the stabilization of the melted core and preservation of the containment integrity for long-term period. During this stage temperature, pressure, radiation and the concentration of combustible gases are decreasing. At the end of this stage the plant will reach the cold stable state. But radiation level may be long lasting.

3. EQUIPMENT IN THE SCOPE OF SEVERE ACCIDENT MITIGATION AND MONITORING

3.1. SEVERE ACCIDENT MITIGATION EQUIPMENT

The equipment needed for monitoring and mitigating severe accidents is defined in SAMGs. Reliably performing mitigation equipment can reduce the consequences of a severe accident.

The mitigation equipment installed in the containment typically includes: containment isolation valves, motor or air operated valves on emergency core cooling injection lines, power cables and penetrations. This equipment needs to reliably perform its intended safety function during and after exposure to severe accident environmental conditions. There is also mitigation equipment that may be indirectly exposed to the consequences of severe accidents (e.g. elevated temperature and radiation values).

Depending on the plant design and SAMG mitigation strategies, the reliable performance of the following systems may be needed:

- Systems ensuring the containment integrity including containment shell, penetrations, isolations valves, hatches, airlocks seals etc.;
- Reactor coolant system (RCS) depressurization;
- Hydrogen mitigation (monitoring and recombination);
- Containment heat removal system;
- Accident monitoring system.

3.2. SEVERE ACCIDENT MONITORING INSTRUMENTATION

Reference [6] states that electrical and I&C equipment needed for severe accident management has to perform reliably under severe accident conditions.

The main function of the accident instrumentation is to provide reliable and unambiguous information even during the extreme conditions of a severe accident. The main parameter for determining inadequate core cooling is typically the core outlet temperature for PWRs, and the reactor coolant level for PHWRs and BWRs. The main parameter for determining the containment integrity is the pressure inside the containment and radiation releases outside the containment. Other parameters indicating potential degradation of the containment fission product barrier include temperature, reactor pressure vessel water level (RPVL), containment sump water level, combustible gas concentration, and radiation level.

Accident measurement channels consisting of sensor/transducer, associated cables, connections, terminal boxes and containment penetrations are typically qualified for DBA conditions. After transition to severe accident conditions, the aforementioned equipment is exposed to conditions above their design limits which could result in loss of the associated measurements channel. According to Ref. [1], this might require extension of the capability of this equipment. Alternatively this can be achieved by limitation of the consequences of the severe accident at the installation positions (shielding).

Accident monitoring instruments designed for DBA conditions may not be able to ensure measurement accuracy over wide ranges of parameters when subjected to severe accident conditions. This is acceptable because the trending of these parameters is more important than obtaining precise values of a specific quantity.

3.2.1. Instrumentation for indicating the status of fission product barrier integrity

The experience during the last forty years has shown that the determination of the integrity of the fission product barriers needs reliable performance of instrumentation and equipment. The indications obtained from monitoring instrumentation during and after severe accident allow the operator to determine when to implement specific mitigating strategies and measures as well as to determine the effectiveness of such strategies and measures. Instrumentation may indicate:

- The possible re-criticality of the reactor;
- The indication of a reactor pressure vessel melt through;
- The location of the core debris/corium;
- The success and effectiveness of water injection (i.e. level and flow rate) into the reactor and/or the containment;
- The success of cooling the core debris/corium and the containment heat removal;
- Factors possibly jeopardizing containment integrity, e.g. flammable concentration of hydrogen, steam explosion, molten core concrete interaction or reaching the containment design pressure;
- Temperature levels that would jeopardize steam generator tube integrity.

Some additional information serves to monitor and estimate the progression of the accident:

- Neutron flux measurements (existing measurements can be used as long as the core is within the pressure vessel);
- Trend of containment pressure and temperature;
- Reactor vessel pressure and temperature;
- Water levels at relevant locations;
- Temperatures in the cooling chains, flow rates in cooling systems;

- Gas concentration in different locations of the containment (hydrogen, carbon monoxide);
- Dose rates inside/outside containment;
- Activity measurements in release paths;
- Positions of isolation valves and actuators.

Reference [7] identifies a set of SAMG accident instrumentation that might be useable to provide information in the event that portions or all of the normal monitoring system fails as well as non-instrumented information sources that may be used to gather needed information. However, monitoring during severe accidents needs to be accomplished by using systems that are designated for severe accident use.

3.2.2. Spent fuel pool instrumentation

Reliable indication of spent fuel pool (SFP) water level is necessary to ensure: (i) water level is adequate to support operation of the normal fuel pool cooling system, (ii) water level is adequate to provide substantial radiation shielding, and (iii) operating personnel are aware of a decrease in water level to the point where actions to implement addition of makeup water are needed. In addition, reliable indication of the SFP water temperature is necessary for determining whether adequate cooling for the spent fuel is being achieved.

3.3. CONSIDERATION OF MATERIAL PROPERTIES

The material properties of the equipment need to be known in order to understand the potential impact on capability of the equipment to perform reliably and to assess the limitations of the equipment. For some materials the direct exposure to radiation or humidity has to be avoided (e.g. PTFE Polytetrafluoroethylene (PTFE) Teflon®, Polyimide). Other materials that could degrade under severe accident environmental conditions need to be carefully evaluated.

For example, PTFE insulated seals and cables are susceptible to exposure to ionizing radiation (alpha and β). These polymeric materials lose their mechanical properties. Consequently, parts of the equipment made of these materials may degrade to the point where the equipment cannot perform reliably. PTFE insulated seals in containment penetrations, when exposed to high γ doses, may result in the loss of containment integrity. In contrast, the use of some materials, e.g. metals, glass, ceramics, high performance polymers which have good resistance to high radiation, is recommended.

3.4. SUPPORTING AND AUXILIARY SYSTEMS

Support systems such as power supplies, air supplies, sampling system piping, component cooling with connection points are necessary to enable the mitigating system to perform reliably. Such vital support systems (excluding I&C and electrical power supply) include:

- Compressed air;
- Heating, ventilation and air conditioning (HVAC) for equipment and personnel;
- Emergency lighting in the plant buildings;
- Communication and security systems.

Some components or parts of supporting and auxiliary systems are located in the auxiliary building, areas adjacent to the containment and are likely to be exposed to elevated radiation and temperature levels. In order to ensure their functionality in long term, either qualification or an assessment of reliable performance is needed.

4. ESTIMATION OF ENVIRONMENTAL PARAMETERS

4.1. MODELLING OF SEVERE ACCIDENTS

Since there is a lack of data from experiments and plant operations defining containment thermodynamic performance, severe accident codes are considered one of the sources to identify containment bounding conditions. Simulation using severe accident codes may aid in determining the necessary instrument ranges (including margins), and instrument mission times.

Characterization of containment thermodynamic profiles is dependent on a large number of severe accident phenomena. Annex I provides examples for calculating containment environmental parameters anticipated during severe accident for boiling water reactors (BWR) and pressurized water reactors (PWR) plant design.

A description of the commonly used severe accident codes is given in Annex II. All severe accident codes have uncertainty in modelling. To compensate for the uncertainty, the commonly used "best estimate plus uncertainty" approach (BEPU) is applied. Additionally, References [8] and [9] provide examples of recent results in calculating the severe accident profiles for light water reactors in the United States. Some post Fukushima modelling data was included in the source deck for simulating the severe accident conditions for loss of offsite power and loss of heat sink. These were developed for evaluating the performance of accident mitigation and monitoring equipment at nuclear power plants.

In order to estimate representative environmental characteristics for the equipment performance during severe accident conditions, the following types of calculations have to be performed:

- Calculation of selected parameters for locations directly subjected to severe accident conditions inside the containment;
- Calculation of selected parameters for locations outside the containment (these locations are subjected to milder environmental conditions than severe accident, but they are affected by the severe accident).

In these calculations at a minimum, the following parameters have to be determined for all selected equipment locations: temperature, pressure, radiation levels, humidity, combustible gas concentration and flooding level.

Based on the results of the modelling, to assess the capability of the equipment to perform reliably, test profiles have to be defined considering the following:

- Temperature vs time;
- Pressure vs time;
- Presence of saturated or superheated steam conditions;
- Dose rate;
- Use of one or two pressure and temperature peaks and their duration;
- Use of chemical spray during the test;
- Use of submergence during the test, total test duration.

State of the art severe accident modelling needs to be considered for the specification of appropriate test parameters or assessment of the survivability of the equipment. Annex I provides examples for calculating the environmental parameters anticipated during a severe accident for BWR and PWR plant designs.

4.2. MISSION TIME

The electrical and I&C equipment mission time is established based on the intended equipment function within the framework of the appropriate mitigation strategy. Overall, accident monitoring equipment is needed beyond achievement of controlled stable state of the plant. The mission time can vary for each piece of equipment. Mission times can be derived from analysis of the different stages of severe accident as described in Section 3.

The equipment mission time may consist of a passive phase, in which the equipment is in standby mode and has to withstand loading conditions without any active operation, followed by an active phase in which the equipment is called upon to execute the required function. The lessons learned from severe accidents that have already occurred show these phases may be long lasting. The relationship between the passive and active phases depends on the equipment intended function. It can be shorter than the duration of harsh environmental conditions resulted from the severe accident conditions. For example, electrical equipment which have moving active mechanical parts (e.g. actuators needed for containment isolation) may have both passive and active phases. This is an important fact to consider as it allows for dividing the severe accident management strategies into several stages of response thereby relaxing the design requirements imposed on dedicated equipment. In contrast, instrument readings necessary for providing continuous information to monitor the accomplishment of mitigating safety functions and reporting status of fission product barriers are needed to function during the entire accident duration.

5. EQUIPMENT DESIGN CAPABILITY UNDER SEVERE ACCIDENT CONDITIONS

5.1. PRINCIPAL CONSIDERATIONS

As stated earlier, References [2] and [3] recommend that electrical and I&C equipment needed to function during and after a severe accident, e.g. monitoring equipment, may be protected against the effects of severe environmental conditions that may result from the severe accident. To protect electrical and I&C equipment it may be physically separated, installed at a safer location, or shielded against the effects of such an event. In case adequate protection cannot be accomplished or is not feasible, the equipment has to be assessed for its capability to perform reliably under severe accident conditions. When assessing electrical and I&C equipment design capabilities under severe accident, the following needs to be considered:

- Availability, accessibility and functionality;
- Uncertainty in the loading parameters for instrument / equipment performance;
- Equipment locations;
- Acceptability of degraded performance of electrical and I&C equipment under harsh environmental conditions (e.g. instrument accuracy, valve stroke time, cable insulation resistance).

Prediction of instrument performance in advance is helpful for future interpretation of measured values under severe accident conditions. Furthermore, possible repair or replacement and sources for alternative signals can be considered in advance. The entire instrument loop (sensor, cable, connections, containment penetrations, etc.) performance may be affected by the severe accident environmental conditions, e.g. when the equipment design limit is exceeded. Degraded instrument loop performance may result in an instrument signal increase, oscillation or complete failure.

Preplanning may include identification of specific alternative signals in the accident mitigation procedures or guidelines and preparation of operator aids for interpreting the readings from degraded instrument channels. In addition, training for operators is needed to recognize when information from alternative signals are acceptable, or when it is apparent that designated severe accident instrumentation is no longer performing reliably. Prediction of the effects of degraded performance of electrical equipment under harsh environmental conditions (e.g. valve stroke time, cable insulation resistance) can also be performed. The results from this evaluation can be used to optimise the selection of mitigation strategies.

The technical report in Ref. [10] provides a methodology for addressing the usability of existing plant instruments during a severe accident. The methodology was applied to two pilot plants, a BWR and a PWR. Ref. [10] concludes that the instrument assessment methodology described is practical and provides guidance on the use of available instrumentation for decision making during severe accident. The report also describes how to identify alternate means for obtaining information supporting mitigating strategies through the use of indirect measurements and operator aids.

5.2. FUNCTIONAL REQUIREMENTS FOR MITIGATING EQUIPMENT

5.2.1. Performance criteria

The sequence and the magnitude of the loading conditions anticipated during severe accidents impose additional challenges to electrical and I&C equipment.

Validation of measurement values can be performed by crosschecking of an instrumentation reading with those of other available alternatives. Comparison of measured values against modelling estimates provides an alternative way for measurement validations. In some cases, a combination of these methods might be necessary in order to gain confidence in the instrumentation reading or equipment function.

In order to achieve reliable performance, electrical and I&C equipment needs to meet specified performance criteria. These performance criteria include: functionality, accuracy and response time. They can be derived from the intended safety functions and may be treated with different degrees of importance. For example, the instrumentation accuracy may be less important than trend indication. Furthermore, the functionality of the instrumentation in a long-term is more relevant than the accuracy attained, because replacement during and after a severe accident may not be possible. However, a minimum level of accuracy is needed for proper decision making in the frame of the mitigation strategy.

5.2.2. Instrumentation measurement range

Determination of an instrumentation measurement range is performed to cover all accident conditions including expected stages of the severe accident.

The range of instrumentation used for monitoring design basis accident conditions covers with appropriate margin the predicted full range of the expected variables. Typically the margin is provided to ensure the instrumentation remains on scale when analytical uncertainties in the predicted range, and additional harsh environment measurement errors are considered. It may be necessary to extend the measurement ranges of existing monitoring equipment to cover ranges of variables that are predicted when a breach of a fission product boundary is expected during severe accidents. As described previously, severe accident modelling may provide insights identifying the appropriate range for such instrumentation.

For example, the pressure instrumentation provided to detect a potential breach of containment pressure boundary typically spans from the range of predicted containment failure conditions from sub atmospheric to ultimate bearing capacity of the containment, including a margin sufficient to account for uncertainties in these values.

5.2.3. Instrument accuracy

The accuracy requirements specified for each severe accident mitigation strategy are based on the level of accuracy needed for decision making. In general, instrumentation can be separated into two categories; those that are intended to determine the exact value of a variable (or status of the variable), and those that are to be used to determine the trend of a variable.

For instrumentation used for severe accident mitigation strategies, trending is frequently more important, although a specific value may still be needed. Accuracy requirements for trending purposes may be sufficient to allow users to determine if the value is increasing, decreasing or staying roughly the same. For spatial orientation and distribution of instrumentation measuring the same parameter, where measurements are provided in different locations, accuracy needs to be sufficient that measurement uncertainties will not cause trend information to be ambiguous. The update frequency needs to be adequate to avoid misleading the operator.

References [11] and [12] describe methods for obtaining information from instrumentation subjected to severe accidents. Specifically, Ref. [11] provides an example for obtaining information from an apparently malfunctioning instrument or for otherwise determining the value of the parameter in question. The report presents a summary of the results of an extensive search for information related to the performance of instruments under severe accident conditions, including the Three Mile Island Unit 2 experience.

5.2.4. Instrument response time

When determining response time for analogue and digital instrumentation, the instrument's intended function needs to be considered. Timely information is needed, but it can also be understood that displayed information will lag behind actual conditions for various reasons.

Since the same equipment will provide information during various stages of severe accidents, the equipment needs to have a response time commensurate with the most demanding mitigation strategy. In general, the early stages of accident progression demand the shortest response time. During the later stages of the severe accident longer response times may be tolerable, since trend information is usually relied upon.

For digital acquisition systems, the variable update rate may dominate a response time. For example, update rates on the order of once per second are normally sufficient for instrumentation directly read by the operator. Where accident monitoring data is used by computers for assisting operator understanding, the data processing time may dictate the response time requirements.

If grab samples are relied upon as a backup alternative to installed equipment, consideration needs to be made for the time necessary for obtaining and analysing the sample.

5.2.5. Instrument mission time

As described in Section 4.2, the instrument mission time is also an important criterion.

6. DEMONSTRATION OF RELIABLE PERFORMANCE FOR SEVERE ACCIDENT CONDITIONS

6.1. BASIC CONCEPT DERIVED FROM DBA QUALIFICATION

The entire measurement or actuation chain needs to be evaluated for its capability to withstand the expected environmental conditions. This typically includes motors, solenoid drives, sensors, transmitters, cable assemblies (i.e. cables, splices, connectors, etc.), terminal boxes, limit switches and penetrations. The monitoring equipment includes components from the sensor to the display to provide the plant operators with necessary information.

Reference [3] states that mitigation equipment and instrumentation for monitoring accident conditions may be designed to withstand:

- Operational conditions and anticipated operational occurrences;
- Induced vibration loads (seismic loads, airplane crash, explosion blast);

- Electromagnetic interferences (EMI) and fulfil the requirements of electromagnetic compatibility (EMC);
- Harsh environment conditions, which are a consequence of high energy line breaks that cause environmental loads (thermodynamic loads, radiation, chemical exposure, combustion processes, submergence).

Reference [4] suggests that qualification for EMI/EMC is not necessary within the “environmental” qualification sequence. The demonstration of EMI/EMC features is rather a separate path in equipment qualification. Thus, EMI/EMC qualification is beyond the scope of this TECDOC. Reference [4] describes methods and practices relating to equipment qualification for DBA conditions. Provisions in this technical report could be used when assessing the capability of the equipment used for severe accident conditions. However, it is necessary to understand the physical limitations of the qualification methods, practices and testing facilities. Because of such limitations, the qualification methods need to be adapted to address the specific aspects caused by severe accident conditions.

6.2. CONSIDERATION FOR SEVERE ACCIDENTS

The assessment of reliable performance needs to evaluate the performance of the equipment while executing its intended safety function when exposed to environmental conditions caused by a severe accident.

Guidance for qualifying electrical and I&C equipment to withstand the environmental effects of design basis accidents and external hazards is well established by two standard development organizations the International Electrotechnical Commission (IEC), and the Institute of Electrical and Electronics Engineers (IEEE). Table 1 provides references to IEC and IEEE standards that provide methods for environmental qualification and for seismic events. These methods such as type testing and analysis may also be applied to equipment that supports mitigation and monitoring for severe accidents.

The IEC and IEEE standards however do not propose specific qualification methods and strategies for demonstrating the reliable performance for severe accident. Nevertheless, the new joint logo standard of IEC/IEEE 60780-323[1], Ref. [13] takes design extension conditions into account. For example, para 5.1 of the standard states that “For all items of equipment that are needed to operate under design extension conditions, demonstrable evidence shall be provided that it is able to perform its function(s) under the applicable service conditions including design extension conditions …” and §7.2.6.3 states that “for such equipment a plant specific severe accident profile may be used for component specific qualification requirements”. More details are provided in Section 6.5.

6.3. ANALYSIS OF EXISTING STANDARDS

In order to develop suitable methods for assessing the equipment capability to perform reliably under severe accident conditions, an analysis of different standards is necessary. The objective of the review is to identify methods, as well as any qualitative or quantitative criteria that can be used for assessing or qualifying the equipment for the environmental effects of a severe accident. Very likely, the existing standards may consider qualification for design basis accident conditions only, and qualification requirements may be given in a descriptive form. Although there may be procedures and methods similar to DBA qualification applied also for a severe accident, the environmental profiles, in particular very high radiation levels during extended time period may substantially differ during severe accident conditions. A review summary is provided in Table 1, quoting from the sources.

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[1] The IEC/IEEE 60780-323 std. (Edition 1.0) published on 19 February 2016 has been developed jointly by IEC and IEEE in order to harmonize methods for the environmental qualification of certain electrical and I&C equipment for nuclear power plants.

TABLE 1. REVIEW OF EXISTING NUCLEAR STANDARDS ON ELECTRICAL AND I&C EQUIPMENT

Standard	Content	Conclusion
YVL 5.2 Electrical power systems and components at nuclear facilities, Section 3.2 Qualification for environmental conditions	The qualification of electrical components that must operate during severe accidents shall be appropriately demonstrated. The qualification of electrical components and cables inside the containment, which must operate especially in the high temperatures occurring during severe accidents (possible hydrogen fires included), shall be demonstrated	Severe accident conditions have to be taken into consideration for equipment qualification. The specific conditions during the severe accident have to be considered.
YVL 5.5 Instrumentation systems at nuclear facilities, Section 2.5.4 Severe Accident	The design of the monitoring instrumentation for severe accidents shall fulfil the following requirements: – The measuring methods chosen shall be suitable for monitoring severe accidents. – The instrumentation shall be independent from all the other instrumentation at the plant. The power supply of the instrumentation (electricity, compressed air, etc.) shall be independent from all other power supplies of the plant.	These are design requirements. The severe accident is not explicitly addressed in the qualification Section. However, the standard refers to postulated accidents, which include the severe accidents. Test shall include aging steps, and steps considering the impact of humidity, pressure and rapid changes in the conditions as well as submerging on the equipment.
YVL 5.5 Instrumentation systems at nuclear facilities, Section 3.1 Qualification	If an automation device is to function in severe reactor accidents, it shall be qualified for this purpose by using suitable methods. The maintenance of the functional performance of automation devices located in the reactor containment during hydrogen fires shall be demonstrated if the equipment needs to operate in accident situations in which the occurrence of hydrogen fires is possible.	Suitable methods have to be applied to demonstrate the capability of the equipment to withstand and to function during severe accident conditions.
IEEE 627-2010 - IEEE Standard for Qualification of Equipment Used in Nuclear Facilities, Section 4.1, Purpose of Qualification	The primary purpose of equipment qualification is to provide reasonable assurance that design and age related common cause failures will not occur to multiple trains of equipment impairing the equipment ability to perform its required function before, during, and after DBEs, as applicable. The overall equipment qualification programme is guided by the quality assurance/quality control programme requirements considered in the design, fabrication and qualification of equipment. Adherence to the quality programme requirements provides assurance that production equipment is the same as, and is traceable to, the qualified design configuration.	Severe accident is not addressed in the standard
IEEE 627-2010 - IEEE Standard for Qualification of Equipment Used in Nuclear Facilities, Appendix A.5	A third consideration for equipment qualification is whether an instrument is required to operate during a design basis event for accident monitoring purposes. IEEE Std 497™ [B17] and RG 1.97 [B47] provide guidance on which types of accident monitoring instruments require equipment qualification.	Severe accident is not addressed in the standard
IEEE 323 (2003), Section 3.1 Definitions	3.11 Harsh environment: An environment resulting from a design basis event, i.e., loss-of-coolant accident (LOCA), high-energy line break (HELB), and main steam line break (MSLB).	Severe accident is not addressed in the standard.

Standard	Content	Conclusion (cont.)
IEEE 323 (2003) Section 6.1.5.2 Design basis event conditions	The postulated design basis event conditions including specified high-energy line break, loss-of-coolant accident, main steam line break, and/or safe shutdown seismic events, during or after which the equipment is required to perform its safety function(s), shall be specified. Equipment shall be qualified for the duration of its operational performance requirement for each applicable design basis event condition, including any required post design basis event operability period.	Severe accident is not addressed in the standard.
IEEE 383 (2003) - IEEE Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations, Section 6.4.4 Design Basis Event Simulation	The design basis event simulation and test procedures shall envelop the environmental and electrical parameters and shall encompass the acceptance criteria as a minimum. Specialty cables such as coaxial, twin axial, or tri axial are often selected for purpose of their extra shielding feature or for added insulation value. In these instances, performance shall be assessed for the specific application instead of the cable's ultimate capability. Any specialized applications using these cables for their high-frequency capability, for example, must be specifically evaluated to define performance criteria.	Severe accident is not addressed in the standard.
IEEE 497 -2016; IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations	Criteria are established in this standard for variable selection, performance, design, and qualification of accident monitoring instrumentation for anticipated operational, design basis events and severe accidents	Severe accident is addressed in the standard.
IEEE 317 (2013), Section 5.1.3 Design pressure and temperature	Note: Under Severe Accident Conditions (SAC), the containment may be subjected to higher pressures and temperatures. Consideration may be given to qualifying the electric penetration to a pressure rating comparable to the containment rating to prevent leakage paths for the severe accident environment and preserve containment integrity.	The mechanical strength of the penetration against pressure loads shall be identical to the containment ratings.
IEEE 317 (2013), Section 6.4 Severe accident conditions	A preconditioned electric penetration design may be tested for Severe Accident Conditions (SAC) of temperature, pressure, humidity and radiation (if not included in the preconditioning) to verify that the electric penetration will maintain containment integrity post-SAC.	The test of mechanical integrity is recommended after the severe accident load has been applied.
IEEE 317 (2013), Section 6.4 Severe accident conditions	The effects of chemical or dematerialized water sprays, submergence (if required), seismic loading, fault currents … conductor operation at rated current and voltage are optional and do not need to be addressed by SAC test.	No functional verification has to be performed during severe accident testing. As stated in the note of Section 6.4 IEEE does not consider the severe accident as qualification test.
NRC JLD-ISG-2012-03, Compliance with Order EA-12-051,Reliable Spent Fuel Pool Instrumentation, Section 3.4, Qualification	Appropriate quality assurance measures may be applied to all instrument channel components to ensure reliability following beyond design basis external events, including seismic events.	Several options for qualification of SFP level instruments are available including test, analysis or other means to show instrumentation can perform its intended function for severe accident conditions.

Standard	Content	Conclusion (cont.)
NRC Order EA-12-051, Order Modifying Licenses with regard to Reliable Spent Fuel Pool Instrumentation, ,Section 1.4, Qualification	The qualification methods, which may include justification based on significant operating history, testing results, or other appropriate means, may apply to the beyond-design-basis initiating event, as well as the potential result of the spent fuel pool remaining at saturation conditions for an extended period. The primary and backup instrument channels shall be reliable at temperature, humidity, and radiation levels consistent with the spent fuel pool water at saturation conditions for an extended period. This reliability shall be established through use of an augmented quality assurance process (e.g., process similar to that applied to the site fire protection programme).	SFP instrumentation may perform its intended function for conditions when the SFP water at saturation for an extended duration. Reasonable assurance that the instrumentation can function may be established through an augmented quality assurance programme.
NEI 12-02, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," Revision 1, Section 3.4, Qualification	The instrument channel reliability shall be demonstrated via an appropriate combination of design, analyses, operating experience, and/or testing of channel components for the following sets of parameters, as described in the paragraphs below: – Conditions in the area of instrument channel component use for all instrument components, – Effects of shock and vibration on instrument channel components used during any applicable event for only installed components, and – Seismic effects on instrument channel components used during and following a potential seismic event for only installed components.	Qualification can be performed via design, analyses, operating experience, and/or test for conditions in the SFP for extended duration.
NRC Order EA-12-049,	Licensees or CP holders must provide reasonable protection for the associated equipment from external events. Such protection must demonstrate that there is adequate capacity to address challenges to core cooling, containment, and SFP cooling capabilities at all units on a site subject to this Order.	This refers to mitigation strategies for severe accidents and such equipment must provide reasonable assurance that it can perform its intended function.
NEI 12-06, Diverse and Flexible Coping Strategies (FLEX)Implementation Guide, Section 11.2,	Design requirements and supporting analysis may be developed for portable equipment that directly performs a FLEX mitigation strategy for core, containment, and SFP that provides the inputs, assumptions, and documented analysis that the mitigation strategy and support equipment will perform as intended. This documentation has to be auditable, consistent with generally accepted engineering principles and practices, and controlled within the configuration document control system.	For portable equipment for mitigation of severe accident, documented analysis show that the equipment can perform its intended function for the environment it is required.
JLD-ISG-2012-01, Compliance with Order EA-12-049,Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Section 6.2, Equipment Quality	NEI 12-06 provides an acceptable method to control the quality of equipment associated with Order EA-12-049 with the following clarifications. Installed structures, systems and components pursuant to 10 CFR 50.63(a) may continue to meet the augmented quality guidelines of Regulatory Guide 1.155, "Station Blackout."	Under severe accident conditions, portable equipment, analysis and maintenance and testing programs should provide reasonable assurance that the equipment can perform its intended function.

Standard	Content	Conclusion (cont.)
	Development of maintenance and testing programs for the portable equipment responsive to Order EA-12-049, following the guidelines of NEI 12-06 and standard industry processes for ensuring equipment reliability, provides an acceptable method to reasonably assure the equipment will be functional.	
RCC-E (2012), Section B7210 Acceptable methods	The analysis test or combined methods are used. The test method for severe accident conditions is similar to the test procedure described in B6000. In other words, the sequence is identical, but the severity levels, methods and acceptance criteria are different.	The test procedure for severe accident is identical to the sequence covering the design basis events.
RCC-E (2012), Section B7230 Successive use of K1 and SA procedure	Seismic resistance will have been demonstrated during the K1 procedure. The test does not need to be repeated during the SA procedure.	Seismic tests do not need to be performed in the severe accident qualification sequence, provided the suitability has been confirmed during a design basis event qualification.
European Utility Requirement Volume 2, Section 4 Design Basis, Section 2.4.8 Equipment Qualification	For systems, structures and components required to mitigate DEC, especially those required to mitigate Severe Accident conditions, the survivability shall be demonstrated.	Severe accident conditions have to be taken into consideration during equipment qualification.
European Utility Requirement Volume 2, Section 4 Design Basis, Section 2.4.8.3 Demonstration of survivability of Safety Category II Equipment	This equipment shall be subject to an assessment to demonstrate that its design provides a reasonable level of confidence that it will operate in the environment under which it is required to perform its function in Design Extension Conditions for the required mission time.	The capability of the equipment to withstand severe accident environmental conditions and to perform the designated function during the accident has to be demonstrated. A comment is included which states that the equipment needed for the DEC mitigation must be identified and provision shall be made for its availability during the course of the event. An additional comment in the standard states that seismic qualification is not required if this equipment is not credited for DBA. In other words, equipment necessary for severe accident monitoring and mitigation only does not need to be seismically qualified.
CD2 IEC60772:2015 Nuclear Power Plants-Instrumentation Systems important to safety - Electrical Penetration Assemblies in Containment structures	Chapter 5.2.7: “The rated capabilities of EPA/ EPA modules required functioning during DBE and DEC shall be defined in the requirement specification. Rated capabilities shall be defined so that the safety function of the connected electrical equipment under DBE or DEC is ensured if required” Chapter 7.6: “The confirmation of the survivability in the case of DEC (e.g. severe accident) shall follow the methodology described in Chapter 7.4 and 7.5 with respect to the specific demands given in the requirement specification. The confirmation envelops the mechanical function for all cases, and electrical function of some equipment if needed for mitigation and/or monitoring of the DEC event. Note: Keeping the integrity of the containment (avoiding the containment breach) has the first priority in mitigating a severe accident).	The specific conditions the penetration assembly is subjected to depend on the reactor model and therefore they are project specific. The methodology which shall be used for the demonstration of the reliability/survivability is identical to those used for DBE. It consists mainly of conditioning of the equipment and accident simulation accompanied by functional testing.

Standard	Content	Conclusion (cont.)
	Since the EPA is part of the outermost containment barrier, it seems reasonable that the sealant system has similar attributes as other pieces of equipment pertaining to the barrier (hatches, mechanical penetration and containment structure). Properties in this context are understood as mechanical tightness and the mission time required. These aspects are relevant for preparing the project specific requirement specification.”	
IEC/IEEE 60780-323: 2016, Nuclear Facilities - electrical equipment important to safety qualification	Chapter 7.2.6.3: “Some equipment needs to be qualified for conditions that are beyond design basis of the plant (e.g., extended station black out, extreme natural hazards, and severe accident). For such equipment a plant specific severe accident profile may be used for component specific qualification requirements. Design bases and design extension conditions should be periodically reassessed in response to events in the region, shared international experience or other findings. To account for these new situations, the following shall be addressed: – Changes in the plant design needed to limit the consequence of these situations on equipment, – Justify that the existing qualification programme covers new requirements or, if it is not the case, perform the qualification programme for addressing the change in the anticipated environments”.	The severe accident qualification is addressed in general, and it should be covered by a plant specific accident profile. Furthermore, changes in the plant design and adaptions of qualification programs shall be respected (“… justify that …”)

The following can be concluded from the review of applicable standards that was presented in Table 1:

- Almost all standards consider qualification for design basis accident conditions only.
- Qualification requirements are given in a descriptive form providing expectations on what is to be the result of the qualification process.
- The French Design and Conception Rules for Electrical Equipment of Nuclear Island (RCC-E, 2012 edition), Section B6000 provides information, which can be directly used for preparation of the equipment qualification programme. It states that the qualification to severe accident conditions is similar in procedure and methodology to qualification for design basis accident conditions (in RCC-E it is called K1 procedure) that can be described by the test sequences, aging (radiological and thermal), seismic tests and accident simulation tests.
- The European Utility Requirements (EUR), Volume 2, Section 4, states that it is not necessary to subject the equipment needed for severe accident mitigation and monitoring to the simulation of seismic loads, whereas the RCC-E allows omitting the seismic test only if similar or identical equipment has already been tested for DBA purposes including a seismic test.
- The IEC/IEEE 60780-323 standard addresses the design extension condition category/severe accident in a descriptive way. The standard indicates that a demonstration of the safety function is needed after the application of specific profiles for severe accidents.

6.4. INFORMATION NEEDED FOR THE ASSESSMENT

The following sources of information need to be analysed:

- The safety analysis report (in the case of a new plant) or supplements to the safety analysis report (in the case of a plant already in operation).
- Specific demands from applicable codes and standards (international and national)
- Specific calculation and assumptions made for the specific plant-based also on software tools developed for the calculation of severe accident purposes.
- Documents describing the environmental conditions of the plant based on the specific plant design in the case of normal operation and severe accident conditions.
- Documents defining the systems needed for the mitigation and the monitoring of the severe accident.
- Documents deriving the specific functional requirements of instrumentation and equipment pertaining to systems described above. This may include the measurement range/operating range, demanded accuracy, mission time, response time, etc.

Sources of information that are needed for defining the scope of equipment subject to assessment for severe accident are shown in Fig. 2.

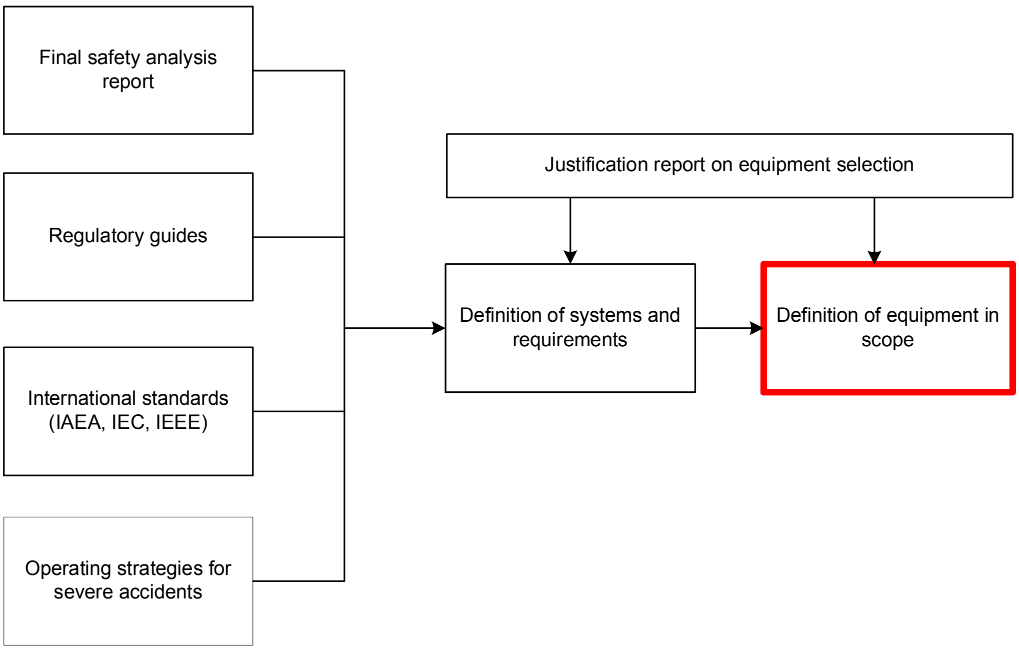

FIG. 2. An example of sources of information that are needed for defining the scope of equipment subject to assessment for severe accidents.

6.5. ASSESSMENT OF RELIABLE PERFORMANCE

6.5.1. General description of the assessment

The process for assessing equipment capability to perform reliably under severe accident conditions involves the following:

- Surveying and evaluating of available information on assessment of reliable performance of equipment as described in international technical reports, codes and standards.

- Describing the assessment process that demonstrates the capability of the equipment to perform reliably.
- Evaluating the impact of specific environmental effects typical for severe accidents, such as temperatures spikes as a consequence of H_2 combustion, high radiation levels caused by the release of active material from the melted core and atmospheric conditions in the containment after quenching of the molten core and corium concrete interaction.
- Evaluating the impact of specific environmental effects at individual equipment installation locations.
- Developing the general approach for assessing the reliable performance of the equipment. This approach may include equipment type testing, assessment of equipment survivability, comparison with previously tested equipment and evaluation of existing margins that may be available from previous qualification testing.
- Identifying alternative measures if the equipment performance is not sufficiently reliable.

The proposed sequence for assessing reliable performance under severe accident conditions may consist of the following steps:

- Performing reference functional tests in order to confirm the safety function under normal operating conditions.
- Conditioning of the equipment using applicable methods in order to simulate the consequences of thermal, radiological and mechanical aging under normal operation conditions (e.g. long lasting vibration and wear).
- Application of accident radiation dose (which may be higher than the dose under design basis condition)
- Application of the p-T profiles including humidity and chemical exposure simulating the accident phase.
- Application of conditions of the post-accident phase (may be up to one year or longer).
- Performing reference functional tests in order to assess the survivability of the equipment during the accident.

Seismic event testing is not described among the steps above, since it is assumed that the procedure regarding the seismic event testing is well established and therefore the equipment is already qualified.

Specific equipment performance acceptance criteria do not necessarily have to be established for the assessment of reliable performance. Rather, the objective of severe accident type testing is to document the expected equipment behaviour under simulated severe accident loads and to compare the performance of this equipment to the expected conditions in individual plant locations.

Qualitative acceptance criteria may be developed, however, it is more important to demonstrate that the equipment remains available, and is providing the required functionality. For example, a reduction in measurement accuracy may be acceptable provided that it can be demonstrated that a particular instrument is able to retain its functions under severe accident conditions for at least 100 days, albeit with degraded accuracy but still capable of providing information on the trends of designated parameters.

6.5.2. Formal assessment process (reports and deliverables)

The formal assessment process may include:

- Development of a programme, which describes the methodology, applied for specific equipment or an equipment type series. It defines which assessment steps are needed to be performed and which methods are to be used for the assessment.
- Preparation of reports that provide a technical basis for performing assessment of equipment performance by analytical methods.

— Development a specific test specification that describes all steps belonging to type testing.
— Selection of appropriate testing laboratories that are able to reproduce the severe accident environmental conditions.
— Conducting the qualification tests in accordance with the test specifications. Each test has to be documented by suitable test protocols or test reports.
— Evaluation of the results gathered in the demonstration, and preparation of a qualification summary report for each equipment subjected to type testing.
— Preparation of a suitability analysis comprising the results of analysis and type testing to demonstrate the reliable performance of the equipment assessed.

If the assessed equipment is not suitable to perform reliably under severe accident conditions it is necessary to:

— Modify the equipment design or complete change of the used physical principles;
— Replace the equipment already installed in the plant;
— Change the installation location;
— Apply protection measures for the equipment in scope (e.g. additional thermal or radiological shielding);
— Propose alternative methods to gather required information for decision making (e.g. indirect measurements).

The quality assurance of the assessment of equipment capability process to perform reliably under severe accident conditions, and manufacturing processes of the electrical and I&C equipment needs to be conducted within the framework of a management system that meets the requirements of Ref. [14], and follows the recommendations of Refs [15] and [16].

For equipment that has already been installed in the plant, the following provisions may be considered when assessing the equipment capability under severe accident conditions:

— Identify potential changes in the plant design to limit the impact of consequence of severe accident conditions on the existing equipment;
— Select the equipment by considering the following aspects:
  - o Whether the equipment design follows quality development processes;
  - o Whether the equipment installed in containment/fuel storage building or in locations where multiple stressors could occur in case of severe accident has already been qualified for LOCA conditions.
  - o Whether the equipment has been installed in a location with only radiation stress or only temperature stress in case of a severe accident.
  - o Whether the equipment is already qualified to required seismic loads with sufficient margin or does not contain material that could be vulnerable to the radiation and high temperature degradation
— Justify that the existing qualification programme covers new requirements or, if it is not the case, perform the qualification programme for addressing the change in the anticipated environments.

When existing equipment does not have a sufficient margin to cover severe accident environmental conditions then an assessment of survivability of the equipment with in the new conditions needs to be performed or additional tests are needed on equipment of the similar representative type (typically same technology, same suppliers, same materials).

The irradiation dose expected during a severe accident includes both γ and β radiation. When existing equipment qualification is being analysed to credit to previous qualification results it is important to note that previous radiation qualification may have been performed solely using γ-radiation. Severe accident conditions may have significantly greater levels of β-radiation that may cause additional degradation. A common practice is to irradiate the equipment by using γ sources to simulate the energy deposition of both γ and β radiation effects. Justification of the equivalence of relative levels of β radiation compensated by additional γ radiation needs to be provided.

6.5.3. Simulating the impact of degradation on equipment function

Electrical and I&C equipment are subjected to degradation due to high temperature, accident radiation combined with humidity (or submergence), and chemical exposure. Following the basic concept set out in Section 6.1, a simulation of the degradation may be covered by the following three demonstration steps:

Step 1: Accident radiation exposure: This step can be performed applying a dose rate that represents the average of the dose rate vs. time function during the accident. Since the severe accident dose rate is typically very high early in the severe accident mission time, and then exponentially decreasing over time following the initial stages of the severe accident, it might be more practical to perform the accident irradiation exposure for a long term mission time using a lower dose rate (simulating the average dose rate over the entire expected mission time) than that which would otherwise be applied to simulate the first days after the onset of the accident, followed by a significantly longer exposure at a decreasing lower rate. Since the average dose rate over the severe accident mission time is also much higher than the dose rate applied under design basis conditions, it is perhaps less important to simulate the effects of the initially high dose rate, and more important to observe instrument performance in handling the effects of total accumulated dose.

Step 2: Chemical exposure: In order to reduce the pressure and the temperature within the containment, spraying is an applicable method. Sprays usually contain boric acid and chemical agents and may contribute to the equipment degradation. The equipment under test may be exposed to spraying during accident simulation (p-T curve) and during the post-accident phase simulation. Carefully selected equipment protective materials, such as stainless steel of appropriate thickness, glass or ceramics can limit or avoid the effect chemical exposure to the equipment.

Step 3: Pressure and temperature loads: Pressure and temperature are applied in the time scale identical to the actual expected event scenario. This is valid for early and middle stages of the accident simulation (e.g. up to 100 or 150 h). In contrast, the equipment that is needed for operation in the long term (e.g. up to one to three years), for example the containment pressure boundary related parts, needs to be tested using an accelerated aging procedure. This procedure might be similar to the simulation of the in service ageing for design basis accident qualification.

Duration of accident conditions covering a period of one month to one year of accident conditions need to be applied, in order to account for approximation uncertainties. The uncertainties in development of those profiles are relatively large; that is why the use of the approximated environmental profiles for test purposes is reasonable. The test profiles are therefore developed to consider these uncertainties.

In some cases, the accident profile may be adapted for the test purposes. For example, using the method of energy deposition (i.e. calculated for bounding cases), a simplified bounding profile can be created. This method is acceptable because the degradation of the equipment is approximately proportional to the energy deposition. The calculated temperature profile used for equipment testing is shown in Fig. 3. and Fig. 4.

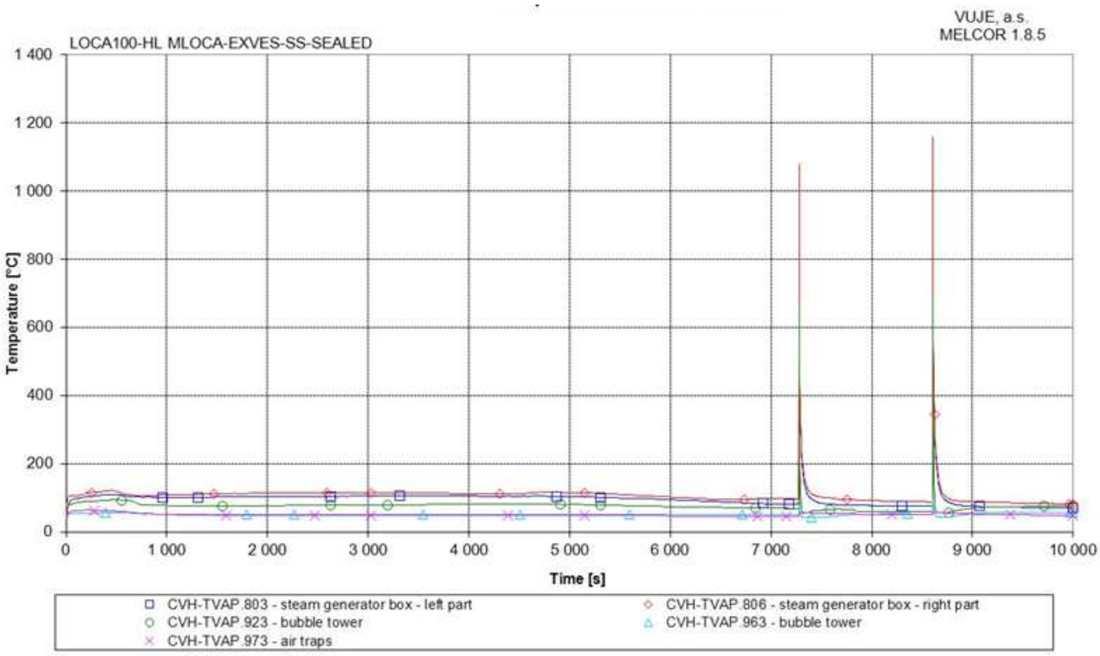

FIG. 3. An example of simulated temperature behaviour in the containment during severe accidents (figure courtesy of VUJE, a.s.).

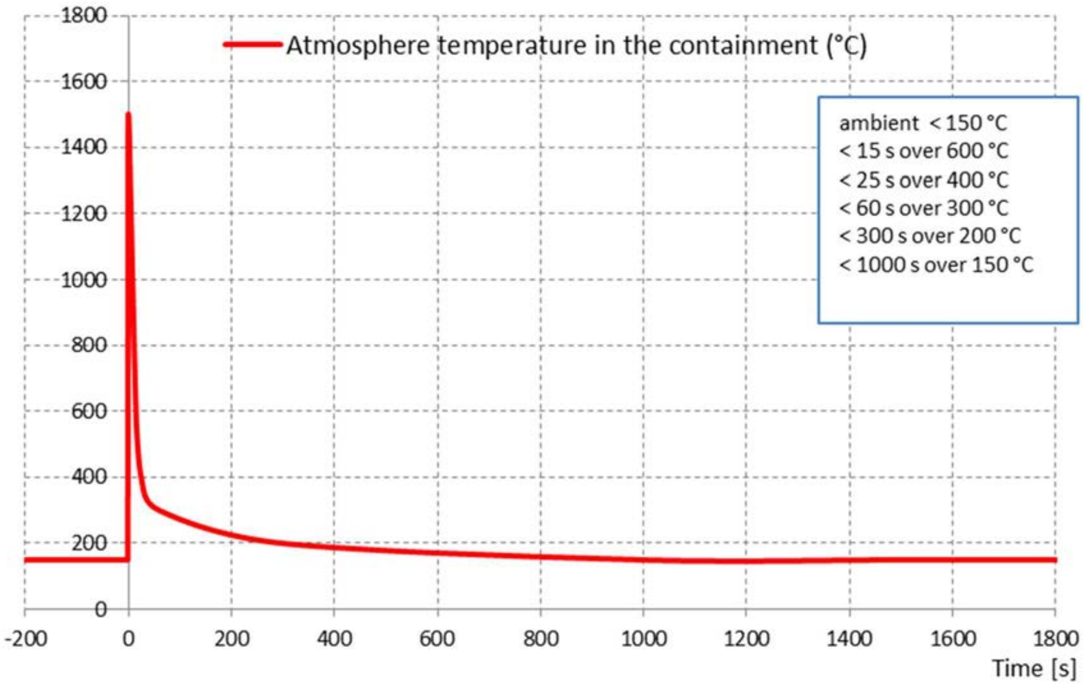

FIG. 4. An example of adapted temperature profile during hydrogen burning in the containment (figure courtesy of, a.s.).

6.6. DESCRIPTION OF RECENTLY IMPLEMENTED PRACTICES

This section provides a description of a robust equipment and instrumentation, specially designed and qualified to withstand severe accident conditions. Typically, this new instruments/equipment can be implemented as part of accident monitoring system at new NPP design, or to be considered for a back fit of existing plants.

6.6.1. Use of robust instrumentation/equipment

6.6.1.1. Accident Level Measurement for pools and vessels inside the containment

In the framework of the EPR^{TM} design, AREVA was developing robust accident level measurement equipment (ALM) for monitoring levels of pools and vessels under conditions of design basis accidents and of a severe accident. The level measurement is installed at the instrumentation bridge of the in-containment refuelling water storage tank (IRWST). The position of the instrumentation bridge is in opposite direction to the core catcher and is exposed to one of the highest radiological loads expected during the severe accident, see also Fig. 5.

During the severe accident progression, the ALM device has to withstand a total integrated accident dose of 5MGy over one year mission time, and a maximum temperature of 156°C combined with saturated steam conditions (duration approximately 12 hours).

Because these extreme ambient conditions may have adverse impact on the signal transmission to locations outside the reactor building, only measurement principles creating electrical signals with sufficient high amplitudes are acceptable. Thus, any radar based principles for monitoring levels of pools and vessels under severe accident conditions were excluded during the design phase.

The selected measurement principle is based on a resistor/reed-relay chain, where a magnetic float actuates the reed-relays corresponding to the fluid level. One of the biggest advantages of this method is the robust electrical signal, since the inner resistance of the chain can be kept low (a few hundred ohms) and the voltage level can be adjusted by the direct current fed into the device. The higher the current is, the higher the voltage amplitude is at the signal output. Since no high frequency signals (e.g. pulses with rise times in the range of nanoseconds) are used, the effects of signal damping of the connected cable is of less importance, and mineral insulated cables can be used for the cable routing inside the containment.

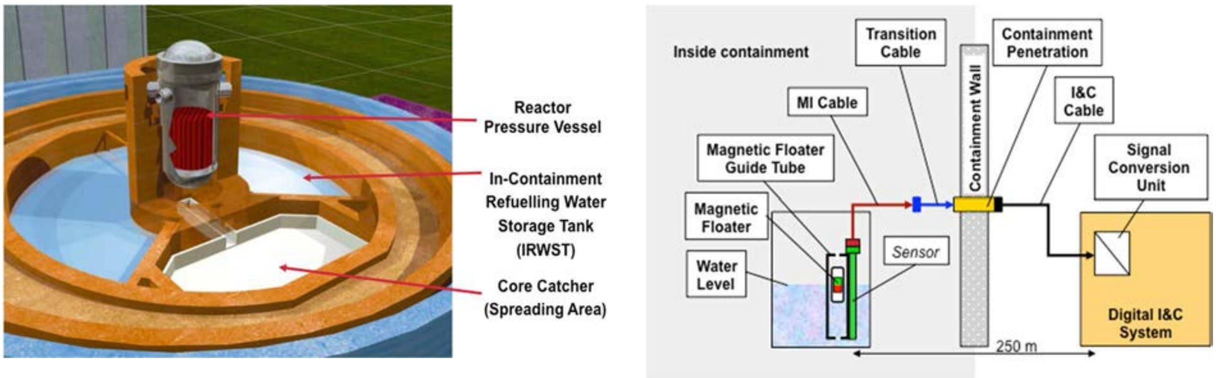

FIG. 5. Principal drawing of the position of the IRWST in relation to the core catcher (left hand side) and overview of the equipment used for the measurement chain of the level measurement (right hand side), (picture courtesy of AREVA).

If the device is expected to be installed in positions experiencing less severe environmental conditions than expected in the vicinity of the IRWST, the device can be equipped with polymer insulated cables, which are easier to install (with reduced risk of damage to the outer sheath during shipping from the manufacturer to the site, and subsequent handling and bending during construction.)

In addition to the cables and their connection interfaces, all parts of the device are made up of inorganic materials, such as stainless steel, ceramic and metallic gaskets. To avoid any impact of high temperature on the electrical connections inside the containment, no soldering procedure is applied during manufacturing.

For demonstrating the reliable equipment performance during the severe accident the assessment followed in general the steps described in Section 6.5.3.

— Step one: Accident radiation exposure: It was analytically proven that the equipment resists the required accident radiation loads and the function is not affected adversely because all parts are inorganic. Furthermore, the greater portion of the total integrated dose during severe accidents is caused by beta radiation. To address this, all the electrical active parts (resistors, reed-relays and solder joints) are shielded by metallic enclosure (stainless steel). This reduces the dose rate load of the reed-relays in the interior of the device so that a loss of function as a consequence of the ionisation of the filling gas can be excluded. This fact was proven experimentally.
— Step two: Chemical exposure: This part of the qualification campaign was also verified on an analytical basis. Evidence was provided that all parts of the device that are in contact with the coolant fluid and the atmosphere, including the connection interface are stable against the chemicals. Because the enclosure is made of stainless steel it is resistant against boric acid (weak acid) as well as against basic chemicals such as lithium hydroxide. Moreover, stainless steel is not affected adversely by gases that can result from the zirconium-water and the corium-concrete interactions (CO and H_2).
— Step three: Pressure and temperature loads: Pressure and temperature loads were adjusted to cover the specific features and material properties of the device:

1. Inorganic gaskets that are part of the connection interfaces may be susceptible to temperature transients because of different thermal expansion coefficients between the gaskets and the enclosure. Because of that, the accident simulation was performed using two transients (peaks) to expose the device to the most severe conditions.

2. Since no organic materials are used for the components of the device and the rated temperature for these materials does not exceed the maximum accident temperature no degradation processes (aging) can occur. This allowed the reduction of the total duration of the test, particularly shortening the duration of the low temperature interval at the end of the test.

3. Because of the fact that ageing processes can be excluded and the resistance against chemical exposure was successfully proven (see step two), no post-accident simulation was necessary to be performed.

Monitoring the function of the level measurement devices during the accident simulation is difficult to be performed, since the fluid used for establishing the level may evaporate into the test vessel atmosphere and precise measurement is not possible. To remedy this, magnetic coils to simulate changes in the level were used as an appropriate solution for obtaining accurate measurement results. Figure 6 presents the specimens used and the test setup.

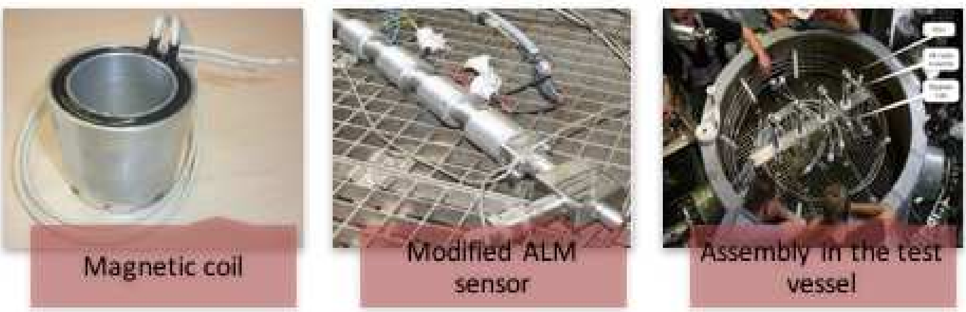

FIG. 6. Design of the specimens, and test assembly used for the simulation of the severe accident (picture courtesy of AREVA).

Because the accident level measurement device is also used for measurements of level for design basis events, the test campaign was performed in accordance with the KTA3505 standard, also fulfilling the requirements of the analogous standard in Ref. [13]. The KTA3505 standard was one of the selected codes and standards for qualifying safety classified I&C equipment for the EPR project. In addition to severe accident tests, the sequence included tests on operational limits, as well as verification of the function under seismic loads and loads caused by an airplane crash. Furthermore, tests complementing the accident test sequence were performed to validate specific features of the level measurement device, such as a test providing evidence that the device is capable of resisting clogging the magnetic float guide tube due to debris in the coolant.

6.6.1.2. Hydrogen monitoring for design basis and severe accidents

In support of the Japan SA-Keisou severe accident monitoring project (see Annexes IV and V), a U.S. vendor is developing a new hydrogen monitoring system to measure the presence and detonation risk of hydrogen gas which can form within the containment or reactor buildings following design basis and severe accident events. The new hydrogen monitoring system, shown in Fig. 7, provides signals that let nuclear plant operators know the hydrogen risk at each critical location where it is installed, including hydrogen concentration, risk of detonation, oxygen concentration, ambient temperature, pressure, and steam/humidity levels. These parameters enable operators to receive information regarding hydrogen gas levels as well as carbon monoxide levels simultaneously.

The system consists of a sensor (gas monitoring unit) to be located within the physical area of interest, and a gas monitoring unit controller/signal processing unit to be located outside the containment and away from the area where the worst-case harsh conditions are expected following severe accident events. The sensor unit is being qualified to function reliably within the very harsh environmental conditions expected to be present under severe accident conditions, and the signal processing unit is being qualified for rugged environmental conditions, but less harsh than those expected to occur inside containments. An accuracy of ± 2% is expected for the system.

The expected qualification radiation is 5MGy (500 MRads) for the gas monitoring unit and 31Gy (3100 Rads) for the gas monitoring unit controller. The sensor is being qualified to function in an ambient environment of up to 700°C (1292 °F) and 1062 kPa (154 psig). The design of the sensing unit uses a proprietary sensing technology to convert hydrogen gas using basic chemical reactions to an electrical signal proportional to gas concentration. This conversion provides for a rapid signal response that is highly selective to hydrogen.

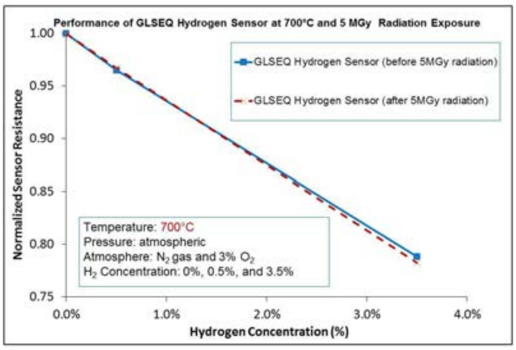

FIG. 7. Gas Monitoring Unit (GMU) for In-containment SA monitoring of hydrogen/carbon monoxide concentration and explosive risk with sensors: hydrogen, oxygen, pressure, temperature, and RH. hydrogen sensor test results for hydrogen monitoring performance at 700°c before and after radiation exposure of 5 MGy gamma radiation. (picture courtesy of GLSEQ, LLC.).

The unit is stable over a wide range of temperatures and extreme environmental ambient conditions, suitable for use both in containment and outside containment, where leakage through piping and electrical penetrations could occur under high containment pressure conditions. The key for its capability to reliably perform under severe accident condition performance is the design of its hydrogen/combustible gas measurement system and electrical isolation system using only glass, ceramic, and metal materials (i.e. no organic materials) that are compatible with the gases present, and the temperature, pressure, and radiological conditions expected for accidents with significant fuel damage.

The sensor has been tested to reliably coexist and function in the presence of caesium iodide, iodine, and methyl iodide. Testing has been performed to determine the unit's capabilities and performance sensitivities to harsh environmental conditions in the presence of gases expected to be present. Tests performed at 700°C indicate the output of the sensor responds nearly equally to measurement of 3.5% carbon monoxide and 3.5% hydrogen. These tests were performed at atmospheric pressure in a nitrogen and 1% oxygen mixture, with significant levels of carbon monoxide present.

The devices are being manufactured to meet the qualification criteria for Class 1E equipment provided in Ref. [13] and standard criteria for accident instrumentations for nuclear power generating stations provided in Ref. [17].

6.6.2. Requalification of electric cable penetrations

This section gives an example of demonstrating reliable performance of already installed cable penetrations at nuclear power plants for severe accident condition. The following assumptions are made:

- The main safety function of cable penetrations during severe accidents is to maintain leak tightness, i.e. to prevent radioactive material release;
- Only some cable penetrations need to retain their electrical functionality (e.g. those penetrations transferring signals from sensors needed for the monitoring of the accident conditions, and those providing motive electrical power and control signals to operate valves, solenoid valves, and other components needed to mitigate the effects of the severe accident);
- The worst case effects of severe accident occur inside the containment. The greatest portion of degradation occurs on the containment side of the penetration. Therefore, the application of severe accident conditions on the containment side may be sufficient.

The suggested steps for demonstrating reliable performance include:

- Testing the functionality, gas leak rate test and electrical properties;
- Pre-aging to simulate long term normal operation: consisting of thermal aging, radiation aging, thermal cycles, vibration aging (if applicable);
- Proving the functionality of the pre-aged penetration specimens;
- Irradiation of the penetration specimens on containment side with severe accident integrated total dose;
- Simulate the thermodynamic temperature-pressure profile of a severe accident and demonstrate continued functionality;
- Analysis of results to prove the reliable performance of the connected measurement and actuator chains.

6.6.3. Protection of the equipment, reduction of mission time

If the reliable performance of the equipment cannot be demonstrated, protecting the equipment from the effects of severe accident conditions is an acceptable method.

Hydrogen burning inside of the containment may lead to exposure of the equipment to high temperature spikes. The amplitude of the spikes depends on the hydrogen concentration, oxygen concentration and the thermal capacity of the surrounding atmosphere (nitrogen and steam content). An example of a temperature loading profile is shown in Fig. 4.

The loading profile appears demanding and not directly applicable for commercially available equipment. However, the following phenomena significantly reduce the effective impact of temperature spikes:

- The limited heat transfer between the atmosphere and the surface of the equipment,
- The thermal capacity of the equipment enclosure materials;
- The comparatively low thermal capacity of the overheated atmosphere.

Any heat transfer processes need a certain amount of time to materialize. The actual pressurized atmosphere inside the containment will have a density of about 1.3 kg/m^3 and a thermal heat capacity of about 1,100 J×kg^{-1}×K^{-1}. This will result in a total volumetric thermal capacity of the atmosphere of about 1,400 J×m^{-3}×K^{-1}.

A widely used construction material, stainless steel, has a density of 7,900 kg× m^{-3} and thermal capacity of 460 J×kg^{-1}×K^{-1}. Therefore the volumetric thermal capacity of the steel is 3,634,000 J×m^{-3}× K^{-1}. The volumetric thermal capacity of the stainless steel is 2596 times higher than volumetric capacity of the atmosphere in the containment. Figure 8 presents an approximation of the resultant temperature functions vs. time in the atmosphere and on the surface of the equipment.

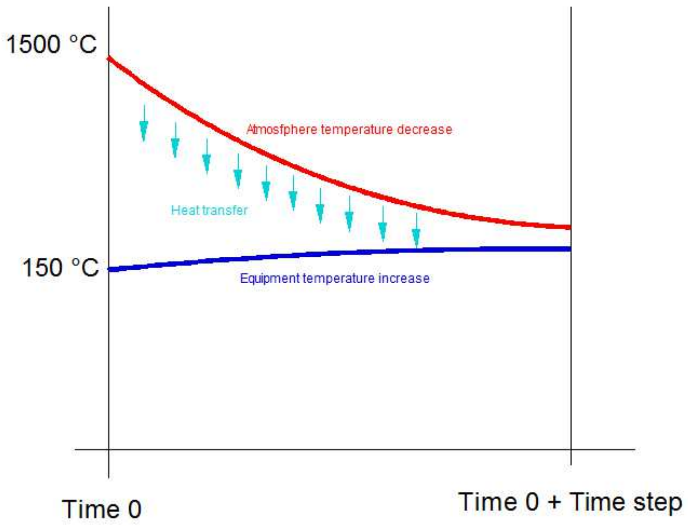

FIG. 8. An example of heat exchange processes between the containment atmosphere and the equipment surface (figure courtesy of VUJE, a.s.).

The approximation presented in Fig. 8 neglects heat conduction processes inside the equipment and the influences of turbulences in the atmosphere.

Figure 9 shows an example of the overall heat exchange processes on a protected radiation detector. The body of the detector is inserted inside of an enclosure (penetration sleeve). The gap between the body of the detector and enclosure is evacuated. Therefore, the only possible heat transfer mechanism between the body of the detector and the enclosure is thermal radiation.

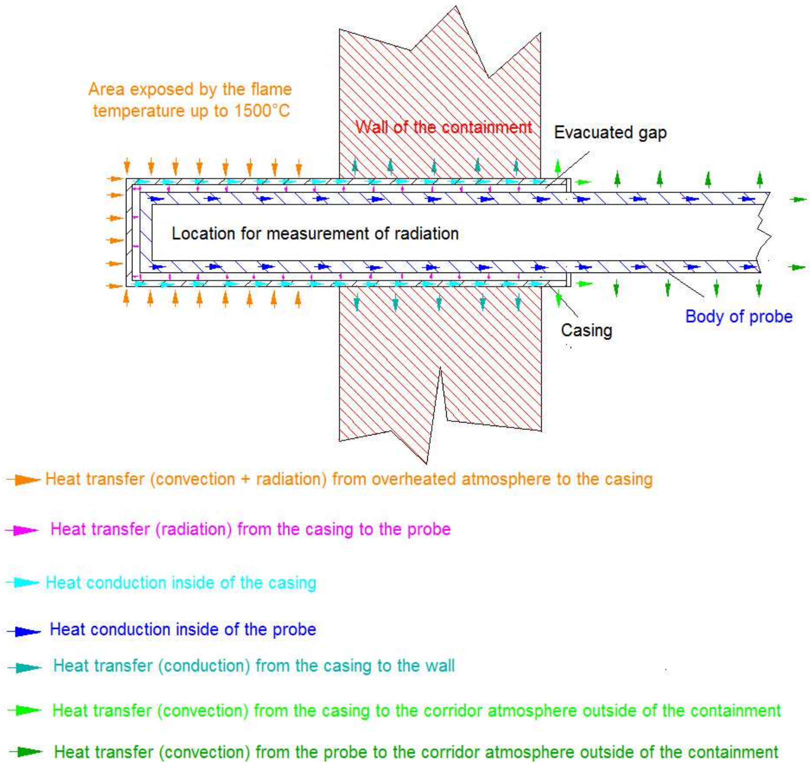

FIG. 9. An example of heat exchange processes in the protected radiation probe (figure courtesy of VUJE, a.s.).

The heat conduction inside the enclosure and the probe body allows cooling of the space where the measurement device is located. The results of the heat exchange of the protected radiation probe are shown in Fig. 10 and Fig. 11. The graphs reveal that the temperature increase in the sensing element is negligible (compare the Fig. 11 curve labeled C1 with Fig. 4).

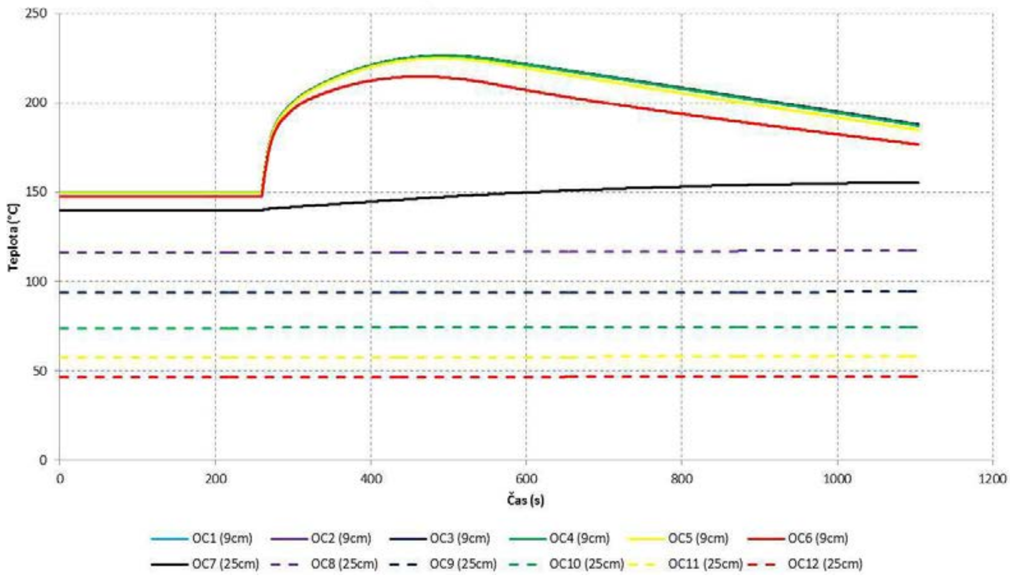

FIG. 10. An example of the casing surface temperature during normalized containment ambience response during hydrogen burning (figure courtesy of VUJE, a.s.).

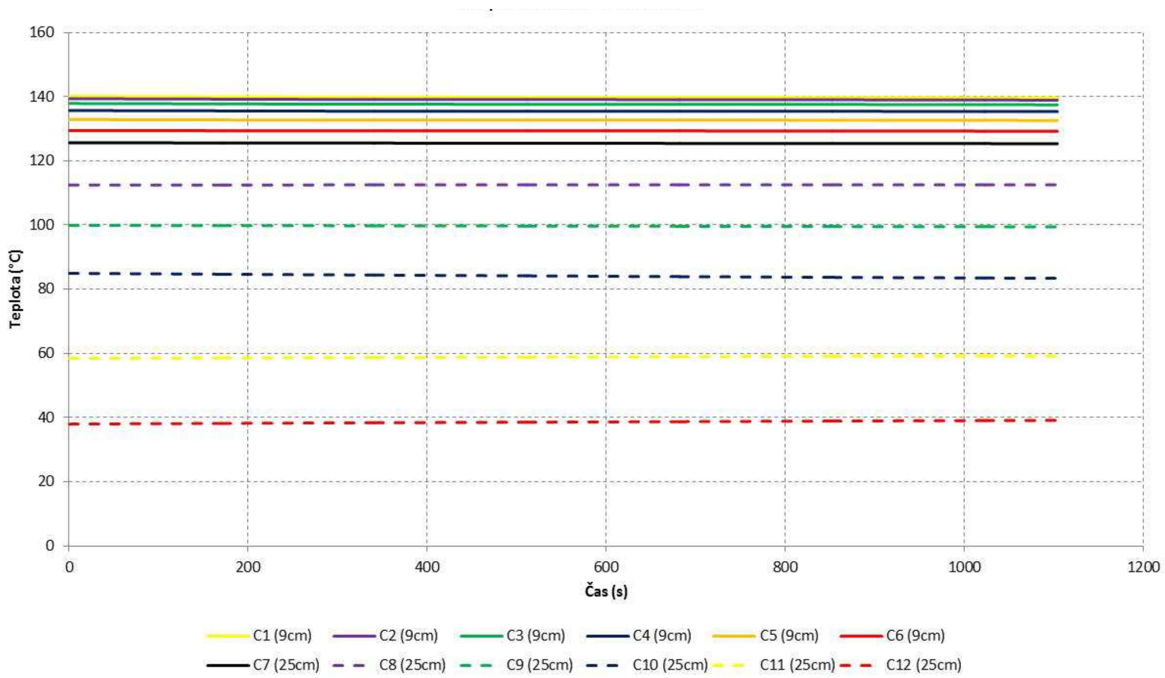

FIG. 11. An example of the probe surface temperature during normalized containment ambience response during hydrogen burning (figure courtesy of VUJE, a.s.).

6.6.4. Separate testing for the most severe environmental parameters

Severe accident profiles may include peak values resulting from hydrogen burning. In this case, it is reasonable to break the test into segments. A functional test of the equipment for peak values may be performed separately, while the test for the entire test profile and appropriate test duration is performed integrally.

In some cases it is reasonable to test the equipment into ultimate failure conditions in order to determine the actual safety margin available. This test can serve to provide an indication of the needs for supplemental measures within the accident mitigation strategies. It is also important to know the equipment can perform for the full mission time during which its operation is needed. This approach is acceptable because it is possible to credit the available equipment design margin for the severe accident.

6.6.5. Japanese national project on severe accident instrumentation

Following the Fukushima Daiichi accident a national project on the development of instrumentation systems for severe accident conditions was launched in Japan in 2012. The objectives of the project were to determine parameters (called SA-Keisou parameters) that severe accident monitoring equipment will need to be capable of withstanding, and develop qualification specifications that determine test conditions under which equipment is to be tested, and (iii) carry out qualification testing of severe accident monitoring instrumentation. SA-Keisou denotes the severe accident instrumentation systems.

Annex IV provides a description of how the SA-Keisou parameters were determined for BWR and PWR designs.

Annex V provides an example of a severe accident classification matrix used for the design of reactor pressure vessel water level instrumentation (BWR design).

Annex VI provides a description of the qualification programme for severe accident instrumentation for nuclear power plants in Japan. The qualification of severe accident instrumentation for nuclear power plants involves the following steps:

- Establishment of environmental conditions;
- Determination of basic specification of severe accident instrumentation;
- Verification of the test methods for severe accident instrumentation.

The qualification specifications determine test conditions under which equipment is tested. The test conditions are used to demonstrate the equipment capability to perform reliably under severe accident conditions. When test conditions could not be accomplished due to limitation of a testing facility, it was necessary to confirm that the acceptance criteria can be met by extrapolation methods or other means. A justification of reasons and use of alternative testing methods have to be provided.

7. SUMMARY AND CONCLUSIONS

The experience during the last forty years has shown that severe accidents may subject electrical and I&C equipment to conditions far outside the original design basis accident conditions. Severe accident conditions may cause rapid degradation or damage to varying degrees up to complete failure.

The electrical and I&C equipment needed to function under a severe accident has to be protected against the harsh environmental conditions. In case adequate protection cannot be accomplished or is not feasible for an existing nuclear power plant, the equipment has to be assessed for its capability to perform reliably under severe accident conditions.

This publication covers relevant aspects of assessing the reliable performance of the electrical and I&C equipment needed for severe accident mitigation and monitoring.

A common technical basis presented in this publication for assessing the capability of electrical and I&C equipment to perform reliably includes:

- Outlining issues related to electrical and I&C equipment capability to perform reliably under severe accident conditions;
- Providing examples of calculation tools for determining the environmental parameters for severe accidents;
- Providing examples of methods that may be applied in Member States to assess reliable performance of electrical and I&C equipment under severe accident conditions;
- Providing examples and methods that Member States may apply to enhance the capability of equipment dedicated for severe accident conditions.

Furthermore, this publication describes the general process for assessing equipment capability to perform reliably under severe accident conditions:

- Surveying and evaluating of available information on assessment of reliable performance of equipment as described in international technical reports, codes and standards.
- Describing the assessment process that demonstrates the capability of the equipment to perform reliably.
- Evaluating the impact of specific environmental effects typical for severe accidents, such as temperatures spikes as a consequence of H_2 combustion, high radiation levels caused by the release of active material from the melted core and atmospheric conditions in the containment after quenching of the melted core and corium concrete interaction.
- Evaluating the impact of specific environmental effects at individual equipment installation locations.

- Developing the general approach for assessing the reliable performance of the equipment. This approach may include equipment type testing, assessment of equipment survivability, comparison with previously tested equipment and evaluation of existing margins that may be available from previous qualification testing.
- Identifying alternative measures if the equipment performance is not sufficiently reliable.

Finally, this publication describes a set of documents needed for the assessment process:

- A programme document, which describes the methodology, applied for specific equipment or equipment type series.
- Reports that provide a technical basis for performing assessment of equipment performance by analytical methods.
- Test specifications that describe all steps belonging to type testing.
- Related test protocols or test reports.
- Summary report for each equipment subjected to the assessment.
- Preparation of a suitability analysis comprising the results of analysis and type testing to demonstrate the reliable performance of the equipment assessed.

The provisions described in this publication will assist in increasing the robustness of the plant electrical and I&C equipment for mitigating a severe accident and enhance the overall plant safety.

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ANNEX I

EXAMPLES OF SEVERE ACCIDENT PROFILES FOR TESTING AND ANALYSIS

I–1. INTRODUCTION

The following Annex presents a methodology to identify the containment bounding environment characterization in severe accident conditions together with two applications to generic western dry-containment PWR and Mark III BWR/6 designs.

The suggested multi-step process falls under the Best Estimate Plus Uncertainty (BEPU) approach to licensing adapted to the field of severe accidents and especially tailored to utility-engineering applications.

Main steps of the methodology comprise the (i) establishment and characterization of a matrix of risk significant severe accident sequences and high-importance / high uncertainty parameters affecting the selected figures of merit (FOMs) for containment environmental characterization, (ii) identification and quantification of the main uncertainty contributions, (iii) epistemic uncertainty propagation for each of the significant scenarios of the severe accident sequence matrix, and (iv) post-processing of the information to reach a set of histogram representations of the selected FOMs ready to be used in equipment qualification testing.

Since harsh condition characterization for equipment qualification cannot be drawn from experimental facilities, and real NPP accident data are too scarce and too plant and scenario-specific to be duly extrapolated, the only remaining option is through accident sequence simulations with nuclear system codes especially tailored to cope with the relevant thermal hydraulic and physicochemical phenomena.

According to Ref. [I–1], already existing methods in making use of safety analysis system codes can be classified into four different approaches:

— Conservative evaluation models2 plus conservative Boundary and Initial Conditions (BIC) and system availability assumptions (conservative analysis);
— Best estimate (BE) evaluation models plus conservative BIC and system availability assumptions;
— BEPU application to evaluation models and BIC assumptions; conservative systems availability assumptions (best estimate binding deterministic analysis);
— BEPU application to evaluation models and BIC assumptions; probabilistic safety assessment (PSA) based system availability assumptions (full best estimate probabilistic analysis).

Since the fully conservative approach characterizing the conservative analysis approach does not provide with the actual safety margins (see Ref. [I–2]) as the real value of the relevant plant parameter provided by the calculation of the code is unknown due to the deliberate pessimistic criteria characterizing the evaluation models, and sometimes can even lead to non-conservative results Ref. [I–3], the use of this approach is no longer recommended by Ref. [I–4]. Nonetheless, it is the national regulatory body which will ultimately have the last say on the suitability of the above-mentioned methods for safety analysis purposes. For instance, U.S. Code of Federal Regulations Ref. [I–5] allows only choosing between option 1 and 3 (in the frame of DBA licensing purposes), whereas Spanish Safety Instructions Ref. [I–6] limit engineering analysis for safety limits and Limiting Conditions for Operation (LCOs) to the use of conservative models, even though option 3 has already been accepted and applied in the Spanish Almaraz NPP power uprate modification under the legal framework allowing for the use of alternative designs and analysis methodologies during fuel outages Ref. [I–7].

[2] In this context, 'evaluation model' can be taken as synonym of thermal-hydraulic system code.

Though that all these methods address licensing related quantities hence careful assessment has to be taken on since severe accident system codes and conditions significantly depart from those falling under before-core-damage scenarios.

In estimating the potential application of the conservative analysis approach to the field of severe accidents, three reasons prevent its due extrapolation:

- The nature of the conservative evaluation models, such as those indicated in US NRC Appendix K to part 50 of the Code of Federal Regulations Ref. [I–5], grounds on previous accurate understanding of the phenomena evolution on which uncertainties can be recognized thereby pessimistically applied. If the scenario evolution identification lacks of a confident base case, then reality may substantially depart from what expected in a non-conservative direction, moreover if large uncertainties apply to a set of interrelated high order phenomena (located at a macro-scale level) as those characterizing the field of severe accidents.
- Conservative evaluation model assigned uncertainties in the frame of DBAs are quantitative hence largely stemming from micro scale (continuum domain) phenomena: departure of the modelled parameter from experiment can be captured thus is limited to a deviation from the exact value of that critical magnitude, e.g. it can be measured in kg/m3, W/(m K), etc. Instead, severe accident phenomena evolution gaps can be rather expressed not only in quantitative yet in qualitative uncertainty statements assigned to phenomena located at meso-scale (component) up to even macro-scale (system) level: corium flow paths identification, in-vessel water ingression after core damage, lower head corium stratification, hot leg / main steam line creep rupture, etc.
- Deliberate conservative burden applied to already extreme conditions such as those of severe accidents might lead to unaffordable SSC design technical specifications.

Second option based on a BE code together with imposed conservative BIC and systems availability assumptions lies on the fact that code uncertainties can be covered by conservatisms applied on the side of BICs and systems availability Ref. [I–1]. Code uncertainties trade-off is very likely whenever uncertainties are confidently believed to be limited according to an accurate state of phenomena modelling, fed and supported by a sufficiently large available experimental database, both of SETs and IETs, such as those collected in before-core-damage phenomena validation matrixes found in Ref. [I–8]. Nonetheless, Far reaching uncertainties in severe accident phenomena may lead to a completely different scenario evolution hence they can hardly be covered just by reducing the functional capacity of the affected safety systems, that is to say, since the set of subsequent affected phenomena evolution stemming from the anticipated error of one particular phenomenon is unknown, specific BICs to conservatively compensate for code uncertainties embedded in realistic phenomena simulations by the BE code are hardly identifiable and applicable. Notwithstanding with the rationales set forth above, one of the few full applications of severe accidents I&C survivability assessment has followed this option and based the calculation of the bounding severe accident environmental envelopes on best estimate without uncertainty calculations Ref. [I–9]. Introduction to Section 2 will provide details and discussion.

The suggested approach, which follows BEPU analysis such as aforementioned third and fourth options differing only on the systems availability nature of the established criteria, relies on deterministic and probabilistic assumptions of systems availability and initiating events. Since beyond design basis accidents are by definition an unbounded class of events, lacking of a set of well-defined framework similar to those collected in the Final Safety Analysis Report of an NPP through which straightforwardly proof that the analysed design modification meets the safety criteria, the safety engineer is constrained to make assumptions on what SA sequences need to be taken into account. An optimal technique to manage the large set of sequences in an objective and traceable fashion is through PSA tool application. And along with PSA, and even within PSA itself, deterministic assumptions will be applied to adapt and extend the initial selected sequences to the objective pursued.

Therefore, estimation of the containment environmental characterization for I&C survivability will be derived from BEPU analysis application in the use of system codes with probabilistic and deterministic criteria for the selection of the accident sequences, BIC and safety system availability conditions.

I–2. KEY ASPECTS OF THE METHODOLOGY

Well established methodologies aforementioned in Section 1 were mainly conceived for DBA licensing purposes and hence their natural field of application falls short when addressing severe accidents particular concerns such as sequence selection and uncertainty assessment.

The suggested process has been specifically tailored aiming at contributing to fill the currently existing gap in accident uncertainty analysis in the frame of design extension conditions (DECs). Theoretical grounds for the setting-up have been drawn from generic suggestions for BEPU approach application as found in Ref. [I–7], whose fundamental features align with some of the most widespread uncertainty calculation methodologies that have been qualified in Ref. [I–10] and collected in Ref. [I–11], such as Ref. [I–12]. Key underlying aspects of the suggested methodology are accident sequence identification and uncertainty assessment and application.

A thorough list of uncertainties is gathered in Ref. [I–13] and can be grouped into the following categories:

- Code structured calculation scheme. Balance equation; numerical methods; machine installation effects.
- Code user effects. Nodalization accuracy; BIC adequacy to as-built real plant data.
- Phenomena modelling assumptions. Constitutive and closure relations comprising a set of interrelated theoretical hypothesis and data fitness against experimental results; thermodynamic and chemical properties.

First category of uncertainty sources has not played a role in traditional BEPU methods applications and its fulfilment merely consists of stating code capability in terms of numerical scheme and quality of the response.

Second type of uncertainty sources will be addressed only partially: on the one hand, utility NPP models would have been qualified and regularly updated and checked; on the other hand and considering the existence of a high-quality continuously updated model of the plant, deviations in BIC needs be limited hence resulting in negligible propagation effects on the results given FOM's large order of magnitude: since severe accident management ultimate goal focuses on containment integrity, magnitudes of interest are either extensive properties (such as pressure) or variables which largely depend on them (such as temperature), both relying on containment large dimensions so that small deviations in BIC will not substantially impact on the results[3].

The importance of nodalization to capture the main phenomena affecting FOM time dependent profile calculation has been extensively stressed, for instance in Ref. [I–14] and Ref. [I–15], so that sensitivity on nodalization schemes has been highly encouraged in BEPU applications. However, phenomena affecting DBA related magnitudes such as DNB or PCT, containment pressure or suppression pool temperature, included in the plant input files handled by utilities or TSOs (Technical Support Organizations), shall have undergone QA (Quality Assurance) process so that their nodalization adequacy shall have already been checked thus suitably extended to severe accident sequence simulations. There still are few other phenomena dealing exclusively with severe accidents and affecting containment characterization FOMs (such as lower head vessel nodalization, potential for non-condensable gases stratification or ex-vessel corium deposition onto the pedestal or reactor cavity)

[3] This statement is clearly not applicable to DBA where typical FOMs are not located at a macro-scale but at a micro-scale, such as PCT or the degree of cladding oxidation.

whose relating nodalization have likely not gone through QA. In this case, suitable nodalization schemes, particular for each of the affected phenomena, shall apply following code manuals (mostly dealing with benchmarking of the SA-related phenomena) and affected expert judgement found in open literature[4] (in experiment validation using the code of interest and for the SA spectrum of phenomena).

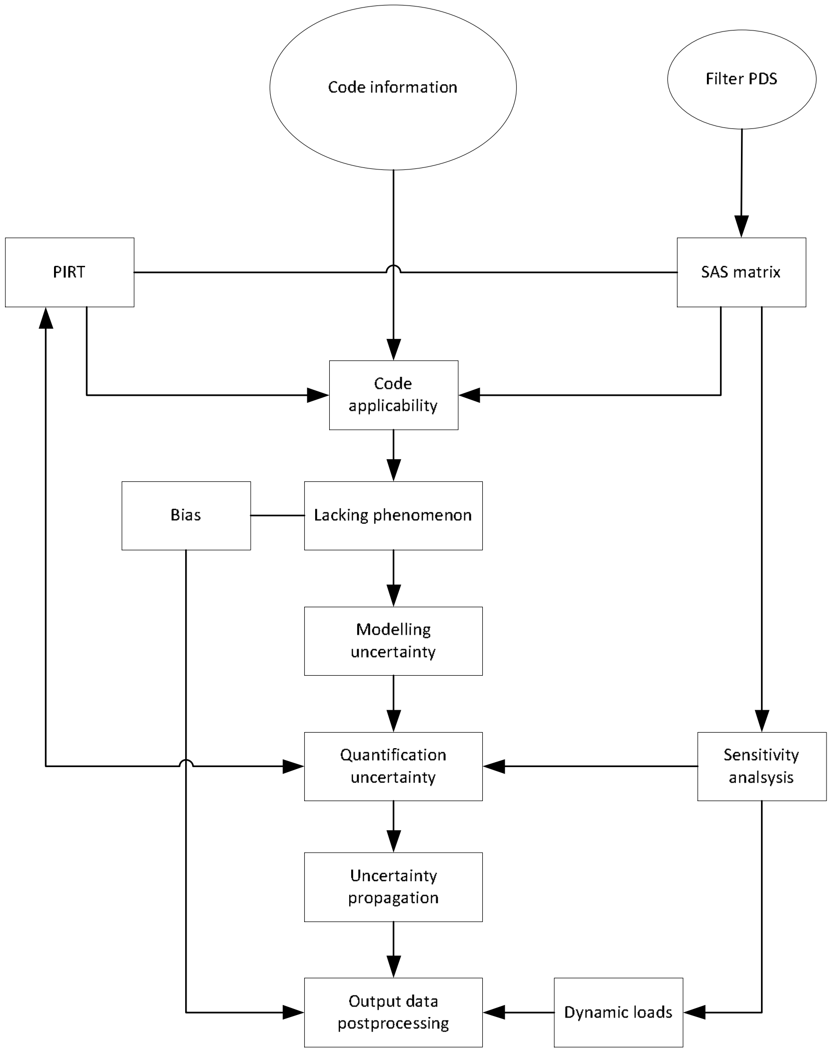

FIG. I–1. Severe accident containment characterization methodology flowchart.

In order to initially identify code uncertainty sources, and related to the third type of the aforementioned categories, uncertainties dealing with before CD phenomena will not be taken into account since their impact is judged to be significantly lower compared to discrepancies found ever since the core heat up phase: impact of deviations in PCT (in magnitude or time), DNBR, or containment / PCV pressure caused by pre CD phenomena will not lead to enveloping (neither as peaks nor as sustained high values) pursued containment FOMs of pressure, temperature, humidity, gases concentration or radiation.

Fig. I–1. depicts the flowchart of the methodology. Rounded-shaped boxes refer to information sources (which can be outputted from a task) hence without associated activity (other than gathering of relevant information) whereas rectangular-shaped boxes constitute the main tasks explained hereafter.

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[4] Open literature hereby designates material related to code-to-code and code-to-experiment studies (Separate Effect Tests) regulatory body reports and other significant reports coming from national laboratories, international agencies and vendors.

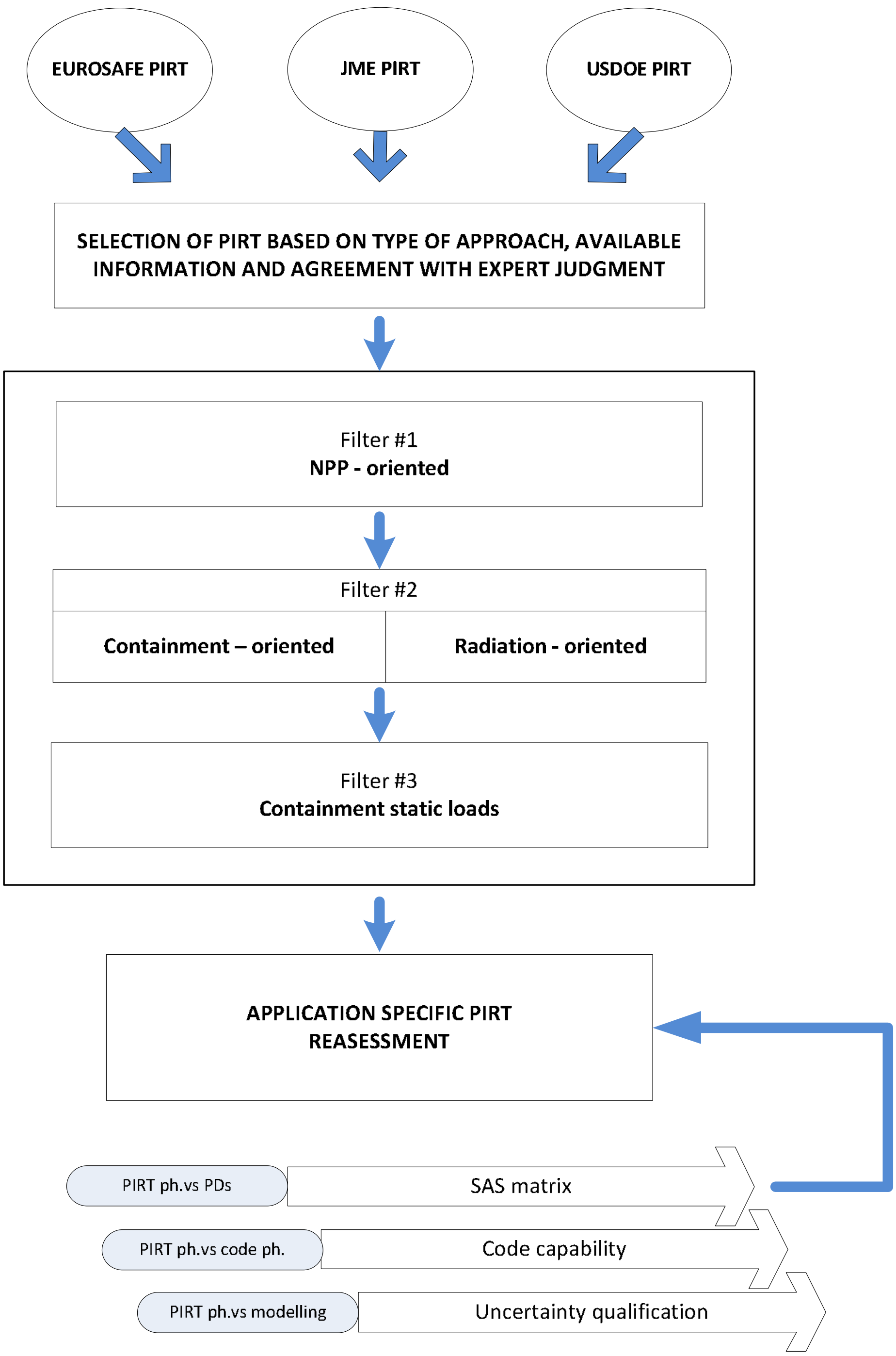

FIG. I–2. In house PIRT adaptation flowchart.

I–2.1. Step 1: Highly important and uncertain phenomena elicitation process

PIRT tool Ref. [I–16] within BEPU approach was first introduced by CSAU methodology to lighten the uncertainty assessment process, on one hand, and the uncertainty propagation process, on the other, since this was achieved through response surface methods so that the number of experimental calculations (number of code simulations) substantially increased with the number of uncertainty sources. PIRT application requires prior definition of goals, scenario (NPP and sequence), ranking means and criteria, and phenomena identification. PIRT goals within this work are (i) to identify modelling sources of uncertainty affecting containment FOMs; (ii) to simplify code applicability assessment; (iii) to check completeness of and to feed back (if needed) SAS matrix.

PIRT large time consuming developing efforts is a very demanding task involving allocation of significant high experienced, cross cutting experts coming from different fields, making the task unaffordable within the frame of the single utility / TSO. Since the current methodology aims at saving these issues and tailoring the process to a viable fashion in accordance with the state of the art and utility (or technical support) safety analysis capabilities, PIRT will be in the first place drawn from open literature and afterwards in-house adapted according to the process depicted in Fig. I–2.

Even if Far reaching qualified PIRTs are found in open literature, there will be the need of ensuring that PIRT development fully reflects plant-specific features and selected accident sequences, so that the utility shall check and reassess PIRT outcomes for agreement. Therefore, application specific PIRT development comprises the following steps: selecting the most suitable open literature PIRT; reassessing PIRT filtered outcomes phenomena; identifying phenomena and SSC related performance gaps; solving PIRT specific drawbacks.

I–2.2.1. Sources of PIRT

So far, three qualified severe accident PIRTs can be found in the open literature, all of them aiming at identifying the main areas of interest where R&D efforts has to focus on. Main features characterising available open literature PIRTs are collected in Table I–1.

- The European expert network for the reduction of uncertainties in severe accident safety issues (EURSAFE) developed a 2-year project gathering more than 200 experts coming from the entire spectrum of affected organizations such as R&D, utility, regulatory, industry and academy, [I–17]. The outcomes of the project were collected under a sequence-generic, plant-generic PIRT and a table on SA topics where further R&D was needed. A total of 916 phenomena were identified and their importance was ranked according to 3 different FOMs: primary circuit, containment and radiation. Among these 916 phenomena, 229 were classified as important and 106 were retained as both important and lacking of sufficient knowledge.
- Japanese Ministry of Economy, Trade and Industry (JME) launched a project in 2012 were national utilities, the Nuclear Fuel Division and the Severe Accident Research Committee of the Atomic Energy Society of Japan, together with USA Electric Power Research Institute and Fauske & Associates, LLC, developed a sequence-specific, plant-specific PIRT (Fukushima Unit 3 reactor and events) focused on ranking SA phenomena both in terms of importance and knowledge, where R&D oriented efforts to bridge current missing gaps not considered in MAAP 5.0.1 version of the code were derived from PIRT outcomes Ref. [I–18]. PIRT was developed looking specifically at Fukushima Unit 3 accident sequence. A total of 1047 phenomena were identified and their importance was ranked according to 4 different FOMs where each of them corresponded to a particular phase of the sequence: PCT, core average temperature (or enthalpy), maximum temperature of the corium located at the lower head, and containment maximum temperature and pressure. Among these 1047 phenomena, 386 were classified as important and 299 were retained as both important and lacking of sufficient knowledge among which 95 were found as lacking in MAAP 5.0.1 code.

US Department of Energy has recently conducted a sequence-specific, PWR/BWR-specific PIRT Ref. [I–19], both for BWRs and PWRs, with the goal of identifying gaps in modelling and experimental data. The PIRT panel was constituted by US experts in LWR operations and safety coming from US DOE staff, national laboratories, industry and BWROG and PWROG.

Phenomena were assessed by appointing radioactive material release as the single, dominant FOM encompassing challenging phenomena to each of the three key fission product barriers such as cladding, primary system and containment. Given that the analysis was not limited to identifying gaps in severe accident phenomenology but also in safety systems performance, an additional functional criterion of potential impact on system availability under BDBE conditions of interest was taken into account. Available results are limited to the 13 phenomena or operation that were identified by the panel as being of high importance and lacking of sufficient knowledge, not being currently addressed by industry, NRC or DOE, together with Human Reliability Analysis and SA instrumentation.

It is worth noticing that the PIRT panel tended to group highly detailed, micro scale phenomena such as identified in EURSAFE and JME PIRT approaches into macro-scale type of phenomena, or category of phenomena, when performing the evaluation process.

TABLE I–1. MAIN FEATURES CHARACTERISING PIRTS

PIRT NAME	ENTRY TYPE / APPROACH	TOTAL ENTRIES	H/H ENTRIES	DRAWBACKS
EURSAFE	Micro scale / Direct phenomena-vs-code-parameter correspondence	916	106	– Not updated after Fukushima – Not accounting for mitigating system sources of uncertainty – Full list of analysed phenomena not available
JME	Micro scale / Direct phenomena-vs-code-parameter correspondence	1047	299	– Sequence-specific – Plant-specific – Not accounting for mitigating system sources of uncertainty
USDOE	System-scale / Coarse phenomena-vs-code-parameter correspondence	Unknown	15	– Not detailed results – PWR/BWR-specific – Sequence-specific – Full list of analysed phenomena not available

I–2.2.2. Filtering process and reassessment

In order to readapt open literature PIRT initial objectives to goals as identified in Section 2.3, PIRT outcomes has to undergo a three step filtering process:

- Non-applicable phenomena in terms of plant-specific SSCs and NPP design has to be neglected;
- Phenomena non affecting containment characterization FOMs or not contributing to bounding values has to be neglected. This is the case, for instance, of uncertainties in containment leakage penetrations or PWR SGTR sequence related, where part of the mass and energy source is discharged and deposited outside containment;
- Phenomena characterising by transferring dynamic loads on the containment barrier up to mechanical failure may be neglected since later evolution after loss of containment isolation will always lead to milder FOM values (pressure, temperature, radiation, and species concentration will decline after containment failure). Containment response on dynamic loads lower than containment failure value shall be further analysed, even though resulting peak values might go beyond component's design maximum capacity. High Pressure Melt Ejection (HPME) related phenomena has be instead considered (i) since it largely affects corium mass and energy distribution thus containment FOMs time dependent profile, and (ii) since it may likely occur without ultimately leading to containment mechanical failure.

Once the open-literature PIRT has been selected and outcomes have been filtered, plant-specific reassessment need to be undertaken furthermore addressing the following aspects:

- NPP and sequence specific phenomena and SSCs gaps identification and solution;
- Open literature PIRT specific drawbacks identification and solution.

Most PIRT specific drawbacks solutions are found just by referring to the other open literature PIRTs. This is the case for instance of EURSAFE lacking of SSC treatment, which can be compensated at least partially by looking up USDOE PIRT; in turn, USDOE PIRT category kind analysis could increase in detail, when addressing phenomena to code modelling conversion, by relating selected categories with specific phenomena as included for instance in EURSAFE PIRT.

I-2.2. Step 2: Severe accident sequence matrix

Severe Accident Sequence (SAS) matrix links PIRT outcomes with risk-significant accident sequences coming from expanded PDSs (PDS*), i.e. scenarios accounting for mitigating systems performance and related human actions usually not included in Level 1-2 interface PSA PDSs. The other input to SAS matrix comes from deterministic criteria implemented following a conservative approach (as explained hereafter) as additional accident sequences not originally present in the set of PDS*s. Fig. I–3. shows an example *m(n+k)* SAS matrix, within which *x* tags a particular phenomenon (arranged by row) taking place in the corresponding sequence (arranged by columns), whereas ○ indicates the contrary.

Since the goal of the SAS matrix is to give evidence on whether each resulting H/H phenomena is represented in the PDS* sequences, phenomena will be arranged in groups so that each of them comprises the same chain of events thereby constituting one single row. SAS matrix will also give the chance to the user of implementing shorter uncertainty input vectors containing only phenomena-related uncertainties to one specific sequence rather than applying an integral uncertainty propagation approach, i.e. using one single uncertainty input vector.

$$
\begin{array}{l}
\\
\textbf{Ph. 1}\\
\textbf{Ph. 2}\\
\textbf{Ph. 3}\\
\textbf{Ph. 4}\\
\textbf{...}\\
\textbf{Ph. m}
\end{array}
\begin{array}{c}
\begin{array}{cccccccc}
\textbf{PDS}^*\textbf{1} & \textbf{PDS}^*\textbf{2} & \textbf{PDS}^*\textbf{3} & \vdots & \textbf{PDS}^*\textbf{n} & \textbf{Det. Seq. 1} & \vdots & \textbf{Det. Seq. k}
\end{array}\\
\left(\begin{array}{cccccccc}
x & x & \circ & \dots & x & x & \dots & \circ\\
x & x & \circ & \dots & \circ & \circ & \dots & x\\
x & \circ & \circ & \dots & \circ & \circ & \dots & x\\
\circ & \circ & \circ & \circ & \circ & x & \circ & \circ\\
\circ & \circ & x & \dots & \circ & x & \dots & \circ\\
\circ & \circ & \circ & \dots & x & \circ & \dots & \circ
\end{array}\right)_{m\mathrm{x}(n+k)}
\end{array}
$$

FIG. I–3. Generic m(n+k) SAS matrix.

Simplifications to the entire list of expanded PDS sequences before directly inputting into SAS matrix will be performed by neglecting transients when clearly bounded by another:

- Transients similar in nature which differ in the availability of one particular safety system or one boundary condition for instance regarding containment isolation state, EFW availability, etc., may be neglected as long as containment output variables follow the same trend and the yielded sequence values are milder throughout the entire sequence.
- PDS sequences may also be discarded whenever the mass and energy source is not fully deposited into the containment against similar PDS sequences. For example, PWR SGTR sequences could be neglected when compared to SBLOCA's as long as the RCS pressure evolution at core damage progression and RPV failure falls under the same range; interfacing system LOCA may be neglected as the RCS discharge of mass and energy partly bypasses the containment to the attached buildings to containment, namely the main Auxiliary Building hosting safety equipment; similar PDS sequences with containment isolation failure might be neglected for the same reason.

Extreme care needs to be taken to avoid neglecting transients whose containment evolution is seemingly milder yet follow a different evolution that might likely result in unexpected worse containment environment conditions. To ensure that no particular transient is mistakenly neglected, default value simulations will always have to be carried out whenever one transient is believed to be bounded by another, checking that containment FOMs follow the same trend (even if elapsed in time) and all of them are significantly covered throughout the entire sequence by the corresponding PDS.

I-2.3. Step 3: Code applicability

The key task is to assess code capability to simulate H/H phenomena and cope with uncertainty implementation and progression.

Code capability rests on overall capacities and detailed modelling aspects related to particular phenomena. Sources of information on code modelling are found in code manuals and relating literature. Overall code capacities to cope with SAS matrix simulations require extensive support from IET benchmarking as well as SET against physical model validation incorporated into the code for augmenting the lumped-parameter approach capabilities of system codes. Checking of detailed code adequacy in terms of H/H phenomena affecting containment characterization for SAS matrix sequences requires mapping plant-specific PIRT final outcomes (after feedback from SAS matrix) with code modelling phenomena.

When demonstrating code capability in the frame of severe accident system codes, the most challenging and distinct aspect to tackle with relies on how to account for gaps in phenomena modelling and how to assess uncertainty stemming from hypotheses assumed in informing phenomena models, i.e. how to assess model uncertainty itself[5].

In the first place, it is necessary to identify the code lacking phenomena and high level hypotheses driving the severe accident evolution. With respect to the former, mapping H/H phenomena against code modelling will straightforwardly highlight lacking phenomena. With respect to the latter, driving hypotheses are not easily recognized as they unless stated explicitly need to be lifted up from modelling equations and/or code results. Nonetheless, enough sources of information are found on code manuals and code to code comparison exercises in open literature for proper identification. One set of hypotheses has to be identified for each time phase and affected component of a generic severe accident sequence evolution. Only that safety relevant, containment condition aggravating lacking phenomena and hypotheses has be taken for further consideration.

If there is a straight way to account for lacking phenomenon through direct assignment to a particular code parameter, a bias will be assigned to that code parameter to compensate for the missing phenomenon. Otherwise that bias will be applied directly on the resulting FOM histogram representations, i.e. we bias the results to compensate for the inherent systematic deviation of the code as a result of the lacking phenomenon.

To address assumed modelling hypotheses, namely assessing modelling uncertainty itself, whenever that hypothesis cannot be judged as mistaken, and as long as it systematically leads to milder conditions derived from phenomena relying on that hypothesis, additional uncertainty assignment will be implemented to account for those alternative scenarios, namely as if widening the model by including alternative hypothesis that can be judged as equally acceptable. In light of the current state of the art on severe accidents phenomena, one particular response of a code can hardly be classified as mistaken. Rather, an attempt to understanding the rationale behind code modelling phenomena and subsequent analysis of the derived link of events up to FOM turns out to be fundamental in accounting

[5] Acknowledging of state-of-the-art uncertainties in the prediction of specific phenomena, some codes have implemented several physical models enabling the user to select the one that better fits to the best of his expertise. By incorporating the different modelling options into the uncertainty quantification, the assumed hypotheses of that particular phenomenon can be accounted for.

for a potential significant source of uncertainty. One practical means to assess the impact of any particular model uncertainty on FOMs is through code to code comparison where the facing code makes use of different assumptions. The idea is to highlight systematic discrepancies in phenomenon prediction to cover them through appropriate uncertainty assignment as long as their impact is safety relevant. For instance, if sharp differences are identified on hydrogen generation prediction between two codes (the one used for the methodology application yielding milder values), and the current state of the art hardly avoids identifying which code gives more accurate results, uncertainty may be implemented in the code to account for that higher hydrogen generation and subsequent phenomena such as the exothermal energy ejected to the RCS.

I–2.4. Step 4: Uncertainty quantification

This key task comprises three fundamental steps: (i) phenomena to code-modelling conversion; (ii) sensitivity analysis; (iii) parameter uncertainty quantification and assignment.

I–2.4.1. Phenomena to code-modelling conversion

PIRT H/H phenomena uncertainty is code-dependent both in terms of value and means of implementation. Depending on the selected PIRT approach, namely on how accurately and detailed PIRT entries have been defined, an intermediate step of relating phenomena to code modelling will be required. If detailed PIRT approach has been taken, turning phenomena into modelling equations will be straightforward, whereas further decomposition analysis will likely be required in case of category phenomenon approach. System code architecture usually organizes phenomena models in interconnected subroutines paving the path to modelling identification and uncertainty assessment. A good example is provided in Ref. [I–20] where a twofold sample mapping, linking PIRT category approach phenomena to code models through associated phenomena within each class of phenomena, is presented.

When performing the connection task between phenomena and code modelling, there is a further applicable filtering process dealing with those phenomena which, according to the way the code specifically models the related thermal hydraulic processes, treats them as fully dependent phenomena, i.e. calculated as a result from other independent phenomena. This post filtering task deals exclusively with the micro scale PIRT approach where very particular questions asked to the expert panel can easily refer to phenomena whose particular code arrangement modelling makes them to be fully derived from relating phenomena addressed in other PIRT entries. Therefore, dependent phenomena have to be carefully recognized to avoid assigning twice the uncertainty of a particular issue.

I–2.4.2. Uncertainty quantification and assignment

Once modelling parameters have been identified, and in order to generate the uncertainty input vector, the support of the distribution and associated probability distribution function (*pdf*) shall be assigned for each of them. Support of the distributions and *pdfs* needs be derived from open literature, especially benchmarking exercises and code manuals provided information according to the status of knowledge. Generation of confidence intervals and suitable *pdfs* concerning information coming from experimental modelling validation if not already included in the report itself, can be found in manuals on statistics and data mining so that there is no need to be addressed in the current work.

Whenever the *pdf* is not confident enough, information theory techniques shall be applied, in particular, the maximum entropy theory establishing *pdfs* as a result of entropy maximization Ref. [I–21] as a measure of uncertainty information and constrained by certain assumptions coming from known data, i.e. a standard Lagrangian method problem (an original function the entropy to be maximized subject to functional constraints).

This approach was first introduced by Ref. [I–22] and subsequently applied by Ref. [I–23] and further on by Ref. [I–24]. For detailed mathematical grounds, the reader is referred to the mentioned works.

TABLE I–2. COLLECTION OF *pdfs* AS FOR THE MOST USUAL CASES APPLYING THE MAXIMUM ENTROPY PRINCIPLE

AVAILABLE INFORMATION	PDF NAME	PDF EQUATION ($\boldsymbol{p_k(k)}$)	NOTES
[a,b]	Uniform	$1/b - a$ if continuous $1/n$ if discrete	b>a is hereby assumed
[0,∞), μ_1	Exponential	$\lambda e^{-\lambda k}$	$\lambda = 1/\mu_1;$
(-∞,∞), μ_1, μ_2	Gaussian	$\frac{1}{\sqrt{2\Pi\mu_2}} e^{-\frac{(k-\mu_1)^2}{2\mu_2^2}}$	–
[a,b], μ_1	Truncated Exponential	$p_k(k) = (1/s)p_x[(k-r)/s];$ $p_x(x) = Ce^{\lambda x}$ for $-1<x<1$	$C = \frac{\lambda}{2sinh\lambda};$ $tanh\lambda = \frac{\lambda}{(\mu_1\lambda+1)};$ $\mu_1 = \frac{\lambda - tanh\lambda}{\lambda tanh\lambda};$ $r = (a+b)/2;$ $s = (b-a)/2;$ $k = r + sx;$
[a,b], $\mu_{1=(a+b)/2}$, μ_2	Truncated Gaussian if $\mu_2 < (b-a)^2/12;$ Uniform if $\mu_2 = (b-a)^2/12;$ Truncated Exp. if $\mu_2 > (b-a)^2/12;$	$p_k(k) = (1/s)p_x[(k-r)/s];$ $p_x(x) = D'e^{\lambda' x^2}$ for $-1<x<1$	$k = \mu_1 + sx;$ $s = (b-a)/2;$ $\mu_2 = s^2 \frac{\ln[0.5\int_{-1}^{1} dx e^{\lambda' x^2}]}{\lambda'};$ $D' = \frac{1}{\int_{-1}^{1} dx e^{\lambda' x^2}};$

Table I–2 collects the *pdfs* as for the most usual cases in terms of available information on sources of uncertainty parameters, provided the range is already known (otherwise sensitivity analysis shall compensate for untrusted support of distributions).

I–2.4.3. Sensitivity analysis

Sensitivity analysis shall be performed to eventually feed uncertainty parameters quantification upon the following reasons: (i) expert judgement or open literature discrepancy; (ii) embedded uncertainties in PDS* underlying PSA models; (iii) assessing of deterministic assumptions; and (iv) dynamic loads transferred to containment.

Whenever the applicant hesitates on the adequacy of code-provided uncertainty ranges, discrepancies are found in open literature, or raised concerns address at the actual impact of a modelling parameter on containment FOMs, specific additional simulations shall be performed resulting, if necessary, in broadening the support of the distribution. Sensitivity calculations are also recommended to identify suitable values for the support of the distributions dealing with lacking phenomena.

Deterministic assumptions on the side of conservatism coming from regulatory requirements or utility criteria might be explicitly addressed whether through additional SAS cases or through uncertainty input vector extension or adaption of the support of the distributions and/or *pdf.* For instance, this might be the case of NRC requirements mentioned dealing with hydrogen generation from cladding oxidation.

The set of uncertainty parameters shall eventually be expanded by including components and actions significant to containment evolution according to PIRT H/H phenomena whose implementation in PSA has been modelled only through best estimate values. Because uncertainty associated bands typical of PSA figures of merit would not be taken into consideration when establishing the list of PDS*s, it would be convenient to accommodate variation of these parameters along their uncertainty ranges as they would have been later included in Level 2 PSA. This is the case of human performance available times for mitigating system oriented actions, for instance, regarding the action of injecting water into the containment to directly reach the reactor cavity / pedestal floor.

Dynamic phenomena not accounted for in PIRT development may be simulated by means of sensitivity analysis even if resulting peak values might likely go far beyond component's design technical limitations to withstand such harsh environmental conditions thus not adding significant value to equipment qualification. Moreover, dynamic phenomena are usually characterized by high uncertainties in code modelling and difficulties for system codes to cope with them, in the end and in line with PSA Level 2 standard practices to deal with dynamic phenomena related release category representative sequences likely making the user imposing the phenomenon occurrence in the simulation. Table I–3. collects the major phenomena transferring dynamic loads on the containment, affected FOMs and related sequences.

Separate simulations through sensitivity analysis may be performed until capturing the dynamic-load phenomena at a containment pressure close to the mechanical failure in order to maximize the subsequent FOM peak value yet without jeopardizing the containment. The resultant peak shall be added to the corresponding FOM histogram representation.

I–2.5. Step 5: Uncertainty propagation

Once the list of key parameters is obtained, uncertainty propagation through a set of simulations by randomly varying the uncertainty input vector will be performed, whose output values of interest will help characterize the containment FOM time dependent profiles. As the uncertainty is not an assigned, fixed value to a specific code, but linked to a particular simulated evolution, uncertainty deviations from code best estimate results will be sequence-specific.

In order to choose the sample minimum size, tolerance limits as a measure of quantitative uncertainty have to be applied. Tolerance interval specifies the range of values within which it has been calculated that a stated percentage of individual values of the population lies within a specified confidence level. For each accident sequence, the one side, first-order 95/95 criterion is applied:

$$0.95 > 1 - 0.95^{n} \quad \text{(I–1)}$$

The value of n that satisfies the above inequality is 59. This is usually referred as the Wilks' formula Ref. [I–25].

Reference [I–26] generalized Wilks' methodology for a set of safety critical parameters where the size of the sample is determined by means of the following equation:

$$\beta = \sum_{j=0}^{N-p} \frac{N!}{(N-j)!j!} \gamma^{j} (1-\gamma)^{N-j} \quad \text{(I–2)}$$

Where β stands for the confidence level, N for the sample size, p for the number of safety critical parameters (i.e. FOMs) and γ for the tolerance interval. There has been a significant debate in the technical community regarding the most suitable application of order statistics, for instance, on whether the sample size for 3 critical parameters are 124 following Guba's approach rather than 59.

Even if, as suggested by Ref. [I–27] and Ref. [I–28], larger sample sizes would compensate for large first order variability (hence additional conservative burden) furthermore reducing the 5% residual value not covered by Wilks' first–order 95/95 criterion application, a sample size of 59 is recommended here following Ref. [I–29] since it is the simplest approach with a clear link to the acceptance criteria, it is fully independent on the underlying probability distribution, and it avoids increasing the number of simulations (since containment characterization relies on 5 FOMs).

Therefore, a 59 sample size will be used to perform 59 simulations ensuring that the 95 percentile is included with 95% confidence.

TABLE I–3. DYNAMIC-PHENOMENA CHALLENGING CONTAINMENT INTEGRITY

Phenomenon	Description / Cause	Affected FOMs	Related sequences	Comments
In-vessel steam explosion	Large rapid energy transfer to RCS and indirectly to containment through core relocation to lower plenum	P	Low-pressure RCS scenarios	Most of system codes do not reproduce steam explosions. Probability of in-vessel steam explosion has been lately neglected according to the status of knowledge.
Ex-vessel steam explosion	Large rapid energy transfer to containment through corium relocation to cavity / pedestal	P	Large water pool presented in the cavity / pedestal at RPV failure	Most of system codes do not reproduce steam explosions.
High Pressure Melt Ejection (HPME)	Combination of corium entrainment, rapid oxidation and direct heat transfer to containment atmosphere	P, T	High pressure RCS scenarios at RPV failure	Related phenomena already accounted for in PIRT.
H2 generation excursion	Large Zr oxidation and exothermal energy ejected to RCS and indirectly to containment with intact core geometry	P, T, [H2]	Reflooding of degraded core	Very hardly to reach peak values on containment FOMs; nonetheless, exothermal energy can be even higher than decay energy and temporary drive containment pressure evolution. Hydrogen peak values likely reached from ex-vessel MCCI.
Flammable gas explosion	Ex-vessel combustible gas combustion and explosion (flame acceleration and DDT)	P, T	Cliff-edge effect: all sequences susceptible to undergo flammable gas explosion, in particular those leading to containment de-inertization as a result of containment cooling whether because of AC power recovery with later actuation of CCS / sprays, containment flooding (after corium quenching) or passive cooling.	Most of system codes do not reproduce hydrogen flame acceleration and DDT; nonetheless, hydrogen explosion simulation is enough for our purposes since containment failure is not an issue.

Though Wilks formula is usually applied to populations, whose elements are spot values, note that as for each time instant a 59 case sample is yielded, the 95/95 criterion is suitably met in time dependent evolutions such as those addressed in I&C survivability tests. Therefore, and for each SAS matrix sequence, the resultant composite bounding sequence, made up of the highest ordered value at each time, will meet the 95/95 criterion. It is worth noting that the goodness of the method requires the independency of the 59 input vectors, which can be checked by making use of the variance-covariance matrix for instance through Pearson or Spearman correlation coefficients both of which have been proven to be adequate, robust indicators. Ready-made functions are available in most of the statistical software programs nowadays used to generate random samples.

I–2.6. Step 6: Output data post-processing

A collection of 59 simulations to specifically meet with the one-sided 95/95 tolerance interval criterion has to be performed for each sequence included in the SAS matrix once fed back from sensitivity analysis. For each containment parameter of interest, the composite bounding sequence has to be compared against each other to get the composite-composite sequence just by reapplying the same process, hence transferring the 95/95 tolerance interval criterion accomplishment from the single case to the full set of sequences.

Once the twice-composite bounding time-evolution FOM has been generated, and since equipment service life expectancy deals with bounding parameter values along limited periods of time, the resultant plot has to be transformed into a histogram representation ordered by columns, where the height of the column represents the largest value within a certain range of values and the width represents the elapsed time during which the containment environment parameter value is found to be within that range.

Bias to account for identified gaps in code modelling phenomena may be applied if no means has been found to directly compensate for their aggravating effect on FOMs. Bias assignment may only affect that specific band within the entire histogram representing the time phase where the code shortcoming is found. For instance, if the code is not provided with a specific set of iodine chemistry related reactions leading to systematic under prediction of radiation levels, the applicant's user may proceed to quantify that gap from open literature and implement it over the affected time window through which those reactions take place as subjected to meeting certain conditions such as specific containment temperature activation, or as long as containment compartment has not been submerged, etc. Consequences of dynamic load related excursions on FOM values not already considered in the uncertainty assessment may be analysed and added to the histogram representation.

Once bias has been assigned, the resultant updating histogram will then meet the goal that equipment qualified against this parameter versus elapsed time values will at least bear the highest 95th percentile with 95% confidence.

I–3. APPLICATION

Simplified application of the methodology is demonstrated hereafter on generic large-dry containment, dry-cavity configuration [6], 4-loop, 3565 MW(th) PWR, and on generic Mark-III containment, 3579 MW(th) BWR/6. For both reactors, the following assumptions on mitigating systems performance —not included in Level 2 PSA models hence in interfacing level PDSs— apply:

- Containment venting system performs well throughout the transients.
- No active or passive combustible gas recombination device is considered.
- Reactor cavity (PWR) / containment flooding (BWR) is available.
- No credit given for ex-vessel IVMR except when explicitly stated.

To simplify the methodology development, all the steps suitable of common application for both plants will be performed once, so that rather than treating each application separately, the different steps of the methodology will be initially carried out for both reactors and only when necessary afterwards specifically treated.

[6] 'Dry-cavity configuration' means that the maximum containment flooding level, i.e. that resulting from discharging the RWST and accumulators in the containment through the RPV, does not reach the lowest reactor cavity opening. This type of configuration is also extended to the generic BWR considered in the application.

I–3.1. H/H phenomena elicitation process

First task consists of reviewing and selecting the most appropriate open literature PIRT. The EURSAFE PIRT will be here used for both applications as a starting point for selecting H/H phenomena, subsequently facilitating the conversion process from phenomena to code modelling parameters as the irreducible nature of PIRT entries will allow establishing a straightforward correspondence thereby without further need of additional decomposition analysis.

The starting point is the 106 out of 916 phenomena classified under the H/H category and gathered in Table 1 in Ref. [I–27]. These PIRT outcomes will then be filtered through three different criteria, namely neglecting non-applicable phenomena whether because of NPP specific features (both in terms of SSCs and NPP design), because of not affecting significantly any containment FOM or not challenging the containment, or because of dealing with dynamic loads transferred to the containment such as hydrogen DDT or PARs ignition. First set of entries on Table I–4 gathers the 39 out of 106 filtered phenomena affecting containment evolution significantly and challengingly.

I–3.1.1. In-house reassessment of PIRT filtered outcomes

The first overarching remarkable aspect of EURSAFE PIRT relies on the relatively minor importance played by in-vessel-related phenomena on the containment importance pillar. EURSAFE PIRT containment FOM H/H phenomena come almost exclusively from the ex-vessel phase. Nevertheless, large uncertainties characterizing the in-vessel phenomena dealing with core degradation and lower plenum relocation will in the end dramatically modify subsequent sequence evolution, mainly through direct impact on corium discharge to the containment, as emphasized in the analysis carried out in Ref. [I–30][7].

Second set of entries on Table I–4 report additional phenomena focused on in-vessel behaviour judged to ultimately impact on containment characterization and lacking in the preliminary PIRT. Identified gaps dealing with plant-specific phenomena, as well as H/H phenomena derived from PIRT specific drawbacks on lacking of post-Fukushima aspects of concern, have also been included in the table with the help of USDOE PIRT.

I–3.2. Severe Accident Sequence matrix

Since the current exercise aims at providing insights on the most significant issues concerning the methodology application, the provided lists of PDSs are not exhaustive. PWR and BWR SAS matrixes are provided in Figs I–4 and I–5. One can notice that second row of PWR SAS matrix related to reflooding is not reflected in any of the PDS* sequences. The user might then be aware on the convenience of including sequences affected by late core damage injection and subsequently feedback the SAS matrix.

HPME related phenomena may also be discarded as a result of imposing RPV depressurization hence leading to neglecting one set of very significant phenomena having large impact on containment characterization. SAS matrix therefore helps highlight whether the selected sequences reflect the entire list of most important uncertainty phenomena affecting our results. As for our application and since we are making use of one single PIRT for simplicity's sake, and the BWR SAS matrix does indeed include degraded core reflooding on PDS* 4, phenomena 6 and 7 of Table I–4 will be taken into account in the PWR case.

---

[7] One might even think of a potential misunderstanding on whether members of EURSAFE PIRT elicitation expert panel have interpreted *primary circuit*, *containment safety*, and *source term* as sequence-phase surrogates.

$$
\begin{array}{l}
\\
\textbf{Ph. 1-5 + 40-46} \\
\textbf{Ph. 6-7} \\
\textbf{Ph. 8 + 47-49} \\
\textbf{Ph. 9-12} \\
\textbf{Ph. 13-18} \\
\textbf{Ph. 19-23} \\
\textbf{Ph. 24} \\
\textbf{Ph. 25-26} \\
\textbf{Ph. 27} \\
\textbf{Ph. 28} \\
\textbf{Ph. 29-39} \\
\textbf{Ph. 50} \\
\textbf{Ph. 51} \\
\textbf{Ph. 52} \\
\textbf{Ph. 53}
\end{array}
\begin{array}{c}
\begin{array}{ccccccc}
\textbf{PDS' 1} & \textbf{PDS' 2} & \textbf{PDS' 3} & \textbf{PDS' 4} & \textbf{PDS' 5} & \textbf{PDS' 6} & \textbf{PDS' 7}
\end{array} \\
\left(\begin{array}{ccccccc}
x & x & x & x & x & x & x \\
o & o & o & o & o & o & o \\
x & x & x & x & x & x & x \\
x & x & o & x & x & x & x \\
o & o & o & x & o & o & o \\
x & x & o & x & x & x & x \\
o/x & o/x & o & o/x & o/x & o/x & o/x \\
x & x & o & x & x & x & x \\
x & o & x & o & o & x & x \\
x & x & x & x & x & x & x \\
x & x & x & x & x & x & x \\
x & x & x & x & x & x & x \\
o & o & x & o & o & x & o \\
o & o & o & x & o & o & o \\
o & o & x & o & o & o & o
\end{array}\right)_{15x7}
\end{array}
$$

FIG. I–4. PWR SAS matrix.

$$
\begin{array}{l}
\\
\textbf{Ph. 1-5 + 40-46} \\
\textbf{Ph. 6-7} \\
\textbf{Ph. 8 + 47-49} \\
\textbf{Ph. 9-12} \\
\textbf{Ph. 13-18} \\
\textbf{Ph. 19-23} \\
\textbf{Ph. 24} \\
\textbf{Ph. 25-26} \\
\textbf{Ph. 27} \\
\textbf{Ph. 28} \\
\textbf{Ph. 29-39} \\
\textbf{Ph. 50} \\
\textbf{Ph. 51} \\
\textbf{Ph. 52} \\
\textbf{Ph. 53}
\end{array}
\begin{array}{c}
\begin{array}{ccccc}
\textbf{PDS' 1} & \textbf{PDS' 2} & \textbf{PDS' 3} & \textbf{PDS' 4} & \textbf{PDS' 5}
\end{array} \\
\left(\begin{array}{ccccc}
x & x & x & x & x \\
o & o & o & x & o \\
x & x & x & o/x & x \\
x & x & x & o/x & o/x \\
o & x & o & o & o \\
x & x & x & o/x & o/x \\
o/x & o/x & o/x & o/x & o/x \\
x & x & x & o/x & o/x \\
o & o & x & o & o \\
x & x & x & x & x \\
x & x & x & x & x \\
x & x & x & x & x \\
o & o & o & o & o \\
o & x & o & o & o \\
- & - & - & - & -
\end{array}\right)_{15x5}
\end{array}
$$

FIG. I–5. BWR SAS matrix.

TABLE I–4. FILTERED PHENOMENA AFFECTING CONTAINMENT EVOLUTION SIGNIFICANTLY AND CHALLENGINGLY

H/H challenging phenomena impacting containment characterization (cont.)		
Ph. No.	Ph. phase & comp.	Phenomenon
EURSAFE PIRT H/H filtered phenomena		
1	Core degradation / Core	Oxidation by air
2		FP releases in highly oxidized fuel
3		FP releases (In and Cd) from AIC control rods
4	Core degradation / RPV	Chemical reactions between vapour species
5		Revolatilization
6	Reflooding / Core and lower head	Degraded core interaction with water (FCI)
7		Oxidation of Zr mixtures and H_2 production

H/H challenging phenomena impacting containment characterization (cont.)		
Ph. No.	Ph. phase & comp.	Phenomenon
8	Lower head behaviour / Lower head	Vaporization of pool materials
9	RPV failure / Lower head	RPV mechanical failure
10		Mass flow rate and pouring history
11	RPV failure / RPV-cavity interface	Corium composition and physical state
12		Breach location and flow path
13	HPME / RPV-containment	Corium / steam two-phase jet
14		Corium entrainment outside the RPV
15		Corium particles generation from corium pool
16	HPME / reactor cavity	Corium particles generation from jet
17		Corium entrainment outside the cavity
18		Corium particles trapping
19		Inner and outer heat transfer
20		Layer configuration
21		AH, liquidus and solidus temperatures
22	MCCI / Debris bed and melt cake	Melt ejection
23		Crust anchorage
24		Jet interaction and breakup between corium and water
25		Top water layer - corium heat transfer
26		Water ingression
27		Jet / plume gas interaction
28		Thermal and mass stratification
29		Aerosol retention
30		Iodine adsorption/desorption to/from surfaces
31		Heterogeneous organic iodide formation in painted walls
32		Organic iodide radiolytic destruction
33	Long term / Containment	Volatile iodine trapping in condensed water
34		Iodine mass transfer between gas and liquid phases
35		Iodine chemistry influenced by boundary conditions
36		Iodine species oxidation-reduction reactions
37		Homogeneous organic iodide formation in liquid phases by radiolytic decomposition of organic material
38		Organic iodide formation in submerged painter walls
39	Long term / Containment	Organic iodide release in dry pools of water
Additional phenomena from reassessment		
Phenomena not considered in EURSAFE PIRT		
40		Core relocation
41		Zr oxidation
42		Zr melt breakout temperature
43	Core degradation / Core	Fuel rod collapse temperature
44		Fuel and control rods melt temperature
45		Gas flow through degraded fuel rods/channels
46		Remaining degraded core at core position
47	Core degradation / Lower head	Heat transfer in in-vessel corium pools
48	Core degradation / Core and lower head	Natural circulation
49		Jet characterization
Identified plant-specific gaps and EURSAFE PIRT specific drawbacks		
50	Long term / Reactor cavity/containment flooding	Time to reach the reactor cavity / drywell floor
51	Entire transient / RCIC/EFW	Operating envelope in severe accident conditions
52	In-vessel / PORVs/SRVs	Operating envelope in severe accident conditions for primary PORVs and SRVs
53	In-vessel / RPV	Creep rupture: hot leg (PWR), SG (PWR), MSL (BWR)

I–3.2.1. Generic PWR sequences

After application of arguments stated in above Section I–2.2.2, the list of PDS sequences taken from generic, simplified PSA application is the following:

— Cold leg SBLOCA. Failure of EFW and F&B. PORV (PZR) manual opening at 649 °C. Failure of Containment Spray System (CSS) and Containment Cooling Systems (CCS).
— Turbine and Reactor trip. Failure of EFW and F&B. PORV (PZR) manual opening at 649 °C. CSS is unavailable and CCS is available.
— Loss of AC safeguards buses. No LOCA through the RCP seals. EFW turbo-pump available. External power recovery after 649 °C. CCS and CSS available. IVMR success.
— Loss of AC safeguards buses without LOCA through the RCP seals. EFW turbine-driven pump unavailable.
— Cold leg SBLOCA without ECCS and availability of EFW in RCS depressurization mode; Containment Fan Coolers and sprays are available.
— Turbine and Reactor Trip. Failure of EFW and F&B. Containment Fan Coolers available but Containment Sprays failed. PORV(PZR) are manually opened half an hour after the onset of core damage. LPIS unavailable.

MBLOCA with HPIS in injection mode. Containment bypass failure through the RWST (failure in HPIS recirculation switch).

I–3.2.2. Generic BWR sequences

The equivalent list for the BWR application is the following:

— MSIV closure. HPCI available (until reaching HCTL limit) and RHR unavailable. Manual ADS at core damage.
— SBO with manual ADS unavailable and HPCI available (until reaching HCTL limit). External power recovery after core damage. All ECCS systems (HPCI, LPCI, CS) are available.
— MBLOCA with manual ADS at core damage and failure of HPCI and LPCI.
— LFW. Loss of high pressure ECCS and RCIC systems and manual ADS at core damage.

I–3.3. Code applicability

I–3.3.1. Overall code capability

The selected severe-accident system code and version for both applications is MAAP5.02 Ref. [I–31]. The Modular Accident Analysis Program (MAAP) is an integral system analysis code for assessing off-normal transients up to containment failure and source term release. MAAP covers the full spectrum of severe accident phenomena during the early in-vessel, ex-vessel, and late in-vessel phase thus including core heatup, relocation, Reactor Pressure Vessel (RPV) failure, corium ejection and Direct Containment Heating (DCH), Molten Corium Concrete Interaction (MCCI), steam explosion phenomena, H2 and fission products behaviour within the RCS and the containment, containment (and auxiliary buildings) thermal hydraulics and failure mechanisms, and source term release characterization. MAAP code has been tailored for many types of Nuclear Power Plants among which PWRs and BWRs and has been extensively benchmarked throughout all its versions against a wide number of SETs and IETs.

I-3.3.2. Detailed code capabilities

Detailed code capability assessment is structured first by means of mapping PIRT phenomena against corresponding code modelling. After analysing entries in Table I–4, no lacking phenomena is found in MAAP5.02 with the low significant exception of phenomenon 39. Nonetheless, phenomena not considered in EURSAFE PIRT on core degradation hence subsequently added follows a more generic approach so that some PIRT entries do not point at one single phenomenon but at an entire category. In particularly this is the case of 'core relocation', 'Zr oxidation', and 'heat transfer in in-vessel corium pools'. Since these sets of phenomena do not specify which of their belonging sub-phenomena are important and lack of knowledge to containment FOMs, a later decomposition analysis has to be performed, on one hand, to ensure that no missing phenomena is found, and on the other, to complete the mapping activity between PIRT outputs and code uncertainty modelling parameters.

According to Ref. [I–18], 95 out of 299 H/H phenomena are not included in versions earlier than 5.01. From these 95 lacking phenomena, 25 are considered inappropriate and 8 may be taken into account when possible. The remaining 62 are already being taken care of in planned enhancements after version 5.03. These phenomena all deal with a more spatially distributed heat source and augmented heat transfer ejected to the RCS through more realistic consideration of downward and sideward corium flow paths, likely leading to an increase of the relocation span, increasing hydrogen generation, decreasing the heat density concentrated in the corium once relocated to the lower head and reactor cavity / pedestal, subsequently decreasing the corium quenching time thus mitigating MCCI. Therefore, the 62 lacking phenomena are expected to lead to a less severe, more realistic representation of corium distribution heat source in- and ex-vessel[8].

A degree of lower severity is limited and intended in terms of a more gradual in-vessel and ex-vessel heat transfer and distribution process, whereas hydrogen generation will be aggravated since larger interfacing areas will favour contact between corium and steam. Similar arguments apply to in-vessel phase peak temperatures, since less hydrogen generation means less exothermal energy taken out from Zr and ejected to the RCS. On the other hand, the 25 lacking phenomena do not affect any of the corresponding PIRT set-of-phenomena outcomes whether because of their very low probability of occurrence, because of not impacting significantly on containment characterization, or because of being rejected by the filtering process stated in Section I–2.2.2. Therefore, lacking phenomena on the side of safety will affect FOM characterization through yielding lower hydrogen generation values during the in-vessel phase and lower peak temperatures within the RPV whose impact on FOM characterization will be addressed through *FAOX* code parameter support of the distribution (see related comments in Table I–5).

Large differences in related in-vessel hydrogen generation were as much as near 300% as calculated by MELCOR compared to MAAP (530 kg versus 190 kg at 6 hours after the initiating event set on March 11, 2011 at 14:46 JST), mostly stemming from debris build-up, candling and configuration in the core region and relocation to the lower plenum, debris configuration in the lower head, and associated heat transfers. Aside from directly affecting hydrogen concentration in the containment, this lower hydrogen generation turned into lower containment pressure of 100 kPa according to the ideal gas law. Therefore, MAAP containment characterization exhibits milder behaviour when compared to MELCOR along the in-vessel phase, since the heat source both decay heat and exothermal heat coming from Zr oxidation tends to remain within the corium rather than being transferred to the RCS (which is also the reason why the corium ejected from the lower head at RPV failure presents an overheated state with higher temperature differences of even 500 K). However, higher RPV gas temperatures predicted by MELCOR upon significant core degradation and hydrogen generation will yet not affect containment temperature since differences in containment pressure as predicted by both codes have been adequately referred to only differences in hydrogen generation[9]. On the contrary, MAAP harsher prediction of ex-vessel corium relocation will have a strong impact on yielding higher containment FOM values. However, as it lies on the side of safety, no uncertainty will be applied here.

Therefore, uncertainty to account for an increase in the amount of hydrogen generated in-vessel has to be included in code calculations to account for aggravating scenarios stemming from alternative modelling hypothesis. A parameter distribution setting up a minimum hydrogen generation rate will be considered with a lower bound distribution support equal to 0, thereby also including no artificial restriction on default hydrogen calculation by the code. The selected way to properly handle this issue

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[8] Once these gaps will be corrected, MAAP in-vessel evolution simulation will likely get closer to MELCOR results, as described and referred to in Ref. [I–32] when pointing out at 5.03 code version.

[9] Notable divergences may instead stem from creep ruptures by high temperature gases leading to main steam line / hot leg rupture hence advanced RPV depressurization, which will be adequately taken into account as reflected in Table I–5 on code uncertainty parameters.

has consisted of implementing uncertainty through a code internal variable called *WH2MIN* which specifies a minimum hydrogen generation rate. Further analysis on uncertainty modelling to keep assessing confidence on the assumed hypotheses and including alternative scenarios has to be conducted for the rest of the accident sequence evolution through a comprehensive review of the driving modelling hypotheses (ex-vessel phase mainly dealing with issues of corium entrainment outside the cavity / pedestal; MCCI and FCI models; FP transport and distribution; and containment thermal hydraulics).

In assessing modelling hypotheses and to estimate the magnitude of the ultimate impact on containment FOMs, it might be highly practical to look at code to code exercise comparisons. Two of the most extensively validated and applied in integral system response severe accident system codes, MAAP and MELCOR, were recently compared (EPRI, 2014) in simulating Fukushima Unit 1 from the onset of core damage up to RPV lower head failure as a result of strong discrepancies in providing initial conditions for ex-vessel core debris relocation and quenching calculation with mechanistic detailed codes as METLSPREAD and CORQUENCH (EPRI, 2014). Despite both codes being benchmarked against similar fuel melt experiments (and likely as a consequence of different assumptions on scaling extrapolation), sharp differences were found in corium characterization relocated ex-vessel. Far reaching divergences [10] in MAAP estimating much more severe rapid overheated debris relocation to pedestal and drywell while MELCOR calculating largely solid, low temperature pouring debris spanning for slightly more than an hour, consequently resulted in significant divergences in containment debris location and evolution[11].

TABLE I–5. MAAP 5.0.2 KEY UNCERTAINTY PARAMETERS FOR UNCERTAINTY PROPAGATION

Parameter	**Description**	**Min**	**Max**	***pdf***	**H/H ph. (cont.)**
	In-vessel Fission Product Release				
FPRAT	Fission product release correlation [-]	-7	7	Discrete Uniform	2,3
FTEREL	Tellurium release flag ("1" if tellurium is released in-vessel, or "0" if it is assumed to be totally bound up with Zircaloy) [-]	0	1	Discrete Uniform	2
ISICRELEASE	Flag to select new EDF control rod material release correlations ("1" takes into account of activity coefficient considering several chemical species in the liquid phase of Ag-In-Cd) [-]	0	1	Discrete Uniform	3
FCSIVP	Multiplier to the vapor pressures of CsI for vapor/aerosol equilibrium [-]	-100	100	Uniform	5
FVPREV	Multiplier to the vapor pressures of CsI for revaporization calculations [-]	0.01	2.0	Uniform	5
FCSHVP	Multiplier to the vapor pressures of CsOH for both vapor/aerosol and vapor/surface equilibrium [-]	0.01	1.0	Uniform	5
	In-Core H2 Generation				
IOXIDAIRLAW	Integer flag to choose oxidation correlation	0	1	Discrete Uniform	1
WH2MIN	Extend in-core metal-water reaction until 100% active fuel-clad is achieved [kg/s]	0	0.2^1	Uniform	*Model Uncert.*
FAOX	Multiplier for cladding outside oxidation surface area [-]	2^2	10^2	Truncated Exponential $\mu=2^2$	41; *Lacking Phenomenon*

[10] As indicated in Ref. [I–30], the reported results and types of differences are typical of the two codes hence can be extrapolated to other scenarios and accident sequences.

[11] Less severe corium relocation related phenomena as simulated by MELCOR when compared to MAAP.

Parameter	Description	Min	Max	*pdf*	H/H ph. (cont.)
IOXIDE	Zr oxidation model: MATPRO, IDCOR, Urbanic or Prater oxidation model [-]	0	1	Discrete Uniform	41
TCLMAX	Clad rupture T [K]	2200	2700	Truncated Gaussian	42
FGBYPA	"0" to allow steam flow in blocked channel and be available for oxidation [-]	1	0	Discrete Uniform	41
EPSCUT	Porosity below which the node is fully blocked [-]	0.0	0.25	Uniform	41
EPSCU2	Porosity below which the collapsed core node is fully blocked [-]	0.001	0.35	Uniform	41
	Core Melt Progression				
LMCOL0 LMCOL1 LMCOL2 LMCOL3	Collapse criteria parameter when no core node surrounding the particular core node has collapsed [-]	48	54	Truncated Exponential μ=53[3]	43
FCRDR	Fraction of the original core mass below which the remaining core is dumped into the lower plenum [-]	0	1	Uniform	46
FCHFCR	Critical heat flux Kutateladze number for In Vessel corium pools [-]	0.0036	0.3	Uniform	47
FUPOOL FDPOOL	Convective heat transfer coefficient multipliers for heat transfer between two crust core nodes, two molten pool core nodes, and a crust and a molten pool node [-]	0.5[4]	2	Truncated Gaussian	47
FFRICX	Gas cross-flow friction coefficient in the core for the in-vessel natural circulation model [-]	0.0	1.0	Truncated Exponential	45, 48
FZORUP	Minimum fraction of Zr that must be oxidized to keep the cladding intact if the cladding is at a user-specified rupture temperature [-]	0.5[5]	0.9[5]	Truncated Exponential	43
FACT	Multiplier to reduce the hydraulic diameter and flow area when an intact fuel node collapses [-]	0.1	1.0	Uniform	45
TSPFAL	Core support plate failure temperature [K]	1500[5]	1700[5]	Truncated Gaussian	40
FPEEL	Fraction of the ZrO2 layer peeled off during reflooding [-]	0.01	1	Uniform	7
EPSPB	Assumed porosity of the particulate debris bed in the vessel lower head [-]	0.26	0.53	Truncated Exponential	47
FEMISD	Emissivity for debris that is within RPV lower head Penetrations [-]	0.4[5]	1	Uniform	47
FEMISP	Emissivity for the RPV lower head penetrations [-]	0.4[5]	1	Truncated Exponential	47
FQUEN	Multiplier to the In Vessel flat plate critical heat flux for lower head debris bed quenching by overlying water [-]	0.036[5]	1	Uniform	6
ECREPF	Strain failure for vessel ductile material [-]	0.00	1.0	Uniform	9
ECREPP	Maximum penetration (or CRD tubes) weld strain at failure [-]	0.001	1	Uniform	9
FHTGAP	Heat transfer effectiveness in the crust/lower head wall Gap [-]	0.1[5]	0.2[5]	Truncated Exponential μ=0.17[5]	9
IQDPB	Flag for the heat transfer model from debris bed particulates to the water in the lower plenum [-]	1	2	Discrete uniform	47
XDJETO	Initial diameter of a corium jet when it hits the water surface in the vessel lower plenum [m]	0.01	0.2[4]	Uniform	49
XGAP0	Initial size of the gap between the debris and the inner surface of penetrations in the lower head [m]	1e-6	3e-4[5]	Truncated Exponential μ=0.1e-4[5]	47

Parameter	Description	Min	Max	*pdf*	H/H ph. (cont.)
XGAPLH	Initial size of the gap between the debris and the lower head steel wall [m]	1e-6	2e-5[5]	Truncated Exponential μ=1.0e-5[5]	47
FASSOXID[6]	Multiplier for steel oxidation area in the core for steel from melted or collapsed top non-fuel nodes or upper plenum internals [-]	0.5[5]	2[5]	Truncated Exponential	41
VFCRCO	Porosity of a collapsed core region [-]	0.3[5]	0.5	Truncated Gaussian	47
FGPOOL	Geometric factor used to define the shape (height) of the in-core molten pool [-]	0.5	1	Truncated Exponential	47
XLFALS	Width of the failure opening when the in-core molten pool side crust has failed such that sideward relocation to the lower head is possible [m]	0.01	0.1[5]	Truncated Exponential μ=0.03[5]	40
FMOVE	Parameter that controls the relocation of solid U-Zr-O material embedded in liquid U-Zr-O [-]	1	5	Uniform	40
FOXBJ	Multiplier for the particulated debris oxidation reaction Fraction [-]	0.5[4]	1	Uniform	7
VFENT	Void fraction of steam in the debris jet entrainment interaction zone [-]	0.1	0.3	Truncated Exponential	40
TEUBS	Melting temperature for control blades and fuel cans [K]	300	2500	Truncated Gaussian	44
FRACAN	Minimum fraction of fuel can that must be dissolved before the fuel can ruptures [-]	0	0.5	Uniform	43
TEU	Core node eutectic melting temperature [K]	2100	2800	Truncated Exponential	44
	Hot Leg Creep Rupture & RPV loss of isolation				
IHTHLG[6]	"0" to allow heat transfer between the counter-current gas flow streams in the horizontal hot leg [-]	0	1	Discrete uniform	53
FAOUT[6]	Fraction of S/G tubes carrying "out" flow in the hot leg natural circulation model [-]	0.1	0.5	Uniform	53
EWLHL[6]	Emissivity of hot leg walls for heat transfer between hot leg and counter-current gas flow [-]	0.4	0.99	Uniform	53
EG	Emissivity of gas [-]	0.4	0.99	Uniform	53
PORV / SRV	Probability of sticking open after *n* cycles[7]	12[7]	703[7]	$[1-(1-p_{so})^n]$[7]	52
	H2 Generation During Melt Relocation to Lower Plenum				
ENT0	Jet entrainment coefficient for a corium jet in lower plenum [-]	0.025	0.06	Truncated Exponential	41
FDDP	Multiplier for jet entrainment particle diameter [-]	0.5	0.75	Truncated Exponential	41
VFENT	Void fraction of steam in the debris jet entrainment interaction zone [-]	0.1	0.3	Truncated Exponential	41
	Vessel Failure				
TEUMISCGAP	Miscibility gap temperature of U-Zr-SS-O system [K]	-25%[8]	+25%[8]	Uniform	9
MMAXSSOXIDE	Maximum amount of stainless steel that may participate in the formation of the heavy metal layer [kg]	-25%[8]	+25%[8]	Uniform	9
EMETALAYER	Emissivity of the metal layer [-]	-25%[8]	+25%[8]	Uniform	9
IEXFMOD	"1" to invoke extensive vessel failure following initial vessel failure [-]	0	1	Discrete Uniform	9

Parameter	Description	Min	Max	*pdf*	H/H ph. (cont.)
IXPSL	"0" to drain the lower crust with the rest of the debris after extensive vessel failure [-] "1" to allow crust to stick to the wall	0	1	Discrete Uniform	9
XLVP0	Initial length of the crack for the vessel lower head creep rupture [m]	0.005	2.0	Uniform	9
XWIDVP0	Initial width of the crack for the vessel lower head creep rupture [m]	0.001	0.25	Uniform	9
FDAMLH	Lower head node damage fraction for extensive failure [-]	0	1	Uniform	9
		DCH			
FENTRC	DCH model [-]: = 0, no DCH = 1, DCH1 model = 2, DCH 2 model = 3, DCH 3 model	0	3	Discrete Uniform	Assumed Hypotheses
TTENTR	Corium debris entrainment time constant [s]	0.1	10.0	Uniform	49
FKUTA	Kutateladze coefficient used in DCH for the minimum gas velocity required for debris entrainment [-]	2.46	3.7	Uniform	13
FOXDCH	Fraction of metal in the entrained debris that would be oxidized, -1 for code calculated value [-]	0	1	Uniform	13
FBNDCH	Hydrogen jet burn completeness [-]	0.8	2.0	Uniform	49
FWEBER	Weber number used for determining the diameter of entrained particles [-]	0.0	100.0	Truncated Exponential	16, 49
FPDIF	Diffusivity of fission products migrating through molten material [m^2/s]	1E-13	1E-10	Truncated Exponential	49
FDENTR[9]	De-entrainment efficiency, -1 for code calculated value [-]	-1	1.0	Uniform	18
FENTR[9]	Debris and water entrainment multiplier out of the cavity [-]	0.2	100	Truncated Exponential $\mu = 10\ \mu_{plant\text{-}specific}$	17
		MCCI			
FHTPB	Multiplier to the nucleate boiling heat flux from particle bed to water [-]	0	1	Uniform	20
HTCMCR HTCMCS HTCMCU	Nominal downward, sideward and upward heat transfer coefficient for convective heat transfer from molten corium to the lower crust for corium-concrete interaction calculations [W/m^2-C]	2000[4]	5000[4]	Uniform	19
ENT0RB	Coefficient in Ricou-Spalding entrainment correlation for the off-gas entraining corium process [-]	0.064[4]	0.12[4]	Uniform	22
ECM	Emissivity of corium surfaces [-]	0.7	0.99	Uniform	19
HTFB	Coefficient for film boiling heat transfer from corium to an overlying pool [W/m^2-C]	100	400	Uniform	25
FGCRXS	Fraction of off-gas from sideward corium-concrete interactions that enters the molten corium pool [-]	0	1	Truncated Exponential	19
IPBRB	This parameter controls whether or not particle bed is formed on top of corium pool when corium jet is relocated from the vessel into the water pool in the reactor cavity [-]	0	1	Uniform	16, 20
XDENTRB	Default diameter of particles entrained by off-gas when melt eruption model is activated [m]	0.002[4]	0.01	Uniform	22
IKCMOXIDE	If activated, oxide thermal conductivity is used to calculate heat transfer rate from corium crust to water using Epstein's water ingression model [-]	0	1	Discrete Uniform	26

Parameter	Description	Min	Max	pdf	H/H ph. (cont.)
FIWNGS	Modeling parameter in the mechanistic dryout heat flux model (water ingression model) [-]	252[4]	308[4]	Uniform	20, 26
ENT0C	Corium jet entrainment coefficient [-]	0.025	0.06	Uniform	24
FCHF[11]	Critical heat flux Kutateladze number for containment corium pools [-]	0.0036	0.3	Truncated Exponential μ= $\mu_{plant-specific}$[12]	19
TSOLSO	Steel oxide solidus temperature [K]	1400	2000	Truncated Gaussian	21
TLIQSO	Steel oxide liquidus temperature [K]	1500	2100	Truncated Gaussian	21
TSCS	Steel solidus temperature [K]	1500	1900	Truncated Gaussian	21
TLCS	Steel liquidus temperature [K]	1550	2000	Truncated Gaussian	21
TFE2ML	Fe2M melting point, where M represents the mixture of other metals [K]	1500	2200	Truncated Gaussian	21
TMMS	Solidus temperature of other metals	1400	2300	Truncated Gaussian	21
TMML	Liqidus temperature of other metals	1500	2400	Truncated Gaussian	21
FNE1	Fe mole fraction at the Fe-rich eutectic point [-]	0.05	0.3	Uniform	21
FNE2	Fe mole fraction at the Fe-lean eutectic point [-]	0.35	0.95	Uniform	21
TE1	Fe-rich eutectic temperature [K]	1000	1650	Truncated Exponential	21
TE2	Metal-rich eutectic temperature [K]	1000	1750	Truncated Exponential	21
FGCSSR	Critical H2 mole fraction below which there is no Fe-steam reaction [-]	0.5	0.75	Truncated Exponential	21
IMLTERP	Flag that activates melt eruption model due to off-gas from concrete ablation [-]	0	1	Discrete Uniform	15
CKABA	Dissociation constant of H3BO3 in water [kg-mole/m3]	1.e-13	1.e-7	Truncated Exponential	35
CHNO3	Mass of HNO3 generated by 1 joule of energy from irradiation of water and air [Kg-mole]	1.e-16	1.e-10	Truncated Exponential	35
FCSOPH	Mass fraction of CsOH that is chemically active as a strong base (high pH) in an aqueous phase in containment sumps [-]	0	1	Truncated Exponential	35
K026/K026B	Kinetic rate constant for Ph.30 related reactions [l/s]	-25%[8]	+25%[8]	Uniform	30
K034/K034B	Kinetic rate constant for Ph.31 related reactions [l/s]	-25%[8]	+25%[8]	Uniform	31
K011/K011B	Kinetic rate constant for Ph.32 related reactions [l/s]	-25%[8]	+25%[8]	Uniform	32
K033/K033B	Kinetic rate constant for Ph.34 related reactions [l/s]	-25%[8]	+25%[8]	Uniform	34
K034/K034B	Kinetic rate constant for Ph.36 related reactions [l/s]	-25%[8]	+25%[8]	Uniform	36
K07/K07B	Kinetic rate constant for Ph.37 related reactions [l/s]	-25%[8]	+25%[8]	Uniform	37
K15/K15B	Kinetic rate constant for Ph.38 related reactions [l/s]	-25%[8]	+25%[8]	Uniform	38
EXP15/EXP15B	Coefficient to the pH for the composite rate constant in Ph. 38 related reactions [-]	-25%[8]	+25%[8]	Uniform	38
	PSA best estimate values				
T(flooding)	Cavity / drywell flooding time to reach the floor [s]	0	10800	Uniform	BE PSA value

[1] Average in-vessel hydrogen rates with intact core geometry taken from Ref. [I–32].

[2] FAOX represents a multiplier for cladding outside oxidation surface area which is increased to account for lacking phenomenon identified in Section 3.3.2. Since this uncertainty needs to be applied as a bias hence valid on a permanent basis, the lower value of the distribution support will be increased to the maximum realistic value as recommended in the user's parameter file. The higher value has been identified, as indicated in Section 2.6.3, by means of sensitivity analysis until reaching the highest peaks collected in Ref. [I–32].

[3] Average value taken from user's parameter file. Best estimate values as suggested by the code will be hereafter taken from user's parameter file as long as they come from extensive benchmarking or have found enough support in the referenced literature.

[4] Code's tendency to lower corium heat transfer has been corrected by adapting the distribution support as indicated in the user's parameter file, thus avoiding setting the average value. This has been done in agreement with Fauske and Associates, Inc., responsible for code development.

[5] The distribution support and / or mean value has been modified upon discussion with Fauske and Associates, Inc.

[6] Only for PWR.

[7] According to Ref. [I–21], large significant uncertainties on BWR SRV and PWR pressurizer PORV performance in extended cycling under high temperature are found. A simplified treatment of valve cycling and failure is carried out by means of the following consideration: Let us call n the number of valve cycles. The probability of a valve to be stuck open in each $m<n$ cycle will be $p_{SO} \cdot (1-p_{so})^{m-1}$. Therefore, the cumulate probability of remaining stuck open in n cycles is equal to the sum of the geometrical series of constant ratio: $\sum_{m=1}^{n} a \cdot r^{n-1} = \frac{a \cdot r^{n} - a}{r-1}$, where $a=p_{so}$ and r<1 with $r=1-p_{so}$. Substituting the ratio and scale factor we obtain that the cumulative probability of failure is $P_{so}=1-(1-p_{so})^{n}$. The value of a single failure, p_{so}, is very uncertain and no references have been found in the open literature. Considering that failures per demand before core damage have been ranged from 2.7e-3 to 5.8 exp–3 (NRC, 2012), we will consider twice the average of these values as a first estimate and then compute the 5th-percentile and 95th-percentile. The applicant's user is encouraged to find more accurate values specific to the type of value distinguishing between PWR pressurizer PORVs and BWR primary SRVs.

[8] For distribution supports not provided in the user's parameter file, sensitivity analyses may be performed and conservative bounds may be assigned if a significant correlation is found. Default values have been here selected to range ±25% around the nominal value as suggested in the user's parameter file. Nonetheless, they highly depend and may be tailored upon literature survey and associated uncertainty.

[9] It strongly depends on plant geometry. Results are very sensitive to this parameter.

[10] Since parameter variation does not rely on uncertainty but on plant configuration, it is preferred to choose a truncated exponential rather than a uniform distribution. Mean value is plant-specific depending on the number and type of openings and paths between cavity / pedestal and containment / drywell and has to be carefully analyzed since it will dramatically impact on ex-vessel corium distribution, MCCI and quenching (hence containment gas concentrations, pressure and temperature).

[11] According to the user's parameter file explanation, "*the new dryout heat flux model removes the necessity of using the model parameter FCHF*". The mechanistic dryout heat flux model is activated when IQDO=1. Single sensitivity calculations with IQDO=0 were carried out and yielded temperatures values were slightly higher (in the order of a few degrees); however, huge differences in the BWR cases were noticed regarding the combustible gases molar fraction, as high as 5 times or even one order of magnitude higher with IQDO=1. Therefore, because of the increased knowledge gained on ex-vessel FCI and in order to be on the conservative side of the relating calculations, applications are encouraged to consider activation of the dry heat flux model by setting IQDO=1.

[12] For Limestone Common Sand concrete, a value of 0.0235 is appropriate to match MAAP with CORQUENCH simulations of (Farmer, 2001). For Basaltic concrete, which has very little off-gas and hence minimal melt eruption cooling, FCHF mean value may be as low as 0.01.

I–3.4. Uncertainty quantification

Table I–5 presents the phenomena to code-uncertainty-parameter conversion, including the encoded name of the parameter, brief description, support of the distribution, selected *pdf* based on the available information and maximum entropy theory methods as collected in Table I–2, and the corresponding PIRT phenomenon. Support of the distributions have been initially taken as provided by the code, based on an uncertainty range identification relying on a wide set of solid, extensive benchmarking according to the latest state of the art.

Except for those additional category-nature phenomena not considered in EURSAFE PIRT, mapping between PIRT H/H phenomena and code parameters will not require further analysis whereas core relocation, Zr oxidation and in-vessel pool heat transfer have first been decomposed into their associated phenomena just like in Ref. [I–20] through arrangement upon time phases and affected components.

Code parameter *WH2MIN* will be used twofold, on one side to increase hydrogen generation as predicted by the code to account for uncertainty modelling resulting in milder containment FOM characterization, and on the other to meet with US NRC 10 CFR 50.34 Ref. [I–5] by imposing an additional condition for continuing hydrogen generation until reaching a quantity equivalent to 100% of cladding reaction.

From the perspective of mitigating systems performance and associated human actions coming from best estimate values embedded in PSA models, only reactor cavity / containment flooding time needed for the water to reach the reactor cavity / drywell floor will be varied from 0 to 3 hours after core damage (realistic information about the expected value may be obtained from Level 2 PSA analysis), and after checking its impact through sensitivity calculations. From the perspective of expert judgement, modifications to the support of the distributions have been implemented as indicated in Table I–5.

With respect to dependent code phenomena found in PIRT analysis, phenomenon 8 does not map with corresponding modelling since it relies on phenomenon 47 dealing with in-vessel heat transfer; phenomenon 29 is not accounted for due to two reasons: on one hand, even if aerosol retention greatly impacts on source term released to the outside environment, we are only interested in characterizing radiation within the containment; on the other hand, we are making use of the so called Alternate Source Term (AST) methodology Ref. [I–33] which does not allow fission product deposition. This argument is also valid for phenomenon 33. Phenomenon 35 is partly dependent when dealing with generic containment properties, whereas uncertainties on PH characterization have been taken into consideration. Uncertainties on containment thermal and mass stratification have not directly accounted for since they are judged as mainly relying on the accuracy of other phenomena such as corium relocation or ex-vessel FCI, and provided nodalization scheme and stratification tracking code capability adequacy. Inherent uncertainties on stratification are not highly relevant on FOM characterization since code capacities to cope with natural circulation current flows and associated heat and mass transfer phenomena have been benchmarked. As already noted, phenomenon 39 is missing in MAAP5.02, but its importance for the selected sequences has been estimated low.

I–3.5. Uncertainty propagation

The selected transient for the PWR exercise is a loss of all AC electrical buses (an SBO-like event) without EFW and not crediting for any human action foreseen in the SAMGs. The selected BWR transient is an LFW without ECCS availability and not crediting for manual ADS performance (close to an SBO like event). The simulations will initially run for 5 days after the onset of the initiating event (time 0). In order to exclusively explore high pressure scenarios at RPV failure, both cases will by default inhibit *hot leg creep rupture* and PORV / SRV stuck-open probability.

Uncertainty input vectors have been generated through simple random sampling, since no significant differences have been observed when comparing it to other sampling methods such as Latin Hypercube sampling as demonstrated in Ref. [I–34] and in Ref. [I–35]. Uncertainty analysis outcomes of pressure, temperature, combustible gases (hydrogen and carbon monoxide) molar fraction, steam molar fraction, and radiation activity in the upper part of the containment (wetwell in the BWR) are depicted in Figs I–6 to I–15. The reference case, namely the default simulation making use of best estimate values, has been highlighted in bold. Even if some variation is registered among the 59 simulations, pressure evolution is driven by the venting system performance as modelled in the code, establishing a range within which containment pressure is limited and cycles along the transient, except for those cases where pressure does not reach the highest opening setpoint during the entire 5-day simulation.

Total radiation activity has been simply computed as the activity generated by the 65 radioisotopes tracked by the code presented in the entire containment without crediting for attenuation by the presence of any kind of shielding or structural walls. Radioactivity input data have been taken from a generic 900 MW(e) PWR for both exercises. The total initial activity amounts to 5^9 Ci, approximately. As already noted, activity has been computed following the AST methodology, i.e. considering that each containment compartment is a single node with airborne nuclides uniformly distributed. In case of PWR containments, quantitative results may not significantly depart from reality yet more complex BWR layouts will require sensitivity analysis to check the need for more detailed geometric nodalization. In either case, a more refined analysis is recommended not only to account for geometry, potential shielding and spatial distance from the emission to the receiver point where the equipment is located, but also to convert activity values into absorbed doses by means of a dose conversion factor for each type of equipment, just as in Ref. [I–36], since dose absorbed by the electrical cables is usually calculated from the human-body dose computed by the code.

Temperature evolution in PWR cases 13 and 47 depicts a slightly positive slope after a 5-day simulation. To ensure bounding until reaching long-term steady stable evolution, both cases were rerun until twice the original simulation time (see Fig. I–16). The value of the results needs to be taken just as a simplified example as they highly depend on specific in-plant characteristics and mitigating systems configuration and performance, for instance, reactor cavity / drywell flooding tank mass, flow rate and expected injection time, number and type of PARs installed in the containment (no PARs were implemented in the applications), rated thermal power and decay power, filtered venting system pressure setpoints, plant geometry affecting code uncertainty parameter ranges such as ex-vessel corium entrainment and relocation outside cavity / pedestal, etc.

I-3.6. Output data post-processing

For each of the transients belonging to the SAS matrix, an associated set of uncertainty simulations has to be generated for each FOM. From each collection of uncertainty parameter time dependent profiles, a twice-composite bounding sequence has to be built up from the entire set of SAS matrix sequences ultimately defining the environmental conditions for containment I&C survivability once transformed into histogram representations. Bias does not apply in the current applications since lacking phenomena have already been taken into account through code parameters. For simplicity's sake, histogram representations will be here directly generated from the single simulated sequence selected for the applications rather than from the twice-composite built-up sequence generated from the full list of SAS sequences. Only containment temperature FOM histograms will be here developed both for the PWR and the BWR application.

Regarding dynamic phenomena as listed in Table I–3, in-vessel and ex-vessel steam explosions do not lead to containment temperature peaks. HPME has already been taken into account and it is in fact responsible for the temperature peaks observed in both applications right after RPV failure. Reflooding of degraded core in the simulated scenarios only occurs after RPV failure thus once the core has fully lost its geometry and no significant hydrogen excursion and oxidation energy release can occur. Therefore, temperature peaks coming from dynamic phenomena whose uncertainty has not been taken into account might only come from hydrogen explosions. Among the 59 PWR and BWR simulations, only minor hydrogen volumetric burnings occur throughout the transients, thus separate simulations may be performed until reaching global hydrogen combustions.

Bounding scenarios until achieving hydrogen explosions leading to containment pressures slightly lower than the containment mechanical failure might initially be generated in two ways: whether by modifying the combustion related parameters within their uncertainty distribution support which for simplicity's sake will be taken from the information provided by the code, such as the ignition temperature as applied in the BWR case, or by tailoring the code uncertainty parameters indirectly affecting hydrogen combustions as applied in the PWR case, namely those included in the PIRT. For this latter case, and in recognizing that scenarios undergoing hydrogen combustion thereby leading to peak values higher than the highest ordered value of the 59 sample are located in the 5/5 probability level-confidence interval, the scenario may be carefully suited to make it become prone to undergo hydrogen explosion.

As shown in Table I–3, sequences with containment deinertization with long-lasting MCCI and FCI are the best candidates to lead to global hydrogen explosion since huge quantities of flammable gases can first accumulate in an inerted atmosphere produced by intense FCI heat transfer, and after later corium quenching increase their concentration as a result of containment deinertization because of the action of the continuous injected water flooding the containment and condensing large quantities of steam.

Temperature and pressure time dependent evolutions of flammable-gas-combustion-related sequences are shown in Fig. I–17 and I–18, while flammable gases concentration are depicted in Fig. I–19 and I–20. It can be noted that global explosion has taken place since concentrations drop to zero right after the temperature and pressure peak. Both peaks have been correspondingly added to the temperature histogram representations depicted in logarithmic scale in Fig. I–21 and I–22.

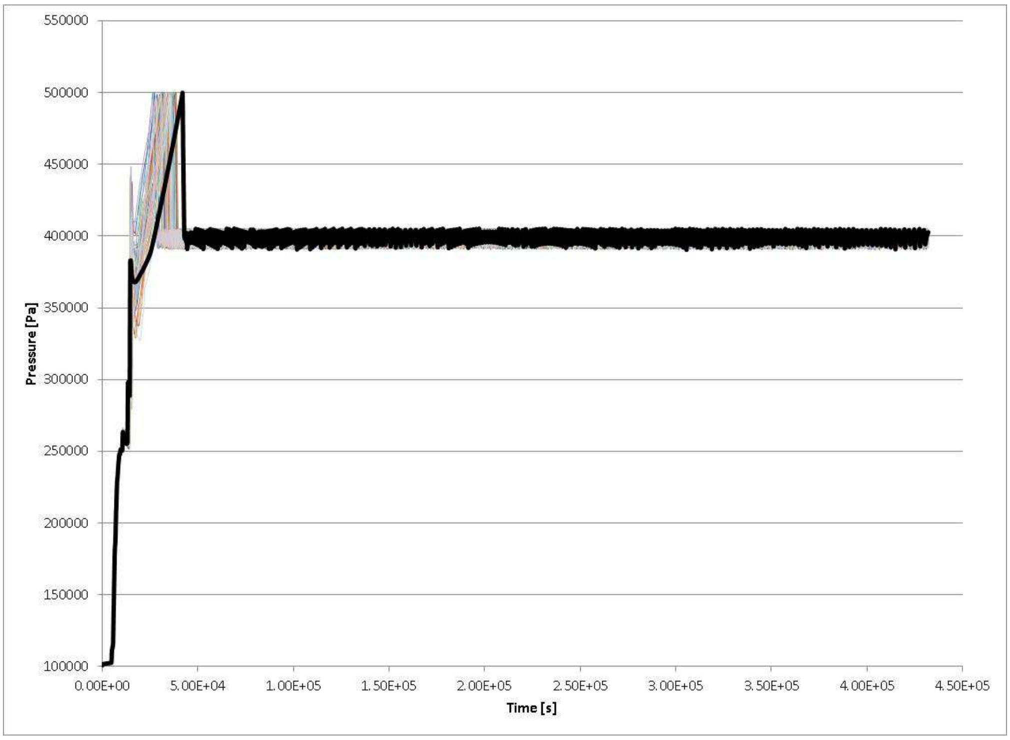

FIG. I–6. Pressure evolution in the containment dome of the PWR analysed case.

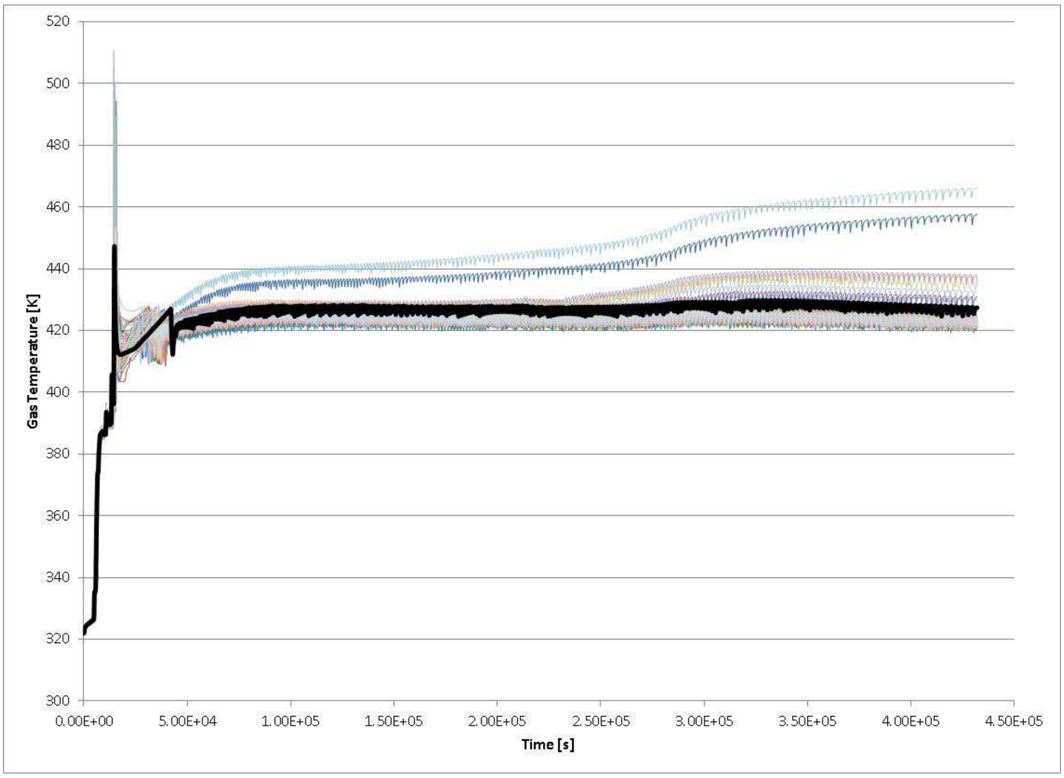

FIG. I–7. Temperature evolution in the containment dome of the PWR analysed case.

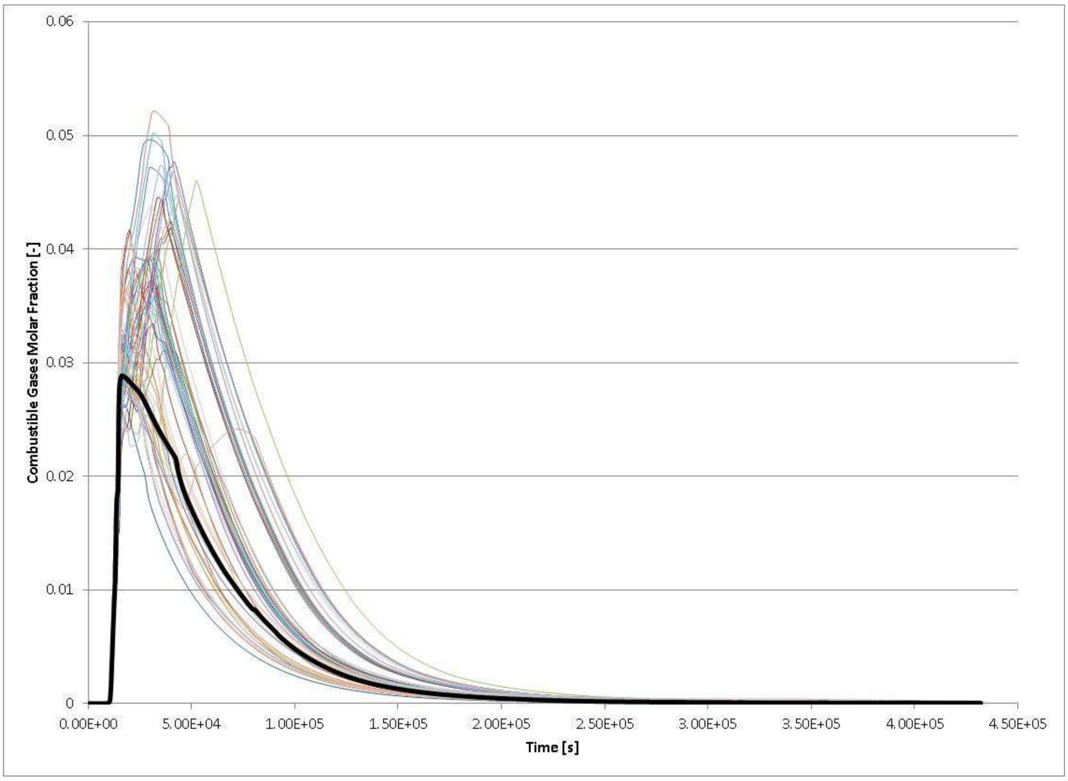

FIG. I–8. H_2 and CO molar fraction in the containment dome of the PWR analysed case.

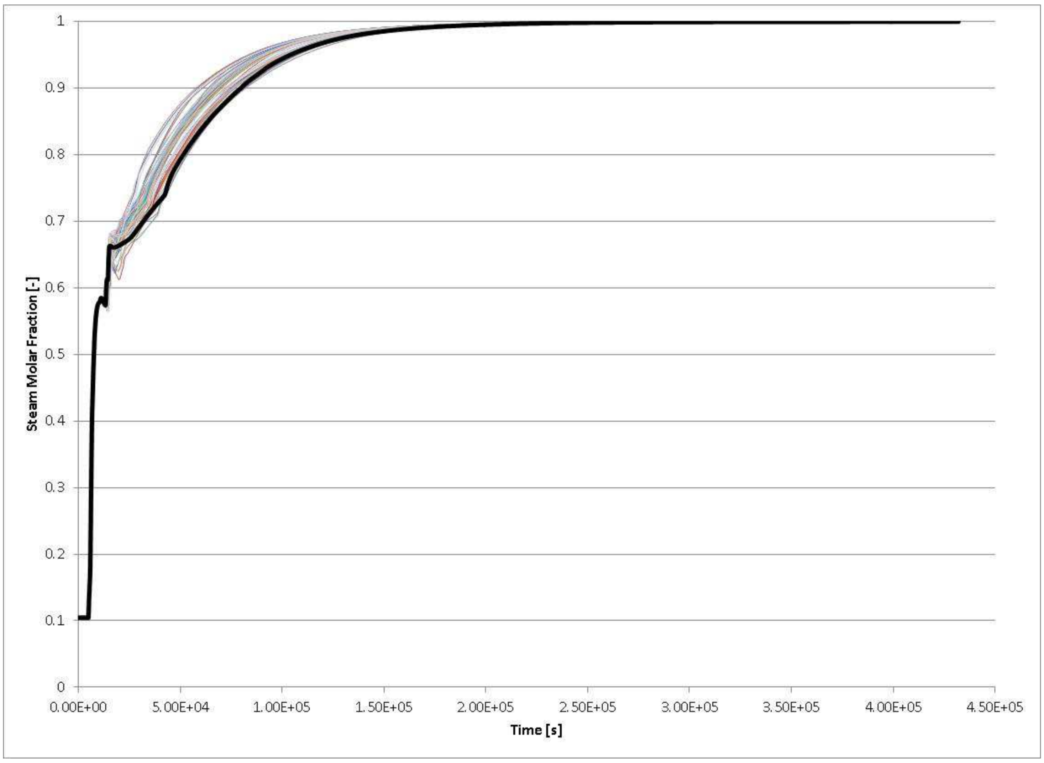

FIG. I–9. Steam Molar Fraction in the containment dome of the PWR analysed case.

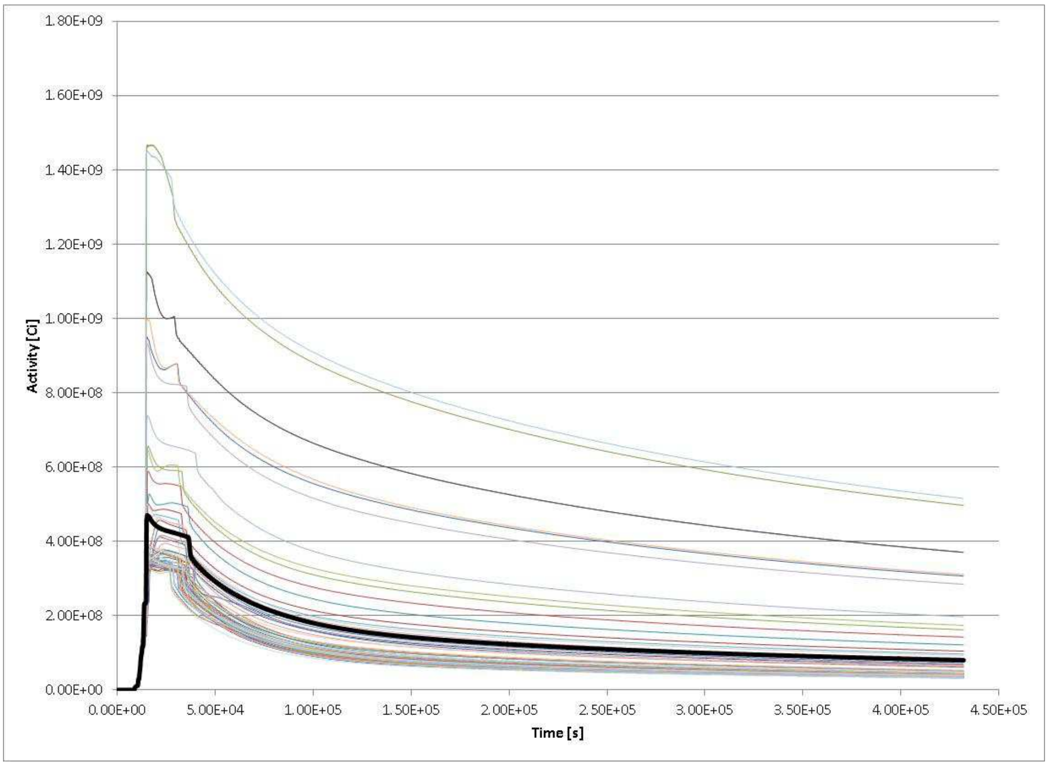

FIG. I–10. Activity in the containment dome of the PWR analysed case.

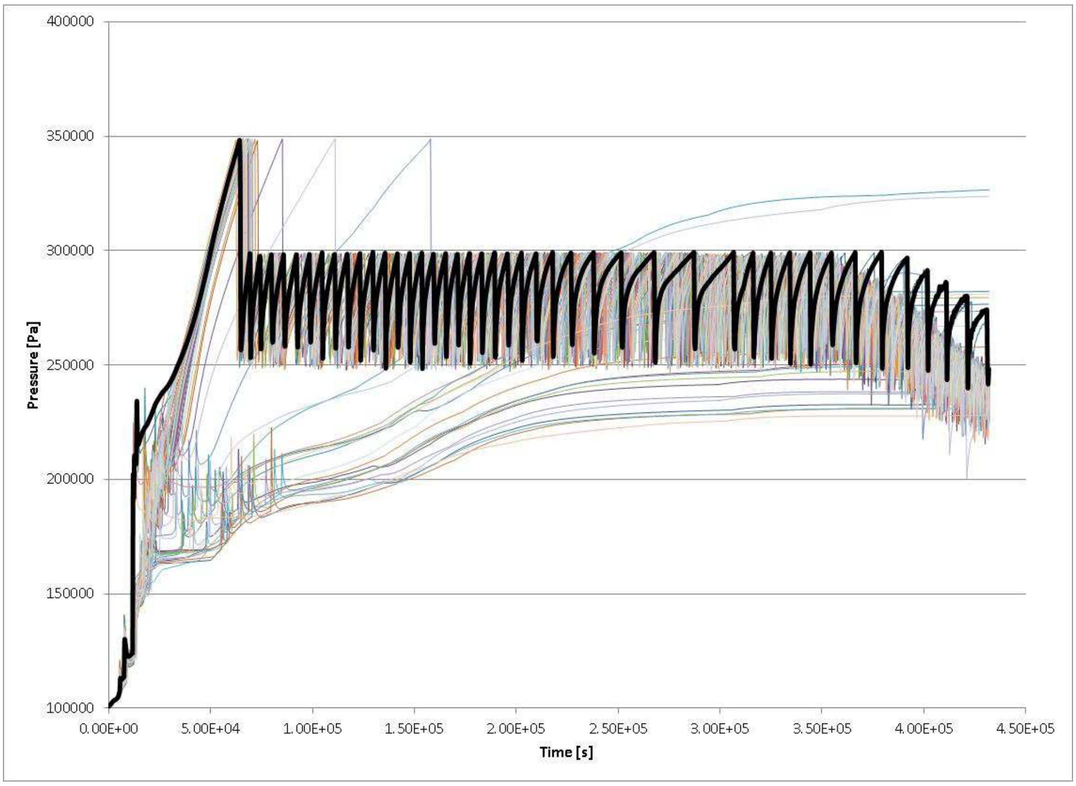

FIG. I–11. Pressure evolution in the wetwell dome of the BWR analysed case.

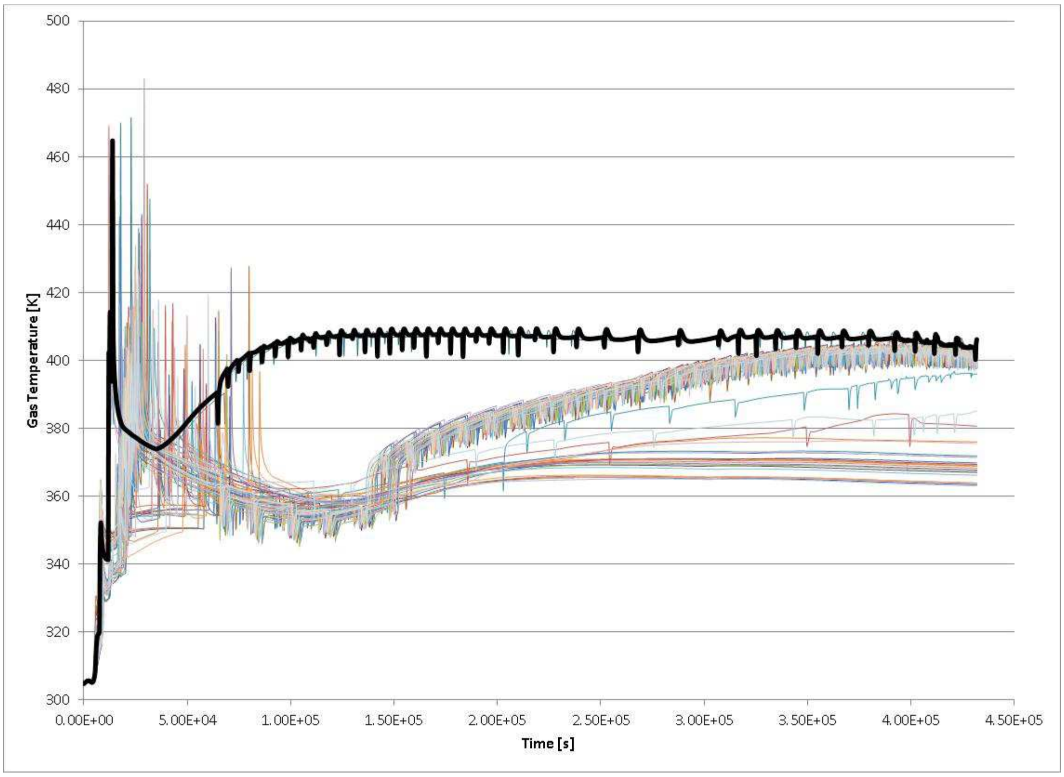

FIG. I–12. Temperature evolution in the wetwell dome of the BWR analysed case.

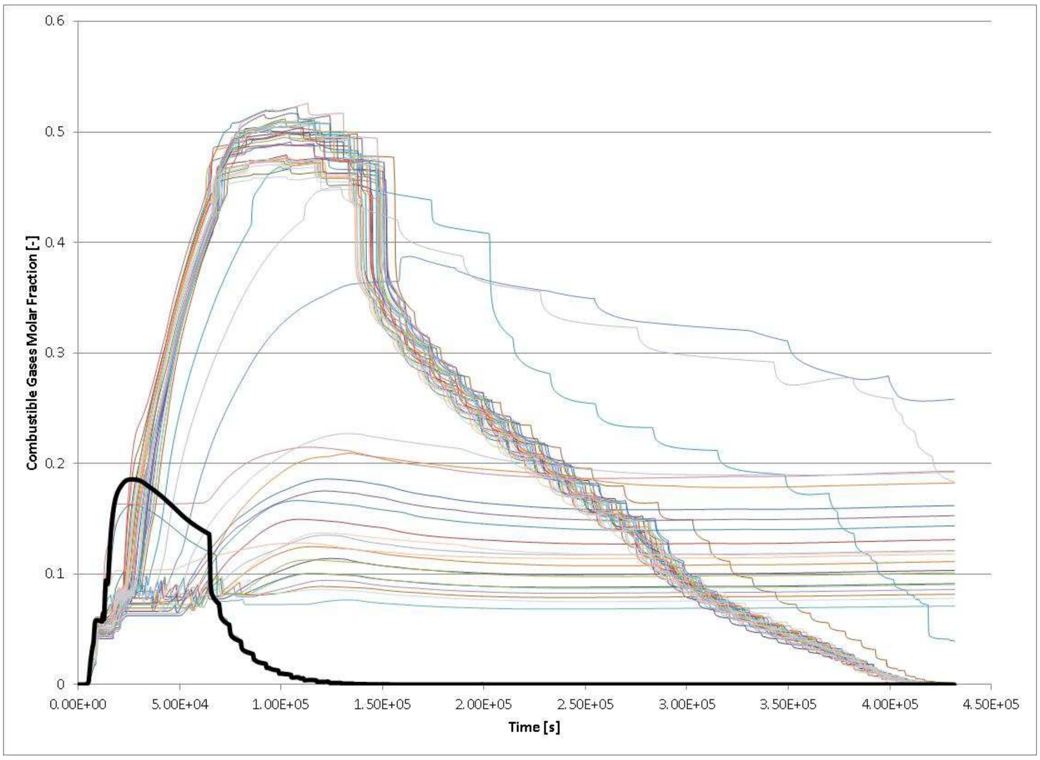

FIG. I–13. H_2 and CO molar fraction in the wetwell dome of the BWR analysed case.

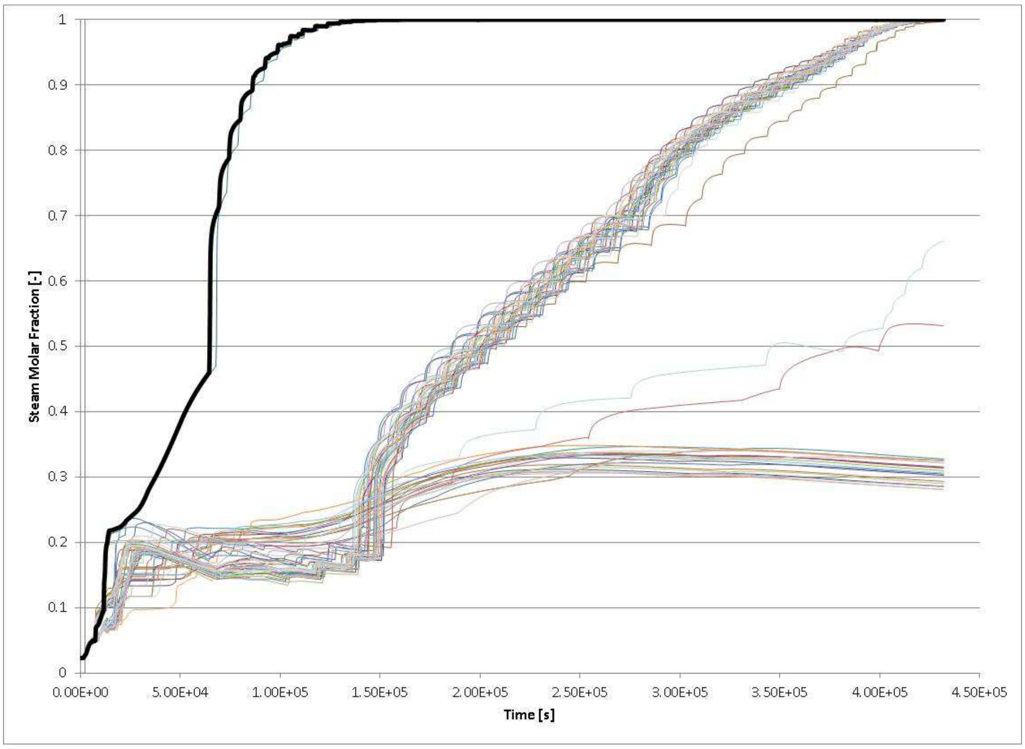

FIG. I–14. Steam molar fraction in the wetwell dome of the BWR analyzed case.

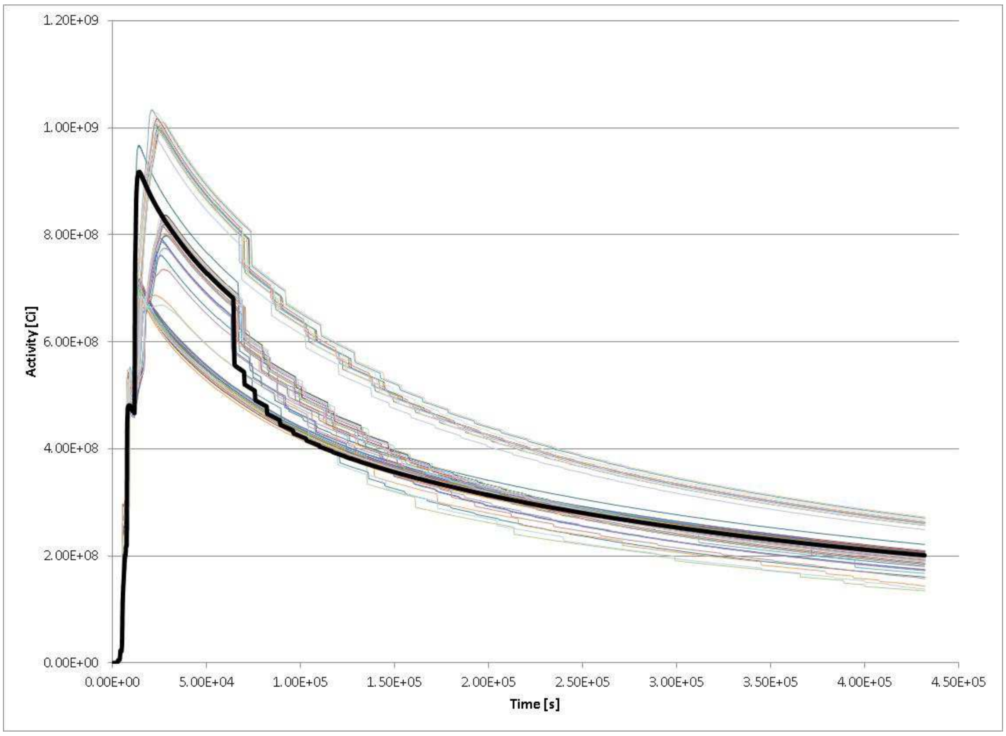

FIG. I–15. Radioactivity in the wetwell dome of the BWR analysed case.

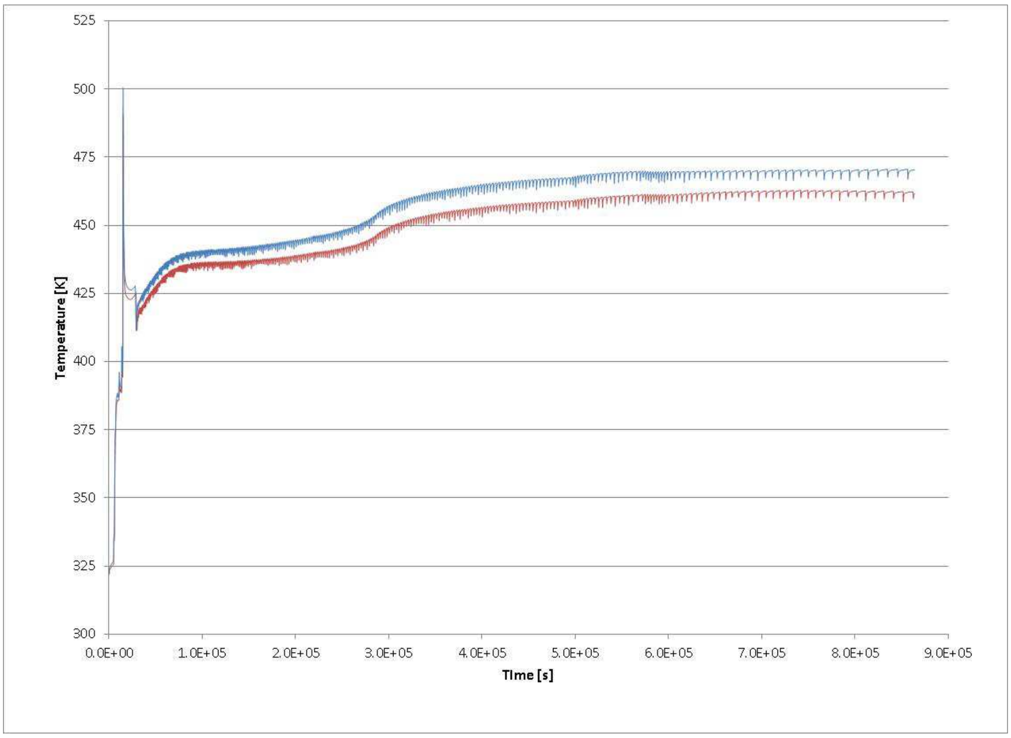

FIG. I–16. Temperature in the containment dome in a 10day simulation of the uncertainty sequences 13 and 47 of the PWR analysed case.

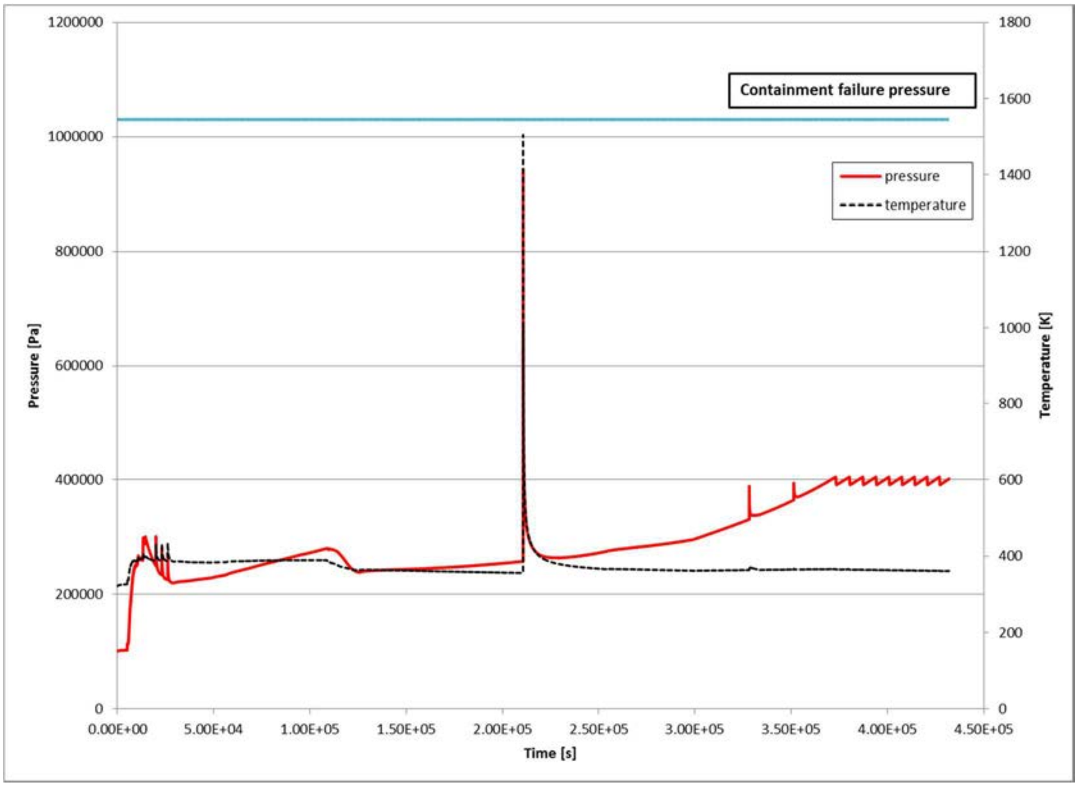

FIG. I–17. Temperature and pressure evolution in the global combustion sequence for the PWR analysed case.

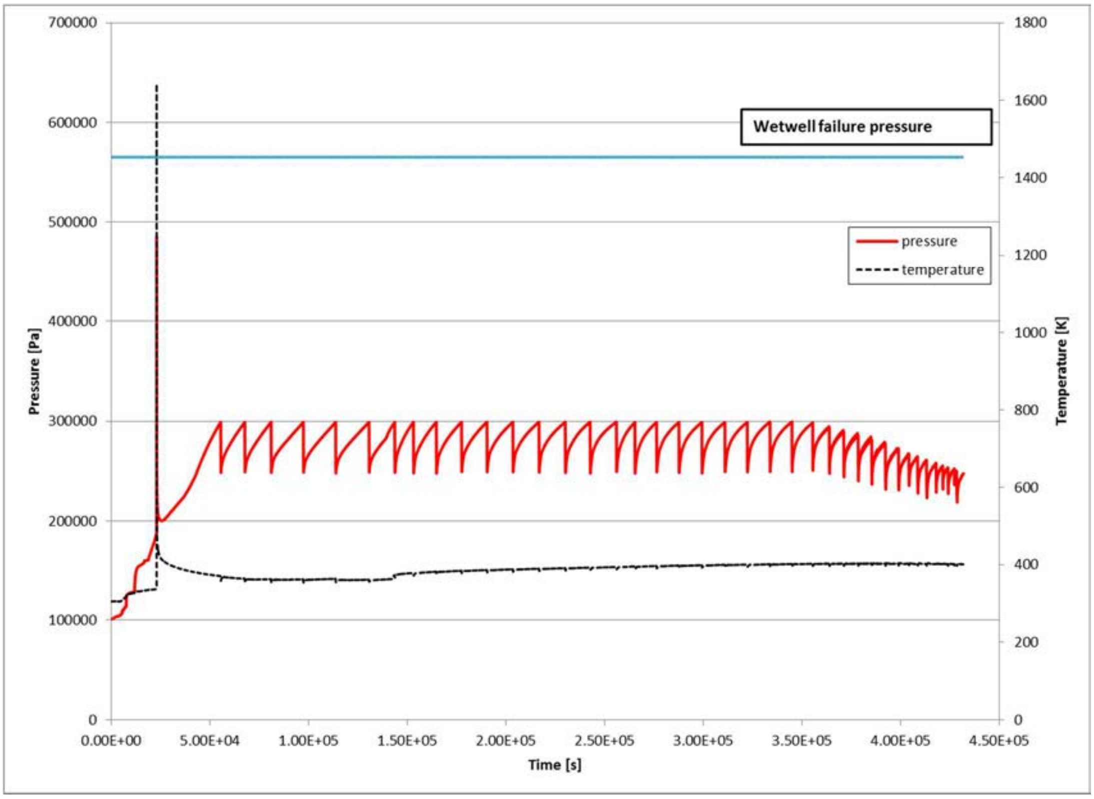

FIG. I–18. Temperature and pressure evolution in the global combustion sequence for the BWR analysed case.

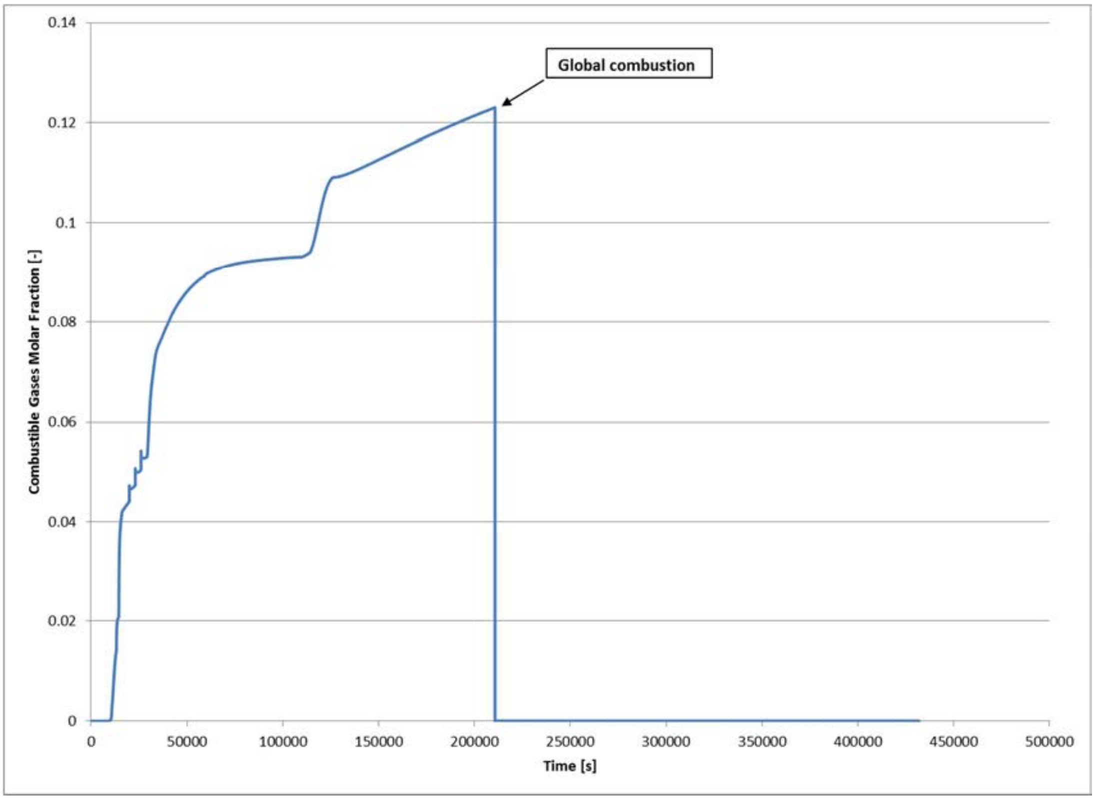

FIG. I–19. Flammable gases concentration in the global combustion sequence for the PWR analysed case.

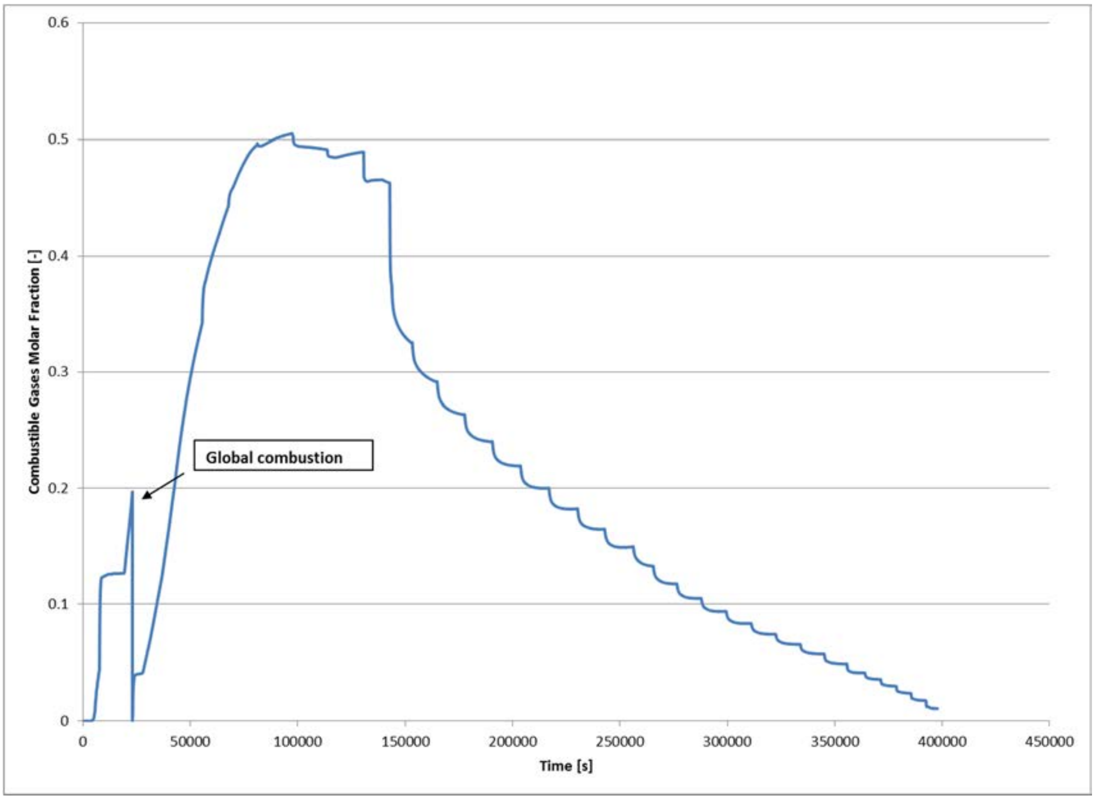

FIG. I–20. Flammable gases concentration in the global combustion sequence for the BWR analysed case.

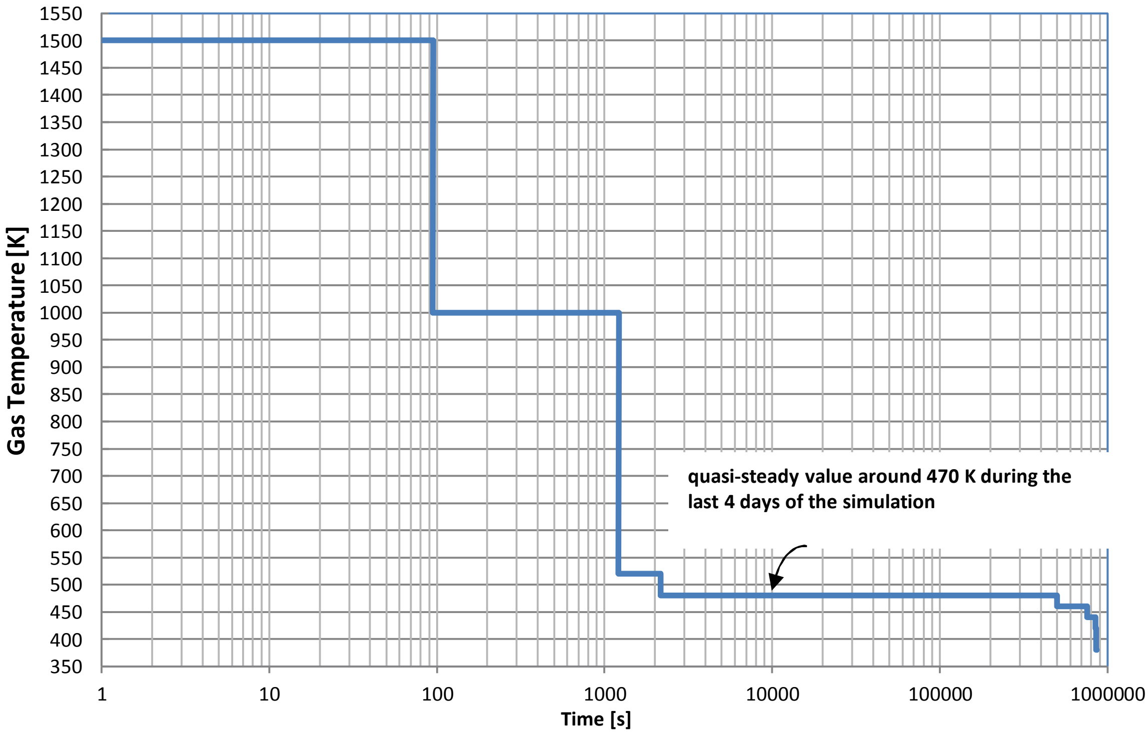

FIG. I–21. Temperature histogram representation for the PWR analysed case.

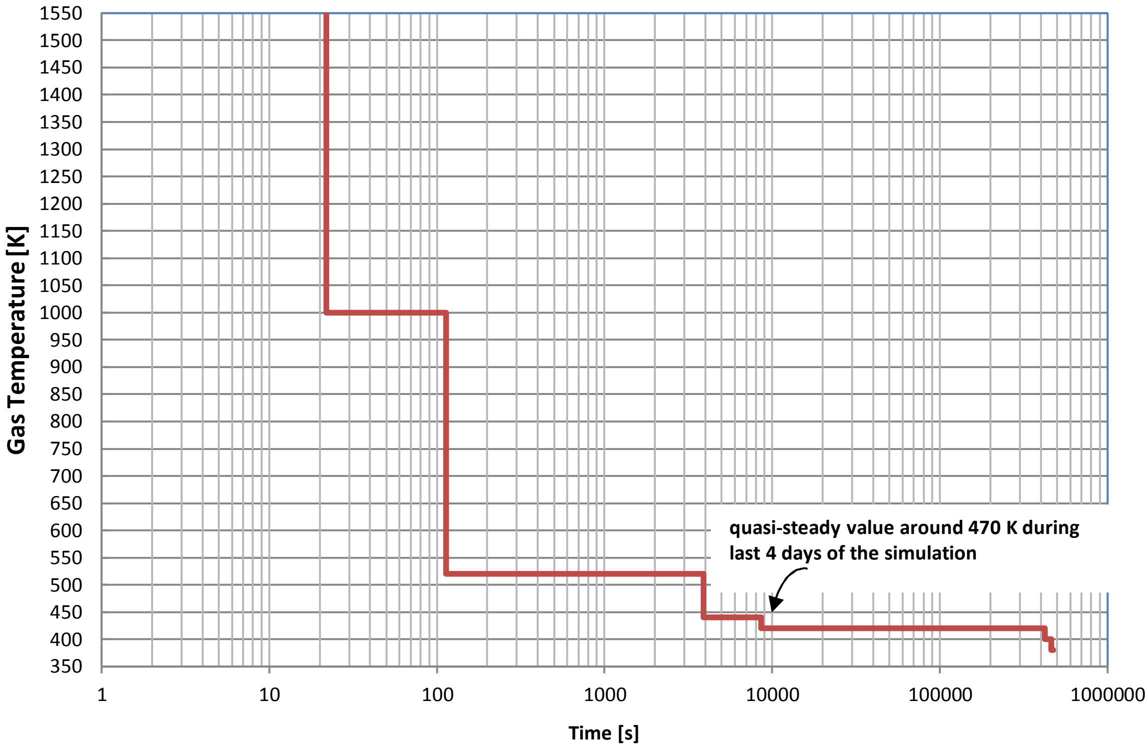

FIG. I–22. Temperature histogram representation for the BWR analysed case.

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ANNEX II

DESCRIPTION OF CODES USED FOR CALCULATION OF ENVIRONMENTAL PARAMETERS

II–1.ASTEC

The Accident Source Term Evaluation Code[12] (ASTEC) severe accident analysis code system is jointly developed by Institute for Radiological Protection and Nuclear Safety, France (IRSN) and Gesellschaft für Anlagen und Reaktorsicherheit (GRS). The code is able to calculate all severe accident related phenomena taking place in the reactor pressure vessel and in the containment during a severe accident; only steam explosion is not modelled. The code system is highly modular; the most important modules are shown in Fig. II–1.

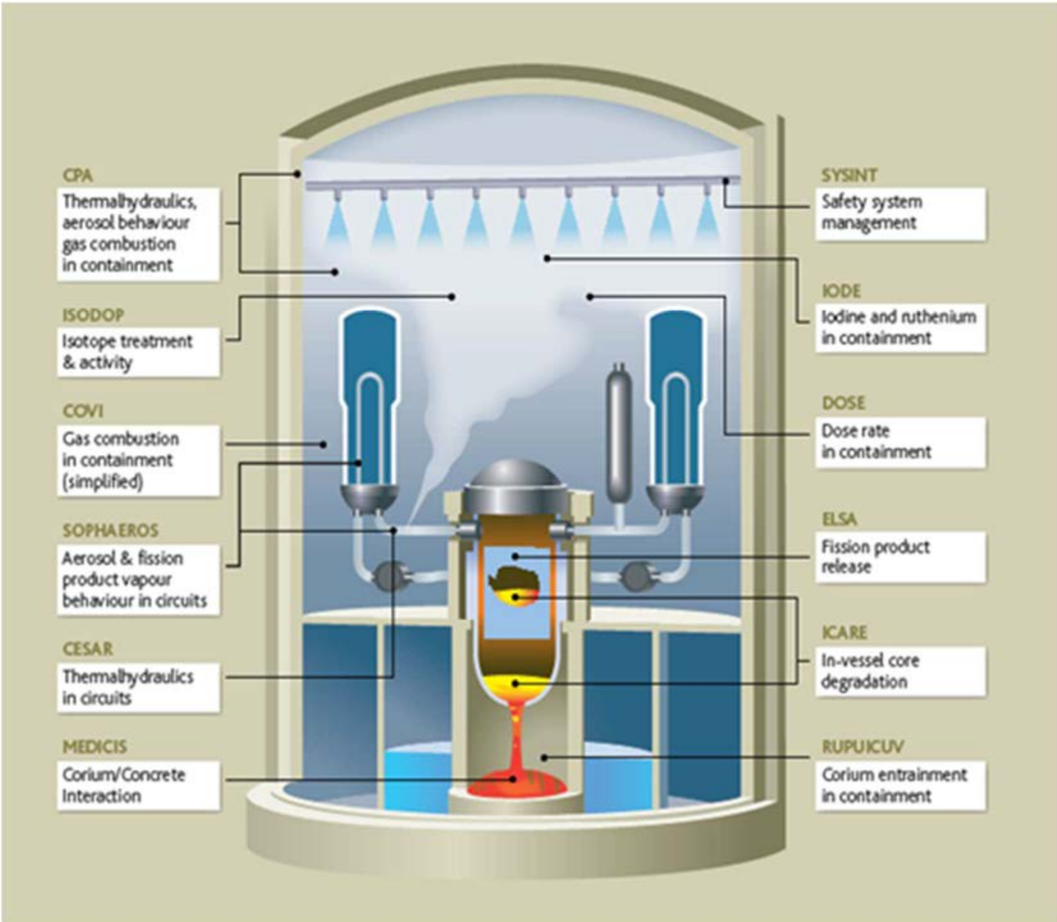


FIG. II–1. Main modules of the ASTEC code (figure courtesy of EC JRC).

The current version is V2, which incorporates improved models for core-reflooding and core cavity flooding phenomena, MCCI, primary circuit iodine chemistry and several other topics proposed by the SARNET[13] R&D network. The code is capable of calculating PWR, VVER and BWR reactors, currently developments are being carried out to incorporate CANDU and SFR (Gen IV sodium cooled fast reactor) types, as well. The code was extensively validated by using experimental data (e.g. PHÉBUS[14]). It was used by IRSN for supporting PSA 2 studies and for the analysis of EPR. An ongoing effort is the Code for European Severe Accident Management (CESAM) Euratom FP7 project, which includes further code developments to provide a realistic modelling of severe accident

[12] Accident Source Term Evaluation Code, IRSN / GRS.
[13] European network of excellence on core meltdown accidents.
[14] European light water reactor accident source term research project.

mitigation measures during SAMG execution. The basic aim is to have ASTEC as a widely accepted European reference code for severe accident analysis and source term calculation.

II–2. MODULAR ACCIDENT ANALYSIS PROGRAMME

The modular accident analysis programme (MAAP) code was originally developed to provide a tool for the fast simulation of severe accident scenarios. In principle, the code is able to calculate all important phenomena potentially occurring during a severe accident. The current version is MAAP5, with enhanced nodalization capabilities (e.g. every steam generator can be modelled individually), as well as a better two-phase calculation algorithm and an enhanced containment model. The MAAP code was developed by Fauske & Associates for EPRI and it is able to determine the following severe accident phenomena:

- Steam generation in the core and core heat-up;
- Cladding oxidation and hydrogen generation;
- Reactor pressure vessel damage;
- Molten core concrete interaction;
- Combustion and explosion of combustible gases;
- Transport of fluids by high speed gas flows;
- Emission, transport and deposition of fission products.

Visualization of a typical MAAP calculation result is shown in Fig. II–2. The code is able to model all important safety systems (e.g. ECCS, safety valves, containment spray, cooling systems etc.) and also takes into account the mitigating actions initiated by the operators.

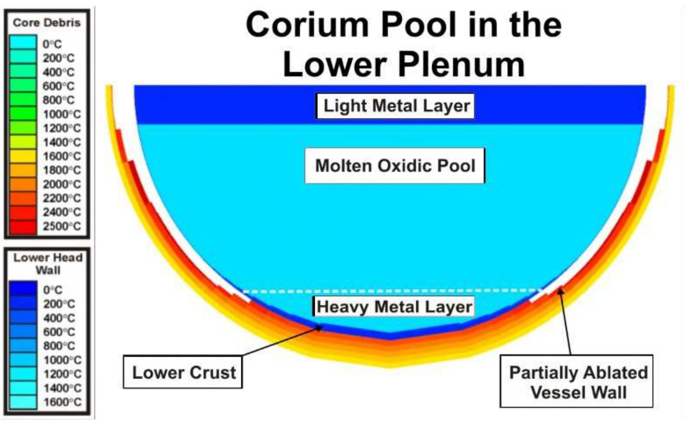


FIG. II–2. Visualization of a typical MAAP calculation result (courtesy of Fauske & Associates).

Following the Fukushima accident, simulator vendors also introduced the enhanced MAAP versions to provide engineering grade severe accident simulation tools in their portfolio. For example, the MAAP HD code developed by Corys (France) is a real time MAAP version that can be integrated into an existing full scope simulator. It is capable of calculating all in-vessel and ex-vessel phases of a severe accident, including containment and the auxiliary building, as well as spent fuel pool. The basic application area is SAMG validation and training. The accuracy of the system is sufficiently high, placing this version into the engineering simulator category.

A similar code using MAAP5 — Advanced Simulation Based Tool for Severe Accident Analysis (PSAHD) has been developed by GSE Systems (USA). An interesting and unique feature of this tool is that it is able to simulate severe accident scenarios simultaneously that may occur simultaneously at several NPP units located at the same site.

II–3.MELCOR

After the Three Mile Island accident in 1979 Sandia National Laboratories (USA) on behalf of the US NRC developed a severe accident code “Methods for Estimation of Leakages and Consequences of Releases” (MELCOR). It is mainly used for modelling of severe accident processes in light water reactors (PWR and BWR) and it is capable of calculating the following severe accident phenomena:

- Thermalhydraulic behaviour of the primary circuit, the reactor cavity and the containment during an accident scenario;
- Heatup, damage and collapse of the active core;
- Molten core concrete interaction (MCCI);
- Generation, transport and combustion of hydrogen;
- Emission and transport of fission products.

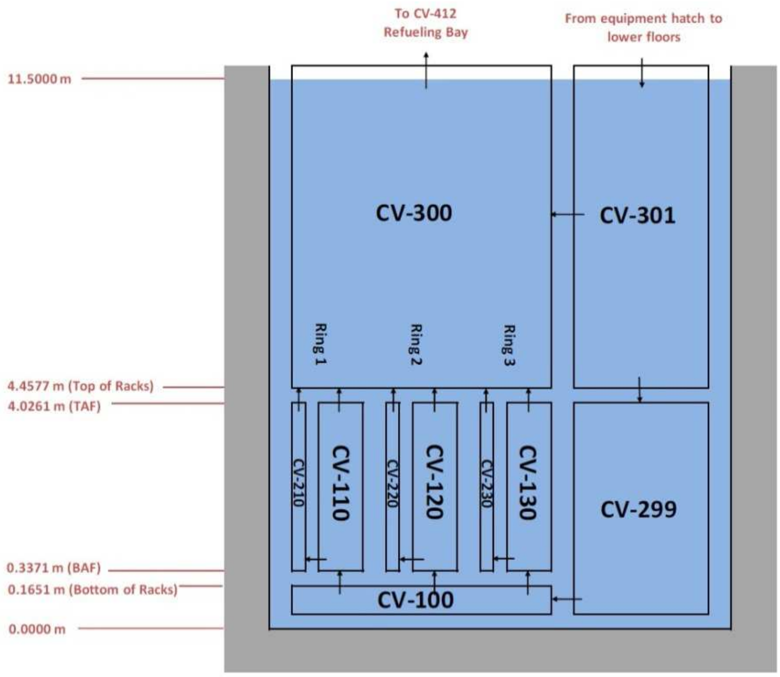


FIG. II–3. MELCOR nasalization for calculating the SFSP of Fukushima Unit 4 (Courtesy of Sandia NL).

After the Fukushima accident the MELCOR was further developed in order to model severe accident processes taking place in the spent fuel pool (SFP). The new code version is able to calculate the three basic SFP states, such as normal operation state, transient state due to a partial loss of coolant, and accident state due to a total loss of a coolant in the pool.

ANNEX III

MAPPING OF ENVIRONMENTAL PARAMETERS INSIDE AND OUTSIDE THE CONTAINMENT DURING SEVERE ACCIDENT

A mapping of environmental parameters outside the containment, in annulus, and reactor building may be affected by high temperature and high radiation is shown in Fig. III–1. This figure illustrates the temperature in the containment, annulus and reactor building to which the electrical and I&C equipment may be exposed during a severe accident.

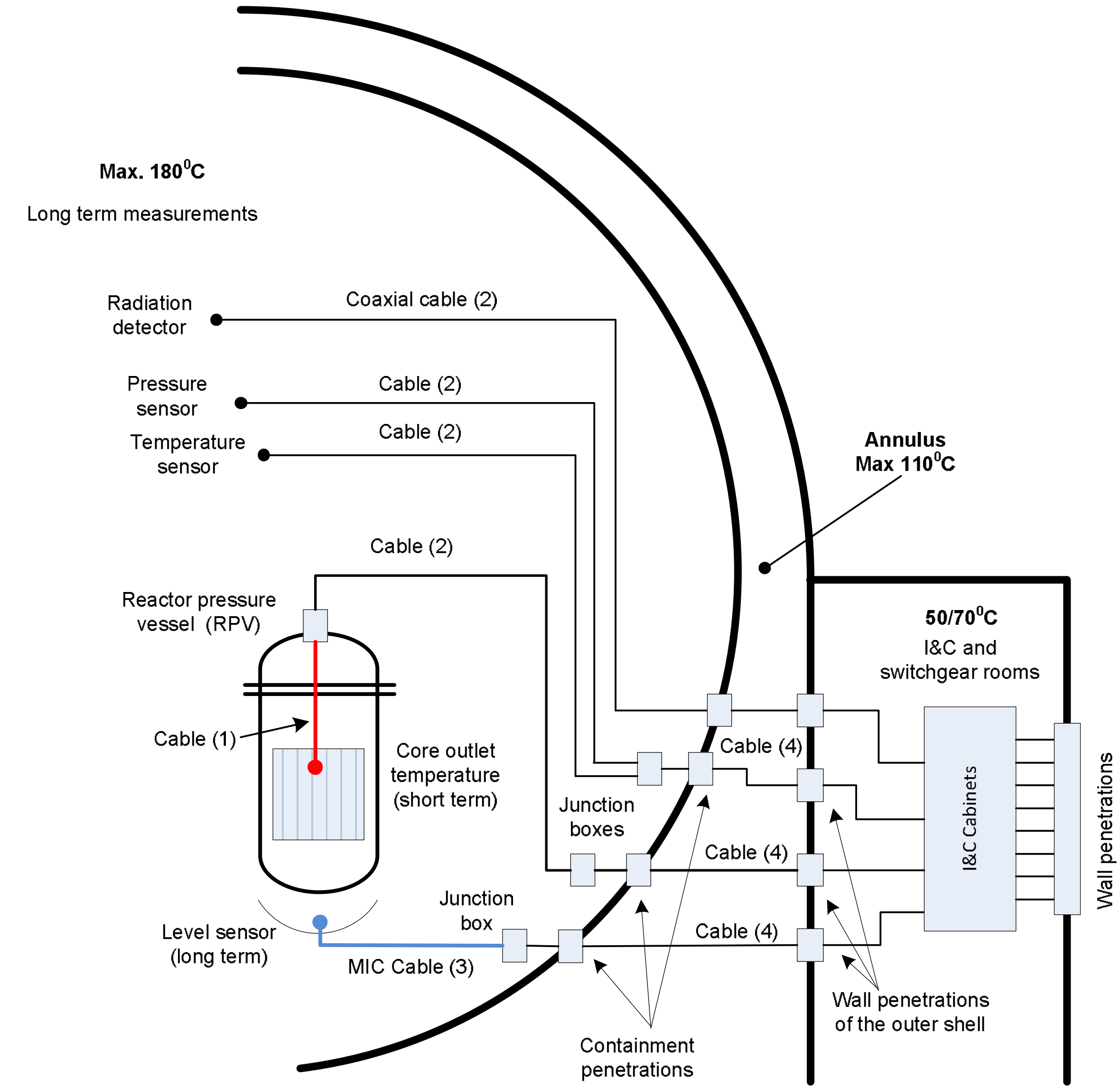
The most exposed instrument cable is cable (1) connecting core outlet temperature to the reactor pressure vessel penetration. As a result of severe accident conditions the temperature inside the vessel may reach and most probably exceeds the 1000°C. In accordance with KTA 3502 the sensor and the cable are designed for temperatures up to 1000°C (wide range instrumentation). However, the sensor and cable certainly starts to degenerate at temperatures higher than 1000°C and therefore this measurement sensor is considered for a short term monitoring only.

Different situation is for temperature, pressure and radiation detectors and their associated instrument cables inside the containment. These measurements are considered for long term function and therefore have to be capable to withstand the assumed maximum temperature of 180°C.

Additional instrument cables for some sensors, e.g. reactor cavity / sump level measurement may be exposed to even higher temperature than 180°C. In case of that mineral insulated cables (MIC) are used to connect a sensor with a junction box inside the containment.

The temperature mapping is estimated based on a known heat source in the containment (e.g. calculations provided in Annex II) and a heat transfer throughout the containment structure, annulus and rector building walls.

The equipment located inside areas adjacent to the reactor building may be exposed up to 70°C and will certainly maintain its capability; however if the operating personnel is instructed (e.g. by severe accident management guidelines) to operate manually an equipment in this room, or to take a direct measurement on instrument terminal points, he may be directly exposed to high ambient temperature and may not be able to perform his task without additional safety measures.



No.	Designation	Temperature (long term)
1	Cable (1) inside RPV	1000°C
2	Reactor penetration (inside RPV/containment)	1000°C/180°C
3	Cable (2) / junction box	180°C
4	Cable (3) / junction box (MIC)	Greater than 180°C
5	Junction box inside the containment	180°C
6	Penetrations – containment /annulus	180/110°C
7	Cable (4) inside annulus	max. 110°C
8	Penetrations – annulus/reactor building	max. 110°C
9	Cable (5) inside reactor building	50/70°C
10	Electronics	50/70°C

FIG. III–1. Example of environmental parameters mapping for a PWR 1000 reactor (figure courtesy of AREVA).

ANNEX IV

LESSONS LEARNED FROM FUKUSHIMA

The earthquake and resultant tsunami which occurred at Tokyo Electric Power Company's Fukushima Daiichi nuclear power plants (the TF-1 accident) in March 2011 lead to failure of the plant power supply. The lack of power supply to monitoring instrumentation made it exceedingly difficult to monitor critical plant parameters needed to establish the status of the reactor. Subsequently, core damage and hydrogen explosions ensued. A national programme for the development of instrumentation systems for severe accidents commenced in Japan (SA-Keisou in Japanese) in 2012 Refs [IV–1] and [IV–2].

In order to develop SA-Keisou, the SA-Keisou parameters are first selected and the design conditions the SA-Keisou can withstand in an accident are determined for BWR and PWR. The SA-Keisou parameters are selected in two steps. In the first step, the potential SA-Keisou parameters deemed to be potentially effective are tentatively identified as "candidate" parameters. In the second step, the final SA-Keisou parameters are determined based on their effectiveness in measuring the plants' condition and the operability of the instrumentation system for monitoring the plants' condition.

The candidate parameters were identified from the analysis of the accident management guide (AMG), the analysis of the TF-1 accident and a survey of world major codes, regulations and standards (see Refs [IV–3], [IV–4], [IV–5], [IV–6] and [IV–7]. Table IV–1 and IV–2 show examples of the selected candidate parameters for BWR and PWR. The letters A, B and C refer to the selection reasons. The notations in the column "Selection Reason" denote the following:

- A: Variable identified based on an extraction from AMG;
- B: Variable identified based on a reflection of overseas reactor plant knowledge;
- C: Variable identified based on an analysis of the TF-1 accident.

For example, the RPV water level (reactor water level) is selected through the analyses of the AMG and the TF-1 accident. The purpose of this measurement is to mainly confirm the core cooling conditions. Finally, 29 parameters were selected as SA-Keisou candidate parameters for BWR. Table IV–1 shows typical parameters that were selected for BWR.

TABLE IV–1. EXAMPLES OF THE SELECTED CANDIDATE PARAMETERS FOR BWR

No.	Parameter	Selection reason	Purpose of measurement
1	RPV water level	A,C	Confirm core cooling conditions, etc.
2	D/W water level	A,B,C	Confirm operation of PCV vent, etc.
3	D/W pressure	A,B,C	Confirm operation of PCV vent, etc.
4	D/W dose rate	A,B,C	Confirm fuel cladding damage and fuel melt, etc.
5	D/W,S/C temperature	A,B,C	Confirm operation of PCV spray, etc.
6	R/B area radiation monitor	A,C	Confirm release of fission products from CV, etc.
7	R/B hydrogen concentration	C	Confirm the integrity of PCV

The "final" SA-Keisou parameters are then selected from the candidate parameters. To select the final set of parameters for the SA-Keisou parameters, the following criteria were used for confirming plant state and equipment operation:

- Parameters that facilitate accident management strategies for preventing RPV/RV damage;
- Parameters that facilitate accident management strategies for preventing RCV/CV damage; and

- Parameters that facilitate accident management strategies to suppress off-site radioactive release when RPV/RV or RCV/CV is damaged.

The SA-Keisou parameters necessary to confirm plant state and equipment operation are then selected. The following are considered for this selection process:

- Selection of parameters that facilitate obtaining specific measurements during each stage of the severe accident; and
- Selection of parameters that facilitate obtaining the above described measurements within the required accuracy and response time.

As a result, several new parameters including an expansion of the instrumentation range were identified. Twenty one parameters were selected for SA-Keisou parameters for BWRs and PWRs respectively. Note that these parameters may change in the future. Table IV–2 shows typical parameters that were selected for PWR.

TABLE IV–2. EXAMPLES OF THE SELECTED CANDIDATE PARAMETERS FOR PWR

No.	Parameter	Selection reason	Purpose of measurement
1	RV water level	A	Confirm core cooling conditions
2	ICIS thimble tube room water level	B	Confirm cooling conditions of debris
3	CV pressure	A	Confirm core damage
4	High range area radiation monitor	A	Confirm integrity of CV
5	CV temperature	A	Confirm CV damage
6	Monitoring post	A	Confirm release of fission products from CV.
7	CV hydrogen concentration	B	Confirm hydrogen initiation and combustion in CV

In the AMGs, the emergency operating procedures have been structured to enable the plant operators to respond after monitoring the plant symptoms that can lead to damage of the core, the RPV and the PCV. A similar process is then applied to emergency procedures applicable to unidentified accident events. It is expected that the severe accident management guidelines (SAMGs) will be formulated based on the same organizational guidelines. For this reason, a set of plant states corresponding to the various stages of accident progression was identified and classified to enable a coherent evaluation of equipment performance in light of its ability to survive the applicable conditions. This classification of severe accident plant states is called the severe accident (SA) classification.

For severe accident monitoring, instrumentation parameters are necessary to provide the information to the plant operator to assess the plant conditions and to permit manual action. The severe accident classifications are defined based on the plant conditions after core damage as follows:

- SA1 is the condition where the reactor core is damaged, but the core fuel remains inside (in-vessel retention);
- SA2 is the condition where a RPV/RV failure has occurred, and the core has relocated to outside the RPV/RV;
- SA3a is the condition where a PCV/CV failure has occurred, but water has been successfully injection within 24 hours after the scram; and
- SA3b is the condition where a PCV/CV failure has occurred and efforts to inject water prior to 24 hours after the scram have failed, but after 24 hours have passed successful injection of water occurs.

The environmental conditions were evaluated using the MAAP code based on the SA classification and the event trees for the severe accidents. The matrix of the severe accident classification was developed. The matrix shows the evaluation results of the environmental conditions, the measurement purposes and the instrumentation parameters per the severe accident classification (SA1, SA2, SA3a and SA3b). The environmental conditions corresponding to the severe accident classification are evaluated. The parameters which indicate the environmental conditions are as follows:

- Temperature;
- Pressure; and
- Accumulative dose.

The anticipated environmental conditions in the RPV and the PCV for BWRs are shown in Table IV–3. The environmental conditions associated with SA2 classification state are a maximum temperature of 300°C, a pressure of 1MPa, and an integrated dose of 5MGy/6 months.

TABLE IV–3. ENVIRONMENTAL CONDITION OF RPV AND PCV FOR BWR

SA classification state	SA1	SA2	SA3a	SA3b
Fuel condition / Fuel position	Meltdown / within RPV	Debris / RPV or PCV	Debris / RPV or PCV	Debris / RPV or PCV
Core condition RPV condition PCV condition Water injection	Damaged Sound Sound Success	Damaged Damaged Sound Success	Damaged Damaged Damaged Success	Damaged Damaged Damaged Failure
Max temperature (^{0}C)	171 (PCV) 302 (RPV water) 500 (RPV gas)	300	700	1000
Pressure (MPa)	0.31 (PCV) 8.62 (RPV)	1.0	1.0	1.0
Radiation (MGy)	5	5	5	5

During the qualification of RPV level instrumentation sensors, the following conditions are tested/evaluated including:

- Threshold temperature;
- Threshold pressure;
- Mission time;
- Resistance to steam;
- Seismic capacity;
- Threshold integrated dose;
- Threshold radiation dose rate;
- Resistance to corrosion;
- Resistance to poisonous substances, for example, Iodine, Cesium Iodide, Carbon Monoxide.

The instrumentation sensors have to withstand the environmental conditions described in Table IV–2.

For two of the instrumentation systems which had been developed and qualified in the Japanese National Project, one is being installed and the other has already been installed into units 6 and 7 at the Kashiwazaki-Kariwa Nuclear Power Plant of Tokyo Electric Power Company Holdings, Inc. Those are the hydrogen concentration monitoring system within the PCV and the water level monitoring system for the lower part of the PCV [IV–8].

The hydrogen concentration monitoring system has a hydrogen concentration instrument currently being installed into the PCV in order to monitor the hydrogen concentration in the anticipating changeable range of the hydrogen concentration level when a reactor core is seriously damaged. The hydrogen concentration instrument is designed in a way so that it enables operators to monitor the hydrogen concentration within the PCV in the Main Control Room (MCR) by using the power supply from a back-up power source.

The water level monitoring system for the lower part of the PCV has a water level instrument which has already been installed into the PCV in order to monitor the water level when injecting water into the PCV. The water level instrument is designed in a way so that it enables operators to monitor the water level within the PCV in the MCR by using the power supply from a back-up power source.

The national programme for the development of instrumentation systems for severe accidents accounts for results from the manufacturer's collaborative effort in the framework of safety enhancement programme for LWR carried out by the Agency for National Resources and Energy in Japan.

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ANNEX V

EXAMPLES OF SEVERE ACCIDENT CLASSIFICATION MATRICES AND DEVELOPMENT OF RPV WATER LEVEL SENSORS FOR BWR

V–1. EXAMPLE OF SEVERE ACCIDENT CLASSIFICATION MATRIX

The following section provides some examples of the severe accident (SA) classification matrix for RPV water level instrumentation for a BWR. This section is reproduced with the permission of Hitachi-GE Ltd, Japan.

An example of the severe accident classification matrix for RPV water level instrumentation for a BWR is shown in Fig. V–1. The purpose of each measurement is described for each severe accident classification. For example, confirmation of core cooling condition is needed for SA1. Therefore, the reactor water level parameter in the RPV is selected to provide this indication.

The symbols shown describe the importance of the SA-Keisou (Instrumentation) parameters: a double circle (a symbol '◎') identifies the main parameter used for providing a measurement, a single circle (a symbol '○') identifies supporting parameters which provide supplemental information with respect to the measurement and a triangle (a symbol '▲') identifies the parameters needed to determine when to execute a strategy or measure, and for confirming the success of the implemented strategy or measure. The water level in the RPV is needed to confirm that the effectiveness of core cooling which is a critical parameter. Note that the table may change in the future.

SA Classification		SA1											SA2									
Measurement Purposes / Performance		Monitoring					Operation Confirmation						Monitoring				Operation Confirmation					
		Confirm fuel boundary damage and meltdown	Confirm core cooling condition	Confirm integrity of reactor pressure boundary (RPV)	Confirm integrity of PCV	Confirm hydrogen initiation and concentration within PCV	Injection to core	Depressurize RPV	Injection to pedestal	Makeup to water source tank	PCV spray	PCV vent	Confirm damage of reactor pressure boundary (RPV)	Confirm cooling of core debris	Confirm integrity of PCV	Confirm hydrogen initiation and concentration within PCV	Injection to core	Injection to pedestal	Makeup to water source tank	PCV spray	Injection to reactor well	PCV vent and filtered vent
1	RPV water level	○	◎	○		○	▲	▲	▲				○									

FIG. V–1. Severe accident classification matrix for RPV water level instrumentation in a BWR (figure courtesy of Hitachi-GE).

V–2. DEVELOPMENT OF RPV WATER LEVEL SENSOR FOR BWR

The instrumentation systems for the severe accidents were developed during a national project. The RPV water level instrumentation system for BWRs was manufactured and tested under a defined set of severe conditions as shown in Ref. [V–1]. The severe accident classification SA1 and SA2 were applied to the reactor water level instrumentation system.

The purpose of the reactor water level instrumentation system is shown in Fig. V–2. Independent thermocouples with heaters are installed inside existing in-core instrumentation tubes and measure the temperature rise caused by heating. This system uses the heat transfer coefficient difference between steam and water. When heated by the same amount, the temperature rise in steam and water differ. By measuring the temperature rise, this sensor can distinguish between steam and water and is able to measure the RPV water level. This method is advantageous, because it uses independent type sensors. Lower position sensors can operate even if the upper position sensors are damaged. Therefore temperatures in the reactor can be obtained by the thermocouple.

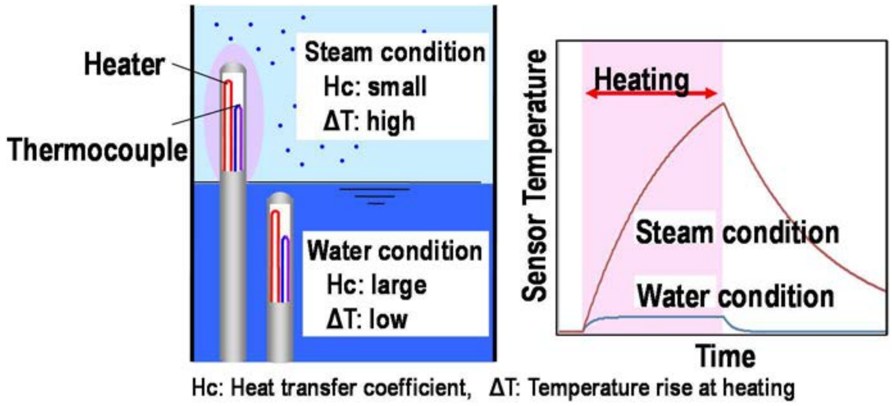

FIG. V–2. Measurement principal for reactor water level (figure courtesy of Hitachi-GE).

The confirmation test for measuring the water level under these severe conditions was implemented using a test facility. The test results are shown in Fig. V–3. and Fig. V–4. As for the test conditions, the water and steam temperatures were changed from room temperature to 300^{0}C in increments of 50^{0}C. Figure V–3 shows the result of the temperature rise by heating to 300^{0}C. The red line shows the temperature rise in steam and the blue line shows the temperature rise in water. The results show temperature rise in steam is 9.7^{0}C and temperature rise in water is 1.9^{0}C after 10s of heating every minute.

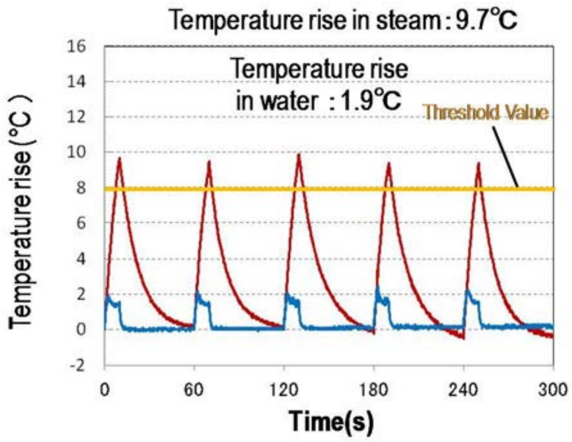

FIG. V–3. Result of temperature rise by heating to 300^{0}C (figure courtesy of Hitachi-GE).

Figure V–4 shows the result of temperature rise when heated to different temperatures. The red points show the temperature rise in steam and the blue line shows the temperature rise in water. At each temperature level, the sensor can distinguish between steam and water by a threshold value of 8^{0}C.

These results show that it is possible to determine if a sensor is immersed in-water or in-steam from room temperature to 300^{0}C. It was confirmed that the system is able to distinguish between the in-water and in-steam, and able to measure the RPV water level. The confirmation and verification tests for the instrumentation systems have been completed.

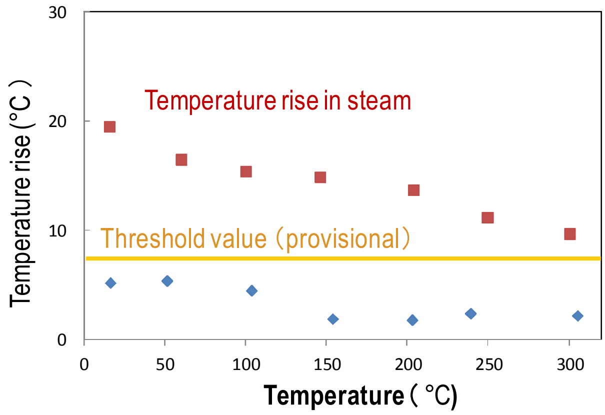

FIG. V–4. Result of temperature rise by heating to different temperatures (figure courtesy of Hitachi-GE).

This study is a part of the collaborative effort of manufacturers that has been carried out in the framework of safety enhancement programme for LWR plants by the Agency for National Resources and Energy in Japan. The results of tni study has been published in Ref. [V–2].

REFERENCES TO ANNEX V

[V–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Monitoring Systems For Nuclear Power Plants, Nuclear Energy Series No. NP-T-3.16, Vienna (2015).

[V–2] SHOHEI WADA, AKIRA MURATA, SETSUO ARITA, Development of Instrumentation System for Severe Accidents, Proceedings of the 22nd International Conference on Nuclear Engineering ICONE22, Prague, Czech Republic (2014).

ANNEX VI

CONDUCT OF QUALIFICATION OF SEVERE ACCIDENT INSTRUMENTS FOR NUCLEAR POWER PLANTS IN THE FRAMEWORK OF JAPANESE NATIONAL PROJECT

VI–1. BACKGROUND

The qualification of severe accident instrumentation for nuclear power plants in Japan is performed in the framework of the Japanese national project. This section is reproduced with the permission of Hitachi-GE Ltd, Japan.

The qualification of severe accident instrumentation for nuclear power plants involves the following elements:

- Establishing environmental conditions;
- Determination of basic specification of severe accident instrumentation;
- Verification test methods for severe accident instrumentation.

Emergency operating procedures have been structured to enable the plant operators to respond to plant symptoms that can lead to damage of the core, the reactor pressure vessel (RPV) in BWR or the Reactor Vessel (RV) in PWR, and the primary containment vessel (PCV) in BWR or the Containment Vessel (CV) in PWR.

In order to establish environmental conditions expected during a severe accident, a set of plant states corresponding to the various stages of accident progression was identified and classified to enable a coherent evaluation whether the equipment is able to survive accident conditions. A classification of severe accident plant states is called severe accident classification.

The accident management guidelines provide accident management strategies for prevention and mitigation of damage to the RPV (RV) and subsequent damage to the PCV (CV). An early water injection after core damage is proposed to prevent damage to the RPV (RV) and subsequent damage to the PCV (CV). Based on lessons learned from Fukushima, a set of severe accident plant states was defined as follows:

- SA1: The reactor core is damaged, but the fuel remains inside the RPV (RV);
- SA2: An RPV (RV) failure has occurred; and the fuel is outside the RPV (RV);
- SA3a: A PCV (CV) failure has occurred (success of the water injection within 24 hours after the scram);
- SA3b: A PCV (CV) failure has occurred (e.g. failure of the ability to inject water within 24 hours after the scram, but after that the water injection is successful).

Various plant states may occur after damage to the RPV (RV); for example SA3a includes accidents similar to Fukushima, and SA3b corresponds to various events which are considered to be beyond SA3a. The difference between plant sub states SA3a and SA3b is whether the water injection at 24h after a reactor trip was successful or not. A classification state for different severe accident conditions is shown in Table V1–1.

A classification of severe accident states intends to help identifying design criteria for accident monitoring instrumentation that is needed for support of the mitigation strategies. For example, although the temperature in the PCV (CV) may be low enough to enable the equipment to function, access to the PCV (CV) to replace inoperable equipment may still be restricted due to high radiation.

TABLE VI–1. DEFINITION OF SEVERE ACCIDENT CLASSIFICATION STATES

Severe Accident Classification State	SA1	SA2	SA3a	SA3b
Fuel condition /Fuel position	Meltdown /within RPV	Debris /RPV or PCV	Debris / RPV or PCV	Debris / RPV or PCV
Core condition	Damaged	Damaged	Damaged	Damaged
RPV condition	Sound	Damaged	Damaged	Damaged
PCV condition	Sound	Sound	Damaged	Damaged
Water injection	Success	Success	Success	Failure

Two reactor designs were considered in the framework of the Japanese national project; BWR and PWR.

For a BWR, the environmental condition to which instrumentation is exposed in the PCV becomes extremely harsh during a severe accident. The environmental condition in the PCV becomes severe when both the ratio of the suppression pool water volume per the amount of power output and the ratio of the PCV capacity per the amount of power output are small. From analytical results, a Mark-II type containment and a reinforced concrete containment vessel were chosen as representative reactor types for which the environmental conditions were established.

For a PWR, a large dry containment vessel type was chosen. The rationale for this selection is the following characteristics: the ratio of the free volume amount of the containment vessel per the amount of power output generations of the reactor core is relatively small, the pressure and temperature within the containment vessel is prone to be high because of its high pressure resistance, and the radiation dose amount within the reactor core is large.

VI–2. DEFINITION OF ENVIRONMENTAL CONDITIONS

The environmental conditions (temperature, pressure, radiation, and humidity) were determined considering real values and data from actual measurements during the Fukushima Daiichi accident (including estimating values), the analysis of the representative accident scenario, and any other relevant analytical data.

The BWR equipped with a Mark-II or a reinforced concrete containment vessel was chosen as a representative reactor type for determining the environmental conditions in different parts of the containment vessel where severe accident instrumentation is located. A Mark-I and a Mark-I improved containment design were also considered as representative types.

Based on the external events for determining the conditions within the PCV for use of mitigating severe accidents, the scenarios were determined considering impact of external events to containment vessel conditions. These scenarios consider possible time delay for executing water injection from the outside as it is shown in Fig. VI–1. The scenarios were analysed by using the MAAP code and the environmental conditions were determined within the PCV and within the secondary containment area. A release ratio of radioactive materials released instantaneously into the PCV was estimated according to NUREG-1465.

The radiation dose rate was evaluated at this condition. The severe accident environmental conditions for BWR are shown in Table VI–2.

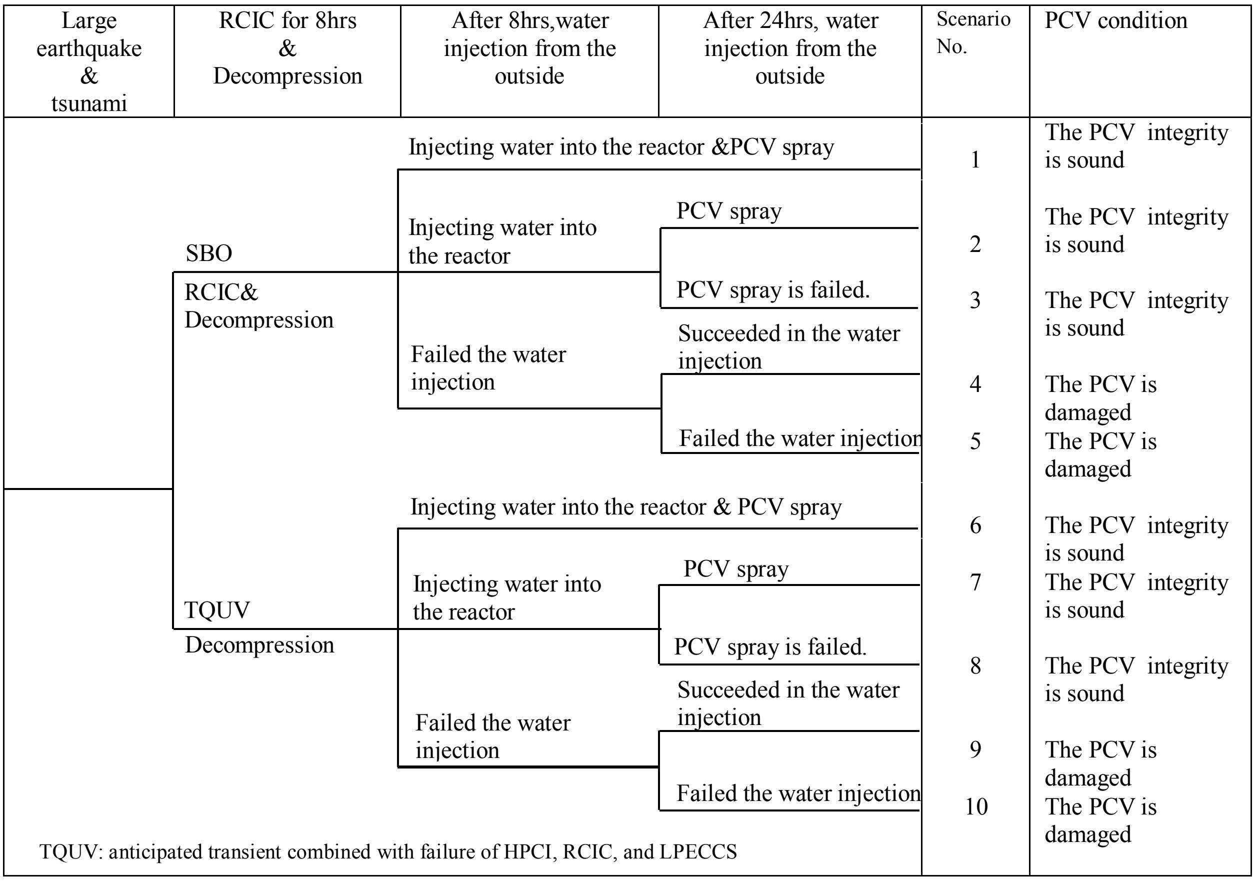

FIG. VI–1.An event tree to establish the criterion within PCV for use in severe accident mitigations.

TABLE VI–2. AN EXAMPLE OF SEVERE ACCIDENT ENVIRONMENT FOR BWR

Plant condition / PCV environmental conditions	SA1 Fuel damage ↓ Melt RPV integrity and PCV integrity is sound	SA2 Fuel melting RPV is damaged PCV integrity is sound	SA3a Fuel melting RPV is damaged PCV is damaged	SA3b Fuel melting RPV is damaged PCV is damaged
– Max-temperature	– 171 ^{0}C	– 300 ^{0}C	– 700 ^{0}C	– 1000 ^{0}C
– Pressure	– 0.31 MPa	– 1.0 MPa	– 1.0 MPa	– 1.0 MPa
– Humidity	– Water vapor	– Water vapor	– Water vapor	– Water vapor
– Radiation	– 5x10^6 Gy/6 months	– 5x10^6 Gy/6 months	– 5x10^6 Gy/6 months	– 5x10^6 Gy/6 months
Environmental conditions				
– Max-temperature	– 66 ^{0}C	– 66 ^{0}C	– 100 ^{0}C	– 100 ^{0}C
– Pressure	–3.4 kPa	– 0.01 MPa	– 0.01 MPa	– 0.01 MPa
– Humidity	–100%	– Water vapor	– Water vapor	– Water vapor
– Radiation	–3x10^5 Gy/6 months	– 3x10^5 Gy /6 months	– 2x10^6 Gy /6 months	– 2x10^6 Gy /6 months
Measurement duration		More than 3 days		N/A

Note: Table VI–2 may be subject to change in future.

The environmental conditions for a representative PWR designed equipped with a large dry containment vessel were analysed by using the MAAP code. The event tree for determination of criteria used in analysis is shown in Fig. VI–2.

The intensity of the radiation source was determined and the radioactive materials released into the containment were evaluated according to different categories such as those which were released into the containment vessel, those which were floating, and those which were deposited on the inner surface of the containment vessel and equipment.

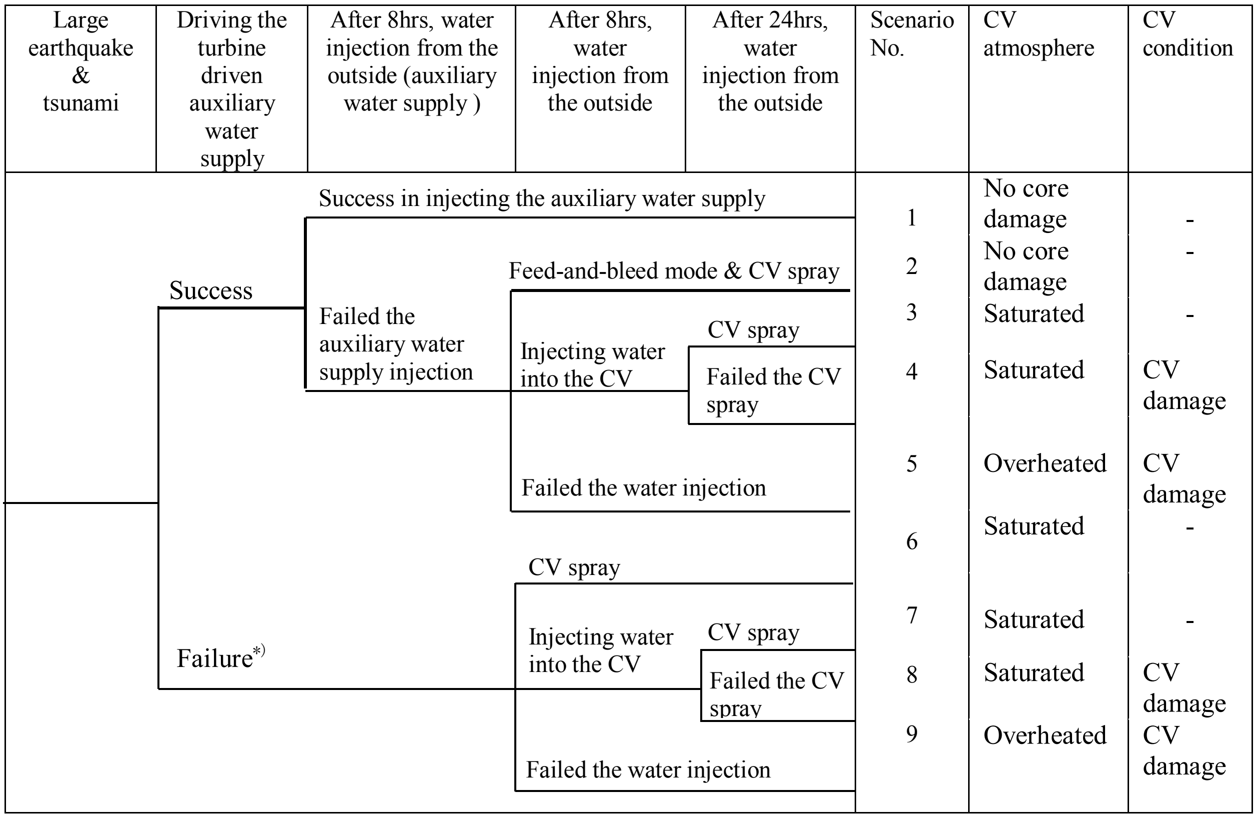

FIG. VI–2. An event tree for pressurizing and overheating event (SBO) within PWR containment vessel.

A turbine driven auxiliary feedwater pump has failed. The event would result in core damage in a time span of several hours with subsequent reactor vessel failure. The mitigating strategies to prevent or to slow down the accident progression by injecting the auxiliary water supply could not help much.

The radiation dose rate and accumulated radiation dose were evaluated considering source term from the corium (molten core materials) released into the containment. The environmental conditions within the containment during a severe accident were identified accordingly. The severe accident environmental conditions for PWR are shown in Table VI–3.

TABLE VI–3. SEVERE ACCIDENT ENVIRONMENTAL CONDITIONS FOR PWR

Plant condition (SA) / Environmental conditions within CV	SA1 Fuel damage ↓ Melt RV integrity is sound CV integrity is sound	SA2 Fuel melting RV is damaged CV integrity is sound	SA3a Fuel melting RV is damaged CV is damaged	SA3b Fuel melting RV is damaged CV is damaged
– Max-temperature	– 190 ^{0}C	200 ^{0}C	– 200 ^{0}C	– 300 ^{0}C
– Pressure	– 0.414 MPa	1.6 MPa	– 1.6 MPa	– Atmospheric pressure
– Humidity	– 100%	100%	– 100%	– 100%
– Radiation	– Below the conventional PAM's environmental conditions	– 2MGy/year (an annular space is 5MGy/year)	– 2MGy/year (an annular space is 5MGy/year)	– 2MGy/year (an annular space is 5MGy/year)
Environmental conditions outside CV				
– Max-temperature	– Ambient temperature	– Depends on the installed location	– Depends on the installed location	– Depends on the installed location
– Pressure	– Atmospheric pressure	– Atmospheric pressure	– Atmospheric pressure	– Atmospheric pressure
– Humidity	–—	– —	– —	– —
– Radiation	–—	– Depends on the installed location	– Depends on the installed location	– Depends on the installed location
A required functional duration		More than 80 hours		N/A

Note: Table VI–3 may be subject to change in future.

VI–3. DETERMINATION OF SPECIFICATIONS FOR SEVERE ACCIDENT INSTRUMENTATION

VI–3.1. Determination of process conditions for each targeting parameter

Severe accident instrumentation parameters were categorized, grouped by process types (liquid, gas, and solid types) for the measurement objects, and then investigated for each of the process conditions to determine the following:

- Required ranges and their rationale;
- Instrument mission time;
- Environmental conditions (temperature, pressure, radiation, humidity).

The instrumentation process conditions such as temperature, pressure, and measuring parameters were set up and organized for every process type in postulated plant conditions according to severe accident classifications.

Instrumentation systems for measuring a specific parameter are composed of several configuration parts; in some cases the environmental conditions may differ because the installed locations for each of the configuration parts may be different.

The basic specifications were investigated at specific locations, which are needed for composing devices/equipment with regards to each severe accident instrumentation parameter. Consequently, the devices/equipment were clearly defined for each parameter as well as where the devices/equipment were installed, e.g. inside and outside of the PCV and containment vessel. The radiation conditions, heat resistance, and waterproofing capability were established for each instrument.

VI–3.2. Establishing basic specifications for severe accident instrumentation

Items defined as basic specifications necessary for a reliable performance of instrumentation under severe accident conditions were identified as follows:

- Measurement range;
- Instrumentation loop accuracy;
- Response time;
- Mission time;
- Redundancy/diversity;
- Independence (including power sources);
- Seismic resistance;
- Environmental resistance;
- With or without indicator /recording;
- Power sources to be used;
- Maintenance.

VI–4. TEST METHODS FOR SEVERE ACCIDENT INSTRUMENTATION

VI–4.1. Methodology for verification

The basic principles related to testing methods are categorized as follows:

- Scope;
- Test planning;
- Test methods;
- Test outline;
- Test evaluation methods.

Testing scope determines the suitability of severe accident instrumentation for use in an actual plant. Thus, the testing scope is set up to confirm that the instruments are produced to meet their design criteria.

VI–4.2. Test planning

Test planning is established considering basic principles as follows:

- Development goals (performance, specifications, environmental conditions, and others) have to be clearly identified;
- Confirmation items categorized as element testing, basic testing, and qualification testing;
- Determination of target range and the confirmation range to be used for establishing the test conditions.

Environmental conditions, such as temperature, pressure, and radiation, under severe accidents are set up for each instrument considering plant conditions (SA1-SA3b) and its installed location. The measuring specifications, such as measuring ranges and measuring accuracies are set up to reflect conditions during a severe accident.

Table VI–4 shows the individual items (instrumentation) to be tested are categorized into three groups: element testing, basic testing, and qualification tests.

The instruments which measure the same kind of parameter may include similar confirmation items because of similar element specifications except for the conditions specific to reactor types.

TABLE VI–4. TEST ITEMS TO BE CONFIRMED AT EACH TEST PHASE

Test	Definitions of test	Test object	Common test items
Element testing	This is an element level (a part unit level) test to confirm the feasibility of the measuring principle and to confirm the resistance of an element and others at the phase prior to the manufacturing sensors.	Sensor elements	1. Confirming the feasibility of basic performance 2. Confirming basic characteristics 3. Confirming influencing impact against test conditions (environmental resistance conditions)
Basic testing	This is a confirming test to verify that a sensor meets the required specifications by testing a unit level prior to conducting qualification testing of a system level which is equivalent to an actual plant.	Sensor and other items	1. Confirming the feasibility of basic performance 2. Confirming basic characteristics 3. Confirming heat resistance, pressure resistance, radiation resistance and others (confirming the stability against environmental resistance conditions, generally, confirming items under simple environmental conditions)
Qualification testing	This confirming test aims to confirm the integrated impacts on an instrument since such impacts cannot be confirmed by testing of a sensor unit	1. Sensors and others items 2. Confirmation, in general, may be evaluated over the entire instrument	1. Confirming the environmental resistance(in general, under the multiplex environment conditions similar to the actual plant environment) 2. Confirming seismic resistance 3. Confirming feasibility of a system*

*Note: This test item has to be conducted only when a confirmation is needed because of the characteristic of a sensor.

VI–4.3. Test methods

Test methods are aimed to verify the achievement of development goals by testing. It is possible to replace or complement these tests with available data, by utilizing relevant past verification results and operational experiences.

The basic test principles are established as follows:

- Test specimens selected for testing are sensors and other devices which are equivalent to actual plants. Tests have to be conducted, in principle, by using multiple test specimens prepared for such tests to increase their performance and testing accuracy.
- The confirming items to assess the suitability are as follows:
  - Measuring ability during a severe accident (performance under assumed operating environmental conditions);

- Environmental resistance (heat resistance, radiation resistance, pressure resistance, etc.);
  - Seismic resistance;
  - System feasibility.
- Simulation of aging degradation performed during environmental resistance tests and results are assessed against relevant factors such as materials used for individual severe accident instruments, effectiveness and efficiency of the tests.
- Testing procedure does not allow upgrading test specimens; the same specimens are used in the testing steps following pre-aging simulation.
- Testing conditions have to preferably cover all test specification items which are to be confirmed. If an item of the test specification cannot be performed because of limiting factors (e.g. test conditions, test duration), appropriate justification and testing within reasonable ranges is allowed. However, the suitability of such test methods need to be demonstrated.

Acceptance criteria are set up with consideration to inaccuracies / uncertainties of the testing facility (e.g. capability and accuracy of measurement devices). Test results of instrumentation which are equivalent to the tested specimens, such as tests performed by manufacturer may be used for this test purpose.

If instrumentation which is known to be equivalent to the tested specimen, and which could maintain its functions under more severe environmental conditions than the proposed test conditions, it is considered confirmed.

If the past verification results do not meet the conditions which need to be confirmed for a tested specimen, additional tests or analyses need to be conducted to confirm instrument's performance and environmental resistance during severe accidents.

Operating experience collected at actual plants for specific instruments can be used to confirm the environmental resistance. If operating experience cannot confirm the instrument environmental resistance, additional tests or analyses need to be carried out.

VI–4.4. Test outline

A test specimen, e.g. a sensor which is equivalent to the actual plant, can be used. The qualification is conducted to cover the entire instrumentation chain including any transducer/converter. As for the instrument parts which are not exposed to severe accident conditions (e.g. transducer/converter installed outside the containment vessel), a confirmation of necessary characteristics (e.g. electric properties) for an actual plant needs to be conducted.

Type tests need to involve several specimens of the same type and kind in order to achieve representative test results. In special and justified cases, a single specimen can be tested.

When selecting instruments (e.g. sensors for severe accident conditions), the following minimum evaluation is needed:

- Instrument performance during severe accident conditions (performance under the assumed operational environmental conditions);
- Environmental resistance (heat resistance, radiation resistance, pressure resistance);
- Seismic resistance;
- Feasibility.

VI–4.5. Test procedure

Test procedures involve several test steps, e.g. specimen pre-aging, testing to harsh environmental conditions, etc. The test procedure typically includes the following confirmation steps:

— Confirmation of specimen performance; this test is conducted under the normal operating environmental conditions in which a test specimen is normally operated. This is to get a reference performance of the specimen (i.e. accuracy, response time, etc.).
— Confirmation of specimen basic characteristics; this involves tests to understand the performance characteristics of a test specimen in a single (in some cases, multiple) environmental impact. In addition, a stability confirmation test may be included. This test is conducted to simulate the impact of the individual environmental parameter (e.g. high temperature, high pressure) that is part of the entire environmental conditions during severe accidents.
— Specimen in service degradation (pre-aging); this test simulates in service aging of a specimen under the normal operating conditions anticipated throughout its qualified life.
— Seismic resistance; a test specimen has to be tested for vibrations simulating expected acceleration under which the specimen is supposed to function. The specimen has to withstand the earthquake loads without any special requirement for its functional performance (some time could be necessary to test functionality during a seismic test); however the specimen has to maintain its functions following the seismic event. As a general rule, seismic qualification is conducted on a test specimen that has been pre-aged unless there is a rational justification that it can be excluded.
— Specimen environmental qualification; a test specimen is tested in similar environmental conditions anticipated during severe accidents. These simulated conditions have to envelop the conditions determined by relevant analysis such as severe accident analysis. Duration of the test needs to be longer than the time for which the specimen needs to function. A test specimen irradiation to the equivalent severe accident conditions can be conducted in the pre-aging phase. Tests for environmental conditions equivalent to the severe accidents, such as temperature/pressure/humidity, can be conducted after the irradiation test. After completing all phases of environmental testing, a confirmation of specimen functional performance is conducted to confirm that the specimen has met defined acceptance criteria.

VI–4.6. Test conditions

Test conditions under which a specimen is tested have to comply with qualification specifications. The test conditions need to be such that to demonstrate the specimen's capability under severe accident conditions. If the test conditions cannot be accomplished due to limitation of a testing facility, in such cases it is necessary to confirm that the acceptance criteria can be met by extrapolation methods or other means. A justification of reasons and use of alternative testing methods have to be provided.

Appropriate consideration of tolerance errors for the instrument chain needs is necessary. For example, if measuring temperatures shows an error within $\pm 5^0$C, the test specimen needs to operate in a way that the tolerance has a margin of more than 5^0C added to the maximum temperature.

ANNEX VII

APPROACH FOR THE ASSESSMENT OF ELECTRICAL EQUIPMENT WHICH HAS TO PERFORM RELIABLY UNDER SEVERE ACCIDENT CONDITIONS (APPLIED IN THE FRAME OF THE OLKILUOTO-3)

VII–1. GENERAL ASSESSMENT APPROACH

An assessment of equipment reliable performance under severe accident conditions has been included in the frame of the Olkiluoto 3 general qualification plan for electrical equipment, which is shown in Fig. VII–1. This general qualification plan identifies all different working areas for which dedicated reports (numbered in square brackets in Fig.1) are prepared as follows:

[1] General qualification procedures of electrical equipment;
[2] Qualification against induced vibrations;
[3] Qualification for harsh environment and increased ambient conditions;
[4] Software qualification;
[5] EMC plan and assessment procedure.

Figure VII–1. outlines two major inputs that are necessary for performing the assessment as follows:

— Plant specific input: This input is derived from “Stipulating Reports” which describe the environmental conditions including the loads of induced vibrations, and the required safety functions of systems, subsystems and equipment needed for the mitigation of the event. As it can be seen from the figure Fig. VII-1, four different events are recognized: seismic event (and/or airplane crash), emergency power mode, loss of coolant accident and severe accident (these two events lead to harsh environmental conditions for the equipment in scope).
— Equipment specific input: The electrical equipment specification covers all properties which are relevant for characterizing the specific type of equipment. Furthermore, it defines the equipment safety class, the anticipated environmental conditions and the required safety function(s).

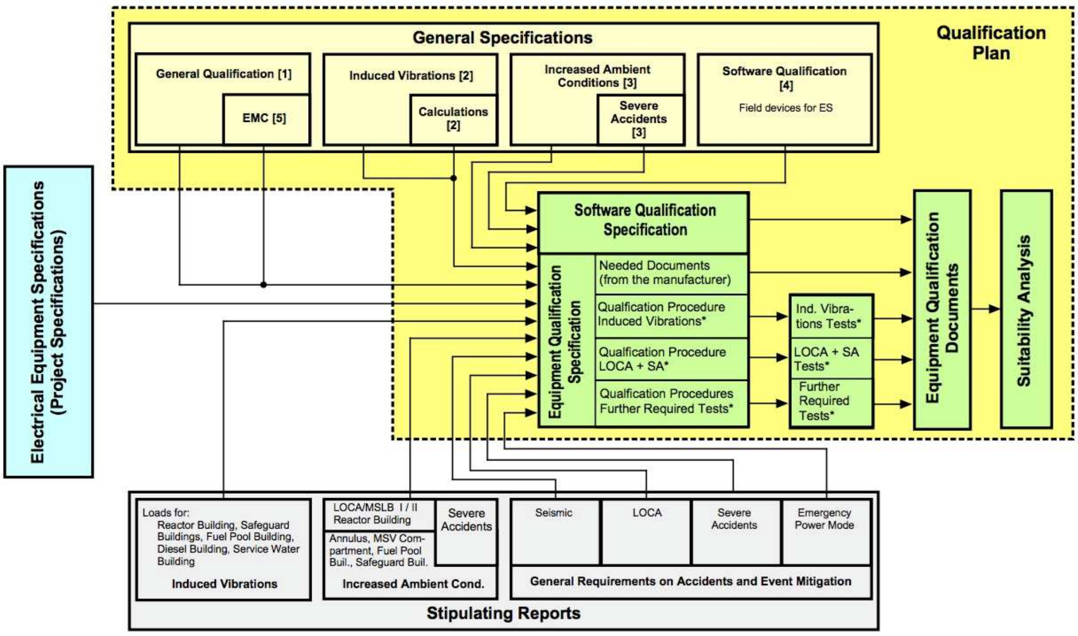

FIG. VII–1. General qualification plan for electrical equipment for Okliluoto-3 (figure courtesy of AREVA).

Based on the input information the qualification documents are elaborated. These documents can be classified into two groups:

(1) Documents that define the main processes and technical boundary conditions for the qualification in the different working fields, such as general specifications for the general qualification process including electromagnetic compatibility, environmental qualification including severe accident, qualification against induced vibrations and software qualification. In case of environmental qualification, they also define enveloping loads during normal operation and accident conditions at particular plant zones. These documents respect the Member States experience as well as standards which are typically issued by IEC TC45/SC45A standards and KTA rules.
(2) Documents that define the equipment specific qualification have to respect all the three sources of information (e.g. stipulating reports, equipment specification, and general specifications). The main document is called equipment qualification specification that provides the equipment specific qualification programme. It describes the following aspects in detail:

- Characteristics of the equipment in scope;
- Qualification requirements (environmental and functional);
- Qualification approach (what has to be considered and how);
- Qualification steps;
- Documentation requirements.

All other documents downstream the suitability analysis, are based on the equipment qualification specification, in particular when the equipment is exposed to severe accident environmental conditions. These documents include detailed test specifications, e.g. for loss of coolant accident and severe accident, and reports elaborated by the test laboratories. The suitability analysis summarises facts showing that the equipment is capable to fulfil the safety function under these conditions.

The equipment manufacturer’s documentation provides supporting information on the quality of the equipment, which together with qualification testing reports provide reasonable assurance that equipment is qualified and suitable for its intended use in harsh environmental conditions. In addition, the equipment manufacturers are subject to a quality assessment and quality audits.

VII–2. CONCEPT FOR COVERING ENVIRONMENTAL CONDITIONS

The effects of operational states and accident conditions to be applied during testing are very much dependent on the equipment location and may vary significantly at different equipment location. In order to optimize the qualification process, the concept of qualification families has been developed has been developed. This concept defines enveloping environmental conditions for normal operational states and accident conditions, as well as intended mission times for equipment installed in specific plant environmental zones (e.g. room areas/buildings). The enveloping environmental conditions are derived from the worst case conditions including margins as specified in codes and standards. Consequently, each KKS [15] location exposed to harsh environment conditions is assigned to a qualification family. If the equipment is intended for mitigation of more than one event, it is dedicated to more than one qualification family.

The areas which can be influenced by the environmental conditions resulting from design basis accidents and severe accidents are:

[15] The abbreviations KKS are the plant identifiers for the specific room areas. The abbreviation KKS is derived from the German term “Kraftwerk–Kennzeichnungssystem”.

— The reactor building (UJA);
— The annulus (space between the containment and the outer shell, UJB);
— The safeguard building (contains the active mechanical equipment, in particular of the containment heat removal system, in the lower part, UJH);
— The fuel building (UFA);
— The main steam valve compartment/main feed water compartment (UJE).

The concept of qualifying the equipment families for different zones of the reactor building, annulus and safeguard building is illustrated in the Fig. VII–2. The letters marked bold red are the families related to severe accidents. Figure VII–2. shows zones in the plant which are affected differently by accident environmental conditions. While equipment in the containment is impacted directly and is exposed immediately to the effects of harsh radiation, temperature, humidity, chemical spray during stages of the accident progression, the equipment located in plant zones outside the containment is affected indirectly as a result of:

— Heat transfer through and radiation from structures and enclosure of active and passive mechanical equipment (pipes, valves, heat exchangers) and radiation. This typically occurs in rooms where the ECCS and containment heat removal system, the containment venting system and specific parts of the HVAC system are installed.
— Assumed pipe breaks in systems, which are needed for the monitoring and mitigation of an accident. The corresponding systems are typically the containment heat removal system and the sampling system. In contrast to conditions caused by the heat transfer and radiation only, these conditions are similar to accidents in the containment because they lead to pressure peaks and humidity loads with potential detrimental factors to the electrical and I&C equipment.

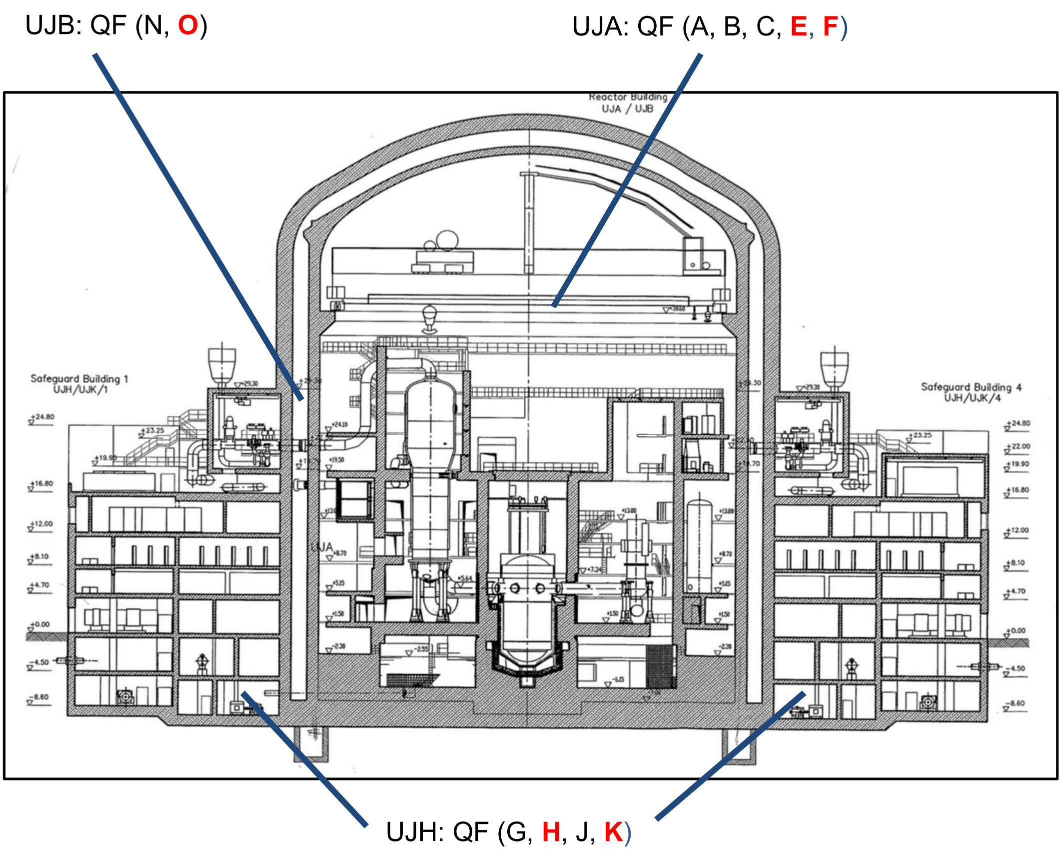

FIG. VII–2. Concept of the qualification families, assignment to room areas; severe accident qualification families are coloured in red (figure courtesy of AREVA).

The locations, where sensing devices (sensors) are installed, require special attention whether these are in direct contact with contaminated coolant. Temperature and radiation loads might be significantly higher than those conditions defined for the qualification family; therefore additional calculations/estimations may be necessary to provide data for the assessment of reliable performance under severe accident conditions.

A qualitative and partially quantitative characterization of severe accident conditions for qualification families are summarized in the Table VII–1. The values shown are the enveloping values. Depending on equipment installation location and equipment protective measures (e.g. shielding), other environmental loads may apply. These measures need to be applied individually for the equipment assigned to a specific KKS plant identifier. This may apply in particular for radiation doses. For example, an impact of β radiation can be significantly reduced by installing metallic enclosures/housing in order to protect sensors, actuators or by using conduits for cables.

TABLE VII–1. QUALITATIVE AND QUANTITATIVE CHARACTERIZATION OF QUALIFICATION FAMILIES

Qualification family/caused by	Area of the NPP (room areas)	Accident environmental conditions	Mission time
E - direct impact of core melt	Inside containment (UJA)	Elevated temperature, pressure, humidity and radiation Maximum temperature 156 °C, saturated steam conditions, Total integrated accident dose 800 kGy	100 Hours
F - direct impact of core melt	Inside containment (UJA)	Elevated temperature, pressure, humidity and radiation Maximum temperature 156 °C, saturated steam conditions, Total integrated accident dose 5000 kGy	1 year
H - break of the containment heat removal pipe during severe accident	Inside safeguard building (UJH)	Elevated temperature, pressure, humidity and radiation Maximum temperature 120 °C, saturated steam conditions, Total integrated accident dose 230 kGy	100 hours
K - consequences of the circulating water in the containment heat removal system	Inside safeguard building (UJH)	Elevated radiation Total integrated dose 180 kGy	1 year
O - consequences of the circulating water in the containment heat removal system No break of the containment heat removal pipe in the annulus is assumed (use of guarded pipes)	Inside annulus (UJB)	Elevated radiation Total integrated dose 180 kGy	1 year

VII–3. DEFINITION OF SYSTEMS AND EQUIPMENT NEEDED FOR SEVERE ACCIDENT

Once and entry point to severe accident management guidelines is reached, a priority is given to implementation of mitigation strategies for protecting the fission product boundaries. This includes implementation of the following measures:

— Avoiding or reducing the radioactive release into the environment;
— Maintaining containment integrity and bringing the plant to the controlled state;
— Restoring the safe state, i.e. subcriticality and long term decay heat removal from the molten core (core debris).

The implementation of above measures requires adequately reliable accident monitoring to check whether the accident progression is already on the severe accident mitigation path. The "accident mitigation path" is a feature of the EPR which is achieved by systems and instrumentation which are designed and qualified for monitoring and mitigation of a sever accident. If a severe accident evolves along with the "severe accident mitigation path" there is no challenge for the last fission product barrier. It cannot be however excluded that the severe accident progression deviates from this path. A reliable function of accident monitoring equipment is essential to implement severe accident mitigation measures so that to bring the plant back to the accident mitigation path.

The equipment which is necessary for monitoring and mitigation of severe accidents for Olkiluoto 3 EPR is identified through a process described in Fig. VII–3. This process identifies the equipment based on general and plant specific prerequisites. General prerequisites are derived from national codes and regulations (e.g. YVL), or international standards. Plant specific preconditions are derived from the plant safety analysis report and severe accident mitigation strategies.

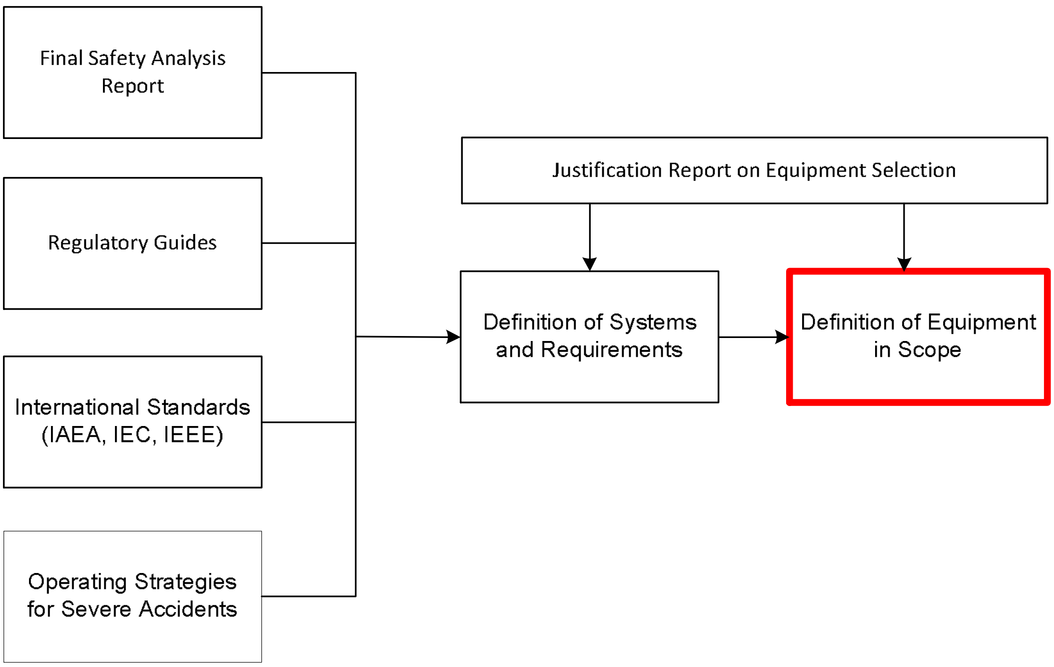

FIG. VII–3. Process for identification of equipment needed for severe accident monitoring (figure courtesy of AREVA).

For Olkiluoto 3 EPR, the following typical systems are identified for monitoring and mitigation of severe accidents:

— Detection and actuation of reactor coolant system depressurization;
— Detection of a corium position;
— Detection of re-criticality of the corium;
— Monitoring the containment integrity;
— Monitoring and mitigation of combustible gases;
— Mitigation of radioactive material releases.

Table VII–2 shows the equipment and its estimated mission time needed for monitoring and mitigation of a severe accident. Different parts of monitoring equipment are assigned to qualification families. A qualification test sequence is determined based on the equipment functional principle, intended safety function and mission time.

TABLE VII–2. THE EQUIPMENT NEEDED FOR SEVERE ACCIDENT MONITORING AND MITIGATION

Description	Task	Mission time	Location
Core outlet temperature	Activation signal for depressurization	0 h	In–core above the core (UJA), QF C
Temperature measurement using thermocouples in venting stack inside the containment	Indication of corium position	24 h	Between spreading area and upper containment (UJA) QF E* (shorten mission time)
Hydrogen concentration measurement	Measures the hydrogen concentration in several areas of the containment	100 h	Containment (UJA), QF E
Measurement containment pressure	Monitoring the containment integrity and checking the success of the containment heat removal	1 year	Containment (UJA), QF F
Measurement the level and temperature in the in-containment refueling water storage tank (IRWST)	IRWST is the source for the containment heat removal system (CHRS)	1 year	Containment (UJA), QF F
Pumps of the CHRS trains	Recirculation of coolant water from the core catcher to the spray system	1 year	Safeguard Building (UJH), QF K

VII–4. AN EXAMPLE OF POLYMER CABLE QUALIFICATION

A mineral insulated cable (MIC) is the preferred solution of a connecting devise when designing a system for severe accident monitoring, in particular when very high radiation load is expected.

Unlike polymer cables that have some advantages in cable routing and handling, attenuation values are higher in the mineral insulated cables.

Mineral insulated cables are also very sensitive against mechanical stress and have higher signal attenuation per length unit. A special attention needs to be paid to the mineral cable connection design. Although mineral insulated cables are robust against harsh environmental conditions such as temperature and radiation, their connection interface is potentially a week point for ensuring proper electrical function; the connection point has to be properly designed.

Polymer cables have some advantages in cable routing and have better electrical characteristics. For instance, attenuation values are lower than in the mineral insulated cables. Owing to limitations of mineral insulated cables, it is worth to qualify polymer cables for applications in which either radiation dose is limited (e.g. through distances from the main radiation source, additional shielding), or a shorter mission time is acceptable. AREVA has decided to qualify their own polymer cable ARENOPYR with twisted pair of wires fort severe accident conditions.

Type series I&C cable ARENOPYR JE-LIHXCHX shown in Fig. VII–4. is intended for connecting temperature sensors, pressure transmitters or limit switches with respective electronic equipment in the cabinet via containment cable penetrations. This cable is designed to transmit a direct current signal as well as low/intermediate frequency signals. Shielding is ensured by a copper wire braid. This type series has improved characteristics for fire; materials are halogen-free and are flame retardant.

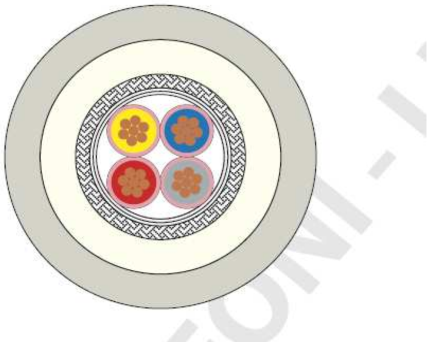

FIG. VII–4. A cross section of ARENOPYR I&C cable JE-LIHXCHX 2×2×0.5 (figure courtesy of AREVA).

In accordance with the Olkiluoto-3 project requirements, this cable type is used in systems requiring a mission time of 100 hours (due to the radiological limitation of polymers materials). A description of accident test qualification sequence is provided in Table VII–3.

TABLE VII–3. EXAMPLE OF SEVERE ACCIDENT TEST SEQUENCE FOR ARENOPYR CABLE, MISSION TIME 100 HOURS

Test step		Remarks
1.	FAT	Mechanical tests: verification of dimensions Electrical tests: dielectric strength test and insulation resistance test.
2.	Incoming Inspection	Was performed by the cable manufacturer and AREVA in the framework of factory acceptance tests.
3.	Thermal Ageing	Accelerated thermal ageing through storage in a heat cabinet: 72 days at 106°C
4.	Radiation ageing	I&C cables installed in side containment; 302kGy at 250Gy/h considering the dose rate effect.
5.	Accident irradiation	Performed using the total irradiation dose for the mission time of 100h (SA MT) 800kGy at 8kGy/h maximum dose rate.
6.	Function tests	Electrical tests: dielectric strength test, insulation resistance test.
7.	SA test	Accident simulation test with saturated steam conditions following the temperature profile specified (peak temperature 156°C for 12 h, total test duration 100h) No chemical spray was injected during the accident simulation test.
	Post SA test	Omitted, the mission time of 100h was tested in real time (step 7); The stability against chemicals (spray) was demonstrated using results from other test campaigns.
	Function tests	During the accident simulation test (step 7): the insulated conductor wires were energized, and insulation resistance measurements were performed.
8.	Final functional tests	Electrical tests: dielectric strength test, insulation resistance test.

DEFINITIONS

The following definitions apply only for the purposes of this TECDOC.
Further definitions are provided in INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary, IAEA, Vienna (2007).

equipment capability to perform reliably. The capability of the SSC to perform reliably under severe accident conditions to be achieved by an appropriate choice of measures including the use of proven components (proven by experience under similar conditions or adequately tested and qualified.), redundancy, diversity (the potential for common cause failure, including common mode failure), physical and functional separation and isolation.

mission time. Time for which the equipment is able to perform (maintain) its intended function, considering the actual environmental conditions.

survivability assessment. Provision of a reasonable level of confidence that equipment will carry out intended function under severe accident conditions for expected mission time.

ABBREVIATIONS

ALM	Accident level measurement equipment
ASTEC	Accident source term evaluation code
BEPU	Best estimate plus uncertainty
BWR	Boiling water reactor
CANDU	Canada deuterium uranium
CCS	Containment cooling system
CD	Core damage
CESAM	Code for European Severe Accident Management
CS	Core spray
CSS	Containment spray system
DBA	Design basis accident
DC	Direct current
DCH	Direct containment heating
DEC	Design extension condition
ECCS	Emergency core cooling systems
ECR	Emergency control room
EDG	Emergency diesel generator
EFW	Emergency feedwater
EME	Emergency mitigating equipment
EOP	Emergency operating procedure
FAT	Factory acceptance test
FCI	Fuel coolant interaction
GRS	Gesellschaft für Anlagen und Reaktorsicherheit gGmbH, Germany
HELB	High energy line break
HPCI	High pressure coolant injection
HPIS	High pressure injection system
HPME	High pressure melt ejection
HVAC	Heating ventilation and air conditioning
I&C	Instrumentation and control
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IRSN	Institute for Radiological Protection and Nuclear Safety, France
IRWST	In-containment refuelling water storage tank
KINS	Korea Institute of Nuclear Safety
KKS	Kraftwerk-Kennzeichnungssystem (plant identifier system)
LP&S	Low power and shutdown
LPCI	Low pressure coolant injection
LPIS	Low pressure injection system
MAAP	Modular accident analysis programme
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
MBLOCA	Medium break loss of coolant accident
MCCI	Molten core concrete interaction
MCR	Main control room
MIC	Mineral insulated cable
MSIV	Main steam isolation valve
NO	Normal operation
NPP	Nuclear power plant
OLC	Operating limits and conditions
PAR	Passive autocatalytic recombiner
PDF	Probability distribution function
PHEBUS	European light water reactor accident source term research project
PORV	Pilot operated relief valve
PSA	Probabilistic safety assessment

PWR	Pressurized water reactor
PZR	Pressurizer
RC	Release category
RCIC	Reactor core isolation cooling
RCP	Reactor coolant pump
RCS	Reactor coolant system
RHR	Residual heat removal
RPV	Reactor pressure vessel
RWST	Refueling water storage tank
SAMG	Severe accident mitigation guideline
SARNET	European network of excellence on core meltdown accidents
SBLOCA	Small break loss of coolant accident
SBO	Station blackout
SCRA	Safety control rod axe man
SFP	Spent fuel pool
SG	Steam generator
SGTR	Steam generator tube rupture
TMI	Three mile island
US NRC	United States Nuclear Regulatory Commission

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