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Severe Accident Mitigation through Improvements in Filtered Containment Vent Systems and Containment Cooling Strategies for Water Cooled Reactors

Proceedings of a Technical Meeting on Severe Accident Mitigation through Improvements in Filtered Containment Venting for Water Cooled Reactors Held in Vienna, Austria, 31 August–3 September 2015

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SEVERE ACCIDENT MITIGATION THROUGH IMPROVEMENTS IN FILTERED CONTAINMENT VENT SYSTEMS AND CONTAINMENT COOLING STRATEGIES FOR WATER COOLED REACTORS

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SEVERE ACCIDENT MITIGATION THROUGH IMPROVEMENTS IN FILTERED CONTAINMENT VENT SYSTEMS AND CONTAINMENT COOLING STRATEGIES FOR WATER COOLED REACTORS

PROCEEDINGS OF A TECHNICAL MEETING ON SEVERE ACCIDENT MITIGATION THROUGH IMPROVEMENTS IN FILTERED CONTAINMENT VENTING FOR WATER COOLED REACTORS HELD IN VIENNA, AUSTRIA, 31 AUGUST–3 SEPTEMBER 2015

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FOREWORD

One of the most important lessons learned from the Fukushima Daiichi accident is that a reliable containment venting system can be crucial for effective accident management during severe accidents, which is especially important for smaller volume containments. Filtered containment venting systems can enhance the capability to maintain core cooling and containment integrity as well as to reduce uncontrolled radioactive releases to the environment in the case of a severe accident. Moreover, such systems increase the flexibility of plant personnel to react to the consequences of unforeseen events.

In 2015, the IAEA hosted the Technical Meeting on Severe Accident Mitigation through Improvements in Filtered Containment Venting for Water Cooled Reactors. This meeting was part of the activities relating to the IAEA Action Plan on Nuclear Safety, which was approved by the Board of Governors in September 2011 as part of the IAEA's response to the Fukushima Daiichi accident.

The meeting brought together representatives of 26 Member States, the IAEA, the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development and the World Association of Nuclear Operators. Experts presented the current status of regulations and requirements for filtered containment venting systems. Member States shared evolving strategies to ensure containment integrity in the event of a severe accident for existing plants in light of events during the Fukushima Daiichi accident. These include assessing the conditions under which venting can be authorized, the decision making process to allow venting, the need for containment venting upgrades, and the design and installation of new and state of the art systems. This publication presents the technical papers submitted and presented at the meeting.

The IAEA expresses its appreciation to the Russian Federation for its extrabudgetary contribution to fund this effort. It also acknowledges the contributions of the experts who participated and submitted papers for presentation. In addition, it would like to thank the session chairs G. Rubio (United Kingdom), T. Matsuo (Japan) and J. Taylor (United States of America). The IAEA is also grateful to S. Guentay (Switzerland) and W. Frid (Sweden) for their roles as session chairs and for their contributions in the preparation of this publication. The IAEA officers responsible for this publication were T. Jevremovic and C. Painter of the Division of Nuclear Power.

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1. INTRODUCTION

1.1. BACKGROUND

One of the most important lessons from the accident at the Fukushima Daiichi nuclear power plant (NPP) is that a reliable containment venting system can be crucial for effective accident management during severe accidents, especially for smaller volume containments in relation to the rated nuclear power. Safety evaluations performed after the Fukushima Daiichi accident have led many States to consider the implementation of filtered containment venting systems (FCVSs) at nuclear power plants where these are not currently applied as an enhancement of the response capability to severe accidents. Containment venting can enhance the capability to maintain core cooling and containment integrity as well as reduce uncontrolled radioactive releases to the environment if the venting system has a filtration capacity. Therefore, FCVSs where installed may be used to prevent containment failure and limit radioactive releases to the environment in the case of a severe accident. Moreover, an FCVS increases the flexibility for the plant personnel when coping with unforeseen events.

It is especially important that the containment remains intact and capable of performing its functions during a prolonged station blackout accident involving severe core damage. Thus, in order to protect the containment against overpressure damage due to steam production from decay heat (slow pressure buildup) and generation of non-condensable gases, notably hydrogen, and to improve the possibility of returning the plant to a safe state after accident involving severe core damage, it needs to be possible to carry out controlled containment pressure relief. The relief system needs to be designed so that, together with other measures taken to protect the containment, it can ensure that the releases to the environment do not exceed the limits established by the regulatory body.

Depending on the type of reactor and accident, controlled venting of containment atmosphere can also provide a means to reduce hydrogen concentration in the containment and thereby reduce the risk of hydrogen combustion. Another mitigating feature of controlled venting is the possibility of reducing containment pressure in order to reduce the driving force for diffuse leakage from the containment. In this way the release of radioactivity to the environment can be reduced as well as leakage of hydrogen into the reactor building in boiling water reactors (BWRs).

As a result of the accident at Three Mile Island in 1979, many NPPs around the world were backfitted to improve the capability to prevent and mitigate core damage accidents. In some States, FCVSs were installed and in others so called hardened containment venting systems were fitted, such as at BWR plants with Mark I containment designs. These developments were presented and discussed at a meeting on filtered containment venting systems organized by the OECD in 1988 [1].

Filtered containment venting became one of the priority issues in connection with stress tests performed following the Fukushima Daiichi accident. Many States decided to install FCVSs, while others are considering fitting of these systems. States that have FCVSs are evaluating the performance of the systems under conditions more severe than those assumed in original designs. As a result of developments following the publication of the OECD report in 1988, as well as insights from the Fukushima Daiichi accident, and in particular EU stress tests, the OECD decided to update the previous report and a new version was published in 2014 [2].

For new plants, including those being designed and currently being built, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3] explicitly includes the consideration of severe accident scenarios and strategies for their management into a plant's design. Specifically, para. 5.30 of SSR-2/1 (Rev. 1) states:

> "In particular, the containment and its safety features shall be able to withstand extreme scenarios that include, among other things, melting of the reactor core. These scenarios shall be selected using engineering judgment and input from probabilistic safety assessments."

Furthermore, Requirement 55 of SSR-2/1 (Rev. 1) on control of radioactive releases from the containment states:

> "The design of the containment shall be such as to ensure that any radioactive release from the nuclear power plant to the environment is as low as reasonably achievable, is below the authorized limits on discharges in operational states and is below acceptable limits in accident conditions."

IAEA Safety Standards Series No. NS-G-1.10, Design of Reactor Containment Systems for Nuclear Power Plants [4], states that:

> "For new plants an energy management system should be incorporated as the primary means of meeting … acceptance criteria for structural integrity for loads derived from the pressures in the containment during accidents…In severe accidents, the systems for energy management in the containment and their support systems … should be independent of the systems used to prevent melting of the core. If this is not the case, the design of the containment should provide a sufficient period of time for measures to recover failed systems for energy management so as to be able to guarantee the operability of the energy management system under severe accident conditions. Venting systems should not be necessary for new plants."

IAEA Safety Standards Series No. NS-G-2.15, Severe Accident Management Programme for Nuclear Power Plants [5], states in para. 2.6 that:

> "At the top level, the objectives of accident management are defined as follows:
>
> — Preventing significant core damage;
>
> — Terminating the progress of core damage once it has started;
>
> — Maintaining the integrity of the containment as long as possible;
>
> — Minimizing releases of radioactive material;
>
> — Achieving a long term stable state."

Moreover, para. 3.67 of NS-G-2.15 [5] states:

> "Examples of possible design changes that can be implemented in existing plants are: a hardened and/or filtered containment vent; passive autocatalytic recombiners; igniters; a passive containment cooling system; reactor cavity flooding; isolation of pathways to the environment that may exist after basemat failure; larger station

batteries or alternate power supplies; and enhanced instrumentation (extended scale or new measurements), such as enhanced instrumentation for the steam generator level. A modification can fulfil several functions. For example, a filtered containment vent can be used to prevent containment overpressurization, but also to release hydrogen (or oxygen) to reduce the hydrogen risk, to prevent unfiltered leakage from existing openings or from a containment that has a pre-existing (relatively) large leakage rate, or to prevent basemat failure — if anticipated to occur — at an elevated containment pressure."

1.2. OBJECTIVE OF THE TECHNICAL MEETING

In the light of recent events, containment venting has become a very important topic for Member States. Actions related to assessing the need for containment venting system upgrades, designing and installing new systems, and building new advanced nuclear plants with the latest passive containment cooling systems are under way in many countries.

With the objective of bringing together experts to discuss the current technologies, industry issues and regulatory actions, the IAEA hosted an international meeting in August 2015 on filter vent technology. Attended by 50 specialists from 26 Member States, the overall objective of the meeting was to provide Member States with:

- A current status of regulatory actions and requirements related to filter containment vent systems;
- An understanding of long term containment response to severe accidents;
- An understanding of strategies to ensure containment integrity for existing plants during severe accidents;
- An understanding of advanced containment cooling and energy management technologies in advanced reactor designs;
- An understanding of filtered containment vent technology.

Twenty-six papers were presented in five Sessions:

(1) Background and Regulations;
(2) Long Term Containment Response to Severe Accidents in Light of the Fukushima Daiichi Accident;
(3) Strategies to Ensure Containment Integrity for Existing Plants during Severe Accidents;
(4) Strategies in Containment Cooling and Energy Management for Advanced Reactor Designs;
(5) Filtered Venting Technology.

2. CONTAINMENT VENTING DURING THE FUKUSHIMA DAIICHI ACCIDENT

In July 1987, Japan's Nuclear Safety Commission established the Common Issue Committee to investigate approaches to severe accidents, probabilistic safety assessment (PSA) techniques and the containment function during severe accidents. As a result, a hardened containment venting system (HCVS) was installed by the Tokyo Electric Power Company (TEPCO) in the 1990s at all units of the Fukushima Daiichi nuclear power plant [6].

The HCVS allows gases to bypass the standby gas treatment system when containment venting is required at high containment pressure. The system has the capacity to vent from either the drywell or from the wetwell. During a severe accident with core damage, venting from the wetwell has priority in order to benefit from the scrubbing of fission products in the water of the pressure suppression pool. Venting of the containment requires opening of motor and air operated valves as well as passive opening of a rupture disc, which is located downstream of these valves and is designed to open if containment pressure is above the pre-set pressure. The rupture disc is thus the last barrier before the external environment (see the IAEA Report on the Fukushima Daiichi Accident, Technical Volumes 1 and 2) [6, 7].

Installation of an HCVS in BWR Mark I containments followed the issuance of Generic Letter 89-16, Installation of a Hardened Wetwell Vent, by the US Nuclear Regulatory Commission (NRC) to enhance the capability of plants to prevent and mitigate the consequences of severe accidents. The HCVS would reduce the likelihood of core melt in accidents involving long term loss of decay heat removal, anticipated transients without scram and station blackout (SBO). As a mitigating measure, the HCVS would provide a means to vent containment with scrubbing of fission products in the water of the suppression pool, thus reducing radioactive releases to the environment.

During the Fukushima Daiichi accident, the operators made several attempts to vent the containments of Units 1, 2 and 3. A timeline indicating venting events is shown in Fig. 1. Manual venting was very difficult due to the lack of alternating current (AC) power and compressed air. Moreover, the local radiological conditions near the vent operating stations were worsening. In Units 1 and 3, venting from the wetwell was finally successful, allowing the plant operators to ultimately decrease the containment pressure; however, it took so many hours to start venting that the leaktight integrity of the containment was lost. In Unit 2, it is unclear whether the operators were ever successful in venting the containments. As a result of the prolonged SBO, the remote control of vent valves was not possible from the control room because of the loss of AC power. In this situation, the vent system isolation valves could only be opened manually and locally by the use of mobile air compressors or compressed air cylinders. In addition, when the decision to vent containments was taken, the high dose rates in the reactor building made it extremely difficult to perform the required manual local actions (see the IAEA Report on the Fukushima Daiichi Accident, Technical Volumes 1 and 2) [6, 7].

A successful and timely containment venting episode would have allowed earlier injection of coolant into the reactor vessel. This could have prevented extensive core damage with the resultant significant fission product release from the core and production of large quantities of hydrogen. As a result of high containment pressure and temperature, the containment integrity was lost which most likely resulted in leakage of hydrogen into the reactor building, causing the subsequent explosions in Units 1, 3 and 4, and the uncontrolled release of radioactive material to the environment.

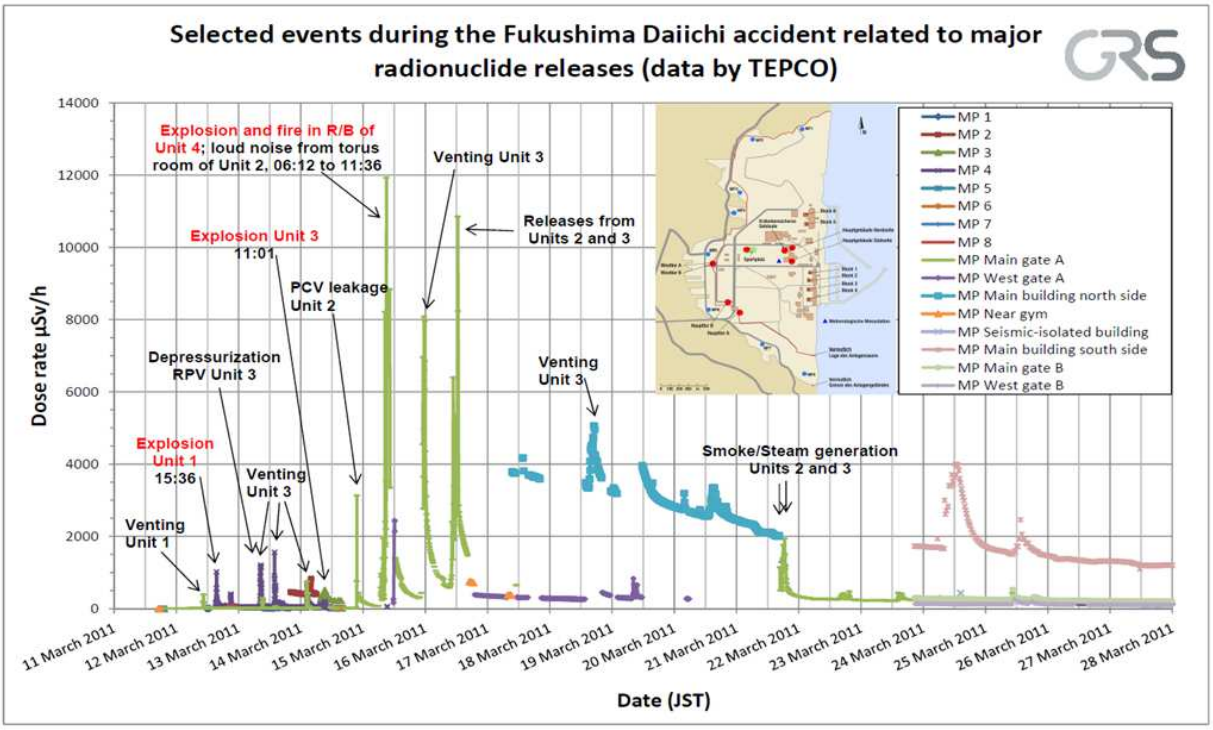

FIG. 1. Timeline of selected events during the Fukushima Daiichi accident related to major radionuclide releases (data provided by TEPCO).

The consensus of experts is that the hydrogen explosion in Unit 4 was caused by the migration of hydrogen from Unit 3 to Unit 4 via an interconnected ventilation system.

The Fukushima Daiichi accident clearly demonstrated that reliable containment venting during severe accident conditions is necessary for maintaining containment integrity, as well as limiting the release of radioactive material to the environment. Reliable containment venting is also important to maintain core and containment cooling.

As a result of lessons from the Fukushima Daiichi accident, the performance of existing containment venting systems in accident scenarios more severe than the design basis has been evaluated and improvements have been made or are imminent. Some States have decided to install FCVSs in their NPPs.

With regard to United States BWR facilities with Mark I and Mark II containments, the US Nuclear Regulatory Commission requires licensees to install or improve the HCVS so that venting functions are also available during severe accidents. These conditions include elevated temperatures, pressures, radiation levels, and combustible gas concentrations, such as hydrogen and carbon monoxide, associated with accidents involving extensive core damage, including accidents involving a breach of the reactor vessel by molten core debris. One of the requirements is that the HCVS is capable of operating with dedicated and permanently installed equipment for at least 24 hours following the loss of normal power or loss of normal pneumatic supplies to air operated components during an extended loss of AC power. Another requirement is that the HCVS be designed to minimize the reliance on operator actions.

3. SUMMARY OF TECHNICAL MEETING

The Technical Meeting covered five Sessions as follows:

- Session I: Background and Regulations;
- Session II: Long Term Containment Response to Severe Accidents in Light of Fukushima Daiichi Accident;
- Session III: Strategies to Ensure Containment Integrity for Existing Plants During Severe Accidents;
- Session IV: Strategies in Containment Cooling and Energy Management for Advanced Reactor Designs;
- Session V: Filtered Venting Technology.

The content and main conclusions of these Sessions are summarized in the following sections based on the participants' presentations. Not all of these presentations were accompanied with the papers provided within this document.

3.1. SESSION I — BACKGROUND AND REGULATIONS

This Session contained 11 presentations covering background, regulations and status of implementation of FCVSs in ten countries. In one presentation, a summary of a recent comprehensive OECD/NEA status report on FCVS was given.

The first presentation, entitled Summary of OECD/NEA Status Report on Filtered Containment Venting, described the content and main conclusions of the OECD/NEA report. The report was published by the Committee on the Safety of Nuclear Installations (CSNI) in July 2014 (NEA/CSNI/R (2014)7) [11]. The objective of the report was to provide an up to date picture of the status of implementation, guidance for their improvement and future design of FCVS in OECD member countries.

The following major FCVS issues are addressed in the paper: purpose and benefits of filtered containment venting, venting strategies and procedures; FCVS design requirements; installed filtering technologies; possible adverse effects relating to the implementation of FCVS; and potential improvements of FCVS. Among many specific aspects of FCVS, the paper draws attention to issues such as radiological consequences and accident management, in particular in comparison with using hardened vent only, hydrogen risk reduction through venting, various filter systems, and venting strategies and operations.

Results of recent source terms evaluations to determine the desirable filter performances are presented and discussed. The decontamination factor (DF) should be > 1000 for radioactive aerosols including for small size (in the 0.1 μm range) and hygroscopic aerosols. The decontamination factor should be > 100 for molecular iodine and > 10 for organic iodides. No requirement is presently considered for the filtration of gaseous ruthenium as its relative contribution to the radiological consequences has to be more thoroughly assessed. There is presently no requirement concerning noble gas retention. Re-suspension/re-volatilisation from filter deposits during a prolonged use of FCVS should not challenge the target DFs. The R&D to assess such processes in FCVS is presently ongoing.

The pipes, supports, and valves of the venting line should be designed for safe and reliable operation for the expected dynamic loads, including hydrogen combustion loads or provisions should be set to keep all parts of the system at a safe distance to hydrogen combustion limits.

With regard to the effective accident management, it is desirable to have a dedicated instrumentation. Such instrumentation must provide the operators and decision makers with reliable information, even in the worst conceivable conditions.

Generally, all contributing countries recognized the potential benefits of the filtered containment venting system (FCVS) to prevent containment failure and limit radioactivity releases to the environment in the case of a severe accident.

The second presentation, entitled Filtered Containment Venting in Sweden, described historical background, regulatory requirements and design of the FCVS which was installed at all Swedish nuclear power plants during 1980s.

After the Three Mile Island accident in the USA in 1979, the Swedish government decided that all Swedish NPPs should be capable of withstanding a core melt accident without any casualties or ground contamination of importance to the population. The releases shall be limited to maximum 0.1% of the reactor core content of ^{134}Cs and ^{137}Cs in a reactor core of 1800 MW thermal power (corresponds to approximately 170 TBq ^{137}Cs), provided that other nuclides of significance, from the use of land viewpoint, are limited to the same extent as caesium.

The design accident scenario for FCVS is total loss of all AC power (station blackout) and — only for pressurized water reactors (PWRs) — steam-driven auxiliary feedwater system for 24 hours. The FCVS is fulfilling the following requirements: passive operation that is activated through rupture disc close to design pressure, possibilities to manually open and close venting, passive operation for at least 24 hours, no manual action needed during the first 8 hours and nitrogen or steam inerted in order to avoid hydrogen combustion. FCVS at all reactors is using the Multi Venturi Scrubber System.

In addition to FCVS, the Swedish reactors were equipped with automatic filling of lower drywell with condensation pool water (only in BWRs with internal recirculation pumps) and independent containment spray and containment water supplying from external water source. In addition, mitigation measures include dedicated severe accident instrumentation and severe accident management guidelines or procedures and knowledge based handbooks for Technical Support Team (staff).

The third presentation, entitled Bangladesh Perspective on the Requirement for Filtered Vent Systems on Rooppur NPP, addressed situation in Bangladesh concerning construction of two 1200 MW(e) Gen-III+ WWER NPP at the Rooppur site. On 2 November 2011, an Inter-Governmental Agreement between the Government of People's Republic of Bangladesh and the Russian Federation was signed for construction by 2021–2022 of a nuclear power plant in Bangladesh. Regulatory infrastructure and nuclear power infrastructure development as well as main safety functions of WWER and techno-economic parameters of Rooppur NPP were described. Preparatory phase construction activities of RNPP were also presented.

The forth presentation, entitled Filtered Containment Venting System For Existing Nuclear Power Plants: Design Parameters and Associated Safety Criteria in Belgium, described activities in Belgium connected with the planned installation of FCVS on all Belgian NPPs.

First, the regulatory framework was presented. The main part of the paper is devoted to description and discussion of the methodology supporting the definition of the FCVS main design parameters and associated safety criteria developed by Tractebel Engineering, on behalf of the licensee Electrabel – GDF Suez, and validated by the Belgian nuclear safety authorities. The deadline of the overall project schedule for the installation of the FCVS is the end of 2017.

The methodology to assess the design parameters relies mainly on the use of the MELCOR 1.8.6 code as well as on the ASTEC V2.0 code through its CPA and IODE modules which have adequate modelling of iodine behaviour. The analysed reference accident sequence was the complete station blackout (SBO). Several of key parameters for the FCVS design were assessed through calculations using best estimate method with conservative bounding conditions.

The FCVS is designed to operate without assistance during at least 24 hours. The permanently installed reserves of water and chemical additives on site shall allow for FCVS autonomy of at least 72 hours after the accident initiating event. Provisions against explosion of combustible gases in the FCVS should be foreseen. The FCVS has to withstand the following external hazards: extreme earthquake, extreme external flooding and extreme weather conditions (extreme winds, lightning, rainfall, temperature range, snowfall).

The fifth presentation, entitled Regulatory Oversight of Filtered Venting and Containment Integrity at Canadian Nuclear Power Plants, consisted of four parts addressing Canadian Nuclear Safety Commission (CNSC), requirements for FCV in Canada, overview of Canadian NPPs, domestic studies/research and development and international collaboration. The focus of the presentation was on the status of filtered containment venting and on plans, and related investigations, for implementation of FCVS for severe accidents at Canadian NPPs. Canada has 19 operating nuclear reactors (all PHWRs) on four sites; multi-unit stations at Bruce, Darlington and Pickering and single unit station at Point Lepreau. After the Fukushima accident, a number of safety improvements were identified by the CNSC, among them hydrogen mitigation equipment and filtered venting for beyond design basis and severe accidents, to ensure protection of the containment. The Point Lepreau single-unit NPP had the AREVA wet scrubber emergency FCVS for severe accidents prior to Fukushima accident. For the multi-unit stations there are various options. At Pickering NPP, the emergency filtered air discharge system (EFADS) will be improved with respect to manual operation and emergency power. At Darlington NPP, the Westinghouse Dry Filter Method (DFM) will be installed. Finally, at Bruce the evaluations to provide the options for a future system are ongoing. With regard to Canadian studies/research and development, the objectives are to evaluate how a filtered containment venting system would help to reduce the overall consequences of a severe accident in CANDU-6 reactors, as well as to examine the essential design considerations for any filtered venting system. Preliminary results of the study were presented.

The sixth presentation, entitled Regulations and improvements in Hungary related to Severe Accidents and Filtered Venting, consisted of three parts. In the first part, the most important regulations related to severe accidents and containment for existing nuclear power plants were presented. In the second part of the paper, the safety improvements performed before the Fukushima Daiichi accident were presented and the third part was devoted to improvements after the Fukushima accident.

Among the safety improvements implemented before the Fukushima accident are installations of reactor vessel external cooling, severe accident hydrogen recombiners, autonomous electric supply system with mobile diesels, post-accident monitoring system and severe accident management guidelines.

Following the EU stress tests, the Hungarian Atomic Energy Authority issued the safety improvement requirements. One of the requirements was to carry out an analysis of the long term (beyond one week) progression of severe accidents to mitigate the severe accident consequences. It was concluded that safety improvements before the Fukushima Daiichi accident provide sufficient measures to avoid early damage of the containment using the spay system, but for long term overpressurization further measures are needed to avoid containment failure. Based on accident analysis, the licensee is planning to avoid the long term overpressurization of the containment by using a severe accident spray system instead of filtered containment venting system. The deadline of the implementation of the severe accident spray is the end of 2018.

The seventh presentation, entitled Regulatory Challenges of Filtered Containment Venting in Indonesia, described plans to build a nuclear power plant in Indonesia. According to the plan, the nuclear energy will support about 5000 MW(e) in 2025. BAPETEN as regulatory body should prepare the regulation infrastructure including the severe accident management. The focus of the paper was on various containment systems for a PWR. The dry containment system and the passive simplified pressurized water reactor concept were described and discussed. Based on the characteristic of these containment systems, the decision concerning FCVS will be taken.

The eighth presentation, entitled Regulatory Requirements and Actions Related to Improve Containment Venting for Laguna Verde NPP, addressed the current status of the requirements established by the Mexican regulatory body and actions underway by the utility in order to improve the containment venting of both BWR Mark II units of Laguna Verde NPP. The process is based on the lessons learned from the international operating experience developed as a consequence of the Fukushima Daiichi accident and on the Order EA-13-109 developed by the United States Nuclear Regulatory Commission. The NRC order requires BWR facilities with Mark I and Mark II containments to make the necessary plant modifications and procedure changes to provide a reliable hardened containment venting system (HCVS) that is capable of performing under severe accident conditions.

The construction process which involves the physical implementation of the hardened containment venting system will be done during the routine outages used for fuel reloading and large maintenance. It is expected that the improvement to the HCVS will be completed during 2018.

The ninth presentation, entitled Status of Spanish Regulations and Industry Actions Related to Filtered Containment Venting Systems, described the current situation at the Spanish NPPs and the regulatory requirements concerning installation of FCVS at the Spanish NPP as a result of evaluation of EU stress tests performed by nuclear power plants. FCVS will be installed in all Spanish nuclear power plants by the end of 2017. The CSN released a guide containing general FCVS criteria to be used for the CSN technical evaluations. The FCVS decontamination factors for aerosols, elemental iodine, organic iodine, and other radionuclides of significance, should be at the level of those offered by the latest technologies available on the market. The FCVS should be designed and constructed in order to minimize the risks associated with the presence of hydrogen in the system. Analyses can give credit to

Passive Autocatalytic Recombiners (PAR) installed in the containment. The FCVS and its instrumentation should have the support systems (power supply, air drive, consumables, etc.) to ensure proper performance of its function autonomously, for at least 24 hours without any site external support and for at least 72 hours only with the site external support of light equipment.

The tenth presentation, entitled Results of Regulatory Review of Activities for Implementation of Containment Filtered Venting for Ukrainian NPPs, gave the status of implementation of FVCS at all Ukrainian NPPs. The requirement for implementation was taken by the Ukrainian regulatory body based on results of the EU stress tests.

Analytical justifications of FCVS were performed using the MELCOR computer code and specific models for reference units. Results of analytical justifications performed by NPPs for non-filtered containment venting and filtered containment venting were presented. Results of analytical justifications of the FCVS for three reference units have been verified in the framework of regulatory review. In the framework of this review, additional calculations (benchmarks) were conducted by the Ukrainian technical support organization SSTC NRS.

The application of different filter types was analysed. Three types of filters for containment venting were considered such as sand filter, venturi scrubber and dry filter.

At present, the conceptual technical solutions were developed by the utility and agreed with the regulatory body for WWER-1000. FCVS for WWER-1000 are under implementation. The technical solution for WWER-440 should be updated according to the state review results.

The eleventh presentation, entitled Provision of Containment Integrity at Russian WWER NPPs under Beyond Design Basis Accident (BDBA) Conditions, described the status of FCVS in the Russian Federation. At present, the Russian Federation has 6 WWER-440 units and 11 WWER-1000 units in operation. None of them is equipped with FCVS. However, the decision has been taken that first FCVS will be installed at Rostov NPP unit 1 and after that FCVSs will be implemented at all other operating WWER NPPs. In order to reveal the weak points in the NPP’s designs in case of BDBA, the stress-tests were carried out to analyse the NPP response. As a result, hydrogen monitoring systems and passive autocatalytic hydrogen recombiners (PARs) were installed at operating WWER units.

Two filtration methods are considered: wet filtration using venture scrubber and dry filtration.

For new NPP designs, taking into account large time reserves (provided by operation of passive safety systems) for enabling additional technical means under accidents resulting in containment pressure increase, the decision on inexpediency of FCVS implementation was taken.

3.2. SESSION II: LONG TERM CONTAINMENT RESPONSE TO SEVERE ACCIDENTS IN LIGHT OF THE FUKUSHIMA DAIICHI ACCIDENT

Three papers were presented covering different aspects of modelling of containment for pressurization and resultant environmental releases, as well as potential temperature stratification in a wetwell pool having implications on the containment pressurization and hence timing of the venting.

The first paper, entitled Behaviour of Containment System under Long Term Station Black Out Condition, introduced in the first part the suite of codes to investigate the thermal-hydraulics and source term behaviour of Tarapur Atomic Power Station (TAPS) BWR, and their assessment and in the second part discussed the key outcome of the containment performance analysis using these codes for the TAPS BWR.

A set of codes used is a mix of US codes; ORIGEN2 (fission product inventory), RELAP5 (primary coolant system thermal hydraulics), own code PHTACT (fission product release from fuel) and SOPHAEROS (fission product transport), and CPA (containment thermal hydraulic, fission product transport) and MEDICIES (molten corium concrete interaction) modules of French ASTEC code. These codes are passively linked together. The portions or all parts of the suite was assessed using i) IAEA Coordinated Research Project (CRP) on 'Benchmarking Severe Accident Computer Codes for Pressurized Heavy Water Reactor applications' on SBO transient in a CANDU6 plant, ii) Severe Accident analysis of Fukushima Daiichi Unit-1 and iii) data from 'Containment Studies Facility'. Both benchmarks showed strengths and limitations of the codes, especially as the complexity of the phenomena increases during the late phase of the in-vessel accident progression, the agreement among the predicted results was more divergent. The Containment Studies Facility is a model containment erected aiming at i) experimental validation of computer codes for containment behaviour under simulated LOCA conditions in Indian HPWRs, ii) assessment of suppression pool efficacy for energy absorption under design basis accident (DBA) and under severe accident conditions, iii) inter-compartmental aerosol transport and aerosol removal by suppression pool and iv) hydrogen transport in the containment and removal by PARS. The containment behaviour under simulated LOCA condition using ASTEC and in-house code CONTRAN was assessed using the data obtained from the facility.

The long term station black-out transient analysis provided TAPS plant response regarding i) containment pressure behavior leading to the venting and during the venting, ii) hydrogen load to the containment and amount released to the environment as well as iii) the fractional release of main fission products to the containment and to the environment without taking credit of containment venting filter system. The environmental fission product release is indicated to be substantially low.

The second paper, entitled Comparative Analysis on Containment Over-Pressurization Scenarios using MAAP code, introduced the analyses performed using MAAP 4 code and comparatively displayed the adequacy of the decay heat removal alternative options by using i) an Emergency Containment Spray Backup System (ECSBS) receiving water from portable equipment and ii) a containment filtered venting system. After a review of international practices regarding effective measures to control over pressurization in a large dry PWR containment, otherwise large environmental activity release from the containment is inevitable, the containment spraying and flooding receiving water from the portable equipment and the containment filtered venting are the two alternative options considered for the potential implementation in Korean PWRs. MAAP 4 analyses for a station black out transient for a reference Korean PWR (OPR100) were performed by considering variations in the selected strategies of each option. Results show that the internal decay heat removal using the first alternative, if successfully prepared, can prevent radiological release and maintain containment integrity as an active way. Although continued flooding of the containment floor terminates core concrete interaction, it is found out that, optimal operating strategy is needed to prevent extensive flooding inside the containment to enable accessibility of instrumentation and control systems. Continued condensation in containment produces hydrogen flammability risk, which requires additional hydrogen mitigation measures. It is found out that filtered venting can mitigate radiological release below 10^{-3} of initial inventory as a passive way, however, due to the lack of any water supply to the containment basement molten corium overlying and interacting with concrete is not coolable and hence mitigation of core concrete interaction is not possible. Therefore, the analyses highlighted overall, pros and cons of each alternative option. The paper suggests that the final design option should be chosen considering the plant specific overall safety effect and design characteristics and other issues such as diversity versus redundancy, long term operability, etc.

The third and the last paper of the Session, entitled An Investigation of Pool Stratification and Vent Heat Transfer on BWR Wetwell External Venting using GOTHIC, described the applicability of the GOTHIC to study pool stratification as well as modelling strategies for dealing with the long transients needed to be considered for assessing vent performance.

Motivation for this study is the postulated stratification of the wet-well leading to the higher than expected pressurization rate of the containment in some forensic analyses for the Fukushima Daiichi events. Thermal stratification, if develops, would diminish the heat absorbing capacity of the pool, leading to earlier and more frequent vent valve cycling and earlier steam bypass through the saturated pool. After introduction of studies assessing GOTHIC capability to predict pool stratification as experimentally observed in three tests; Browns Ferry hot shutdown test, POOLEX test and Monticellop SRV test, for those the code decently reproduced the experimentally observed pool stratifications, the GOTHIC code was used to investigate the containment response during a long term loss of AC power with periodic wetwell venting, including the possibility of suppression pool stratification. The results indicated that under boundary conditions selected for the analysis; i) the sstratification is possible for steam discharge from turbine exhaust spargers; ii) stratification would promote early venting and increase vent cycling; and iii) a small amount of SRV discharge tends to keep the pool well mixed. The study also indicated the need for additional testing for the GOTHIC lumped model pool mixing correlation for pool temperature approaching T_{sat} with resulting in steam bypass for considering multiple concurrent steam sources.

3.3. SESSION III: STRATEGIES TO ENSURE CONTAINMENT INTEGRITY FOR EXISTING PLANTS DURING SEVERE ACCIDENTS

Session III contained two presentations.

The first presentation, entitled Experience with Containment Venting at Malaysian Research Reactor, had three parts. In the first part, the goals, organization and facilities of Nuclear Malaysia were presented. The second part of presentation was devoted to description of one of the facilities, namely the PUSPATI TRIGA Reactor (RTP), in particular its ventilation system. It was stated that current RTP confinement is adequate to provide sufficient and effective control of effluents during normal operational conditions. Finally, in the third part of the presentation the National Nuclear Energy Policy and Action Plan was described. Malaysia needs assistance in safety assessment, especially in the severe accident area.

The second presentation, entitled Strategies to Ensure Containment Integrity for ABWRs in Kashiwazaki-Kariwa Nuclear Power Station, described strategies to ensure containment integrity for ABWRs in Kashiwazaki-Kariwa Nuclear Power Station in Japan, reflecting the lessons learned from the Fukushima Daiichi accident.

First, with considerations of multiple failures of the Fukushima Daiichi accident, defense in depth (DID) was enhanced by applying more diverse safety measures. Various safety measures were implemented based on these policies to ensure containment integrity.

In addition to tsunami and earthquake, 40 natural events and 20 human induced external events were evaluated, following the US NRC and IAEA recommendations and guidelines.

A filtered containment venting system was installed at the plant. The system consists of the aerosol filter and the iodine filter. The aerosol filter is composed from three basic components: the scrubber nozzles; the mixing element, which can mix the gas and scrubber water and atomize bubbles to enhance decontamination factor; and the metal filter, which captures aerosol in the vent gas and separate droplets from the vent gas. The decontamination factor (DF) for the aerosol filter is greater than 1000. The silver zeolite iodine filter is located downstream of the aerosol filter. The DF for organic iodine is more than 50.

3.4. SESSION IV: STRATEGIES IN CONTAINMENT COOLING AND ENERGY MANAGEMENT FOR ADVANCED REACTOR DESIGNS

Four papers were presented covering different aspects of new reactor designs coping with potential containment pressurization during a severe accident and discuss the need or no need for external filtered containment venting system.

The first paper, entitled Strategies to Ensure Containment Integrity for EC6, introduced drivers for design enhancements with respect to the standard CANDU 6 design and described briefly the enhanced CANDU 6 Advanced Design Features, The enhanced CANDU 6 is developed based on the several new criteria dealing with the prevention and mitigation of severe accidents as considered in the revised IAEA Safety Standards for the design of new nuclear power plants, SSR-2/1 (Rev 1). The Enhanced CANDU 6 (EC6) is a 740 MW(e) reactor, evolved from the well-established CANDU line of reactors. The EC6 design consists of multiple lines of defence of preventative and mitigative features preventing uncontrolled radioactive releases during a severe core damage accident. The EC6 design addresses accident prevention, accident mitigation, severe accident resistance and recovery, and post-accident control and monitoring to meet the international standards for safety goals for new plants in terms of Core Damage Frequency (CDF) and Large Release Frequency (LRF). Hydrogen mitigation system by active hydrogen igniters and passive autocatalytic recombines and controlled pressure relief by means of an emergency containment filtered venting system (ECFVS) are the two key features regarding mitigating and preventing over pressurization in the containment and environmental release of fission products among many other design features of EC6. The proposed ECFVS is an AREVA's combined Venturi Scrubber FCVS. The Combined Venturi Scrubber Unit is connected at one end to the containment via vent piping and isolation valves and at the other end to the exhaust stack via a line equipped with the throttling orifice.

The second paper, entitled The Response of CAP1400 under the Condition of Fukushima Accident, presented MAAP analysis considering an accident scenario with initiators similar to those of Fukushima accident; earthquake, external flooding and loss of off-site power affecting availability of safety and control systems. The CAP1400 is an innovation plant, a Chinese design principally based on AP1000 but with a power level of 1400 MW(e), equipped with passive safety systems. The passive safety systems eliminate or decrease the number of active safety support systems compare to typical nuclear plants, such as safety AC diesel generator, cooling water etc. Furthermore, the plant is equipped with systematic severe accident prevention and mitigation strategies to establish effective defence against large fission products release, especially cooperating with the SAMG.

The MAAP study demonstrated that the design features enables even under the severe set of boundary conditions the containment pressure to remain under the critical level for which necessity of incorporation of a containment venting filter system is therefore not warranted. The analysis results show that the core could be cooled by the passive systems and the containment would be pressurized but the failure probability is still very low that the risk of large release is not a concern. The analysis has provided specific enhancement provisions, such as; 6 more PARs, needs for assurance of water and power sources after 72 hours, improvements of the environment monitoring system, and severe accident management guideline.

The third presentation, entitled ATMEA1 Strategy for Containment Cooling, described main design features of the ATMEA1 reactor, which integrates accumulated knowledge available in AREVA and Mitsubishi Heavy Industries (MHI), regarding EPR, KONVOI, N4, TOMARi3 and APWR. ATMEA1 is equipped with dedicated systems for severe accident mitigation, specifically; dedicated severe accident (SA) batteries feeding necessary monitoring systems and key valves as well as ensuring main control room (MCR) habitability, dedicated depressurization system to avoid core melt at high pressure, core catcher which spreads corium and prevents basemat degradation, hydrogen recombiners to avoid pressure increase in containment, pressure resistant containment and annulus for enabling radiological releases into the environment to remain below the acceptable level, severe accident heat removal system which cools the corium in the spreading area on the long term. Due to the design precautions containment pressure will not reach any critical level requiring venting, and hence filtered containment venting is not foreseen, however, may be implemented if required by a utility/safety authority due to other reasons.

The fourth and the last paper of the Session, entitled Long Term Containment Protection Strategies for the AP1000 Plant Design, depicted design features of the AP1000 specifically regarding coping with postulated severe accident conditions involving core melt. Containment protection strategies principally rely on; i) containment pressure control primarily by passive containment water cooling (PCS); and ii) containment venting into spent fuel pool (SFP) to address low frequency MCCI scenario in SAMGs. Containment venting is regarded being scenarios with very low probability, those associated with vessel failure, molten core concrete interaction (MCCI) leading to long term pressurization by noncondensable, flammable gas generation and failure of PCS. The spent fuel pool as the filtration media, if in place, provides adequate aerosol decontamination. AP1000 design precludes a need for external containment venting filter as may be the need for generation 2 plants under certain circumstances. Paper discusses in detail key features of AP1000 regarding of the passive containment cooling during a SBO transient, in-vessel retention of corium, hydrogen control features, and venting needs during molten core-concrete interaction.

3.5. SESSION V: FILTERED VENTING TECHNOLOGY

Seven papers were presented covering modelling aspects of filtered containment venting, as well as descriptions of experimental programmes and results in support of verification of components of certain FCVS designs, pool scrubbing aspects of jets and also depicted experimental studies of certain phenomena that may have potential to produce higher activity attenuation.

The first paper, entitled KAERI Activities on the Filter Containment Venting System Development, described in the first part efforts to design and develop an experimental facility which is aimed at verifying the performance of scaled-down version of Korean design of FCVS components, namely; a pool venturi scrubber, cyclone separator, particulate filter, and a molecular sieve filter. The test facility consists of a test vessel, aerosol/iodine generation and measurement systems. Tests are planned to qualify the performance of the FCVS, specifically for the thermal-hydraulic behaviour, removal of aerosol, and elemental and organic iodine as well as re-suspension of aerosols and re-volatilization of iodine. Additionally dynamic FCVS behaviour under a high-pressure gas ejection from the containment will be experimentally studied. The second part of the paper depicted the effects of thermal hydraulic conditions on iodine scrubbing and hydrogen concentration during the postulated FCVS operation using a station blackout (SBO) scenario undergoing in a Korean OPR1000, a target nuclear power plant in the Republic of Korea. The MELCOR computer code is the simulation tool. The predicted decontamination factors for metal iodide aerosol, specifically Cesium Iodide (CsI) aerosols, as a result of gas sparging in the scrubbing solution of the FCVS were presented. The possibility of hydrogen combustion during the initial operation phase of the FCVS was also discussed.

The second paper, entitled The European PASSAM (Passive and Active Systems on Severe Accident source term Mitigation) project: Experimental Studies for Atmospheric Source Term Mitigation with focus on Filtered Containment Venting Systems, described briefly the project objectives and the structure and some outcome of the current project. The project is aiming at experimental studies of certain phenomena in relation to FCVS that might produce potential for reducing radioactive atmospheric releases to the environment. The four year (2013-2016) project is conducted within the 7th framework programme of the European Commission (EC) and coordinated by IRSN and conducted by nine organizations in six countries: IRSN, EDF and university of Lorraine (France); CIEMAT and CSIC (Spain); PSI (Switzerland); RSE (Italy); VTT (Finland) and AREVA GmbH (Germany).

Experimental programmes specifically deal with potential improvements of existing source term mitigation devices (pool scrubbing systems; sand bed filters plus metallic pre-filters), as well as innovative systems which might enable achievement of larger reduction in the source term (Acoustic agglomeration systems; High pressure spray agglomeration systems; Electric filtration systems; Improved zeolite filtration systems; Combined filtration systems).

The third paper, entitled Advances on Understanding of Pool Scrubbing for FCVS based on the PASSAM project, produced a detailed account of experimental information obtained so far on: i) more detailed understanding of pool hydrodynamics coupled with aerosol retention; ii) aerosol retention in jet injection regime; iii) the effect of submerged structures and surfactants on the pool hydrodynamics and aerosol retention; iv) effect of submerged structures and additives on the organic iodine retention in water; v) pool hydrodynamics in the jet injection regime, and vi) for the medium and long term retention of iodine in water under FCVS conditions. Models have been developed to describe the investigated phenomena, and this work continues. It is foreseen that the data from the project will be implemented in severe accident codes, such as ASTEC, in the form of models and correlations.

The fourth paper, entitled Design of Containment Filtered Venting System (CFVS) for Tarapur Atomic Power Station (TAPS)-1&2 to limit the containment pressure below design pressure during design extension conditions, presented first analytical efforts to determine the boundary conditions for CFVS operation for selected accident scenarios, and the design objectives and goals of a CFVS, especially regarding its performance in filtering aerosol particles, gaseous elemental iodine and organic iodides from the vent gas. A dedicated test facility was designed and erected to provide aerosol and gaseous iodine species in a prescribed carrier gas at desired conditions and test a scaled down model of a full size unit, containing specifically a venturi based wet scrubber and mist removal located above the water pool. The water in the scrubber is doped with chemicals to attain the desired gaseous iodine removal efficiency. Tests conducted at selected conditions provided evidence that the CFVS will achieve the targeted decontamination factors. Thermal-hydraulic and radiological impact analyses conducted for TAPS1&2 plants demonstrated that the venting, initiated from the common chamber, which receives gas from one suppression pool not only depressurizes the containment under design extension conditions and enables the containment pressure to remain below the design pressure of 8 psig (0.55 Bar-gauge), but also limits the long term land contamination well below the Safety Goal of 1.014 Bq. CFVS CsI removal efficiency is assumed to be 100 in the study.

The fifth paper, entitled Filtered System for Purging Vapor-gas Discharges from Containment of WCR at Severe Accidents, introduced policy of "Rosenergoatom" Corporate Group OJSC regarding backfitting certain operating and future WWER designs (WWER-440 (RU V-213), WWER-1000 (RU V-187, RU V-320, RU V-338)) with filtered pressure vent and gas filtration (FPV) systems. OJSC's beliefs regarding expected deficiencies of western venting filter systems especially associated with the gaseous iodine filtration motivated design and testing of a filter system using a high heat resistant inorganic "Thermoxid-58" sorbent based on titanium dioxide. The tests conducted at SF NIKIET (in the Russian Federation) and abroad (Hanford in USA and Karlsruhe in Germany) as early as 1990s presented high purging efficiency of the bed for aerosols and volatile forms of iodine include organic iodides. The paper depicted in detail filtration characteristics of the filtration bed for aerosols, elemental iodine and organic iodide at different conditions. The paper presented various requirements specified for FCV systems for Rostov NPP Unit No. 1 and Kola NPP Unit No. 4, among which of interest to note is that the required decontamination factor for organic iodide is 100. The paper presented current improved FCVS configuration.

The sixth paper, entitled Characterization of the Performance of Wet Scrubbers used in FCVS, presented a summary of PSI research projects in the area of iodine and aerosol retention in water during the last three decades. Extensive investigations using a model version of a commercial containment venting filter and a dedicated test facility were briefly summarized providing key details on aerosol and gaseous elemental iodine and organic iodide retention. Recent experimental investigations on swell level and droplet generation and retention in the model FCVS system provided invaluable information for assessment of the system level code RELAP5, and an in-house developed code. Key details of the bubble hydrodynamics during the jet formation and break up in the sparer nozzle were introduced. PSI's extensive investigations with an aim of establishing a chemical process for a fast and efficient retention and decomposition of organic iodides in water was introduced. Aliquat 336 in combination with sodium thiosulfate provided highest retention of organic iodide in water with potential to suppress revolatilization of iodide ions in water under FCVS conditions.

The seventh and the last paper of the Session is entitled Lessons Learned from Analytical Re-Evaluation of a Venturi Scrubber Venting System Implemented in German NPPs through COCOSYS Analyses. After a short introduction of the wet scrubber – metal fiber containment venting filter system as implemented in some German reactors and its developmental test programme carried out in Java test facility, the results of the current re-assessment of the FCVS system are introduced. The analyses were performed using GRS COCOSYS code and under operational conditions predicted to occur for a Konvoi plant for two core melt accident scenarios. The analyses indicated that containment pressure development leading to the containment venting and during the venting was mainly dependent on the sump water inventory and operator actions. Hydrogen risk in the venting filter exhaust gas piping was shown to be sensitive to the operation of the off-gas ventilation system, which delivers the gas to the common FCVS-off gas channel and the stack. Thermal-hydraulic response of the FCVS predicted an initial water level increase in the scrubber due to steam condensation and followed by a continuous decrease due to the accumulated decay heat, which eventually requires water refilling of the FCVS. Comparisons of the COCOSYS results for the total aerosol and iodine load, associated decay heats and decontamination factors for aerosols and iodine are made with those requested by the German Reactor Safety Commission (RSK). The comparisons show that although conservative estimates were made in the COCOSYS analysis regarding modelling of iodine which yields deliberately: i) higher iodine concentration in the containment from boiling-sump; and ii) lower retention of iodine in sump (by silver reaction) the calculated decontamination factors still fulfil the required decontamination levels.

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SUBMITTED PAPERS

SESSION I

BACKGROUND AND REGULATIONS

SUMMARY OF OECD/NEA STATUS REPORT ON FILTERED CONTAINMENT VENTING

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Abstract. The Nuclear Energy Agency (NEA) is a specialised agency within the Organisation for Economic Co-operation and Development (OECD) based in Paris, France. The Committee on the Safety of Nuclear Installations (CSNI) within NEA assists member countries in maintaining and further developing the scientific and technical knowledge base required to assess the safety of nuclear reactors and fuel cycle facilities. In May 1988, CSNI organised an experts meeting in Paris to facilitate international exchanges on Filtered Containment Venting System (FCVS) concepts, research and development, and their analysis. This meeting also discussed the risk objectives, challenges to deployments of FCVS and procedures for the implementation. Following the Fukushima accident, the CSNI approved another activity for the documentation of a status report on FCVS in OECD countries, guidance for their improvement and future design. The report, "Status Report on Filtered Containment Venting", NEA/CSNI/R(2014)7 was published in July 2014. Post Fukushima activities in OECD/NEA member countries included stress tests of existing nuclear power plants. One of the many follow-up actions of the stress tests was for Nuclear Power Plants (NPPs) operators to consider the implementation of filtered venting of the containment to prevent significant radiological releases. The basic premise of FCVS is that a catastrophic failure of the containment structure can be avoided by discharging steam, air and non-condensable gases like hydrogen to the atmosphere. However, unfiltered containment venting could result in significant radioactive releases to the environment. These releases may be largely reduced by the implementation of filtering systems on the venting lines, a concept that is already integrated in some NPPs. From a safety perspective, filtered containment venting is desirable for the on-site management of the accident and for the protection of populations from potential radiological exposures, by reducing releases as much as technically feasible. In NPPs without containment venting systems, the incorporation of FCVS could be considered as part of severe accident management (SAM) measures to enhance the response capability to severe accident (SA) events. In addition, some countries are considering upgrading existing FCVS and their operation procedures for safe and reliable use in conditions which were not necessarily fully addressed at their design stage (e.g. robustness to hazards and hydrogen combustion loads, prolonged or repetitive use during a SA and manual operation without power supply). The CSNI report details safety design and qualification requirements for FCVS, filtered containment venting strategies for emergency operating procedures and SAM domains, implemented filtration technologies, source term evaluations performed in view of FCVS and provides guidance for the improvement of existing systems and for the design of future systems. The main outcomes of the report are presented in this paper.

1. INTRODUCTION

Filtered containment venting is one of the actions that can protect the containment and the facility while mitigating radioactivity releases to the environment during a Severe Accident (SA) event. The basic premise of Filtered Containment Venting System (FCVS) is that, independent of the state of the reactor, the catastrophic failure of the containment structure can be avoided by discharging steam, air and non-condensable gases like hydrogen to the atmosphere. Unfiltered containment venting could result in significant amount of radioactive releases to the environment, depending on the time of release, however, these radioactive releases may be largely reduced by the implementation of filtering systems on the venting lines. From a safety perspective, filtered containment venting is desirable for the on-site management of the accident and for the protection of populations from potential radiological exposures, by reducing releases as much as technically feasible.

Several different types of FCVS have been deployed at Nuclear Power Plants (NPPs) worldwide since the 1980s, in response to the Three Mile Island accident. Since then the understanding and calculation tools required to assess radioactive releases during the course of a severe accident have progressed significantly as have filtration technologies. In light of these developments, the CSNI organized an experts meeting in May 1988 [1], in Paris, to facilitate international exchanges on FCVS concepts, research and development, and their analysis. This meeting also discussed the risk objectives, challenges to deployments of FCVS and procedures for their implementation.

The Fukushima Daiichi NPP accident demonstrated that, in the absence of alternatives to reduce the containment pressure build-up, due to steam and non-condensable gas accumulation, the venting of the containment can become an essential accident management measure for the preservation of its structural integrity. The accident also highlighted: (1) that the design and operation requirements of the existing venting systems have to be reassessed both in terms of the strength and pressure resilience when containment venting is initiated and (2) the importance of having a filtration capacity to reduce environmental releases for safe and reliable use in situations resulting from extremely damaging events.

Stress tests after the Fukushima accident led many countries to consider the implementation of FCVS and strategies at their NPP where these are not currently applied as an enhancement of the response to SA events [2–4]. Also, they led some countries to consider improvement of existing venting systems and of operational procedures to enhance robustness to hazards and hydrogen risk, safe use and efficient filtration for prolonged or repetitive use in a SA event.

FCVS implemented before Fukushima were mainly designed to manage long term pressure build-up in the containment. New FCVS, on the other hand, may be designed to deal with more challenging conditions such as managing early phases of an accident, cycling or long term use during severe accident events. The robustness (including a design capable of withstanding several external events), safe use, and reliability of FCVS for such conditions should be further assessed either to improve existing systems or to propose upgraded design requirements for future systems.

The objective of the paper is to summarize the status of Filtered Containment Venting (FCV) presented in the recently published OECD/CSNI status report [5]. This report describes the current status of the technology and venting strategies in several countries, the envisaged developments for possible improvements to filtration technologies, general design requirements and specific design recommendations, notably for the qualification of filter technologies for reliable function and performance in SA situations.

2. PURPOSE AND BENEFITS OF FILTERED CONTAINMENT VENTING

The main purposes of the FCVS is preventing containment overpressure failure and maintaining the containment pressure below its design value while minimizing radioactivity releases to the environment during a SA event. This is achieved by venting the containment in a controlled manner, through a filtration system that captures a significant portion of radioactivity and prevent large releases that could occur compared with either unfiltered venting or containment failure.

FCVS are typically to be used in SA as part of the overall applied Severe Accident Management (SAM) strategy for PWRs and BWRs, while they are also used in Design Basis

Accidents (DBA) for some Pressurized Heavy Water Reactors (PHWRs) (CANDUs) [5]. In addition, a FCVS can be used in some cases as an additional means of hydrogen risk reduction and, in the case of smaller BWR containments, for decay heat removal. Thus, an FCVS reinforces the function of nuclear containment by minimizing radiological health consequences and preventing large amounts of land contamination. That said, it is worth emphasizing that the FCVS is not the only available system in NPPs to ensure containment integrity in the event of a SA.

Following the Fukushima accident, there was renewed interest in some countries in evaluating the necessity and the consequences of installing engineered filters for use during SAs. In particular, studies and analysis are ongoing in the United States, as part of the regulatory assessment, to determine whether FCV is indeed a viable cost-beneficial mitigation strategy. Such an analysis will differentiate between densely and sparsely populated areas in which NPPs are located. Each country should draw its own conclusions about the cost-benefit ratio of a FCVS taking account of the specific characteristics of individual sites.

2.1. Hydrogen risk reduction

Venting the containment in a timely manner can reduce hydrogen migration to the reactor building or any other adjacent areas. This, in turn, can reduce the risk of any hydrogen explosion as reportedly occurred in Fukushima. The extent of hydrogen migration depends on pressure reduction due to venting and on effluent flow rate. A high effluent flow rate, while reducing the hydrogen build-up in the reactor building, may temporarily increase the hydrogen concentration in the vent line and potentially increase the risk of hydrogen combustion in the line.

2.2. BWR-specific benefits

Early venting prior to core damage is being explored as a means of keeping the Reactor Core Injection Cooling (RCIC) available for a longer period of time in BWRs with a Mark I containment to optimize the operation of the RCIC. With the RCIC running for an extended period, core degradation and vessel failure can be delayed.

Venting can also remove decay heat more effectively than other methods of heat removal for smaller BWR containments. Some BWRs have the provision for dry-well and wet-well venting. Generally, wet-well venting with the suppression pool temperature below saturation is the preferred option when there is a breach of the Reactor Coolant System (RCS) because it provides an additional benefit of scrubbing fission products. Prior to breaching of the RCS, dry-well venting may provide better heat removal from the upper dry-well and be of benefit for dry-well head seal leakage concerns.

Venting steam from a boiling wet-well in a BWR accident with complete loss of heat removal from the wet-well is another benefit. In German BWRs, such a strategy is implemented for Beyond Design Basis Accidents before any core degradation is reached.

Venting before failure of the reactor pressure vessel is another strategy that may have some associated benefits, particularly for BWRs. Such a strategy would concern accident scenarios with no initial or induced break in the RCS. Nevertheless, it seems challenging at this time to predict with any accuracy the timing of vessel rupture which can occur quite rapidly after core melt.

2.3. Radiological consequences and accident management

The most important benefit of a FCVS with respect to radiological consequences, when compared to the use of hardened vent only, lies in the fact that a FCVS can greatly reduce such consequences, both on-site and off-site, in the case of a SA event at a NPP. In particular, cesium and iodine releases can be expected to be much less with filtration when compared to a situation with unfiltered releases. Thus, a FCVS can facilitate on-site accident management as well as limit radiological impact on the population in the short term. FCV can also reduce the extent of land contamination and perhaps simplify the treatment of contaminated land in the long term. In this way, installation of FCVS may also lead to higher public acceptance of nuclear power plants.

Filtration inside the containment itself can be of high value by limiting to a large extent the amount of radioactive aerosols exiting the containment. In BWRs, wet-well venting and in other reactors, pre-filters or filters placed inside the containment (e.g. for French reactors and some of the FCVS for German reactors, see Fig. 1) offer this possibility. When available, such filtration reduces the challenge to any external filtration placed downstream in terms of aerosols and out-of-containment dose loadings.

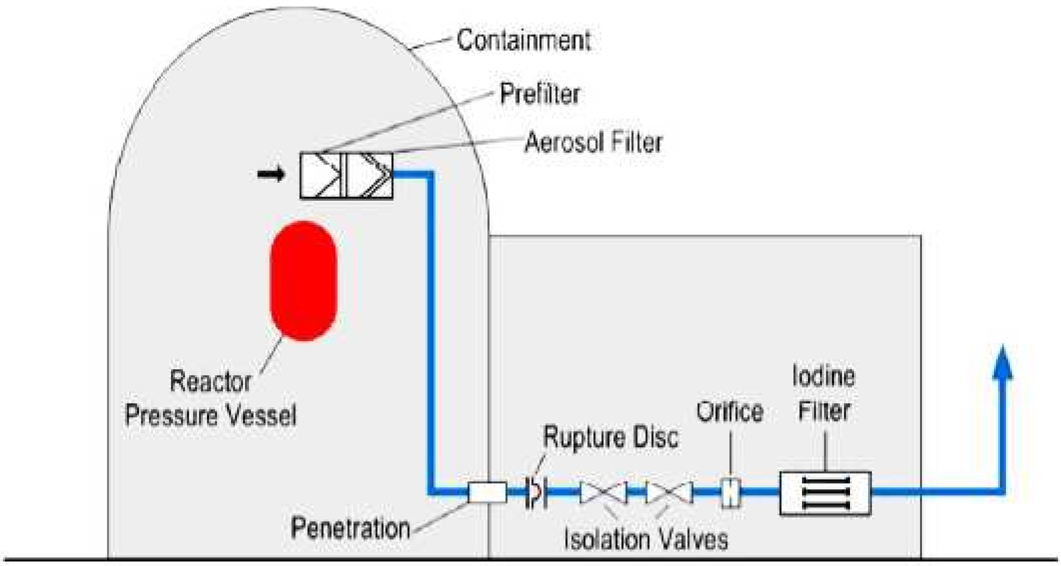

FIG. 1. Example of FCVS implementation at a German NPP [5].

3. VENTING STRATEGIES AND PROCEDURES

Venting strategies considered in various countries show different timings for vent initiation (a few hours up to a day or two after the beginning of the accident) and a range of pressures at initiation (between 2 and 9 bars) depending on the reactor type, the containment strength and volume (faster pressurization of smaller containments), and applied SAM procedure (e.g. contribution to the containment pressurization of steam produced by fuel-coolant interaction at the vessel rupture when the reactor pit is flooded).

Venting strategies and procedure definitions are generally based on analyses of accident progression in the reactor pressure vessel, in the containment, and in the reactor building taking into account all release pathways, the effect of the venting and the assessed margin to containment failure. Performing such analyses and assessment remain challenging.

The possibility of early venting to prevent early challenge to the containment integrity is considered for certain reactor designs in some countries [5]. It is expected that the venting would then operate prior to any significant activity release to the containment. The possibility to use FCVS to cope with large early activity release is considered in some countries but excluded in others [5], due to the challenge it represents in terms of filter capacity and efficiency since activity deposition in the containment would be limited in the early period. In countries excluding the use of FCVS for early venting, other Severe Accident Management Measures (SAMM) are being designed to eliminate the risk of early containment failure.

If no early challenge to the containment integrity is expected, it is desirable to maintain the containment closed as long as possible to maximize the benefit from activity deposition inside the containment and the extra time it offers for evacuation and off-site protection measures.

In some cases, venting strategies also consider the possibility of multiple or long venting as long as no other cooling system is recovered to evacuate the containment heat. The Fukushima accident showed that in situations with prolonged station black-out and loss of heat sinks, the application of such strategies will certainly be beneficial.

FCVS may be passively (e.g. by failure of a rupture disc at a certain pressure level) or actively actuated depending on the system implemented. In any case, the opening and closing of the venting line should be made easy and certain. Thus, provisions including radiological protections, operations from a remote location, and reduced intervention times are generally incorporated.

4. FCVS DESIGN REQUIREMENTS

In this section a summary of FCVS design requirements which should be considered for their implementation is provided. FCVS should allow reliable containment decay heat transfer and pressure decrease for expected SA thermal-hydraulic conditions, including conditions resulting from energetic events (e.g. hydrogen combustion). FCVS should be designed for efficient filtration of radioactive materials at SA temperature, flow and pressure ranges encountered in the FCVS.

The pipes, supports, and valves of the venting line should be designed for safe and reliable operation for the expected dynamic loads, including hydrogen combustion loads or provisions should be set to keep all parts of the system at a safe distance to hydrogen combustion limits. Sufficient aerosols (tens to several 100 kg), gaseous iodine (few g to tens of g) and heat load (a few to hundreds of kW) capacities should be set depending on the expected vent timing and duration.

The system should be designed to have sufficient autonomy to cope with prolonged or multiple uses. For instance, the autonomy time without maintenance is expected to be at least 24 hours for liquid-type filters (provisions should be made to operate such systems for longer times). The system should be designed for a safe maintenance and operation during severe accident events and to be manageable in post-accident operations. The system should be designed to be robust to external hazard, in particular to seismic hazards.

Source term evaluations are generally performed to determine the desirable filter performances. These evaluations showed that the Decontamination Factor (DF) should be > 1000 for radioactive aerosols including for small size (in the 0.1 μm range) and hygroscopic aerosols. The filtration of 0.1 µm-size for hygroscopic aerosols is more challenging. Such a DF would allow reducing the on and off-site long term radiological consequences (due to radioactive aerosols containing ^{134}Cs and ^{137}Cs isotopes) to manageable levels. A larger decontamination factor >10000 may be desirable when considering early venting with high filter loads and/or to further reduce long term land contamination due to Cs isotopes.

The decontamination factor should be > 100 for molecular iodine and >10 for organic iodides to reduce the on and off-site short-term radiological consequences (due essentially to the 131I isotope) to a level compatible with on-site workers and public protection measures.

No requirement is presently considered for the filtration of gaseous ruthenium as its relative contribution to the radiological consequences has to be more thoroughly assessed. The required R&D to tackle this issue is presently ongoing.

There is presently no requirement concerning noble gas retention. On one hand, no reliable technology exists to date for efficient retention of these species in SA situations and, on the other, the benefit of reducing their releases in the environment has to be balanced with drawbacks that could result on-site from activity accumulation in the system designed for their retention.

Re-suspension/re-volatilisation from filter deposits during a prolonged use in a SA should not challenge the target DFs. The R&D to assess such processes in FCVS is presently ongoing.

5. INSTALLED FILTERING TECHNOLOGIES

The well-known existing filtration technologies consist of liquid scrubbers, deep-bed filtration and different sorption systems. Highlighting of any particular FCVS technology or product in this paper is simply a reflection of the fact that such a system is currently available and information has been provided by the corresponding designer in the OECD status report [5]. Detailed description of the systems cited in this section is given in the status report. Other systems are being developed and will be commercialized in the future.

There are presently two main types of systems implemented in NPPs:

Dry solid filters: solid filtration is generally ensured by layers of metal fibre filters (as for instance in the Dry Filter Method developed by Westinghouse. In France, Électricité de France (EDF) implemented a metal pre-filter on the venting line inside the containment which is supplemented by a sand-bed filter placed outside of the containment;

Liquid scrubber filters: filtration and steam and heat removal is ensured by a liquid pool scrubber, usually coupled with a droplet separator and deep-bed fine aerosols filters. For instance, Westinghouse developed the FILTRA-MVSS system (multi venturi scrubber system); CCI [a] developed a system which is implemented in Swiss NPPs and AREVA developed a high speed sliding pressure venturi (HSSPV) scrubber system. The pool of the CCI system is chemically-doped to improve gaseous iodine retention.

Critical aspects for the dry solid filters are the filter clogging and the reduced filter efficiency in wet conditions. Thus provisions have to be made to avoid clogging and the presence of water in the filter in SA situations. Critical aspects for the liquid scrubber filters are the filter efficiency and its maintenance for a prolonged use close to saturation conditions in SA situations. Thus provisions have to be made for a prolonged and safe use in SA situations.

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[a] Name of an engineering and manufacturing company.

It should be underlined that there exists some flexibility in the implementation of some of the proposed systems at NPPs. Part of, or the complete system, may be implemented inside the containment (Fig. 1). In-containment filtration (or pre-filtration) offers the advantage of limiting radioactivity transfer outside the containment and on-site worker exposure. This may ease on-site interventions following venting.

There is also the possibility of adding sorbent stages at the system outlet to retain remaining radioactive species, in particular gaseous iodine species. Some of the recently developed FCVS integrate additional sorbent stages with this objective. Such systems are also described in the report [5].

The status of the implementation of FCVS in OECD countries NPPs is provided in Table 1. Note that multi-Unit CANDU stations in Canada are equipped with specific emergency filtered air discharge systems (EFADS) which are designed to filter aerosols and gaseous iodine for Design Basis Accidents [5]. Part of the information concerning the existing filtration-systems performance is proprietary and was not disclosed by FCVS designers. However, two major aspects can be underlined concerning existing systems:

Most of the available systems were designed on knowledge-bases which were existing in the late 1980s [1, 6, 7]. Some have been updated, depending on the system design and implementation, based on the consideration of relevant R&D results and plant safety reviews, notably additional filtration stages to modify the overall filtration efficiency and provisions to improve their safe use during SA.

Given the possible extension of the domain of FCVS use, as envisaged following the Fukushima accident, the demonstration of the system performance should be consequently extended to more challenging conditions.

It was shown, through extensive qualification campaigns (notably the ACE tests [8, 9]) performed for postulated SA conditions, that existing systems and technologies allow the realization of desirable DFs for radioactive aerosols (DF > 1000) and molecular iodine (DF > 100). However, besides aerosols and molecular iodine, more attention is currently being given to organic iodides [10] and iodine-oxide particles [11] as they may contribute significantly to the source term in some accidents. Some recent developments were made to improve the filtration of organic iodides. Systems that may provide under prescribed conditions, DF in the range of 5 to 50, have also been developed.

Some attention is also given to volatile ruthenium-oxide filtration in some R&D projects since it is suspected that in some specific SAs where oxidizing conditions would prevail in the Reactor Pressure Vessel and the RCS (steam-rich or air flows), significant volatile ruthenium oxide fractions could reach the containment [12].

FCVS implemented before Fukushima were mainly designed to manage long term pressure build-up in the containment and as stated earlier, new systems may be designed to deal with more challenging conditions such as the management of early phases of an accident, cycle or long term use during SA conditions. The reliability of the filters for such conditions (high filter loads and resulting thermal and radiolytic effects, long term operation under challenging conditions) need to be further assessed, particularly if control of aerosols/iodine re-vaporisation/re-volatilisation becomes a requirement.

TABLE 1. STATUS ON THE IMPLEMENTATION OF FCVS IN OECD COUNTRIES [5]

Country	NPPs	No FCVS	HSSPV	Metal + sand-bed	DFM	FILTRA-MVSS	SULZER CCI	EFADS	Comments
Belgium	7 PWR		○						Implementation of HSSPV+ planned in 5 units between 2016-2018, under evaluation for remaining two units which were just granted operation beyond 2015
Brazil	3 PWR		○		○				
Bulgaria	2 WWER 1000		●						
Canada	19 PHWR		●		○			●	HSSPV for Point Lepreau single unit. DFM planned in 4 Darlington units. 4 EFADS shared among 18 multi units reactors, EFADS designed for DBA. Plans to have emergency power so EFADS can be used in BDBA.
Czech Republic	4 WWER-440	■							No FCVS planned
	2 WWER-1000	■							Under assessment in conjunction with SAMM for corium cooling (in or ex-vessel)
Finland	2 WWER 440	■							FCVS is not feasible in WWER-440 due to the steel shell of containment that is vulnerable to sub-atmospheric pressures; EPR plant under construction will be equipped with FCVS
	2 BWR		●						
France	58 PWR			●					Metal pre-filter inside containment, sand bed filter outside containment
Germany	6 PWR		●		●				DFM in 2 PWR units
	2 BWR		●						
Japan	24 PWR		○		○				DFM proposed in some PWRs; HSSPV and other scrubber system (TEPCO design) planned for BWRs. For BWRs and PWRs, additional FCVS planned as "Specialized Safety Facility"[b]
	26 BWR		○						
Mexico	2 BWR	■							Hardened CVS from wet and dry well (Mark II) under implementation
Netherlands	1 PWR		●						
Romania	2 PHWR		●						
Slovakia	4 WWER-400	■							Under assessment with other SAMM
Russian Federation	6 WWER-440 11 WWER-1000	■							Under assessment for some WWERs after Fukushima. Not considered for other reactors (RBMK, …)
Slovenia	1 PWR				●				DFM installed in 2013 in Krško NPP, used in passive mode with rupture disk
Korea, Republic of	19 PWR	□							Only Wolsong-1 Unit equipped with HSSPV. Other units are planned to be equipped with FCVS by 2020, development of a Korean system under consideration
	4 PHWR	□	●						
Spain	6 PWR		○				○		PWR: FCVS implementation is planned by 2016 BWR: Hardened venting available. Filter implementation is planned in 2017
	1 BWR	□							
Sweden	3 PWR					●			
	7 BWR					●			
Switzerland	3 PWR		●				●		HSSPV type at Gösgen
	2 BWR					●	●		F-MVSS type at Mühleberg
Ukraine	2 WWER-440 13 WWER-1000	□							
USA	69 PWR	■							Preparation of guidance documents on Hardened CVS for BWR Mark I & II. Implementation by 2018 or earlier[c].
	35 BWR	■							

■ no FCVS □ planned but design not yet selected ● installed ○ planned *
Taiwan, China: 2 PWR and 4 BWR

[b] Refers to systems to suppress large releases caused by containment failure as a result of extreme external events.
[c] The rulemaking effort for containment vent filtration and accident management strategies is in progress. Implementation of an external filter is contingent on the outcome of the rulemaking effort.

6. POSSIBLE ADVERSE EFFECTS RELATING TO THE IMPLEMTENTATION OF FCVS

This section briefly discusses the potential adverse effects related to the implementations of a FCVS.

6.1. Unintended release of radioactivity

The FCVS hardware and procedures can have the desired effects only if the systems are designed and operated as intended. For example, the timing and the criteria used for vent opening are important with regard to the potential and mode of containment failure. Delayed venting at high containment pressure (e.g. in excess of design pressure or primary containment pressure limit) may increase the likelihood of penetration failure and for BWRs, head flange leakage. Further, it may fail to prevent more catastrophic containment failure. Early venting at low containment pressure, on the other hand, may lead to unintended early release of radioactivity prior to implementing adequate protection measures for the population. Thus, for radiological consequences to remain acceptable, early venting should be considered only when the radioactivity levels in the containment are low enough and/or if the FCVS is adequately qualified for efficient reduction of releases.

6.2. Potential for containment function impairment

In the event the vent cannot be closed, a pressure decrease in the containment may result in an inflow of air. Moreover, venting may have the unintended effect of de-inerting the containment as any inert gas will be purged in the process. The combined effect of de-inerting the containment and of steam condensation is likely to create conditions that are conducive to hydrogen combustion, if there is inflow of air due to vent opening.

6.3. Qualification against external events

An external filtering device should be qualified for relevant external events dependent on the NPP site location and the relevant risks there. A system which is not seismically rugged may contribute to the risk of containment impairment in case of an earthquake. Likewise, a system which is not designed for and/or protected from other external hazards such as high flood level, external fire, or extreme meteorological events, may contribute to the risk of containment impairment.

6.4. Hydrogen risk

Hydrogen in the FCVS (vent piping, stack, etc.) can form combustible mixtures in the presence of oxygen due to air ingress from vent actuation. This situation can be aggravated by steam condensation if the vent line is cold and poses a risk when a FCVS has not been initially designed to withstand the dynamic loads that would result from hydrogen combustion (if the latter has not been excluded by other means). If a common vent line is used, as in some PWRs where the vent line ends in a common collecting space in the building exhaust system or in the entrance to the chimney, the possibility of hydrogen combustion there has to be analysed.

6.5. Operation mode and operator exposure

It is questionable whether an automatic start of the venting by failure of a rupture disc at a certain pressure level is an easily manageable option when a requirement to terminate venting arises. Such an automatic start would involve an automatic termination or manual termination. The automatic termination seems to be risky as instrumentation logic and power sources for such action may not be available in a SA. This may lead to a long-lasting and uncontrolled open containment which should be avoided. On the other hand, the Fukushima accidents showed that manual operation of the FCVS may be required in a Long Term Station Blackout accident. The requirement for manual termination would, in turn, require physical accessibility of equipment and proper protection of the operators.

6.6. Filter loading

There are different vent procedures under discussion, continuous or intermittent venting, and FCVS capable of operating up to 72 hours. Also, venting that can be started a few hours to more than a day after the reactor vessel rupture. Early and/or long term venting might lead to some constraints for the filters related to the risk of overloading them with radioactive products which can diminish the filtration efficiency and reduce the venting rate.

7. POTENTIAL IMPROVEMENTS OF FCVS

The CSNI status [5] identified a number of improvements that can be recommended for FCVS and these are briefly discussed in the following sections.

7.1. Venting strategies and operations

As said earlier, venting strategies have shown different timings and ranges of pressures for vent initiation. One challenge to implementing such strategies is to be able to realistically assess the actual margin to containment failure. Factors that should be considered include penetration failure during SA conditions, leakage through concrete walls at pressures above the design pressure, opening (often reversible) of flanges, hatches, and other structural components, and the possibility of sudden pressure escalation due to energetic events in the containment.

If there is no early challenge to containment integrity, it is desirable to maintain the containment closed as long as possible. The success of such a strategy will depend on the effectiveness of long term accident management actions based on outside means (e.g. portable diesel generators to start up safety systems, etc.). Learning from the Fukushima accident, incorporation of a pressure limit for vent initiation (e.g. slightly above the design pressure of the containment) should be considered.

If early venting initiation is required during the accident, it has implications for aerosol loading of the filter. It is desirable, whenever possible, to use some form of pre-filtration inside the containment to limit the amount of radioactivity exiting the containment (e.g. wet-well venting in BWRs and pre- filtration or filtration inside the containment for dry containment venting). In this respect, it is important to ensure that the FCVS effectiveness is not compromised by, for instance, clogging of the filters. Also, measures are needed to maintain the aerosols within the filters after the use of FCVS is terminated.

Generally, to determine the efficacy of the venting strategies, it is important to analyse, in a systematic manner the accident progression in the reactor pressure vessel, containment, reactor building, and all release pathways, taking into account the impact of the vent process. Recent documented analyses in the US for BWRs with a Mark I containment have shown an obvious impact of venting on the accident progression as well as the mode and timing of failure.

It is understood that venting should only be initiated if, at the time of opening, there are no challenging conditions for the system termination that might not be overcome. Then there is the question of whether manual venting operation is acceptable considering the risk to the workers. There should be provisions to reduce these risks as much as possible. The option of having redundant systems to make the venting termination more reliable should be investigated.

In the case of early venting, the risk of hydrogen combustion in the FCVS can be high for de-inerted containments or containments that may not be inerted to start with. New systems could be designed and operated so as to ensure that flammability limits of gases passing through the system are not reached; on the other hand, they could be designed to withstand dynamic loadings resulting from hydrogen combustion. Specific solutions already exist in some plants to reduce the risk, e.g. inerting of the vent line (requiring inert gas supply), heating of the line to prevent steam condensation (requiring power supply), or a separate exhaust line up to the exit of the chimney. If recombiners are installed in the containment, the risk of hydrogen combustion in the FCVS will be reduced as less oxygen would be available. Also, the vents can be designed to minimize the potential for hydrogen gas migration and ingress into the reactor building or other buildings. It may be desirable to avoid sharing a FCVS line between two or more units.

Finally, to reduce the risk of failure of the overall venting process, investigation is required to determine whether the implementation of more than one vent line (possibly connected to different containment locations) is advantageous. In general, the redundancy of such systems should be further assessed.

7.2. Filter systems

With respect to the filter performance, more than 99.9% (DF > 1000) retention for aerosols and 99% (DF > 100) for molecular iodine was considered sufficient when the first FCVSs were implemented in the 1980s and 90s (such DF values were set as target values by some regulators). With the possible evolutions of FCV strategies (notably if FCVS are considered for the management of early phases of an accident, with fast pressurization, and for prolonged use in long term phases of an accident), and given the preference in some countries to further reduce an accident's radiological consequences, higher DFs may be desirable (DF > 10000 for aerosols). Existing technologies may already perform at that level.

However, there is a concern that the filter efficiencies of certain filter types as quoted by filter vendors may not be independently validated by data, notably for a wide range of aerosol sizes and for a wide range of thermal loadings. In particular, performance of existing and innovative filter technologies in more challenging conditions needs to be validated with regard to their claimed efficiency.

Retention of organic iodine compounds is less than 80% for some existing systems (DF < 5), and there is an interest in having higher organic-iodide retention since, as recent research results have shown, these compounds may dominate the in-containment gaseous iodine inventory. There is undoubtedly a lack of qualification of the filtration technologies for these compounds. Work is in progress in the MIRE/PASSAM[d] projects to gain additional knowledge on organic iodide filtration [13].

Another perceived deficiency of the current filtering technology relates to filtration of noble gases which could be beneficial both in easing on-site interventions and in reducing population and environment exposure during the accident. It might be of interest to investigate further potential retention of these species. The benefit of reducing their releases has to be balanced with drawbacks that could result on-site from radioactivity accumulation due to their retention.

In core-melt accidents when conditions are oxidizing ruthenium, re-volatilization from RCS deposits could result in significant gaseous ruthenium tetroxide fractions inside the containment which may contribute significantly to radiological consequences. Current filtration technology does not appear to provide the capability to filter ruthenium tetroxide. Work is in progress in OECD/STEM[e] programme to gain knowledge on ruthenium re-vaporization and in MIRE/PASSAM to gain knowledge on gaseous ruthenium tetroxide filtration [13].

The impact of high dose rates and thermal loads on the filter effectiveness in the course of the venting process needs careful evaluation. The impact of certain energetic events that could occur in the containment on the filter effectiveness also requires careful evaluation. In particular, detailed phenomena concerning the migration of radioactive species for high loadings and prolonged or intermittent use in SA conditions should be investigated.

7.3. Other areas

Additional effort should be placed in ensuring the availability of power sources for FCVS operation. These power sources could then require implementation with diverse redundancy.

For effective accident management, it is desirable to have a specific instrumentation dedicated to the measurement of hydrogen, radioactive aerosols and iodine concentrations in the containment and in the FCVS exhaust line. Such instrumentation must provide the operators and decision makers with reliable information, even in the worst conceivable conditions.

Finally, it is recommended that the FCVS be seismically qualified so that it does not contribute to the risk of containment impairment in the event of an earthquake. Also, provisions should be made so that it does not threaten the proper functioning of other important safety systems in such situations.

8. CONCLUSION

Generally, all contributing countries recognized the potential benefits of FCVS for emergency response, reduction of the extent of land contamination, the mitigating effect on public health and increased social acceptability of nuclear power plant installations while documenting this status report. FCVS should, however, be considered in conjunction with

[d] MIRE — French Release Mitigation R&D Programme dealing with FCVS; PASSAM — European Severe Accident Mitigation R&D Programme dealing with FCVS.

[e] Source Term Evaluation and Mitigation project.

other SAM strategies (e.g. no large benefit is expected for containment by-pass scenarios which have to be managed by other SAMM).

FCVSs implemented before Fukushima were mainly designed to manage long term pressure build-up in the containment; the new FCVSs may perhaps be designed to deal with more challenging conditions (management of early phases of an accident, cycling or long term use in SA conditions). The robustness (including a design withstanding several external events), the safe use and the reliability of FCVSs for such conditions should be further assessed either to improve existing systems or to propose upgraded design requirements for future systems.

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FILTERED CONTAINMENT VENTING SYSTEM FOR EXISTING NUCLEAR POWER PLANTS: DESIGN PARAMETERS AND ASSOCIATED SAFETY CRITERIA IN BELGIUM

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Abstract. Following the post-Fukushima stress tests performed in Belgium, Filtered Containment Venting Systems (FCVS) are foreseen to be installed on Belgian Nuclear Power Plants (NPPs) as ultimate means to cope with slow pressurization of the containment that could occur during potential Severe Accidents (SA). After an introduction about the Belgian context, information about the regulatory framework is provided, as well as the main objectives of the FCVS. Then, the main part of the paper discusses the methodology supporting the definition of the FCVS main design parameters and associated safety criteria developed by Tractebel Engineering, on behalf of the licensee Electrabel – GDF Suez, and validated by the Belgian nuclear Safety Authorities. This methodology followed a series of subsequent steps starting with the definition of the safety referential applicable to the FCVS and leading to the identification of a reference sequence, namely the Complete Station BlackOut (CSBO). Then, key parameters for the FCVS design were defined (e.g. pressure range, venting flow, stored decay heat, stored mass, Decontamination Factors (DF)) along with the associated safety criteria. Several of these key parameters were assessed through a well-defined set of supporting calculations. This set included both best estimate representative cases as well as conservative bounding cases performed with the state of the art SA codes MELCOR 1.8.6 and, subsequently, ASTEC V2.0 (CPA/IODE modules), in a chaining methodology. The aim of this approach was to take advantage of the iodine chemistry models features in the CPA/IODE modules of ASTEC to calculate the iodine speciation in the containment and in the FCVS, without having to run a full scope ASTEC calculation. Finally, an overview of the assessment by the Belgian TSO, Bel V, as well as the follow-up actions and next steps are reported, with the aim to provide information about the current status of the actions and implementation of the requirements related to the FCVS.

1. INTRODUCTION

Belgium has 7 Pressurized Water Reactors (PWRs) on 2 sites, i.e. Doel (4 units) and Tihange (3 units). Currently, there is no FCVS installed in the Belgian NPPs, but studies are ongoing to install such systems at the units that will operate beyond 2015, as shown in Table 1.

The installation of the FCVS at the older units, Doel 1/2 and Tihange 1, was envisaged in the framework of the Long Term Operation (LTO), and was foreseen at the other units as result of the stress test after the Fukushima Daiichi accident. After the Belgian government decision (mid 2012) to permanently shutdown Doel 1 in February 2015 and Doel 2 in December 2015, the related FCVS feasibility studies were stopped. However, analyses for the implementation of FCVS were recently restarted for the Doel 1/2 units, since a long term operation (beyond 2015) has been proposed by the new government.

Design studies for the implementation of FCVS are currently being completed for every NPP of Tihange and for the units Doel 3 and Doel 4 (5 of the 7 Belgian PWRs, i.e. the units which will operate beyond 2015). The licensee has chosen a wet-type design filtration system and plans a progressive installation starting in 2016 and depending on the NPP outages. The deadline of the overall project schedule for the installation of the FCVS is the end of 2017.

The paper focuses on every NPP of Tihange and on the units of Doel 3 and 4. For Doel 1/2 units, if the LTO is decided and authorized, a FCVS could be installed before 2018/2019. The same procedure followed for the previous units is progressing for the Doel 1/2 units, thanks also to the experience gained in the previous process.

TABLE 1. CHARACTERISTICS OF BELGIAN UNITS

Unit	Type	Thermal power MW(th)	Designer	Containment	Date of first criticality	Expected shutdown date
Doel 1	PWR (2 loops)	1312	Westinghouse	Double containment (steel and concrete)	1974	*2015**
Doel 2	PWR (2 loops)	1312	Westinghouse		1975	*2015**
Doel 3	PWR (3 loops)	3054	Framatome	Double containment with inner metallic liner and annular space in underpresure	1982	2022
Doel 4	PWR (3 loops)	2988	Westinghouse		1985	2025
Tihange 1	PWR (3 loops)	2873	Framatome / Westinghouse		1975	2025
Tihange 2	PWR (3 loops)	3054	Framatome		1982	2022
Tihange 3	PWR (3 loops)	2988	Westinghouse		1985	2025

**2025 proposed*

2. REGULATORY FRAMEWORK

Before the Fukushima Daiichi accident, the regulatory framework for consideration of installation of FCVS was the Periodic Safety Review (PSR).

Following the safety assessment performed in the framework of the stress tests of the Belgian NPPs, the Belgian Federal Agency for Nuclear Control (FANC, i.e. the Belgian Nuclear Safety Authorities) and Electrabel – GDF Suez (i.e., the Belgian Nuclear Utility) agreed on an action plan to further enhance the safety of the Belgian NPPs against Fukushima-type events [1]. Main actions required were to perform a study for the implementation and then implementation of FCVS on all Belgian units.

Article 21.4 of the Belgian Royal Decree of 30 November 2011 [2], requires that the containment shall be protected from overpressure in a severe accident. This requirement is highlighted in the Western European Nuclear Regulators Association (WENRA) reference level F.4.5 [3].

There is however no further national requirement or specific guidance related to the implementation of FCVS.

Recently FANC published guidance about the safety demonstration for new nuclear installations, in which quantitative radiological safety objectives (SO) are set for accidents (SO_2) and severe accidents (SO_3), caused either by internal events or by external hazards [4]. Although the scope of this guidance does not include existing nuclear installations, during their future PSRs these safety objectives might be considered as a reference.

3. OBJECTIVES OF THE FCVS

The main objective of the FCVS is to preserve the containment integrity to avoid any uncontrolled radioactive release towards the environment due to a potential SA provoking a slow pressurization of the containment above or close to its design pressure.

Additionally, as containment venting is inherently associated with radioactive releases, a second objective is to limit the latter ones as far as reasonably achievable by filtering the fission products passing through the venting system during SA conditions.

The FCVS is however seen as an ultimate mitigation mean and should only be relied upon as the last resort in the frame of a global SA mitigation strategy, in accordance with the emergency plan of the Belgian NPPs.

4. METHODOLOGY FOR ESTABLISHMENT OF FCVS DESIGN PARAMETERS AND ASSOCIATED SAFETY CRITERIA DEVELOPED BY THE LICENSEE

The first step taken in this methodology was to categorize the design parameters in two groups, namely either requiring specific supporting calculations or not.

The latter group concerned on one hand general specifications, for instance related to the global design, the Emergency Planning of the Belgian NPPs or the foreseen SA mitigation strategy. On the other hand, this group also comprised the design parameters related to the opening and the closing pressure of the FCVS, assessed based on the available fragility curves of the containment buildings, as discussed in Section 4.2.

This section focusses then on the former group, i.e. on design parameters requiring specific supporting calculations and for which specific methodologies had to be developed by Tractebel Engineering (on behalf of the licensee, Electrabel – GDF Suez), as detailed in Section 4.3 and Section 4.3.4. As shown in Table 2, two assessment methods were defined for these parameters, one relying solely on MELCOR calculations and one relying on both MELCOR and ASTEC-CPA/IODE calculations. The second method was of particular importance to more properly evaluate the impact of the iodine chemistry than solely relying on the MELCOR code, historically used in Tractebel Engineering and for which full input decks of Belgian NPPs have been developed.

More details about the methodology were presented by Tractebel Engineering in [5], only a short overview is reported below.

TABLE 2. DESIGN PARAMETERS REQUIRING SUPPORTING CALCULATION, SAFETY CRITERIA AND ASSESSMENT METHODS

#	Design parameter	Safety criteria	Assessment method
1	Venting mass flow rate	7.6 kg/s	From MELCOR 1.8.6 calculations
2	Mixture temperature and composition	217 °C and explosion risk to be avoided	
3	Decay heat in FCVS	50 kW	From MELCOR 1.8.6 and ASTEC V2.1 (CPA/IODE) calculations
4	Mass retained in FCVS	100 kg	
5	Decontamination factor (DF)	>1000 for Aerosols >100 for Inorganic Iodine >10 for Organic Iodine	

4.1. General specifications

The considered FCVS is a wet-type filter with the following general specifications [6]:

1) The actuation of the vent shall require manual operation, based on a specific criterion associated to the measurement of the containment pressure;
2) The opening of the venting valve shall be possible from the control room as well as locally, via a manual actuation;

3) The access to the rooms and systems needed for the correct operation of the FCVS should be guaranteed;
4) The shielding should be sufficient to limit the dose to a maximum of 50 mSv per operator and per intervention that is estimated to last for around 1h 30;
5) The FCVS only operates during beyond design basis events. Therefore, the single failure criterion does not apply. System redundancy is thus not foreseen except for the containment isolation function;
6) The FCVS must be available during all plant operating states except during the periods when maintenance activities have to be performed on the FCVS;
7) After the first opening of the containment isolation valves, the FCVS is designed to operate without assistance during at least 24 h;
8) The permanently installed reserves of water and chemical additives on site shall allow for a FCVS autonomy of at least 72 h after the accident initiating event;
9) After a period of maximum 10 days from the accident initiating event, it is assumed that the conventional safety systems (for example cooled recirculation) would have been recovered ensuring a long term controlled stable state of the NPP;
10) Provisions against explosion of combustible gases in the FCVS should be foreseen;
11) The FCVS is actuated when the “venting opening pressure” is reached, and stopped when the pressure decreases below the “venting closing pressure”. These plant specific thresholds are determined based on containment design pressure and fragility curves. Multiple venting phases are thus considered if the containment cooling system is not recovered;
12) The FCVS has to withstand the following external hazards: extreme earthquake, extreme external flooding and extreme weather conditions (extreme winds, lightning, rainfall, temperature range, snowfall).

4.2. Criteria assessed based on fragility curves

The safety criteria related to the opening pressure was assessed based on the fragility curves of Belgian units with the objectives to limit the containment failure probability to maximum 5 %, in accordance with the Westinghouse Owner Group (WOG) Severe Accident Management Guidances (SAMG). The criteria for the closing pressure was also assessed based on these fragility curves but with the objective to reach a pressure that does not threaten the containment integrity.

This approach resulted in setting the opening pressures at 4.5 bars abs. or 5 bar abs. depending on the NPP and the closing pressure at 3 bar abs. for all NPPs.

4.3. Criteria assessed based on MELCOR 1.8.6 calculations

4.3.1. Determination of SA scenarios to be used

The reference scenario leading to a SA consists in the complete loss of all the systems able to inject water into the primary circuit and/or containment, i.e. a Complete Station BlackOut (CSBO) with the loss of all internal and external electrical power sources. The

CSBO has therefore been considered as the initiating event for both the representative[f] and bounding scenarios for the assessment of the FCVS.

This selection is reinforced by the analysis of the Belgian Level 2 Probabilistic Safety Assessment (PSA) model, dealing with internal events, that shows that within the sequences which would most likely lead to the containment venting pressure, 96 % of those are bounded by the CSBO. Moreover, as the current Belgian PSA models do not take into account external hazards[g], one should expect that CSBO-like initiating events would actually have an even larger contribution which reinforces even further the selection of the CSBO sequences.

4.3.2. Venting mass flow rate

The venting mass flow rate is directly linked to the FCVS flow area and in definitive, the choice of the FCVS flow area should be based on the desired venting period. A venting period of about 12 h is considered appropriate for the bounding case. As the decay heat of the representative scenario at the time of venting is lower than in the bounding case, the venting period of the representative scenario will be shorter. This consideration led to set the design criterion related to the venting mass flow rate to a value between 5 and 10 kg/s for all units, which is thus to be seen as the required value to be reached and handled by the FCVS vendor in all operating conditions.

4.3.3. Mixture temperature and composition

The temperature and composition of the vented mixture along with the maximum flammable gases concentrations can be obtained once the relevant scenarios are identified. This information is directly extracted from the MELCOR 1.8.6 calculations.

For the mixture temperature, a value of about 200 °C is obtained, which is of importance notably for the heat-up and potential evaporation of the FCVS scrubbing liquid.

Besides, thanks to the use of Passive Autocatalytic Recombiners (PARs) installed in all Belgian NPPs, almost all the hydrogen and the majority of the carbon monoxide produced during the SA have been recombined before the first opening of the FCVS. However, flammable mixture could be found in the FCVS system during later ventings. Indeed, the depletion of oxygen prevents the PARs from recombining the H_2 and the CO. Such a risk will be taken into account by the FCVS supplier based on the results of the supporting calculations.

4.3.4. Criteria assessed based on MELCOR and ASTEC-IODE calculations

As presented in the previous sections, MELCOR 1.8.6 has been used for the quantification of FCVS design parameters by modelling specific accidental sequences using full scope plant models.

However, MELCOR 1.8.6 only possesses a simple iodine pool chemistry model with limited use and underestimating the possible concentration of CH_3I in the containment atmosphere. Organic iodides like CH_3I are assumed the most dangerous as they are very volatile and because, unlike I_2, they are not adsorbed on surfaces and the efficiency of their filtering or scrubbing in water solutions is generally low [7].

It has therefore been decided to use an ASTEC-CPA/IODE (and SYSINT for cases with containment sprays) or a stand-alone ASTEC-IODE model to obtain a better picture of

[f] Most likely leading to FCVS opening, accounting for alternative means investigated in post stress test studies.
[g] Internal hazards such as fire and flooding are in the process to be implemented in the PSA.

the iodine speciation and amount in the gas flowing through the filter and consequently to allow a better determination of the associated Decontamination Factors (DFs).

Although not the most realistic case, the conservative hypothesis is made that all the iodine exiting the primary circuit is released as gaseous I_2, Similarly, the amounts of IO_3- evaluated by ASTEC-IODE and sent to the FCVS are considered under gaseous form as well.

Tractebel Engineering developed a methodology for the speciation of the iodine source term by means of chaining MELCOR and ASTEC-IODE. More details can be found in [5].

4.3.5. Decay heat and mass retained in FCVS

The speciation of the iodine source term by the use of the ASTEC-IODE code allowed relaxing the source term sent to the FCVS by considering more properly the retention of iodine inside the containment. The resulting source term considered for the evaluation of the decay heat and of the mass retained in the FCVS is composed of the following categories:

(a) Aerosols → computed by MELCOR 1.8.6;
(b) Inorganic iodine → I_2 and IO_3^- computed by ASTEC-IODE;
(c) Organic iodine → CH_3I computed by ASTEC-IODE.

The results of this assessment were that values of 50 kW and 100 kg were found to be required as minimum values respectively for the decay heat and for the mass retained in the FCVS with the conservative assumption that the FCVS acted as an absolute filter for all aerosols and iodine species.

4.3.6. Decontamination factors

The source term speciation was also of primary importance regarding the assessment of the DF of the FCVS. Indeed, these DF are ensured in practice by different aspects/devices of the FCVS for the three categories composing the source term (i.e. aerosols, inorganic iodine, and organic iodine) and can therefore be quite different.

As there exists currently no Belgian regulation setting limits of releases in case of SA (or any associated limit), which would have supported the definition of those DF, the following methodology, relying on a marginal gain approach and accepted by the Belgian Safety authorities, was developed to cope with the large uncertainties existing on radiological evaluations:

1) The evaluation of a global effective dose due to the venting of the containment was performed with simplified assumptions. The goal was indeed not to derive any realistic absolute value of such a dose but solely to determine a single parameter able to regroup at once the impact of the three categories composing the source term and their associated DF;
2) This evaluation was then performed for successive increases of the filtration efficiency.

In conclusion of this marginal gain approach, the minimum DF to be required could be set as follows, which have later on be found in agreement with international benchmarks:

(a) Minimum DF for aerosols = 1000;
(b) Minimum DF for inorganic iodine = 100;
(c) Minimum DF for organic iodine = 10.

Besides, for the aerosols, a complementary estimation was made on a longer term regarding the impact of ^{137}Cs on the ground contamination. This evaluation comforted the minimum required DF of 1000 for the aerosols by comparing the obtained values to the Swiss limits of Decontamination, Relocation and Interdiction 0.

Moreover, the FCVS shall respect the following conditions:

(1) The DFs must be guaranteed on the whole pressure range for which the FCVS could be used and must be guaranteed over the whole venting periods;
(2) The FCVS design shall ensure that fission products are retained in the filters (taking into account several venting periods) and that the decay heat is evacuated on the long term.

5. ASSESSMENT BY BEL V

The methodology for establishment of FCVS design criteria has been defined by the licensee (i.e. Electrabel – GDF Suez and its architect-engineer Tractebel Engineering) and assessed and validated by Bel V, the TSO of the Belgian Nuclear Safety Authorities.

The purpose of Bel V's assessment has been to enable decision making regarding acceptance of the methodology developed by the licensee for the definition of the design parameters and the associated safety criteria. In this framework, a series of technical meetings were organized between the licensee and Bel V.

In particular, for the design parameters requiring supporting calculations, the assessment by Bel V covered the verification of the adequacy of codes and code versions, plant modelling methodology, plant data, accident scenarios and associated assumptions, and calculation results.

For this purpose, another series of technical meetings were performed for presentations to and assessment by Bel V of severe-accident-code model development, validation and use, and for the audit of related documentation according to Bel V's practice. According to that practice, documents presenting at least the code description and the qualification report, which gives a summary of the code qualification elements, have been received before the audit. While some recommendations were provided to the licensee, the general conclusion of the review of the different aspects covered during the audit was that Bel V had no objections against the use of MELCOR and ASTEC and the associated plant models for the FCVS design criteria evaluation.

This phase of the project, i.e. the finalization of design criteria, has been completed before the selection of the FCVS vendor by the licensee and before the detailed design of the FCVS for each unit.

6. FOLLOW-UP ACTIONS AND NEXT STEPS

In the course of the discussions regarding the FCVS design criteria, numerous questions have been raised by Bel V. Answers to those questions have been provided by the licensee through different notes, letters and presentations at meetings.

The evaluation by Bel V of the final selection of FCVS design (and vendor) consisted of verifying whether the final choice was compliant with the well-defined set of quantitative and qualitative criteria described in this paper (Section 4). The rationale of the selection was also explained by the licensee on request of Bel V, with the aim to provide evidence that a robust solution was chosen. This was requested since Bel V was not involved in the final selection process itself, but could only verify whether the resulting selection was acceptable.

In the next step, the selected FCVS vendor was requested to submit to the licensee a preliminary version of the complete FCVS qualification file. A review process has been foreseen before the submittal of the final version of this file. The complete FCVS qualification file has been reviewed by Bel V (a similar process has been followed from the licensee side). The check of the adequacy of the actual design, with respect to the design requirements, has been carried out for different subjects:

- filter system performances;
- functional/operational design: these include a proper operation of the filter system during either normal operation or a severe accident, including the absence of cliff-edges in the filter design;
- mechanical design.

A meeting between the vendor, the licensee and Bel V, which took place in the vendor's premises, aimed at addressing in more detail all remaining issues and further questions of the licensee or Bel V, and to consult confidential documents, reports and background data.

The assessment by Bel V of the remaining issues about qualification file is almost completed. The detailed design phase is in progress and the overall project schedule for the installation of the FCVS at 5 of the 7 Belgian PWRs is planned by the end of 2017.

7. CONCLUDING REMARKS

The present paper describes the methodology developed by Tractebel Engineering and accepted by the Belgian Safety Authorities supporting the definition of the FCVS main design parameters and associated safety criteria.

The methodology presented relied mainly on the use of the MELCOR 1.8.6 code as well as on the ASTEC V2.0 code through its CPA and IODE modules. The main reason to rely on these modules was to take profit of their added value in terms of iodine behavior modelling without performing full scope ASTEC calculations. This approach was realized thanks to a chaining methodology that allowed the post-processing of MELCOR data to be used as CPA and IODE inputs.

The obtained values for the FCVS main safety criteria have been presented along with some information about the assessment by Bel V, follow-up actions and next steps foreseen in the ongoing safety assessment of the FCVS that are foreseen to be installed on Belgian nuclear power plants.

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REGULATORY OVERSIGHT OF FILTERED VENTING AND CONTAINMENT INTEGRITY AT CANADIAN NUCLEAR POWER PLANTS

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Abstract. In the case of a severe accident (SA) at a nuclear power plant, filtered containment venting (FCV) is an option that can, in the absence of other means, to reduce the build-up of pressure in containment from steam and incondensable gas accumulation, while mitigating radioactivity releases to the environment. In providing an additional line of defense, a FCV, would be an important tool to help manage an accident, protect the surrounding population, and reduce on- and off-site contamination. The focus of this presentation is to summarize the current status and near future plans for the operating Canadian NPPs, regarding venting options. The paper will discuss how the FCV fits into the larger context of severe accident management to protect containment integrity and unfiltered releases into the environment. The incorporation of FCV into the regulatory framework of the CNSC will also be presented.

1. INTRODUCTION

The accident at the Fukushima Daiichi nuclear power in 2011 was one of the world's worst nuclear accidents. As a result of the damage to different systems when the site was hit by a major earthquake and tsunami, the power plant lost the ability to keep its three operating nuclear reactors cool. Three units eventually melted down, lost the integrity of their reactor containment systems, and emitted large quantities of radioactive fission products into the surrounding environment.

This nuclear event prompted nuclear operators and government regulators around the world to evaluate the safety of their nuclear infrastructure. The Canadian Nuclear safety Commission (CNSC), Canada's nuclear regulator launched a review of all major nuclear facilities in Canada. Led by a multidisciplinary CNSC task force, the review confirmed the ability of Canadian facilities to withstand and respond to credible external events, such as earthquakes. However, to strengthen defence in depth, a four-year action plan was instituted to enhance emergency preparedness, response capabilities and to improve CNSC's regulatory framework and processes.

In case of severe accidents in CANDU reactors, releases steam and non-condensable gases into an intact containment — due to failures of fuel channels, the calandria vessel, and potentially the calandria vault — would cause the containment pressure to increase beyond the design values. In CANDU reactors, periodic or deliberate venting, in conjunction with water make-up to various systems is postulated to keep the containment pressure under control and to prevent calandria vessel/containment failure as well as help mitigate the progression of a severe accident. In providing an additional line of defense, to protect containment integrity, FCV would be an important tool to help manage an accident, protect the surrounding population, and reduce on- and off-site contamination.

2. THE REGULATORY FRAMEWORK IN CANADA

The CNSC has a regulatory framework that emanates from the Nuclear Safety and Control Act (NSCA) and associated regulations. The CNSC's regulatory documents provide greater detail on requirements set out in the NSCA and regulations on a broad range of topics, such as safety analysis, site evaluation, licensing, life extension of existing nuclear power plants, emergency planning, and severe accident management programmes.

3. NUCLEAR POWER PLANT PROFILE

The CANDU reactors in Canada are designed and focused on safe operation, and accident prevention and mitigation to minimize risk to the public and the environment. Each nuclear power plant in Canada has redundant and diverse safety systems designed to prevent accidents and reduce their effects, should they occur. All of these systems are maintained and inspected regularly to ensure plants meet or exceed the CNSC's safety requirements

The overview of CANDU plants are listed in Table 1. There are currently 5 stations in Canada 4 of which are in operation. Canadian nuclear power industry continues to supply over 15 % of Canada's electricity.

TABLE 1. OVERVIEW CANDU PLANTS IN CANADA

<table>
<tr><th rowspan="2">Station</th><th rowspan="2">Units</th><th rowspan="2">Power/ MW(e)</th><th colspan="2">History</th><th rowspan="2">Status</th></tr>
<tr><th>Dates</th><th>Service Life</th></tr>
<tr><td rowspan="8">Bruce</td><td>A1</td><td>904</td><td>1977/2012</td><td rowspan="2">Refurbished and returned to service</td><td>Operating</td></tr>
<tr><td>A2</td><td>904</td><td>1977/2012</td><td>Operating</td></tr>
<tr><td>A3</td><td>904</td><td>1978/2003</td><td rowspan="6">In service within design life</td><td>Operating</td></tr>
<tr><td>A4</td><td>904</td><td>1979/2003</td><td>Operating</td></tr>
<tr><td>B1</td><td>915</td><td>1985</td><td>Operating</td></tr>
<tr><td>B2</td><td>915</td><td>1984</td><td>Operating</td></tr>
<tr><td>B3</td><td>915</td><td>1986</td><td>Operating</td></tr>
<tr><td>B4</td><td>915</td><td>1987</td><td>Operating</td></tr>
<tr><td rowspan="4">Darlington</td><td>1</td><td>935</td><td>1992</td><td rowspan="4">In service within design life</td><td>Operating</td></tr>
<tr><td>2</td><td>935</td><td>1990</td><td>Operating</td></tr>
<tr><td>3</td><td>935</td><td>1993</td><td>Operating</td></tr>
<tr><td>4</td><td>935</td><td>1993</td><td>Operating</td></tr>
<tr><td rowspan="8">Pickering</td><td>A1</td><td>542</td><td>1971/2003</td><td>Refurbished and returned to service</td><td>Operating</td></tr>
<tr><td>A2</td><td></td><td>1971</td><td></td><td>Safe storage state</td></tr>
<tr><td>A3</td><td></td><td>1972</td><td></td><td>Safe storage state</td></tr>
<tr><td>A4</td><td>542</td><td>1971/2003</td><td>Refurbished and returned to service</td><td>Operating</td></tr>
<tr><td>B1</td><td>540</td><td>1983</td><td rowspan="4">In service within design life</td><td>Operating</td></tr>
<tr><td>B2</td><td>540</td><td>1984</td><td>Operating</td></tr>
<tr><td>B3</td><td>540</td><td>1985</td><td>Operating</td></tr>
<tr><td>B4</td><td>540</td><td>1986</td><td>Operating</td></tr>
<tr><td>Gentilly 2</td><td>1</td><td></td><td>1983</td><td></td><td>Transitioning to safe storage</td></tr>
<tr><td>Point Lepreau</td><td>1</td><td>705</td><td>1983/2012</td><td>Refurbished and returned to service</td><td>Operating</td></tr>
</table>

4. REQUIREMENTS FOR FCV IN CANADA

Regulatory oversight and requirements in Canada for FCV are not explicitly mandated however, issues with containment integrity and the releases into the environment are regulated by the CNSC through regulatory documents and other associated documentation embedded in its regulatory framework and processes. In Canada, Containment is an essential requirement for NPPs in Canada and there are therefore systems in place to limit the release of radioactive material following a nuclear accident. Based on the CNSC Fukushima Task Force recommendations, more advanced FCV system are valuable, but the important aspect is that the integrity of reactor containment is maintained, and off-site releases are minimized.

Following the Fukushima Daiichi accident in 2011, a CNSC Fukushima Task Force compared how a similar, large external event would impact nuclear power plants in Canada, and developed several recommendations on how their resilience could be improved. Filtered venting, particularly using designs that could be deployed during a SA, was recommended as a means to improve containment performance and prevent unfiltered releases of radioactive products. Expectations for FCVS going forward are that designs be able to handle the large gaseous discharges, aerosol loads, and fission product loads that could be expected during a SA, and also consider the potential for hydrogen combustion downstream of the event and, where applicable, multi-unit accidents. The goal of these recommendations was to strengthen reactor defence in depth.

In Canada, the regulatory documents that apply in the context of FCV are the following:

(1) Regulatory document REGDOC 2.5.2, Design of Reactor Facilities: Nuclear Power Plants [1] which superseded Regulatory Document RD 337and is applicable for new plants outlines the following:

- Section 8.6.9, Containment pressure and energy management:
  - (a) The design shall enable heat removal and pressure reduction in the reactor containment in operational states, DBAs and DECs. Systems designed for this purpose shall be treated as part of the containment system, and are capable of:
    - — minimizing the pressure-assisted release of fission products to the environment;
    - — preserving containment integrity;
    - — preserving required leak tightness.
  - (b) Plant design shall be capable of meeting safety goals:
    - — Severe Core Damage Frequency: Sum of frequencies of all events < $1x10^{-5}$ per reactor year;
    - — Large Release Frequency: Sum of frequencies of all events < $1x10^{-6}$ per reactor year(^{137}Cs, 1014 Bq);
    - — Small Release Frequency: Sum of frequencies of all events < $1x10^{-5}$ per reactor year (^{131}I, 1015 Bq).
  - (c) Containment system shall have design capability to:
    - — control hydrogen concentration to prevent deflagration;
    - — remove fission products, hydrogen and other combustible gases.

(2) Regulatory document, REGDOC 2.3.2, Accident Management: Severe Accident Management Programmes for Nuclear Reactors [2] (which superseded Regulatory Guide document GD 306) which are outlined in:

- Section 3.1 Severe accident management goals:
  - (a) maintaining containment integrity;
  - (b) minimizing the release of radioactive products into the environment.
- Section 4.2 Accident analysis:
  - (a) identify the challenges to fission product boundaries in different reactor states, including shutdown states;
  - (b) verify that SAM actions would be effective to counter challenges to protective barriers.
- Section 5.1 Preventive and mitigating actions:
  - (a) controlling the containment pressure and temperature;
  - (b) controlling the concentration of flammable gases;
  - (c) controlling radioactive releases.

The new regulatory documents (REGDOCs 2.3.2 and 2.5.2) include amendments to reflect lessons learned from the Fukushima nuclear event of March 2011, and to address findings from the CNSC Fukushima Task Force Report.

5. FILTERED VENTING AND ACCIDENT MANAGEMENT

In order to fulfil REGDOC 2.3.2 requirements, the Industry have developed Severe Accident Guidelines (or SAGs), to enable and control conditions in containment. They are SAG-1: Injection into the HTS, SAG-2 , Control Moderator Conditions, SAG-3: Control Shield-Tank Conditions, SAG-4: Reduce Fission Product Releases, SAG-5 Reduce containment Hydrogen, SAG 6 and 7 to reduce containment conditions; and inject water into containment respectively. It is worthy of note that SAGs 4, 5 and 6 may involve filtered venting as part of their strategies to control containment conditions and maintain its integrity.

6. FILTERED DISCHARGE SYSTEMS IN CANDU REACTORS IN CANADA

Prior to the Fukushima accident, Canadian multiunit CANDU stations had emergency filtered air discharge systems (EFADS) employed as part of the overall containment design to help avoid releases of radioactive particulates and iodine into the environment during design basis accidents. This is based on deterministic requirements of single and dual-failure criteria of loss of coolant accident and loss of cooling injection for containment performance analysis using assessments of accident progression and safety system performance through Level 1 Probabilistic Risk Assessment (PRA).

In Beyond Design Basis, assessment of accident progression and containment performance is through Level 2 PRA. The following are considered; safety goals, internal and external events, including seismic, fire and flood and other extreme weather conditions, as deemed necessary.

During the Point Lepreau NGS's 2008-2012 refurbishment, an emergency filtered containment venting (an AREVA HSSPV-type) system was installed in addition to the installation of passive autocatalytic recombiners (PARs) to manage combustible gases and maintain containment integrity in the event of a severe accident.

Some of the technical specifications of the seismically qualified FCV system are given as follows:

- Manually and remotely operated with no external power;
- Design pressure of 400 kPa(g);
- Maximum design temperature of 200 °C;
- Design load of aerosols up to 300 kg;
- Decay Heat removal greater than 16 kW;
- Operate passively to relieve containment pressure with retention rates of:
    - — aerosols greater than 99.9 %;
    - — molecular cesium and fission products(CsI, RbI, CsOH, RbOH) greater than 99.5 %;
    - — Elemental iodine > 99 %;
    - — Organic iodine of approximately. 80 %.

7. FILTERED DISCHARGE SYSTEMS POST-FUKUSHIMA AND FUTURE PLANS

In order to strengthen reactor defence in depth, identification of safety enhancements was considered post Fukushima. Some of them were:

- make-up water capability for various coolant reservoirs;
- installation of PARs to reduce hydrogen concentration;
- installation of emergency containment filtered venting.

In this regard an upgrade to the EFADS so that it can function during BDBA scenarios for multiunit CANDU stations was considered. It is expected that provision for emergency power as well as manual operation of values would be retrofitted into the EFADS. In addition, consideration is being given to the installation of advanced filtered designs. For example, Darlington NGS is presently proposing a Westinghouse Dry Filter Method FCV system during its impending refurbishment. In view of this, it is necessary that an assessment of these forms of filtered venting be assessed for their benefits in Canada during a severe accident. Bruce NGS has presently provided future tie-in into its containment pending the decision for an appropriate design for its FCV system.

8. DOMESTIC STUDIES/RESEARCH AND DEVELOPMENT

To disseminate scientific information and inform the Canadian public when required, CNSC engages in domestic research and development. One such proactive research related to containment integrity and filtered venting is the evaluation of a filtered containment venting system and how it would help to reduce the overall consequences of a hypothetical severe nuclear accident in CANDU-6 containment [4].

The following essential design considerations for any FCV system were considered:

- flow capacity;
- mass loading capacity;
- heat loading capacity.

Ongoing parametric studies are being performed using MAAP-4 CANDU code on vent line size, vent initiation time, filter efficiency, filter clogging, and heat load. In addition, an investigation of the consequences of release would also be performed using the ADDAM Code.

Preliminary results reveal that FCV with core debris cooling could help, establish a containment heat sink, relieve temperatures and pressures from within, provide a controlled filtered release pathway to the environment and retain fission products in the filtering media (pool, scrubber, etc.). In addition, vent line size should be sized to handle the large pressure spike that occurs after the calandria vessel failure.

9. INTERNATIONAL COLLABORATION

CNSC works with IAEA, OECD/NEA and other regulators in accident management and its consequences [5, 6]. For example CNSC participation and contribution to some WGAMA Projects related to FCV and containment integrity are:

- Containment filtered venting;
- Hydrogen mitigation;
- Fast running tools;
- Severe accident management programme (SAMP).

10. CONCLUSION

Maintaining containment integrity is essential to minimizing radioactive releases in accident management. In Canada, effectiveness of existing systems have been confirmed during post Fukushima assessments. However, safety improvement opportunities were recommended of which most of them have been completed or in the process of being implemented.

The benefits from having FCV as an option provides an additional defence in depth, protects containment, prevents releases and operates independent of the state of the reactor. Due to these benefits, FCV is a system that has been applied in one form or another by the Canadian utilities.

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[5] OECD NUCLEAR ENERGY AGENCY, Status Report on Filtered Containment Venting, NEA/CSNI/R(2014)7, July 2014.

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REGULATIONS AND IMPROVEMENTS IN HUNGARY RELATED TO SEVERE ACCIDENTS AND FILTERED VENTING

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1. REGULATIONS IN HUNGARY

At that time when Paks NPP was designed and commissioned (1982–1987), the beyond design basis events had not been analyzed, so there were no analyses if these events can lead to a severe accident situation or not, as this was not required at that time. The requirements regarding to the beyond design basis events appeared just 15 years later in the domestic nuclear safety regulations.

The most important regulations (without being exhaustive) related to severe accidents and containment for existing power plants are the following which can be found in the Nuclear Safety Codes Volume 3:

3.2.2.0300. The following categories of the extended design basis shall be distinguished:

(a) DEC1: complex accident not resulting in melting of the fuel in the active core and the spent fuel pool;
(b) DEC2: a severe accident resulting in a considerable melting of fuel.

3.2.2.3930. The DEC analyses and designs shall identify all reasonably achievable measures by which severe accidents can be prevented. Irrespective of the success of the identified measures, preparations shall also be made for severe accidents. As part of the analyses and design, all reasonably implementable solutions by which the consequences of severe accidents can be limited shall also be identified.

3.2.2.4300. Implementation of specific design solutions or preventive accident management capabilities shall ensure that the occurrence frequency of the following accidents with catastrophic energy release in the reactor vessel or within the containment is infinitesimal: a) reactivity accidents with prompt criticality, including heterogeneous boric acid dilution, b) steam explosion, and c) hydrogen detonation.

3.2.2.4310. In order to minimize uncertainties and to increase the robustness of the nuclear power plant unit, during the demonstration of practical screening ability, demonstration based on physical impossibility shall be preferred to demonstration on a probabilistic basis.

3.2.2.4400. During the design the accident management functions and the accident management systems for pressure reduction and hydrogen removal shall be determined to such extent that high pressure processes leading to fuel melting and early containment damages can be avoided.

3.2.2.4500. The mitigating functions for accident consequences and systems providing these functions shall be determined to such extent that in a severe accident the molten core can be retained and cooled down within the containment.

3.2.2.4610. The required means of accident management shall be designed and accident management guidelines shall be devised for the efficient mitigation of the consequences of beyond design basis events analyzed in detail, including severe accident processes resulting in a complete fuel meltdown, in such a way that any hazard posed to the environment and the population remains below a predefined, manageable level if the procedures and means of accident management work successfully.

3.2.2.4620. The special design requirements set for safety systems shall be applied to the means of accident management only to the extent reasonably achievable. The means of accident management shall not adversely affect the fulfilment of the design basis safety functions.

3.3.2.1100. It shall be ensured that the physical-chemical properties of the materials used in the containment prevent significant hydrogen production during events resulting in TA2 to 4 operating conditions.

3.4.6.0200. For the performance of physical barrier and monitored release function of the containment:

[…]

(c) in the case of DEC2 operating conditions, the release of radioactive materials shall be limited both in time and quantity in order to ensure that:

(ca) sufficient time is available for introducing population protection measures, if necessary;

(cb) the long term contamination of large areas can be avoided.

(d) the concentration of radioactive aerosols and radioiodine shall be reduced in the containment atmosphere.

[…]

(h) heat removal from the containment, protection of the structure against overpressure and the handling of flammable gases generated shall be ensured in all operating conditions, (i) the cleaning of the atmosphere of the containment or the filtering of gaseous media released from the containment shall be provided.

(k) the destructive effect of core melting on the structural integrity of the containment shall be prevented or shall be limited to the extent reasonably achievable.

3.4.6.1300. The containment as a system shall comprise the following:

(a) all important parts of the primary circuit;
(b) systems capable of controlling pressures and temperatures;
(c) isolation elements;
(d) instruments serving for the management and removal of fission products, hydrogen, oxygen and other materials released into the containment atmosphere.

3.4.6.1400. The heat removal system of the containment shall ensure the quick reduction of the pressure and temperature in the containment following a loss of coolant event, and then it shall ensure their maintenance at a reasonably achievable low level, assuming single failure.

3.4.6.1600. Technical solutions shall be applied in the design of the containment for DEC1 and 2 operating conditions, for the monitoring and control of pressure and temperature in the containment, as well as for the management of combustible gases. The leaktightness of the containment shall not decrease in a significant extent during a reasonable period following such events.

3.4.6.1800. The containment isolation shall be possible even in DEC1 and 2 operating conditions. If an event leads to environmental release by bypassing the containment, the consequences must be mitigated. If an event leads to bypassing the containment, design solutions that prevent any damage to the fuel elements with high certainty shall be provided.

3.4.6.2000. The cleaning of the containment atmosphere shall be so performed that the systems which provide the management and monitoring of the fission products, hydrogen, oxygen and other material potentially released into the containment atmosphere, ensure the reduction of the quantity and concentration of the fission products released into the environment, assuming single failure, as well as the control of concentration of hydrogen and oxygen in the containment atmosphere, in order to provide the integrity of the containment.

For new nuclear power plants there are more strict requirements (few examples without being exhaustive):

3a.2.2.7200. At least the following events shall be practically excluded by design solutions or the implementation of preventive accident management capabilities, i.e. it shall be demonstrated that their occurrence is physically impossible or the frequencies of their occurrence are less than 10^{-7} /year with high certainty:

[…]

(c) all loads appearing in the short and long run, which may jeopardise the integrity of the containment, in particular, the dropping of a heavy load, steam and hydrogen explosion, interaction between the molten core and concrete loadbearing structures, and containment overpressurzation, […];

(e) loss of coolant with open containment, which may cause core dry out.

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

(a) regarding operator interventions:

[…]

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

[…]

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

2. SAFETY IMPROVEMENTS BEFORE FUKUSHIMA RELATED TO SEVERE ACCIDENTS

Due to the changes in the regulations, HAEA (Hungarian Atomic Energy Authority) prescribed at the end of the first PSR (Periodic Safety Review, 1998) to perform a level 2 PSA which is suitable for the evaluation of the consequence-mitigating strategies (regarding high probability events, and events are leading to significant radiological releases) until the end of 2003 and to finalize the severe accident management strategy until the end of 2004. The aim of the implementation of the severe accident management strategy is not only the compliance with the requirements, but to ensure the safety operation after the lifetime extension as well. As the result of the performed level 2 PSA and further analyses, a number of measures had been decided, the main ones the following:

— installation of reactor vessel external cooling;
— installation of severe accident hydrogen recombinators;
— implementation of the Severe Accident Management Guidelines;
— installation of an autonomous electric supply system with mobile diesels for severe accidents;
— installation of the severe accident monitoring system;
— For the first unit all measures completed until 2011, for the rest of the units all measures completed until the end of 2014.

The aim of the external cooling of the reactor vessel is to retain the reactor vessel integrity in a severe accident situation. According to this modification the vessel can be cooled by natural circulation using the water in the trays of the bubble condenser. Emptying the bubble condenser trays, the water first accumulates on the floor of the containment, and from the bubble condenser corridor through an emptying armature gets into the pipeline of the TL03 ventilation system, and through 4 smaller pipes gets into the room A005 (reactor cavity) under the reactor pressure vessel.

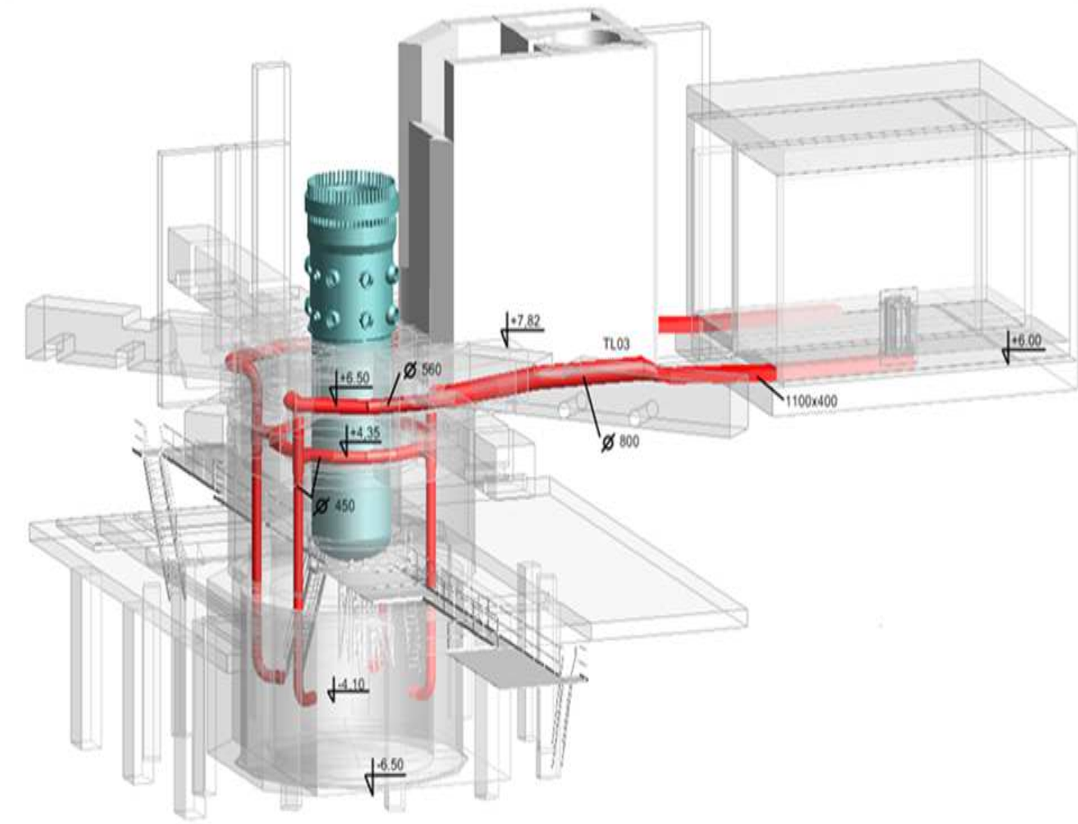

FIG. 1. Pipes for reactor vessel external cooling.

The performed Level 2 PSA concluded that the severe accident situations in most cases lead to early containment damage, as the number and the surface of hydrogen recombinators is not enough to handle the generated hydrogen in a severe accident situation. The installation of more recombinators with bigger surface was essential to avoid the containment damage and provide that the maximal pressure of the containment do not exceed the 3.35 bar limit even in a severe accident situation. The optimal number of the recombinators in the containment had to be calculated. As the result of the calculations further 60 high capacity accident hydrogen recombinator had been installed beside the originally installed 16 design basis hydrogen recombinator in the containment to avoid the damage of the containment integrity and reduce the radiological release to the environment. The installation of the accident hydrogen recombinators had been finished at all four units in 2011.

The severe accident monitoring system is operable during a severe accident situation and provides several important measures as:

- containment water level;
- reactor cavity water level;
- containment pressure;
- containment temperature;
- containment hydrogen concentration;
- spent fuel pool water level;
- reactor hall dose rate, and release of radioactive materials (by severe accident gamma measurement).

The function of the accident autonomous electric supply system is to provide electric supply in case of a station blackout, when no safety electric supply is available. The autonomous electric supply system for severe accidents provides electric supply for the depressurization of the primary circuit, for the equipment of the reactor pressure vessel external cooling, and for the severe accident monitoring system. It consists of 4 new mobile diesel aggregators and their electrical connections to the safety electric supply system. Until the start of the mobile diesels (max. 90 min.) two 8,5 kW power UPS/unit provides electrical supply for the equipment. The establishment of the autonomous severe accident electric supply had been finished at the fourth unit in 2011.

3. SAFETY IMPROVEMENTS AFTER FUKUSHIMA RELATED TO SEVERE ACCIDENTS

After the severe accident at Fukushima Daiichi, all European countries having nuclear power plants had performed the Targeted Safety Reassessment — the so-called stress test — for the call of the European Council. The most important result of the stress test was that the design basis of Paks NPP is adequate, and complies with the related legislation requirements and international practice. The Hungarian Atomic Energy Authority issued the safety improvement requirements related to the Targeted Safety Reassessment at the end of 2011. One of the requirements was to carry out an analysis of the long term (beyond 1 week) progression of severe accidents to mitigate the severe accident consequences. Based on the analysis results, appropriate measures that is suitable to prevent the long term, slow over-pressurization of the containment (filtered venting, internal containment cooling) has to be developed and implemented.

The designated safety improvements before Fukushima provide sufficient measures to avoid early damage of the containment using the sprinkler system, but for long term over-pressurization further measures are needed mainly related to the external cooling of the reactor vessel, since the steam generated during the cooling of the vessel flows into the containment and increases the pressure and temperature. The increase of the pressure without cooling the airspace of the containment leads to containment failure.

Instead of filtered containment venting the licensee is planning to avoid the long term over-pressurization of the containment by using the water accumulated in the containment floor through the sumps and returns it to the localization shaft by a severe accident sprinkler system. The severe accident sprinkler system is scheduled to start after the depressurization of the primary circuit and the operation of the reactor vessel external cooling, assuming the operation of the severe accident hydrogen recombinators.

The analyses showed that the severe accident sprinkler must start before reaching 3.2 bar containment pressure, and right after the start of the reactor vessel external cooling. In that case the sprinkler not only avoids the slow over-pressurization of the containment and provides water to the external vessel cooling by condensation, but also decreases the release of the radioactive fission products from the containment. The earlier it starts, the more efficient the sprinkler, the more it reduces the pressure in the containment and reduces the release of the radioactive fission products to the environment. If it can be provided that the injected water drops size are smaller than 5 mm and the distance of the freefall is bigger than 30 m, then depending on the actual pressure in the containment the pressure can be reduced to atmospheric in 3 to 6 h, also it can maintain this pressure level in long term. The deadline of the implementation of the severe accident sprinkler is the end of 2018.

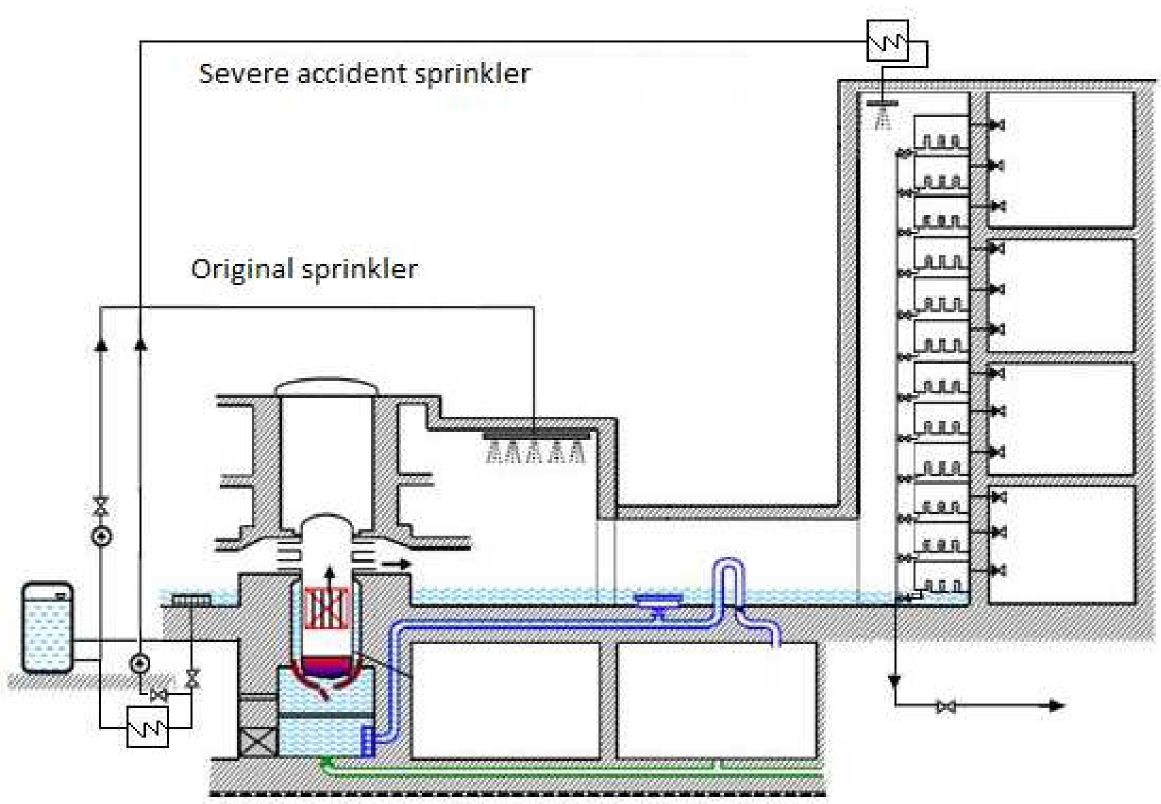

FIG. 2. The original and planned severe accident sprinkler.

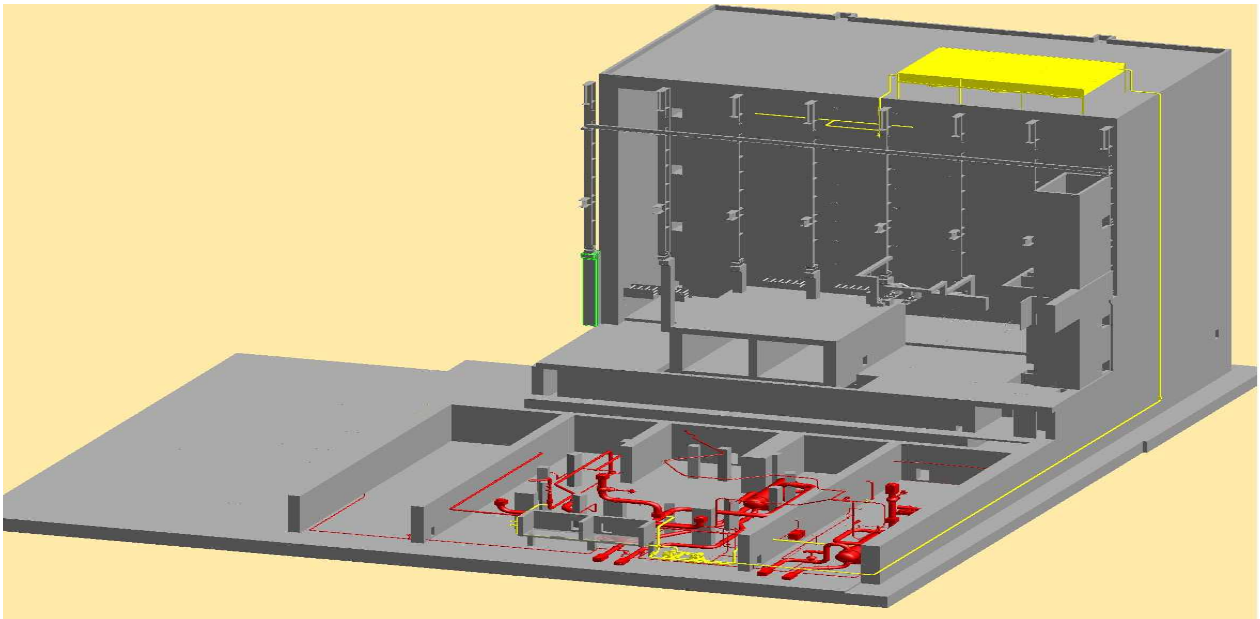

FIG. 3. The location of the severe accident sprinkler system.

REGULATORY CHALLENGES OF FILTERED CONTAINMENT VENTING IN INDONESIA

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Indonesia has planned to build the nuclear power plant (NPP) that mentioned inside the white paper of energy policy year 2015 from Ministry of Energy and Mineral Resources. Nuclear energy will support about 5000 MW(e) in 2025. BAPETEN as regulatory body should prepare the regulation infrastructure including the severe accident management. The severe accidents mean accident conditions more severe than a design basis accident and involving significant core degradation.

BAPETEN Chairman Regulation number 3 year 2011 about Design Safety of Nuclear Power Plant [1] in Article 6 stated that ' ...the application of the concept of defense in depth in the design of a nuclear power plant' and article 7 '. there are five levels of defense':

(1) The first level, prevent deviation and anticipated operational occurrences through conservative design construction and high quality operation;

(2) The second level, detection and control deviations from normal operational states in order to prevent anticipated operational occurrences at the plant from escalating to accident conditions';

(3) The third level, controlling of design basis accident to keep reactor condition under controlled and provided that are capable of preventing damage to the reactor core with engineered safety features and accident procedures;

(4) The fourth level, controlling of severe condition to ensure that radioactive releases are kept as low as reasonably achievable, including propagation prevention of accidents, severe accident mitigation, accident management;

(5) The fifth level, mitigate the radiological consequences of radioactive releases that must have emergency procedures for on-site and off-site emergency response.

And in Article 90 about the containments system stated that '...must be provided to ensure, or to contribute to the fulfilment of safety functions at the nuclear power plant:

(a) confinement of radioactive substances in operational states and in accident conditions;

(b) protection of the reactor against natural external;

(c) radiation shielding in operational states and in accident conditions.

Also in Article 95 paragraph 4, stated that '...containment system must be designed able to heat transfer from reactor containment through decreasing of pressure and temperature, and maintained pressure and temperature as low as reasonably achievable after high energy fluid release on design basis accident.

(d) There are several concepts for containment system in use or in advanced stage of design. This paper only discusses PWR containment system. In Figure 1 and 2, the primary containment envelope is a steel shell or a concrete building with a steel liner that surrounds the nuclear steam supply system. The containment encompasses all components of the reactor coolant system under primary pressure. It is designed as full pressure containment. It is able to withstand the increases in pressure and temperature that occur in the event of any design basis accident, especially a LOCA. The atmospheric pressure in the containment envelope is usually maintained at a substantial negative gauge pressure during normal operations by means of a filtered air discharge system (i.e. a fan and HEPA filter).

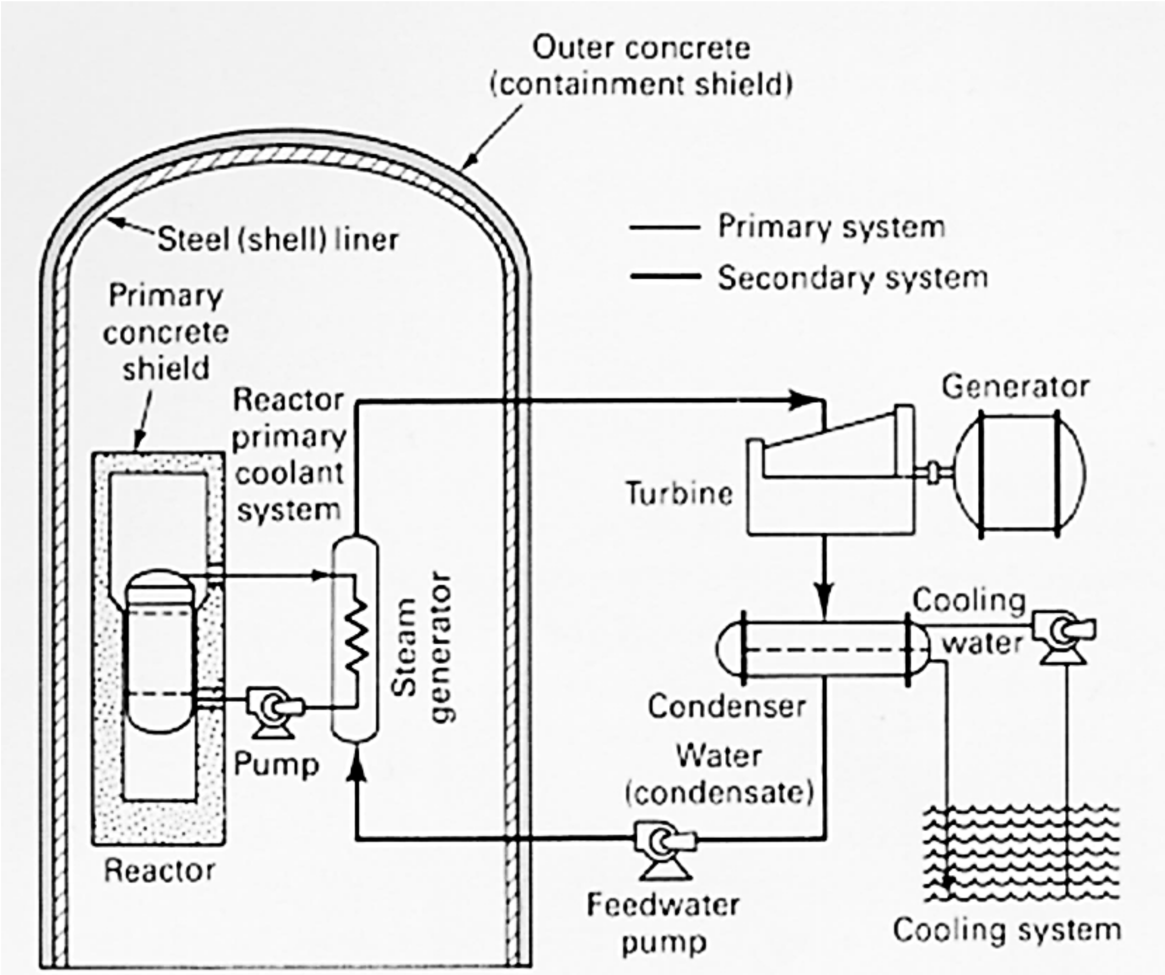

FIG. 1. PWR typical

Energy management in the building can be accomplished by an air cooler system or by a water spray system. In addition, the free volume of the containment and the structural heat are used to limit peak pressures and temperatures in postulated conditions for pipe rupture accidents. The initial supply of water for the spray system and for the emergency core cooling system is held in a large tank. When this water has been used, suction for both the spray system and the emergency core cooling system is switched to the containment building sump.

Water that is recirculated to the reactor vessel is sometimes cooled by means of heat exchangers. In most designs the recirculation water for the spray headers which is also used to limit contamination of the containment atmosphere is cooled by means of heat exchangers. When pipes rupture in systems other than the reactor coolant system, only the spray system is operated in the recirculation mode.

The containment vent to prevent containment overpressure failure can be used to mitigate the consequences of a severe accident. There are two situations in which it would eventually require venting, specifically, events involving either failure of the passive containment cooling system (PCS), or reactor pressure vessels (RPV) failure followed by unmitigated core-concrete interaction (CCI). Filtered vent is design alternative would involve the installation of a filtered containment vent, including all associated piping and penetrations. This modification would provide a means to vent containment to prevent catastrophic overpressure failures as well as a filtering capability for source term release. The filtered vent would reduce the risk associated with late containment failures that might occur after failure of the PCS. Note, however, that even if the PCS fails, it is expected that air cooling will limit the containment pressure to less than the ultimate pressure under most environmental conditions.

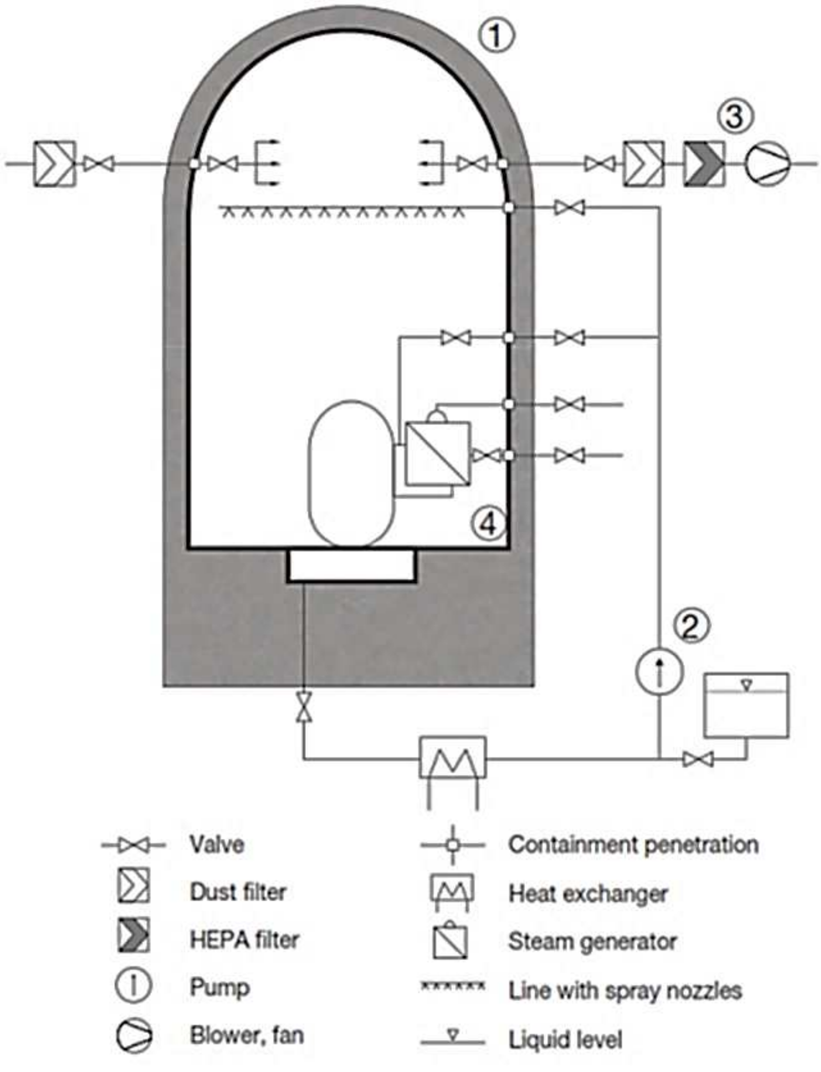

FIG. 2. The dry containment system for a PWR.

In the passive simplified pressurized water reactor concept (Figs 3–4), the containment vessel consists of a metallic shell surrounding the nuclear steam supply system. While the operational systems rely on proven pressurized water reactor technology, the safety systems for such reactors work passively, and do not depend on active components and safety grade support systems. In accidents, the residual heat is transferred via steam to the containment atmosphere, either through the leak or through the passive core cooling system, which uses the in-containment refueling water storage tank as a heat sink. The in-containment refueling water storage tank is also used as a water source to provide the safety injection in the event of a LOCA, and flooding of the reactor cavity for external cooling of the reactor pressure vessel in the event of a severe accident.

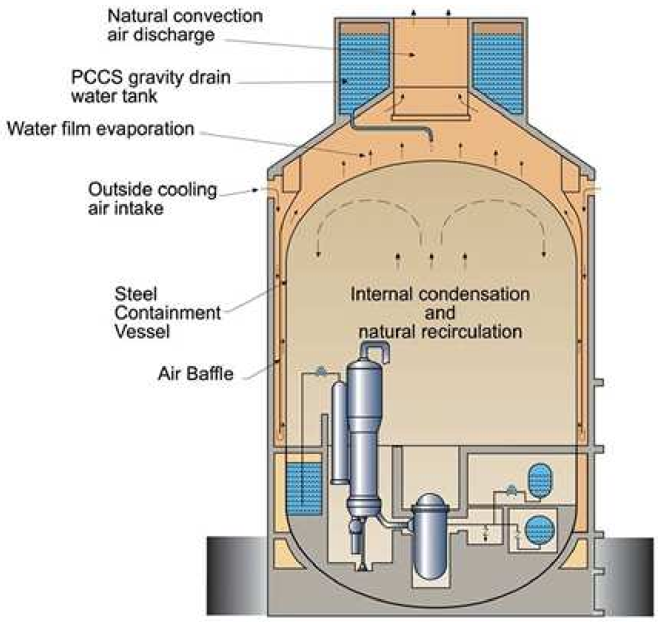

FIG. 3. AP1000 PCCS.

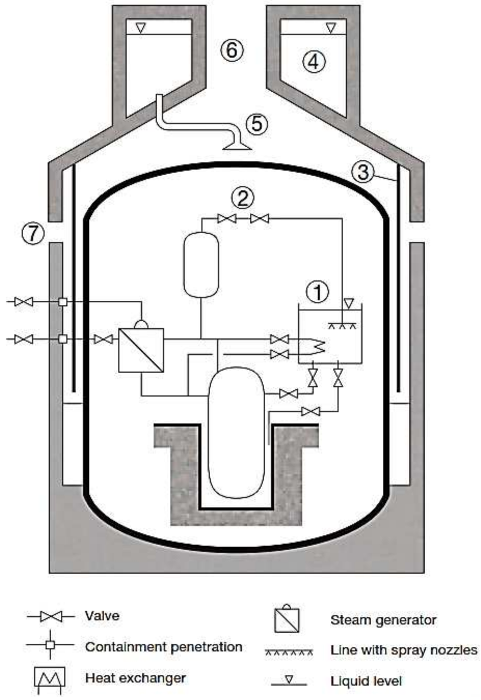

FIG. 4. Schematic diagram of a passive simplified pressurized water reactor.

Containment energy management is provided by passive external containment cooling, either by means of passive air circulation in the annulus or supported by external gravity spraying of the containment vessel. The design features of the containment promote flooding of the containment cavity region in accidents and submersion of the lower head of the reactor pressure vessel in water. The liquid effluents released during a LOCA through the break are also directed to the reactor cavity. After collection of the water in the lower part of the containment during an accident, a water level is reached that ensures that the water is drained back via sump screens into the reactor coolant system

Regulatory framework nuclear reactor strategy in Indonesia by independent process based on operating experience, adequacy of licensee's programmes, proposed safety improvements, and new requirements. BAPETEN should build integrated safety review (ISR) for assessment of necessary repairs and replacements, determination of reasonable and practical safety improvements, including measures to mitigate consequences of beyond design basis accidents, consideration of modern international standards and practices through cost-benefit analysis risk-informed, performance based regulatory approach.

Plant design shall be capable of meeting safety goals like severe core damage frequency (SCDF): 1E-4 to 1E-5, large release frequency (LRF): 1E-5 to 1E-6 (< 1% ^{137}Cs). Containment shall maintain its role as barrier against releases for a period of approximately

24 hours and uncontrolled releases of radioactivity after this period. That will answer using filtered venting containment system. Containment system shall have design capability to remove heat to preserve containment integrity, control hydrogen concentration to prevent deflagration, remove fission products, hydrogen and other combustible gases.

In design basis should have deterministic requirements based on single and dual failure criteria, loss of coolant accident and loss of cooling injection for containment performance analysis, assessment of accident progression and safety system performance through Level 1 PRA. Pressure suppression through dousing to prevent containment failure for single-unit stations. Emergency filtered air discharge Systems (EFADS) to maintain containment below atmospheric pressure.

In beyond design basis should assessment of accident progression and containment performance through Level 2 PRA, safety goals, internal and external events, including seismic, fire and flood, other extreme weather conditions, as deemed necessary, identification of safety enhancements, make up capability for various coolant reservoirs, passive autocatalytic recombiners (PAR), emergency containment filtered ventilation, effectiveness of existing systems confirmed by post Fukushima.

Emergency containment filtered venting (ECFV) system design to prevent containment failure and limit radioactive releases to atmosphere, installed for single-unit stations, design options for multiunit stations, emergency filtered containment venting technical specifications, manually and remotely operated (no external power), design pressure of 400 kPa(g), maximum design temperature of 200 °C, design load of aerosols up to 300 kg, operate passively to relieve containment pressure and remove fission products (FP) with retention rates of aerosols > 99.9 %, molecular cesium > 99.5 %, FPs (CsI, RbI, CsOH, RbOH) > 99.5 %, Elemental iodine > 99 %, Organic iodine approx. 80 %. Those parameters should be analyzed by assessment center in all aspect like safety and security if it will be applied on binding regulation.

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REGULATORY REQUIREMENTS AND ACTIONS RELATED TO IMPROVE CONTAINMENT VENTING FOR LAGUNA VERDE NPP

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Abstract. Currently in Mexico, there are two operating nuclear reactors at Laguna Verde Nuclear Power Plant. They are of the BWR type, with Mark II containment, located near to the Gulf of Mexico, and managed by the state owned utility. This paper describes the current status of the requirements established by the Mexican regulatory body and actions underway by the utility in order to improve the containment venting of both units. This process is based in the lessons learned from the international operating experience developed as a consequence of the Fukushima Daiichi accident. There is a description of the regulatory framework, the attributions of the regulatory body, the terms and definitions related to the containment venting systems. Furthermore, there is a description of the action plan developed by the utility to use the NEI 13-02 as a guideline for the design of a reliable hardened containment vent capable of operation under severe accident conditions. Finally, a timetable is presented with the expected dates for the conclusion of the activities related to the improvement of the containment venting.

1. INTRODUCTION

The National Commission on Nuclear Safety and Safeguards, CNSNS, is the regulatory body for nuclear matters in Mexico, and is part of the Secretary of Energy. It was created in 1979 as part of a reorganization of the national nuclear administration. Currently in Mexico, there are two power generating reactors at Laguna Verde Nuclear Power Plant (LVNPP), these are of the boiling water type (BWR), with Mark II containment, located near to the Gulf of Mexico, in the State of Veracruz. Both reactors are managed by the Federal Commission of Electricity, which is the state owned utility for electricity generation. The first reactor at LVNPP is in operation since 1990, and the second reactor went on line in 1995. The Secretary of Energy, under the technical guidance of the regulatory body, was the institution that granted both licenses of operation. As a consequence of the Fukushima Accident, the regulatory body required to the utility the implementation of the lessons learned developed by the international operating experience through the execution of stress tests designed to check that LVNPP is able to sustain unexpected external events such as earthquakes, flooding, or extreme weather conditions; furthermore, the utility was required to develop a programme to implement alternatives to avoid fuel failure, install spent fuel pool instrumentation, develop Severe Accident Management Guidelines, and install have a reliable hard venting system in the containment.

The information described below is based, for the case of the regulatory framework, in the documentation generated by the regulatory body staff; for the action plan and the timetable, documents related to these topics were provided by the utility.

2. REGULATORY FRAMEWORK

The Mexican regulatory body, before the Fukushima Accident, had already required to the utility in relation to the management of severe accidents, the development of Probability Safety Analysis (PSA) levels 1 and 2, and an internal flooding analysis. These were done by the utility and reviewed by the regulatory body in 2001. Because of the Fukushima Accident, CNSNS required reviewing and updating these analyses, the utility and their technical support organization started this process in April 2013. Another requirement post-Fukushima was to implant mitigation strategies in order to maintain or reestablish de capability to remove heat from the core, the containment and the spent fuel pool, in scenarios in which large areas of the plant had suffered fire or explosions that caused impediment in their operation. This requirement follows paragraph 50.54 (hh) (2) of the US NRC 10CFR Code of Regulations. Following the experience of the Western European Nuclear Regulators Association (WENRA), and through its participation in its equivalent, the Foro Iberoamericano de Organismos Reguladores Radiológicos y Nucleares (FORO), the regulatory body required to the utility to carry out stress tests, in order to assess the safety margins of Laguna Verde NPP.

Additionally, the regulatory body requested to the utility the implementation of a programme that follows the US NRC orders that were developed as a consequence of the Fukushima Accident. This requirement was formalized in a letter dated November 4, 2013. This document requests that the utility, in order to develop its programme, should consider the following documents: EA-12-049 "Issuance of order to modify licenses with regard to requirements for mitigation strategies for beyond-design-basis external events", EA-13-109 "Modified NRC Order for Containment Venting Systems", EA-12-051 "Issuance of order to modify licenses with regard to reliable spent fuel pool instrumentation" [1, 2].

In this context, for the EA-13-109 Order, the regulatory body specified to the utility to consider the US NRC document SECY-12-0157 "Consideration of additional requirements for containment venting systems for boiling water reactors with Mark II containments" of March 19, 2013, and use as guide, the methodology described in the Nuclear Energy Institute document NEI-13-02 "Industry Guidance for Compliance with Order EA-13-109" revision 1 [3].

3. ACTION PLAN TO DESIGN A RELIABLE HARDENED CONTAINMENT VENT SYSTEM (HCVS)

The action plan to design a reliable HCVS considers two phases. Phase 1 is related to the design of the HCVS located at the Wetwell of the Primary Containment. Phase 2 considers the similar process for Drywell of the Primary Containment. Both processes are similar, the inputs, criteria and requirements are described below:

3.1 Design an HCVS to minimize the dependency in the operator actions, considering radiation conditions to determine the impact and capacity of the venting and the analysis of the electrical and pneumatic capacity for long operation of multiple cycles of the valves, and its instrumentation during the first period of 24 hours.

3.2 Design an HCVS to minimize occupational risks such as extreme heat, and radiological conditions that could impede action by the operators, this includes shielding calculations and other actions of radiological control so to have acceptable radiation levels for the operators access, and calculations for the temperature conditions due to the loss of ventilation in parts of the HCVS during Extended Loss of Alternate Current Power (ELAP).

3.3 Calculations for boundary conditions of the instrumentations of the HCVS, controls and indicators must be accessible and functional under a range of plant conditions, including severe accident, ELAP or inadequate cooling of the containment.

3.4 Design an HCVS in order to have the capacity for venting vapor/energy equivalent to 1% of the rated thermal power licensed, this includes the pipe diameter calculation.

3.5 Design an HCVS that has an effluent discharge point above the main structures of the plant, this includes the calculations for HCVS pipe protections against missiles generated by external events.

3.6 Design the HCVS to include characteristics that minimize the accidental cross flow of vented fluids within a unit and between units; this has to consider the site characteristics.

3.7 Design the HCVS to be operated manually during long operations from Main Control Room or another remote control station. This includes system status monitoring from a specified panel. Calculations for the number of open/close cycles needed during the first 24 hours of operation. This includes the calculations for the temperature and radiological conditions at the primary containment and alternate controls of the HCVS, and the assessment of the pressure indications at the containment and level indication at the suppression pool.

3.8 Design the HCVS capable of being operated manually (from a shielded location, with manually operated valves). This analysis should determine the location for the alternate control of the valves and the support equipment, such as nitrogen bottles, compressors and batteries.

3.9 Design de the HCVS to operate with qualified equipment and permanently installed after 24 hours of the loss of either the electrical or pneumatic power during an ELAP. This design requirement is related with the calculation of the electrical and pneumatic capability of requirement (3.1) already mentioned above.

3.10 Design the HCVS that includes means for an inadvertent actuation. This involves the assessment of the set point of the rupture disc required to avoid the inadvertent actuation.

3.11 Design de HCVS to include means for radioactive effluent monitoring that could be released due to operation. This task is based in calculations in order to develop the selection criteria of the monitors.

3.12 Design the HCVS to sustain severe accident conditions. Design parameters analysis to establish or verify the boundary conditions of the containment during a severe accident.

3.13 Design the HCVS to assure that flammability limits of gases that go through the system are not reached. The system must sustain dynamic loads resulting from hydrogen deflagration or detonation. The system´s piping has to be designed against hydrogen deflagration or detonation. A calculation must be done to measure the purge needed in the HCVS in order to reduce the gases concentration under the flammability limits. An analysis should be done in order to avoid detonation/deflagration phenomena in the piping of the HCVS, this includes the reduction of potential migration of hydrogen from the reactor building or other structures.

3.14 Design of the HCVS should include provisions for operating, testing, inspection and maintenance activities in order to assure its reliable operability. Testing and inspection requirement should be developed.

3.15 The route of the HCVS from end to end, including the second barrier of containment isolation must be designed according to the design basis of the installation. In this sense an analysis should demonstrate the correct safety grade of all HCVS structures and components.

3.16 Development, implementation and updating of the procedures needed for the safe HCVS operation, this includes the system's operation with non-emergency power, backup power, and during ELAP.

3.17 The utility should provide training programme for the operation personnel in the correct operation of the HCVS. This training should include system's operation with non-emergency power, backup power, and during ELAP.

4. TIMETABLE FOR THE IMPLEMENTATION OF THE HCVS

As it was described above, the regulatory process for the actions related to the implementation of the HCVS in Laguna Verde NPP started in November 2013, since then; there have been various meetings between the regulatory body and utility´s management and technical staff. The design process for the Phase 1 (Wetwell) started in February 2014 and it is expected to end in July 2016. This process for Phase 2 (Drywell) will start in January 2015 and it is expected to end in December 2017 (if there are no changes in the requirements).

The acquisition process for components needed for the implementation of the Phase 1 (Wetwell) started in December 2014, these components are for example: instrumentation equipment needed for pressure and temperature indicators in the HCVS and the Wetwell, level indicators of the suppression pool, equipment for monitoring radiation levels at the HCVS, mechanical components such as valves, rupture discs, piping and structural components for the construction of the system. This process for Phase 2 (Drywell) is expected to start in January 2016.

The design change process related to the elaboration of modification packages, blueprints, construction procedures, for the implementation of Phase 1 (Wetwell) started in December 2014 and it will end by October 2016. This process will be for Units 1 & 2 of LVNPP. For Phase 2 (Drywell) will start in January 2016, for both Units, and will end by October 2016 in the case of Unit 1, and March 2018 for Unit 2.

The construction process which involves the physical implementation of the HCVS will be done during the routine outages used for fuel reloading and large maintenance. Expected dates are:

Implementation of Phase 1 (Wetwell) for Unit 1, Fuel Reload 18, January 2017. For the Unit 2, Fuel Reload 15, April 2017.

Implementation of Phase 2 (Drywell) for Unit 1, Fuel Reload 19, May 2018. For Unit 2, Fuel Reload 16, September 2018.

This calendar, although subject to changes, is used by the regulatory body staff to develop its own action plan in order to carry out the specific technical evaluations and inspections of the whole process of design and implementation. Any further change or development, whether is regulatory or technical, that comes from the internal or external operational experience, or safety analysis will be attended in order for LVNPP to have a reliable hard venting system.

5. CONCLUSION

This paper described the current status of the requirements established by the Mexican regulatory body and actions underway by the utility in order to improve the containment venting of both units of LVNPP. This process is based in the lessons learned from the international operating experience developed as a consequence of the Fukushima Daiichi accident, and these are based in the Order EA-13-109 developed by the United States Nuclear Regulatory Commission. The design action plan developed by the utility was described for Phases, 1 (Wetwell) and 2 (Drywell); finally, a timetable was presented with the expected dates for the conclusion of these activities related to the improvement of the containment venting.

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STATUS OF SPANISH REGULATIONS AND INDUSTRY ACTIONS RELATED TO FILTERED CONTAINMENT VENT SYSTEMS

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1. INTRODUCTION

After the Fukushima Daiichi nuclear power plant accident on March 11, 2011, the European Union launched a plan in order to analyse the capacity of nuclear power plants in the member countries to manage a similar situation. This plan was called "stress tests" and its objectives were to assess the ability of European nuclear power plants in operation to handle situations beyond their design basis, while identifying the existing margins and the possible measures to improve the managing of the postulated situations.

Western European Nuclear Regulators Association (WENRA) drafted a proposal for the “stress tests” which was approved by the European Nuclear Safety Regulators Group (ENSREG) and distributed to the member countries for its implementation. The provisions included at the ENSREG’s document were required to the Spanish nuclear power plants by the Spanish Nuclear Safety Council (CSN) through Complementary Technical Instructions (ITC).

CSN, after evaluating the facilities’ stress tests reports, required the installation of Filtered Containment Venting Systems (FCVS), in agreement to the operators’ conclusions.

FCVS will be installed in all Spanish nuclear power plants by the end of 2017.

2. CURRENT SITUATION

In Spain the operating seven nuclear reactors (6 PWR and 1 BWR) produce around 20% of the country electric power:

(1) BWR:
- Cofrentes NPP: GE BWR-6, Mark III.

(2) PWR:
- Almaraz I and II NPP: Westinghouse PWR, 3 loops, 2 units;
- Ascó I and II NPP: Westinghouse PWR, 3 loops, 2 units;
- Trillo NPP: KWU PWR, 3 loops, 1 unit;
- Vandellós II NPP: Westinghouse PWR, 3 loops, 1 unit.

The eighth reactors (Santa María de Garoña NPP: GE BWR-3, Mark I) is currently shut-down but the operator has applied for the renewal of the plant operating permit, which is currently being evaluated by the CSN.

2.1 BWR

2.1.1 Cofrentes NPP

Regarding the prevention against overpressure conditions in containment, Cofrentes NPP has a Hardened Containment Venting System (HCVS) which is provided with two air-operated isolation valves, taking suction from the wetwell. This system was installed in 1993.

After the Fukushima Daiichi accident, the operator has implemented some design modifications in order to increase the reliability of the system operation in case of SBO and/or earthquake, such as modifying isolation valves (so they can be manually operated from a radiation shielded local area) or increased nitrogen supply seismic requirements.

HCVS has a dual function and accordingly is considered in EPG/SAMG:

— Preventive venting: HCVS first function is to prevent severe accident. Venting provides heat removal capacity. Additionally, protecting the containment from over-pressurization preserves the containment integrity and allows the core cooling systems to keep on operation.
— Mitigating venting: when the severe accident has happened, HCVS provides containment overpressure protection, preserves the suppression pool scrubbing capability, allows the containment flooding and provides a way for combustible gases control.

2.1.2 Santa María de Garoña NPP

Santa María de Garoña NPP is currently in final shut-down since July 2014, but the facility has applied for the renewal of the plant operating permit.

The facility is provided with a HCVS which was installed in 1992. The system has two venting paths, one from the wetwell and the other from the drywell. Each path has one air operated isolation valve. Then both paths are joined in the same pipe which is connected to the stack and is also provided with a motor operated valve.

HCVS functions are analogous to those described for the Cofrentes NPP case.

2.2 PWR

Westinghouse PWR NPPs (Almaraz I and II, Ascó I and II, and Vandellós II).

Westinghouse PWR NPPs in Spain are provided with the controlled hydrogen purge subsystem. The vent flow is prefiltered by HEPA and active charcoal filters. This subsystem only actuates in case of LOCA and it is foreseen to only operate at low containment pressures.

The Severe Accident Mitigating Guidelines (SAMGs) include provisions for containment venting, but they are based on systems which are not designed for severe accident conditions.

2.2.1 Trillo NPP

Currently Trillo NPP is not provided with any venting path.

3 POST-FUKUSHIMA ENHANCEMENTS

3.1 Post stress tests Complementary Technical Instruction (ITC) and FCVS design criteria

In March 2012, the CSN, after evaluating the facilities' stress tests reports, issued a ITC requiring the installation of FCVS in all the Spanish NPPs before December 31, 2016 (the precise time for implementation will depend on refueling outages dates), except Cofrentes NPP. The latter was required to analyse the convenience for installing a FCVS taking into account the suppression pool retention capability and finally the CSN required the facility to install the system before December 31, 2017.

Before the installation of the new systems, operators have been requested to send to CSN a detailed proposal of the design modifications (design of the systems, procedures, etc.) for the regulatory approval.

In November 2013, the CSN released a guide containing general FCVS criteria to be used for the CSN technical evaluations. This guide was transmitted to the operators to be used as a reference. These criteria were established to ensure that the FCVS and the support systems that maintain its functionality were designed with the adequate capacity and a high degree of availability and reliability.

These criteria take into account the special characteristics of this system as:

(1) an effective way to prevent the occurrence of a severe accident and to preserve the containment integrity;
(2) as a system which mitigate the release of radioactive products to the environment for severe accidents.

The system is provided with equipment performing safety functions (e.g. containment penetrations and isolation valves). This equipment must be designed and manufactured for that purpose, according to the corresponding design basis regulation.

As shown below, the CSN criteria for FCVS were classified in two groups: issues related to functional design and issues related to qualification.

3.1.1 Issues related to functional design

Below are detailed the issues related to functional design within the CSN criteria guide:

(1) The FCVS should be designed and manufactured to meet the expected functional objectives:
   (a) The system should be able to keep the containment pressure in adequate values.

   These values should be calculated taking into account: the structural capacity of the containment, the ability of the containment penetrations to operate with acceptable leakage levels and the requirements of maximum pressure for the containment systems to be able to deal with the accident, considering the reactor trip was effective.

   The design pressure of the FCVS should be consistent with these values. The criteria for initiation and completion of FCVS should be collected in emergency procedures or guides.

   (b) If the function of venting is required to prevent the occurrence of severe accidents (events with failure of the residual heat removal capacity), and as an alternative method for controlling hydrogen in containment, the system should be able to meet its objectives in the range of pressures that such functions are carried out.
   (c) The FCVS decontamination factors against aerosols, elemental iodine, organic iodine, etc., should be at the level of those offered by the latest technologies available on the market.

   In the event that an operator selects a FCVS technology which does not fit the criteria above, an additional evaluation considering the following issues should be done by the operator and validated by the CSN (quantitative criteria have not been established):

   — Population doses;

— Land contamination.

(d) The filter capacity should allow the autonomous operation of the system without manual intervention in the filtration equipment for a period of time consistent with the needs of venting. If this is not possible, the owner should demonstrate the feasibility of manual actions required to recover the filter capacity and that the unavailability of the FCVS due to the intervention of the filter will not compromise the appropriate performance of the system while the FCVS is required.

(2) The FCVS should be designed and constructed in order to minimize the risks associated to the presence of hydrogen in the system, facilitating the flow of hydrogen and preventing the presence of hazardous hydrogen concentrations inside the system. Analyses can give credit to Passive Autocatalytic Recombiners (PAR) installed in the containment.

(3) The FCVS should be provided with appropriate instrumentation to verify the correct operation of the system and the adequate control of the release.

(4) The FCVS and its instrumentation should have the support systems (power supply, air drive, consumables, etc.) to ensure proper performance of its function autonomously, for at least 24 hours without any site external support and for at least 72 hours only with the site external support of lighting equipment.

(5) The FCVS should be designed and constructed so that the manual operation of the venting system is possible by trained personnel using procedures developed for this purpose; the system components that require or may require manual local action should remain accessible under the physical and radiological conditions expected in the foreseen accidental situations.

(6) The FCVS should be designed and constructed to allow the proper management of the generated waste, considering both handling and subsequent storage. This should be done by properly trained personnel using procedures developed for this purpose.

(7) The FCVS should be designed, manufactured and located so that it can be verified by periodically testing in accordance to procedures developed for this purpose. These periodic tests together with a suitable maintenance programme (including the appropriate spare management), should adequately ensure that the system maintains its reliability and functionality.

(8) The FCVS should be designed so that the risk of undesired opening and closure failure is minimized.

3.1.2 Issues related to qualification

Below are the issues related to qualification within the CSN criteria guide:

(1) The FCVS should be seismically designed and supported in accordance with the design basis of the facility (SSE) and using nuclear industry standards. The analysis must verify the additional margin considered applicable in the stress tests process.

(2) While it is out of service, the FCVS should be able to withstand external or internal events analysed in the stress test process, maintaining its functionality for future uses (for example, it must withstand extreme environmental conditions if its located outside, and it should be located at a height consistent with the analysed flooding level). A formal qualification procedure is not required, but it should be demonstrated (analytically, or by verifiable usage experience or when it is not possible by other means, as by expert judgment) with a high confidence level that SSCs will be able to support such a condition maintaining its future functionality.

(3) The FCVS should be able to withstand, while operating, the predicted environmental conditions in the scenarios postulated during the stress test process

maintaining its functionality for as long as required (if it is located outdoors, it should be able to withstand the conditions analysed in the stress tests and if it is located inside a building, it must be able to withstand the environmental conditions that can be reached during a severe accident). A qualification procedure is not required, but it should be demonstrated (analytically, or by verifiable usage experience or when it is not possible by other means, by expert judgment) with a high confidence level that SSCs will be able to support such a condition maintaining its future functionality.

3.2 Plans of implementation

3.2.1 Almaraz I/II and Trillo NPPs

Both Almaraz I/II and Trillo NPPs have decided to install wet filters which will be in operation in the three facilities by the end of 2017.

The decontamination factors (DF) provided by the filters are 10000 for aerosols, 300 for inorganic iodine and 10 for organic iodine.

The operators have proposed a FCVS provided with two manually operated valves, placed outside containment, which would be operated from an adjacent room by a stem extension device. This issue is being currently under evaluation at CSN.

3.2.2 Ascó I/II and Vandellós II NPPs

Both Ascó I/II and Vandellós II NPPs have decided to install a wet filter (first stage) with a molecular sieve (zeolite filter, second stage). It will be installed in the three NPPs by the end of 2016.

The decontamination factors (DF) provided by the filter are 10000 for 0.5 µm aerosols size (100000 for 1–2 µm aerosols size), 1000 for inorganic iodine and 10 for organic iodine.

The operators have proposed a FCVS provided with two manually operated valves, placed outside containment, which would be operated from an adjacent room by a stem extension device. This issue is being currently under evaluation at CSN.

3.2.3 Cofrentes NPP

Cofrentes NPP has developed the specifications for the FCVS but hasn't decided the technology yet.

For the specifications development, Cofrentes NPP has taken into account the particles retention in the suppression pool, and has analysed with MAAP code the size of the aerosols released through the vent, concluding that it would be very small (smaller than 0.5 microns).

The operator has also made a calculation of the radioactive emissions to the environment (with RASCAL code), using the source term obtained in each MAAP simulation as an input.

As a result of the previous analysis, the operator has concluded that to be able to reduce doses to the environment below 50 mSv, is necessary to combine both mitigation strategies and a FCVS with a decontamination factor (DF) for small particles (below 0.5 µm) of 100 for aerosols and 200 for inorganic iodine.

Cofrentes NPP also studied the dose reduction associated with an additional organic iodine filter, concluding that it would be almost negligible.

Currently, the operator specification proposal is under evaluation at the CSN.

3.2.4 Santa María de Garoña NPP

Santa María de Garoña NPP is currently in shut-down condition, but the operator has applied for the renewal of the plant operating permit. One of the requirements for granting the operator renewal is to have an FCVS installed before the start-up of the facility.

The operator has already sent to the CSN an application for the authorization of the FCVS design modification, which is currently being assessed at the CSN.

The operator has decided to install a wet filter (first stage) with a molecular sieve (zeolites filter, second stage). The decontamination factors (DF) provided by the filter are 10000 for 0.5 µm aerosols size (100000 for 1-2 µm aerosols size), 1000 for inorganic iodine and 10 for organic iodine.

Currently, Santa María de Garoña NPP is provided with two venting paths: one from the drywell and the other from the wetwell. Each path has one air operated isolation valve. Then both paths are joined in the same pipe which is connected to the emergency stack and is also provided with a motor operated valve.

The operator proposes to comply with NRC order EA-13-109 "Issuance of Order to modify Licenses with regard to Reliable Hardened Containment Vents capable of operation under Severe Accident conditions". Then, Santa María de Garoña NPP would implement the following additional means to improve the isolation valves reliability:

- The tubing sections between the nitrogen bottles and the system valves will be assessed/replaced by others seismically qualified.
- The nitrogen bottles will be relocated in a new electric auxiliary building (post-Fukushima enhancement) outside the reactor building.
- The Motor Control Centre that feeds the motor operated isolation valve will be relocated at the new electric auxiliary building allowing its connection with the portable diesel generators.
- The motor operated isolation valve can be locally acted as it is provided with a clutch and a wheel. Additionally it will be provided with a new clutch and wheel set so the valve can be remotely operated.

It will be possible to operate the FCVS from the Control Room, from the new electric auxiliary building and locally.

4 CONCLUSIONS

The accident at Daiichi Fukushima nuclear power plant highlighted both the importance of having a reliable containment venting and, in order to reduce environmental releases, the importance of this venting to be filtered.

The Spanish Nuclear Safety Council (CSN) has required the Spanish nuclear power plants to install FCVS. The installation process must be finished in all the facilities by the end of 2017.

All the Spanish NPPs, except Cofrentes NPP (this facility is currently developing FCVS specifications), have already chosen a FCVS technology:

- Ascó I/II, Vandellós II and Santa María de Garoña NPP will install a wet filter (first stage) with a zeolite filter (second stage);
- Almaraz I/II and Trillo NPP will install wet FCVS technology.

Before installation of the new systems, operators are requested to send to CSN the design modification detailed proposals (design of the systems, procedures, etc.) for the regulatory approval.

RESULTS OF REGULATORY REVIEW OF ACTIVITIES FOR IMPLEMENTATION OF CONTAINMENT FILTERED VENTING FOR UKRAINIAN NPPS

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Abstract. The stress test assessment for all Ukrainian NPPs was performed in 2012. The assessment also included evaluation of NPP vulnerability during progression of severe accidents. Results of the stress tests showed that containment parameters (e.g. pressure, temperature, activity, etc.) could substantially increase. This results from the generation of steam and non-condensable gases during severe accident progression (in-vessel and ex-vessel phases). This could lead to failure of the isolation function due to hydrogen combustion or containment overpressure (in case of loss of all containment cooling systems). Based on results of the stress tests, the requirement for implementation of containment filtered venting for all Ukrainian NPPs has been stated by the Ukrainian Regulatory Body (SNRIU). This requirement has been reflected in the Comprehensive Safety Improvement Programme (measure "Implementation of Containment Venting System"). The filtration system for high-radioactive gas-steam mixture discharge from the accident confinement area has become de-facto a mandatory feature in modern NPP designs. However, for the old-design power units currently in operation, implementation of such a system is a complex engineering task. An especially challenging task is the implementation of a filtering system at WWER-440 power units.

1. GENERAL INFORMATION

There are currently 15 power units operated at four NPP sites in Ukraine with a total installed electrical power of 13,835 MW, which constitutes approx. 50% of the total installed electrical power of all power plants in Ukraine.

There are three types of WWER units operating in Ukraine: WWER-440/213 (Rivne NPP Units 1 and 2), WWER-1000/320 (Zaporizhzhya NPP Units 1-6, South-Ukraine NPP Unit 3, Khmelnitsky NPP Units 1, 2 and Rivne NPP Units 3, 4) and WWER-1000/302 (South-Ukraine NPP Units 1, 2).

After the accident at Fukushima-1 NPP (Japan), the SNRIU Board approved an Action Plan for a targeted safety reassessment and further safety improvement of Ukrainian NPPs in the light of the Fukushima-1 accident and an Action Plan for a targeted safety reassessment. One of the actions defined in the Action Plans was a targeted safety reassessment of operating nuclear facilities at NPP sites (stress tests).

In the framework of the stress tests, the operators analysed in detail:

- external extreme natural events (earthquakes, flooding, fires, tornadoes, extremely high/low temperatures, extreme precipitations, strong winds, combinations of events);
- loss of electrical power and/or loss of ultimate heat sink;
- severe accident management.

At the operating nuclear power plants, the stress tests focused on nuclear fuel in the reactor cores, spent fuel pools, fresh fuel rooms and dry spent nuclear fuel storage facility (Zaporizhzhya NPP).

Based on the completed analyses, a number of recommendations for decreasing vulnerability of the Ukrainian NPPs were developed. These recommendations were considered and reflected in the Comprehensive Safety Improvement Programme. Implementation of containment filtered venting was one of these recommendations.

2. DESCRIPTION OF CONCEPTS FOR TECHNICAL SOLUTIONS AND RESULTS OF PRELIMINARY FCVS JUSTIFICATIONS

At present, the technical concepts for containment filtering venting have been developed for each type of Ukrainian NPPs. Many justification activities have been performed for development of the technical solution. The main goals of these activities were to:

- evaluate the possibility of CFV implementation at NPPs;
- select the CFV type possible for installation at NPPs;
- preliminary evaluate the CFV efficiency.

3. ANALYSIS OF FILTER TYPES FOR UKRAINIAN NPPS

The application of different filter types was analysed for Ukrainian NPPs. The following types of filters for containment venting were considered:

- sand filter;
- Venturi scrubber;
- dry filter.

Based on this analysis, the application of a dry filter and Venturi scrubber was selected as possible for Ukrainian NPPs.

4. FCVS JUSTIFICATION

According to stress test results, an in-depth analysis of the need for containment venting and justification of its implementation have been performed. All main types of Ukrainian reactors were covered. There are WWER-1000/320 (reference plant was ZNPP-1), WWER-1000/302 (SUNPP-1) and WWER-440/213 (RNPP-1).

The main aims of in-depth analysis were to:

- estimate the possibility and conditions for pressure reaching the maximum design limit;
- assess the possibility of containment failure prevention using containment venting system;
- evaluate the main requirements for containment filtering venting system.

Analytical justifications of FCVS were performed using the MELCOR computer code and specific models for reference units. Those models were based on severe accident models created for SAMG development and justifications.

Results of such analyses are briefly described below.

4.1. FCVS for WWER-1000/320

The following accidents were considered under FCVS for WWER-1000/320:

- large break loss-of-coolant accident with loss of power supply;
- total station blackout.

Results of severe accident calculations (Figs 1–2) showed the possibility of containment failure due to reaching the containment pressure limit (5 kgf/cm^2 (abs.)).

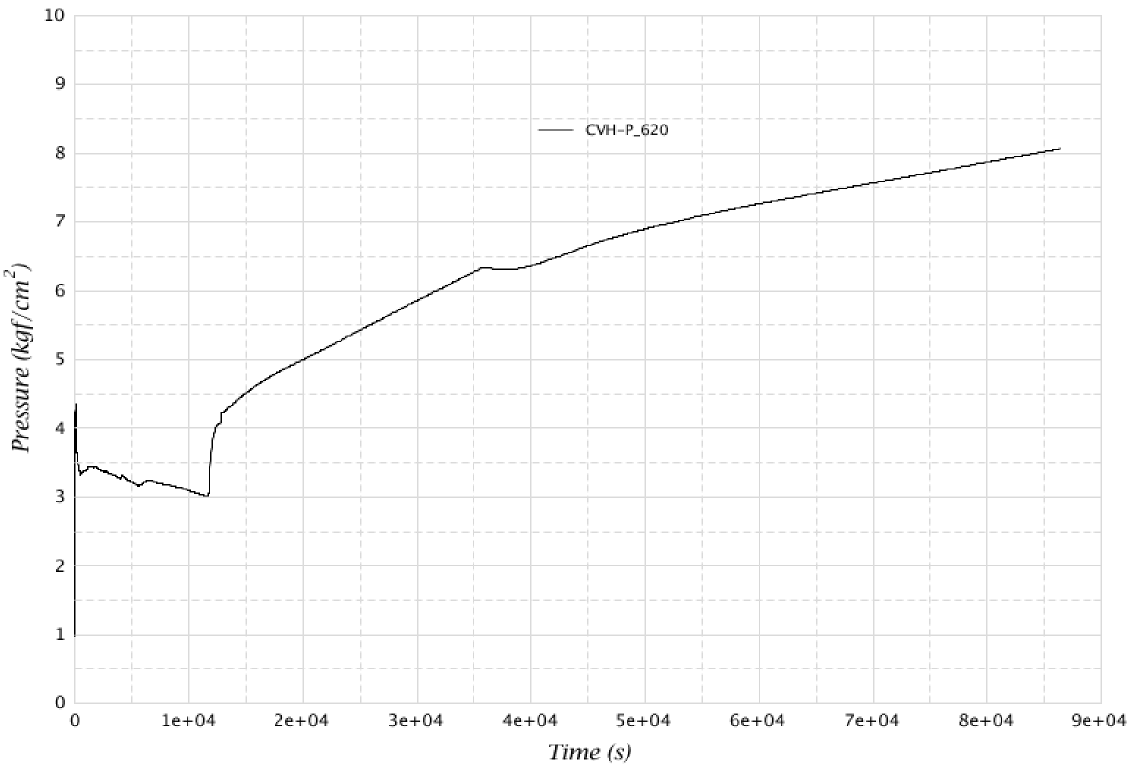

FIG. 1. Pressure in WWER-1000/320 containment under LB LOCA with loss of power supply.

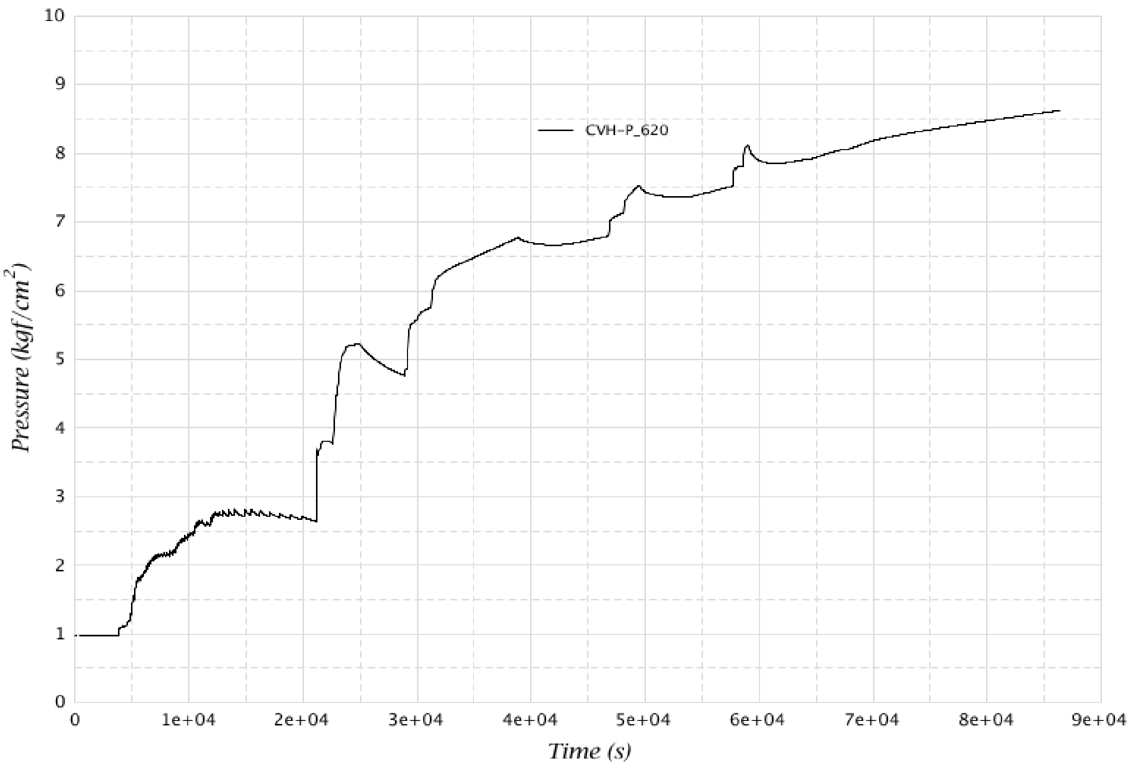

FIG. 2. Pressure in WWER-1000/320 containment under total station blackout.

Results from evaluation of the minimal diameter for containment venting system showed that implementation of FCVS with dump pipelines of less than 100 mm did not prevent increase in containment pressure. Implementation of FCVS with dump pipelines with a diameter of more than 100 mm prevents increase in containment pressure and prevents containment failure due to overpressure (see Figs 3–4).

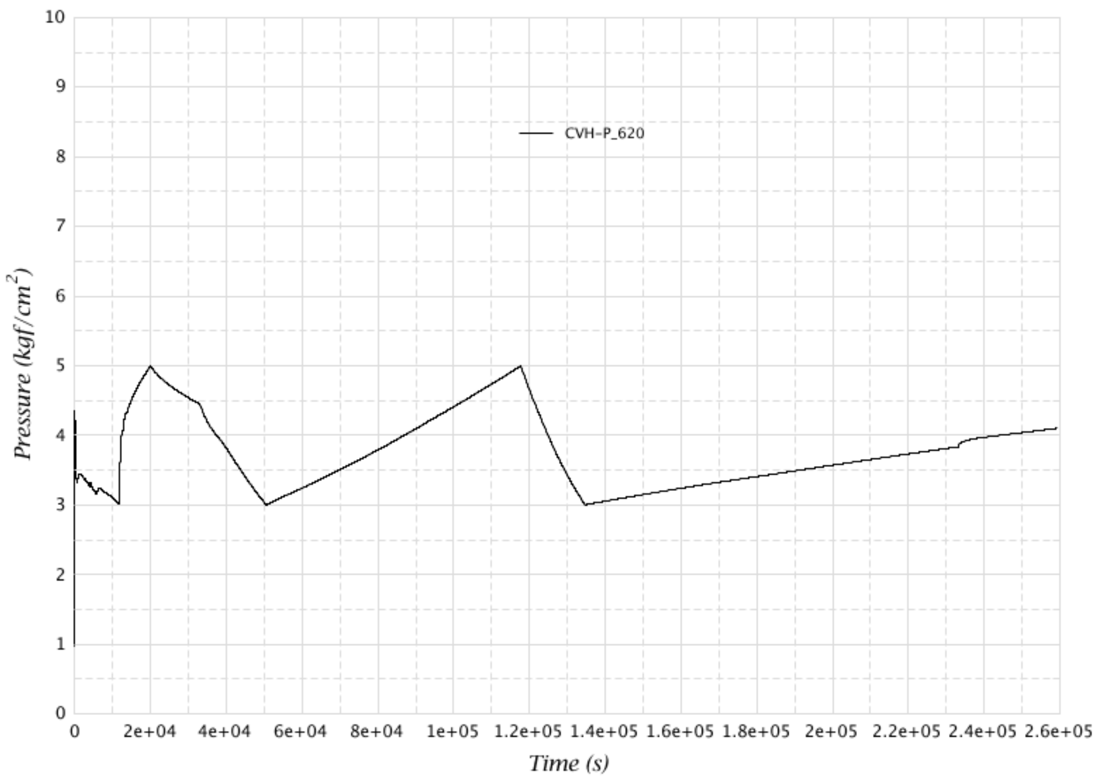

FIG. 3. FCVS application for WWER-1000/320 containment under LB LOCA with loss of power supply.

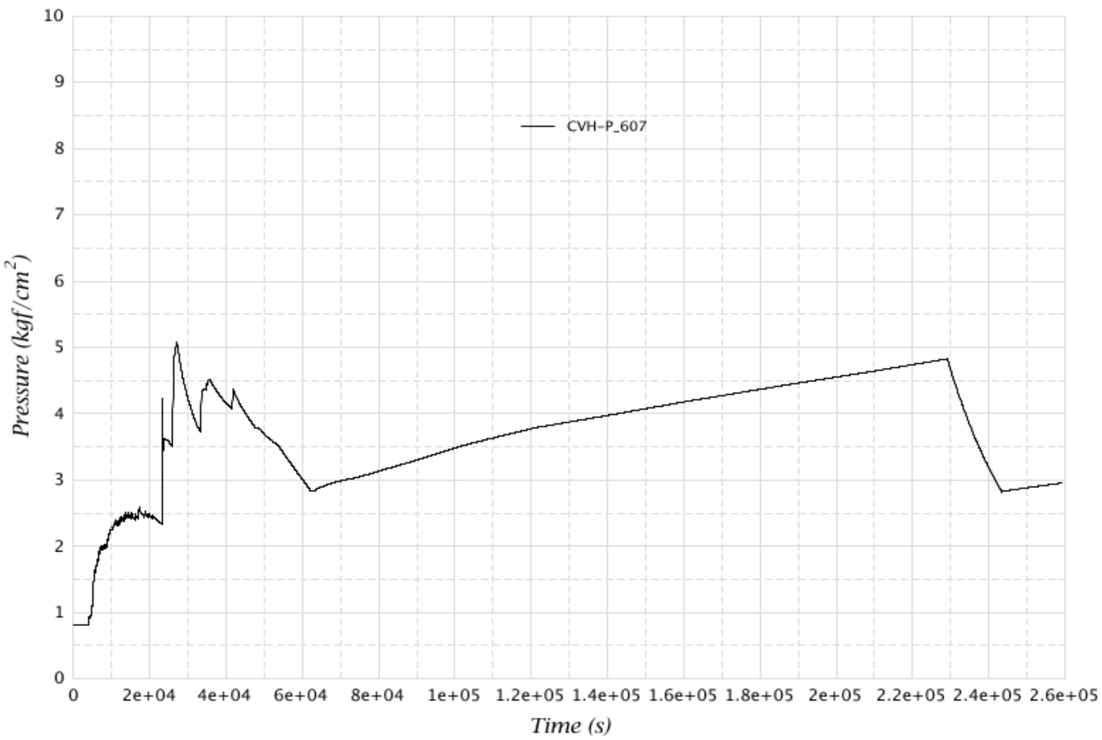

FIG. 4. Pressure in WWER-1000/320 containment under total station blackout.

According to results of analytical justification of FCVS for WWER-1000/320, the following preliminary requirements have been developed:

— minimum diameter of dump pipeline — more than 100 mm;
— opening pressure ——5 kgf/cm^2 (abs.);
— closing pressure — 3 kgf/cm^2 (abs.);
— cycles of operation — 2 (for 3 days);
— total time of operation — 14 hours.

4.2. FCVS for WWER-1000/302

According to the conceptual technical solution of FCVS for WWER-1000/302, containment venting implementation was subdivided into two main steps: implementation of non-filtered venting systems and installation of filters on non-filtered venting system. Two stages of analytical justifications were performed.

Results of analytical justifications for non-filtered containment venting showed:

- possibility to reach the maximum design pressure for WWER-1000/302 containment (Fig. 5);
- containment venting pipeline must be more than 100 mm;
- according to analysis of radioactive impact on the environment due to containment venting activation, the non-filtered venting should be activated only if containment pressure is sufficiently higher than design pressure and continues to increase, and there is no possibility for its decrease.

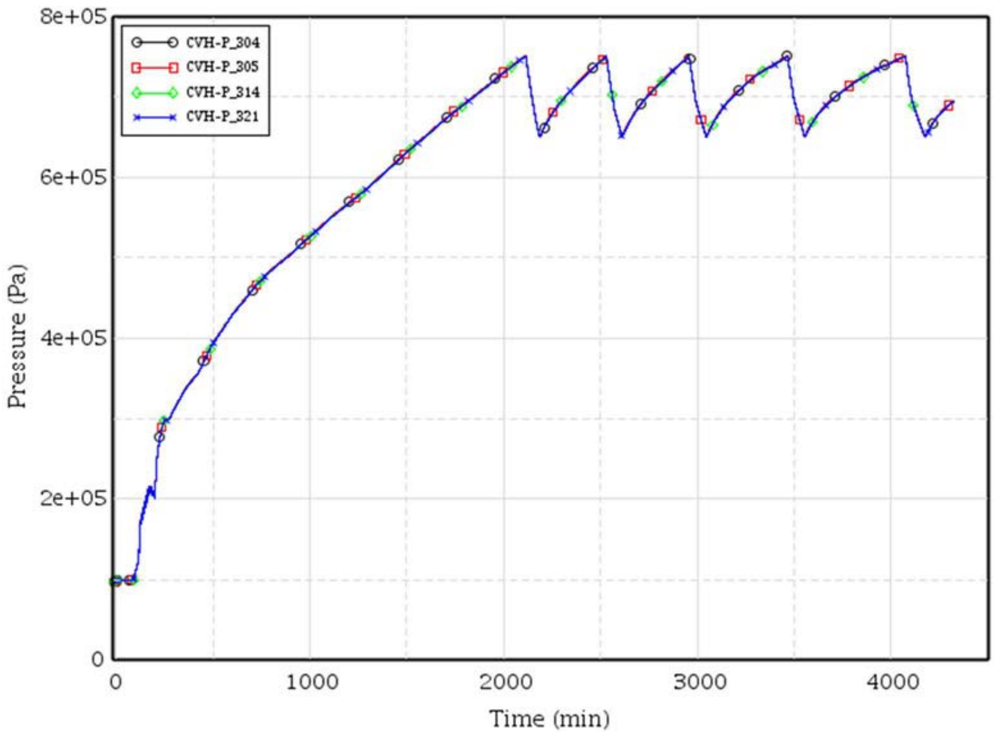

FIG. 5. Pressure in WWER-1000/302 containment under total station blackout with primary side feeding.

Results of analytical justifications of filtered containment venting showed:

— possibility to prevent reaching of maximum design pressure for WWER-1000/302 containment (Fig. 6);

— containment filtered venting pipeline must be more than 100 mm;
— opening pressure —5 kgf/cm^2 (abs.);
— closing pressure — 3 kgf/cm^2 (abs.);
— cycles of operation — up to 3 (for 3 days);
— total time of operation — 10 hours.

According to justifications for WWER-1000 (both for 302 and 320), the necessity and possibility of CFV implementation were showed.

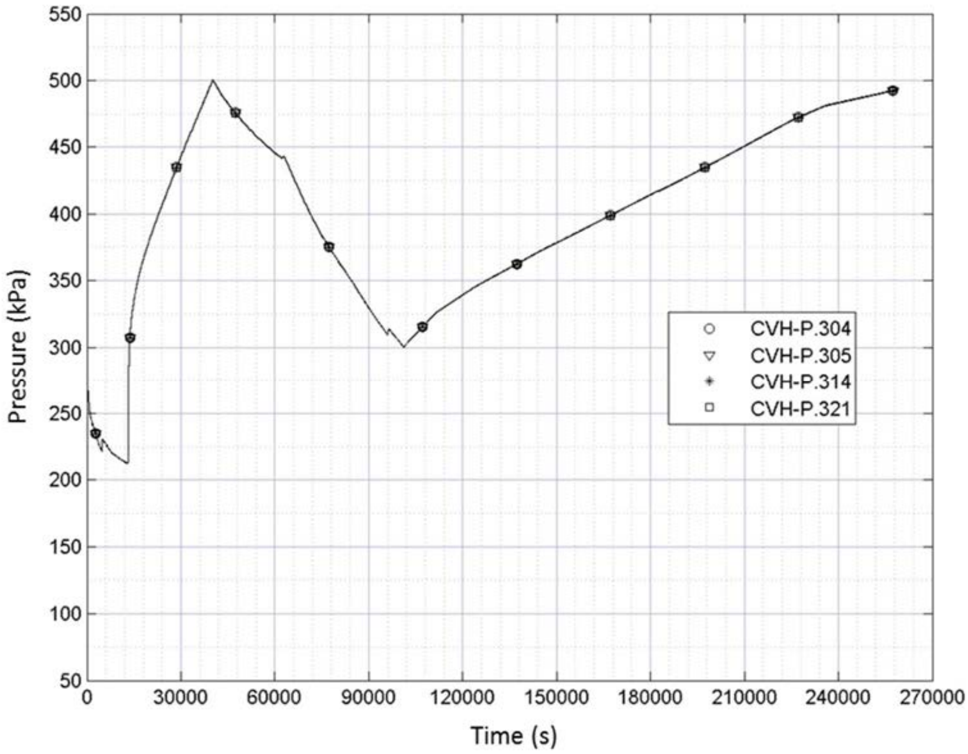

FIG. 6. FCVS application for WWER-1000/302 containment under LB LOCA with loss of power supply.

4.3. FCVS for WWER-440

Results of the best-estimate analyses of severe accident progression for WWER-440 (see Fig. 7) did not confirm the possibility of containment pressure increase above the maximum design limit (2.5 kgf/cm^2 (abs.)).

Only a number of conservative assumptions can lead to exceeding the design limit for WWER-440 containment (see Fig. 8), such as:

— conservative assumption about cavity concrete content;
— failure to take into account containment leakage (more than 16% of initial mass per day);
— consideration of possible water presence in cavity before reactor vessel failure.

Based on results of analytical justification, the conceptual technical solution has been developed for WWER-440 containment filtered venting. This solution foresees implementation of the filtered containment venting system based on existing exhaust ventilation system.

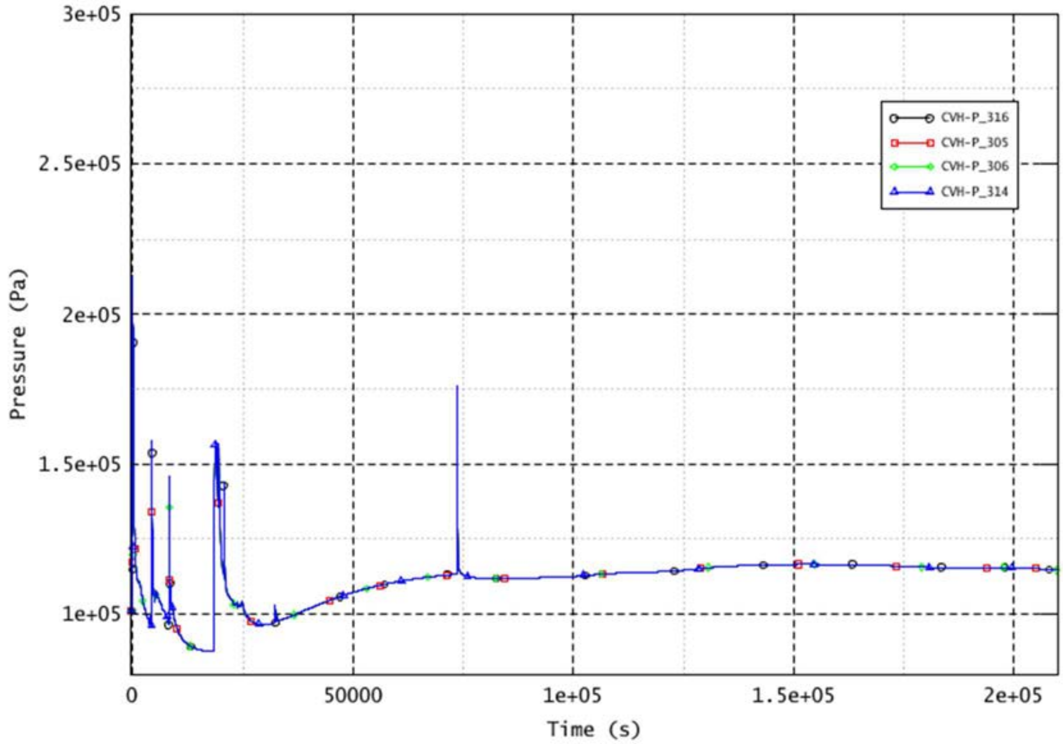

FIG. 7. Pressure in WWER-440 containment under LB LOCA with loss of power supply (best-estimate analysis).

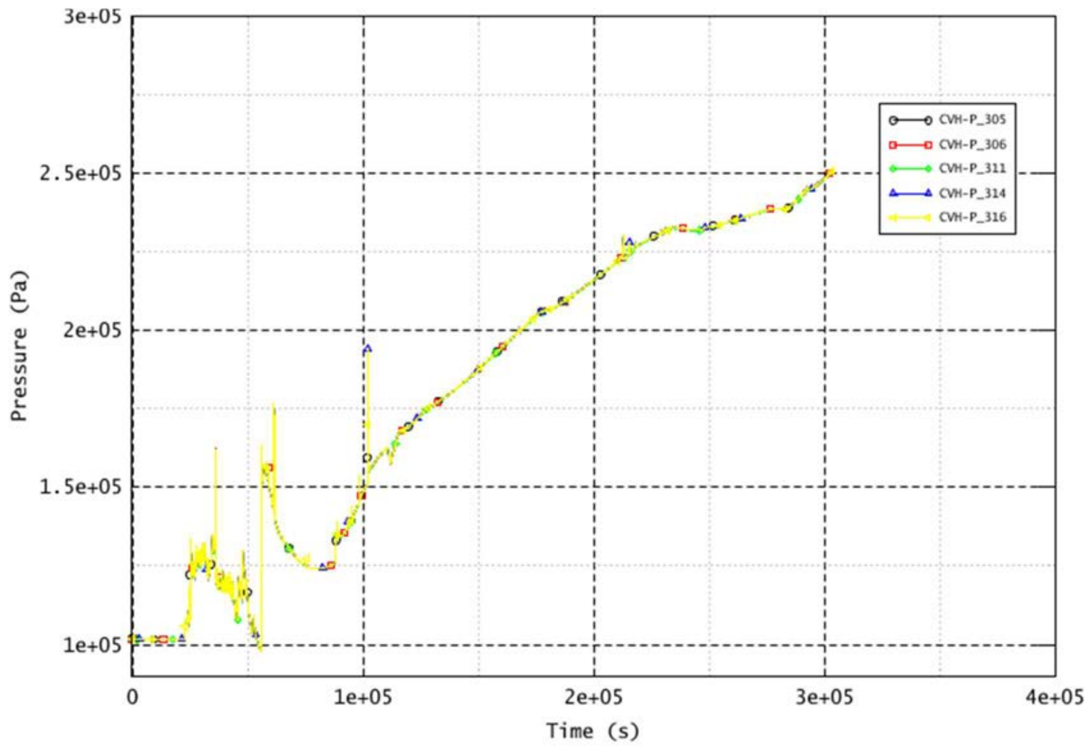

FIG. 8. Pressure in WWER-440 containment under LB LOCA with loss of power supply (conservative analysis).

5. FINDINGS OF STATE REGULATORY REVIEW

Results of analytical justifications of the FCVS for three reference units have been verified in the framework of state review. The main results of the review were related to analysis of calculations and technical justifications for correctness and review of technical solutions for validity (including operation algorithm).

Under review of calculations and technical justifications for correctness, additional calculations (benchmarks) were conducted by SSTC NRS experts. The aims of these benchmarks were to check the correctness of justification results and confirm the validity of SSTC NRS expert comments and suggestions for the reviewed documents.

Analysis of the technical solutions is needed for confirmation than the implemented measure will solve the safety problem and will be the best solution for the selected NPP.

Findings of state review are briefly described below.

5.1. State review findings for WWER-1000/320

The setpoint of FCVS venting stop must be updated or additionally justified. The existing value (3 kgf/cm^2) can lead to deep containment vacuum (see Fig. 9). Besides, it is necessary to take into account that higher setpoint will lead to decrease of FCVS operation and, as a result, to lower radioactive release to the environment. SSTC experts proposed to foresee additional investigation and justifications at FCVS implementation stage.

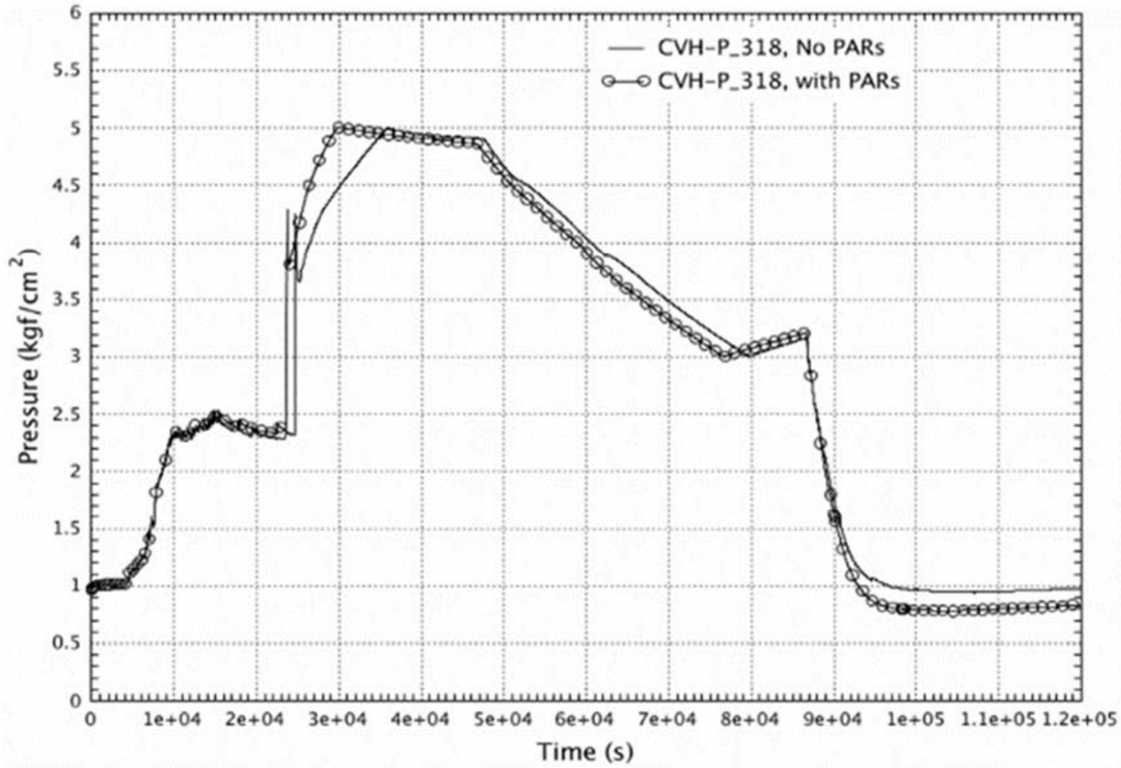

FIG. 9. Influence of spray activation under FCVS operation on containment pressure.

Review of the results showed overestimation of the aerosol mass and energy deposited on venting system filters. Benchmark calculations confirmed that these values were overestimated by more than 30%. The Utility checked the decay heat model, recalculated all analyses and conducted a number of sensitivity analysis. According to new results, maximum aerosol heat power and maximum aerosol mass deposited on filters were updated.

Review results showed that FCVS justifications were performed without taking into account PARs. PAR operation leads to generation of additional heat (due to hydrogen recombination), resulting in decrease of FCVS efficiency. In addition, PAR accounting leads to:

- earlier FCVS setpoint reaching (see Fig. 9);
- increase of containment temperature (see Fig. 10);
- decrease (below the PAR operation limit) of oxygen concentration in the containment atmosphere (see Fig. 11);

Additional investigation considering PARs was performed by the Utility. Results of these calculations will be taken into account under FCVS implementation at units with WWER-1000. Venting size increased to 125 mm (previous 100 mm).

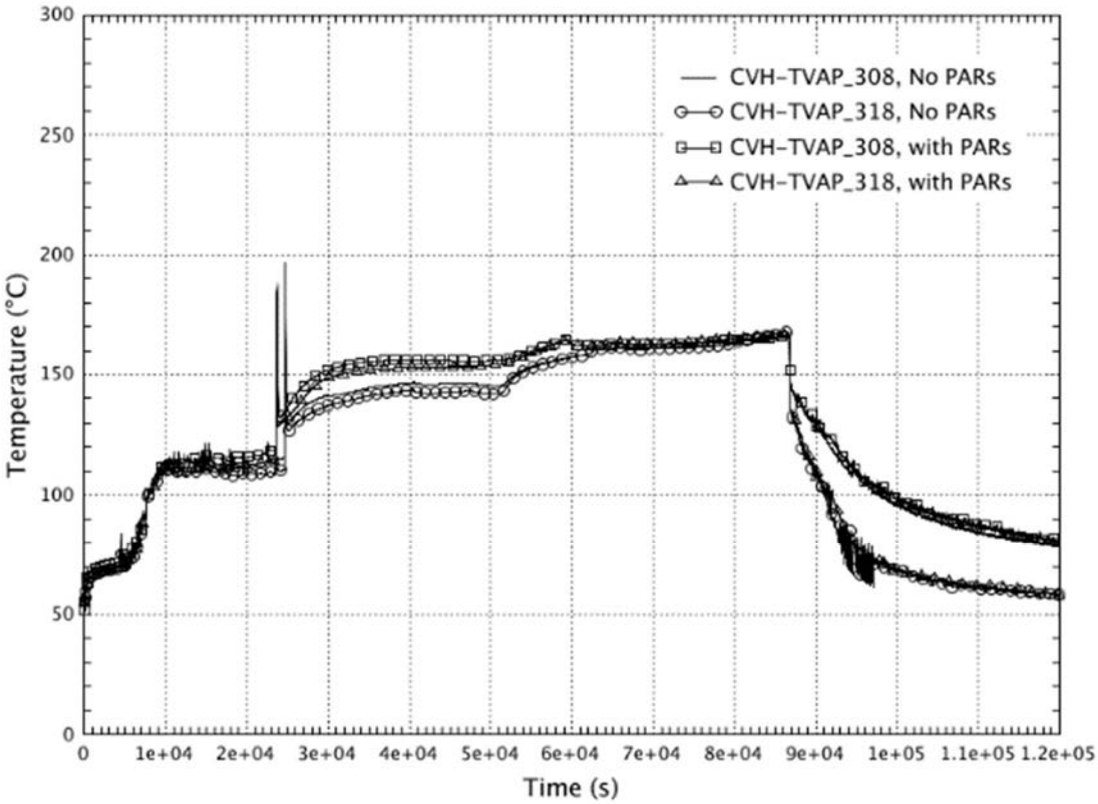

FIG. 10. Influence of PAR installation on containment temperature.

5.2. State review findings for WWER-1000/302

In general, results of the state review for WWER-1000/302 materials are very similar to results for WWER-1000/320.

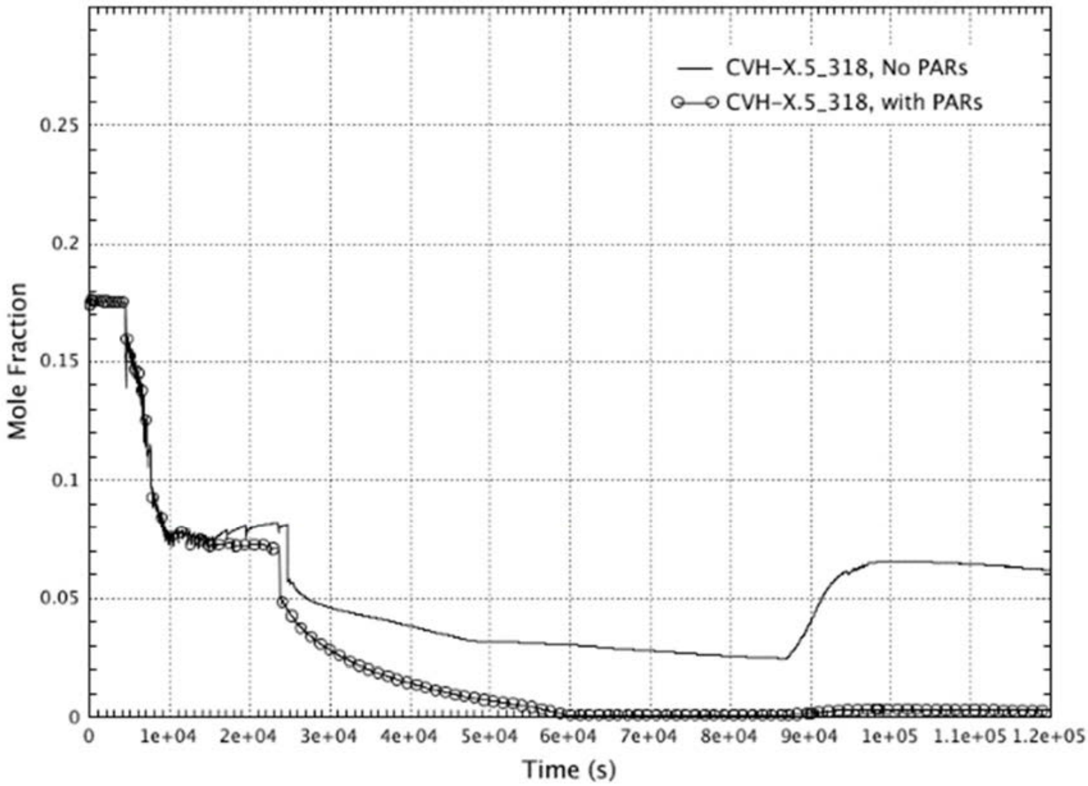

FIG. 11. Influence of PAR installation on oxygen concentration in containment atmosphere.

5.3. State review findings for WWER-440

The developed conceptual technical solution is based on international experience (first of all, on the solution for Kozloduy NPP), but it did not take into account WWER-440/213 features, containment design (see Fig. 12) and high containment leakage (more than 16 % of initial mass per day). The majority of expert comments were related to this aspect.

SSTC NRS experts proposed to investigate the possibility of FCVS use to reduce radioactive release. The benchmark analysis was conducted to investigate the efficiency of FCVS activation at early phases of severe accidents (e.g. immediately after core damage). Results of this benchmark analysis showed reduction in radioactive release in comparison with the case without FCVS activation (see Fig. 15). It was recommended to carry out in-depth analysis for this aspect in analytical justification.

SSTC NRS conducted benchmark with FCVS modelling at outlet of air traps (see Fig. 12). In this case, additional radionuclide scrubbing through water on bubble-condenser trays is possible. Results of benchmark confirmed expert assumption. The total radioactive release was decreased (see Fig. 13) due to additional radionuclides scrubbing on bubble-condenser trays (see Fig. 14) and due to dumping through FCVS quite “clear” atmosphere from air traps.

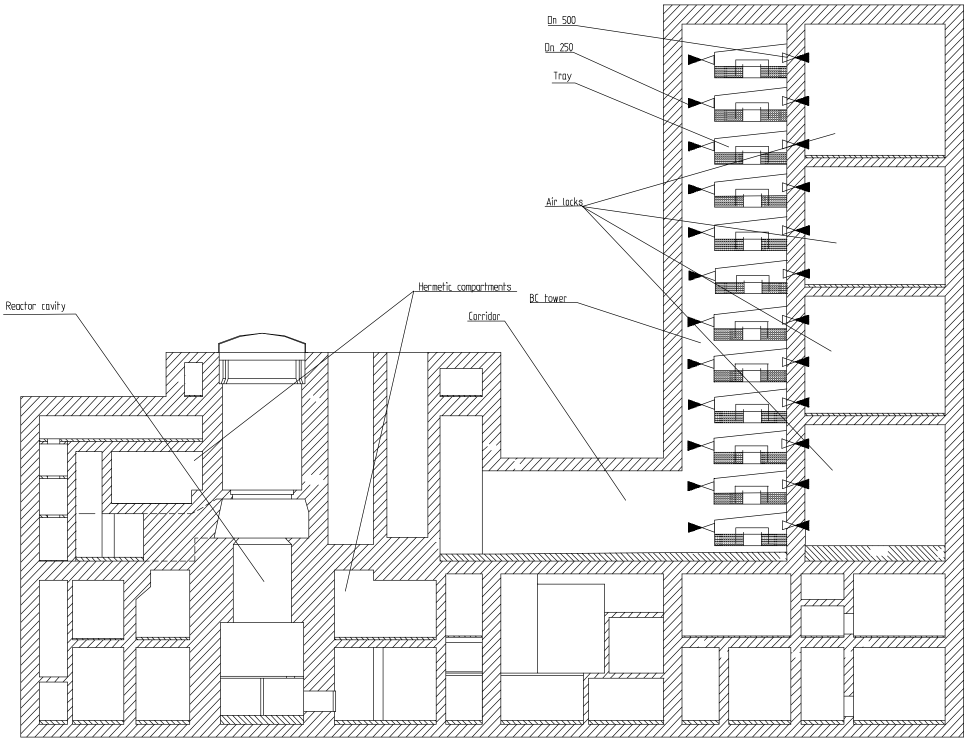

FIG. 12. Schematic view of the WWER-440/213 containment.

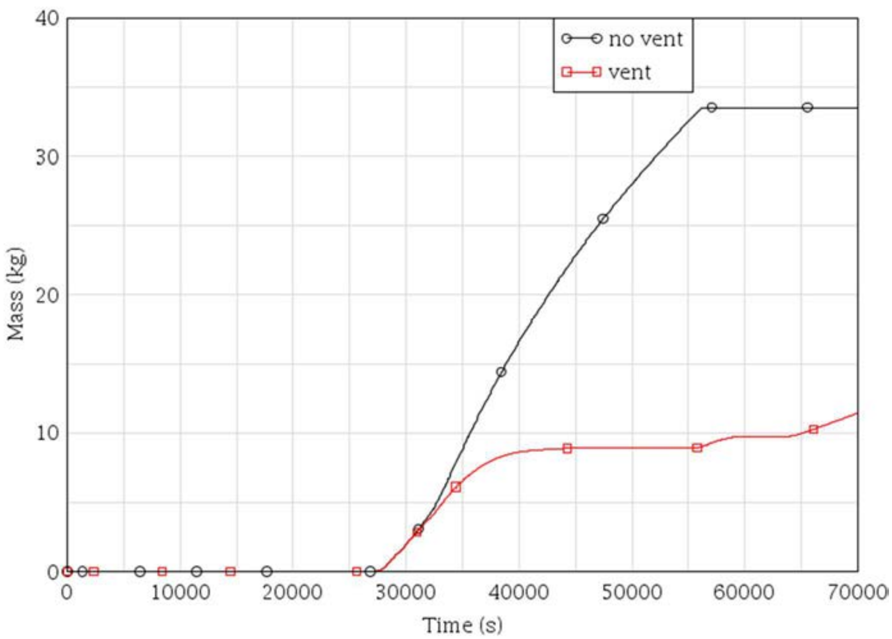

FIG. 13. Mass of radionuclides released to environment under total station blackout on WWER-440.

It is necessary to note that three main "post-Fukushima' modifications are planned for Ukrainian WWER-440. They include in-vessel retention modification, PAR installation and FCVS implementation. According to SSTC NRS opinion, coupled analyses should be performed for these modifications due to their interference. At present, the FCVS conceptual technical solution for WWER-440 and analytical justification are being updated by the Utility.

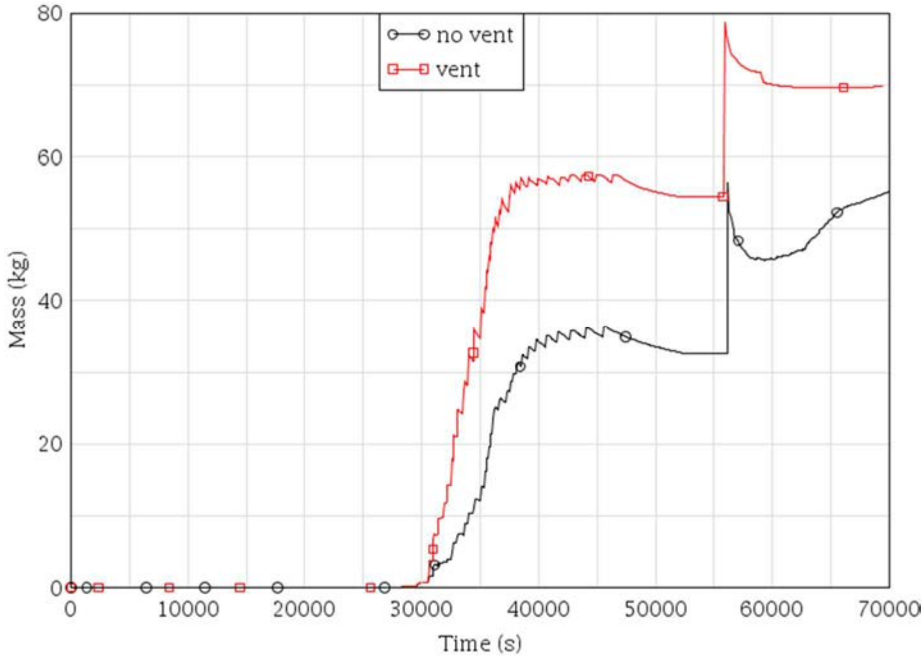

FIG. 14. Mass of radionuclides scrubbed on BC trays.

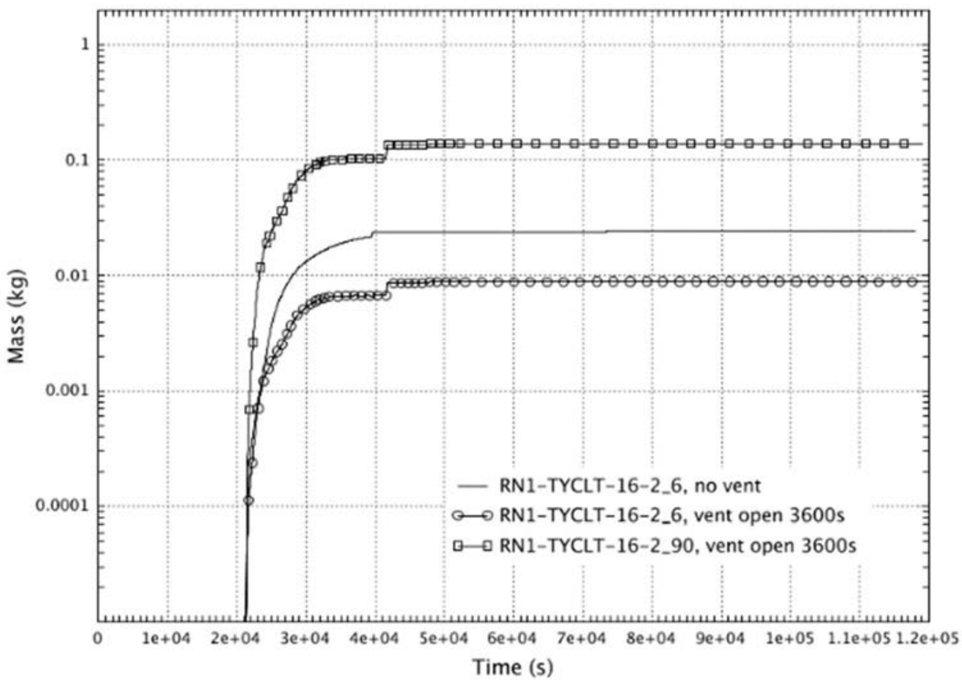

FIG. 15. Efficiency of CFVS operation for release reduction for WWER-440.

6. CONCLUSIONS

The FCVS is one of the main modernization foreseen for the Ukrainian NPPs. It was included into the Comprehensive Safety Improvement Programme (measure “Implementation of Containment Venting System”). This measure is one of the requirements for NPP long term operation.

At present, the conceptual technical solutions were developed by the Utility and agreed with SNRIU for WWER-1000. FCVS for WWER-1000 are under implementation. The technical solution for WWER-440 should be updated according to the state review results.

PROVISIONS FOR CONTAINMENT INTEGRITY AT RUSSIAN WWER NPPs UNDER BDBA CONDITIONS

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1. GENERAL INFORMATION

The Fukushima Daiichi accident demonstrated that, in the absence of alternatives to reduce the containment pressure build-up (due to steam and incondensable gas accumulation during an accident), the venting of the containment becomes an essential accident management measure for the preservation of its structural integrity. This accident has led to considerations as regard to possible implementation of filtered containment venting at those NPPs where these systems have not been applied, as an enhancement of their capability to withstand severe accident situations.

At present, the Russian Federation has 6 WWER-440 units and 12 WWER-1000 units in operation. Up to now, none of them are equipped with FCVS.

The reason for that is the fact that there are no strict Regulatory Body requirements demanding to provide FCVS in the NPP design. Only some indirect references can be found in the existing requirements applied for operating Russian NPPs, for example the principal safety document named “General Safety Rules OPB-88/97” reads:

- It. 4.6.2 The reactor and NPP systems and elements containing radioactive substances shall be accommodated entirely in leak proof compartments for localizing radioactive substances released within their limits during design basis accidents. In this or any other case of arrangement it is necessary to ensure that during normal operation and design basis accidents the corresponding reference exposure doses as well as allowable release rate and content of radioactive substances in the environment are not exceeded. The necessity and acceptability of the directed release of radioactive substances under beyond design basis accidents shall be substantiated in the design.

or in “Rules of Design and Operation of NPP Confinement Safety Systems NP-010-98” one can find the following requirements:

- 3.7.1. If in the nuclear power plant design a provision is made for the discharge of the working medium from one compartment within the accident confinement area to its another compartment, or for discharge of the working medium beyond the accident confinement borders (apart from discharge of the working medium through steam passive condensers), then such accident confinement areas shall be equipped with the safety and (or) relief devices (dump valves, rupture diaphragms, release (nonreturn) valves, etc.) with decontamination of the working medium being discharged from the accident confinement area.

- 4.5.3. The confining safety systems shall be designed with allowance for pressure raising as result of combustion of the hydrogenous mixtures. The systems (component elements) and compartment can be protected from destruction by means of pressure release devices, fire barriers, hydraulic locks.

Initially, for earlier WWER reactor plant designs the beyond design accidents accompanied with a prolonged loss of ultimate heat sink or electric power supply like those happened at Fukushima NPP were not considered in the design bases. So, it was very important to perform these BDBA analyses for operating WWERs to reveal any potential weaknesses in the NPP designs. These activities were carried out in the scope of the so called "stress tests".

In order to obtain the most objective appraisal of the NPP response to BDBA in addition to events happened at Fukushima plant:

- loss of power supply, including major station black-out (SBO);
- loss of ultimate heat sink;
- combination of the above-said events.

the following additional accident conditions were examined:

- Large break LOCA combined with loss of all AC power sources;
- Long term blackout with full core unloaded to SFP;
- Long term blackout during power operation;
- Long term blackout during refueling.

The BDBA analyses were carried out both for power operation conditions and for shut-down states.

2. RESULTS OF BDBA ANALYSES FOR OPERATING WWER-1000

Figures 1 & 2 present the calculated containment pressure behavior for the worst BDBA cases without accounting for additional technical means. These results demonstrate the need in improvements of operating WWER NPP safety.

3. MEASURES TO IMPROVE SAFETY OF OPERATING WWER NPPS

Based on the results of "stress-tests" and calculation analyses performed, Rosenergoatom has developed the Programme for mitigation of BDBA consequences at NPPs aimed at improvement of BDBA management and mitigation of their consequences. According to this programme, additional so-called "post-Fukushima" technical means were suggested for operating WWER-1000 such as:

- mobile diesel-generator unit (MDGU) MDGU-2.0 MW and MDGU-0.2 MW, capable to provide electric power supply to the equipment as required to overcome BDBA.
- mobile pump units (MPU):
  — MPU 150/900 for boron solution supply to the reactor ;
  — MPU 500/50 for service water supply to consumers;
  — MPU 150/120 for water supply to steam generators;
  — MPU 40/50 for boron solution supply to spent fuel pool and/or water pumping from «bottom elevations» in case of flooding.

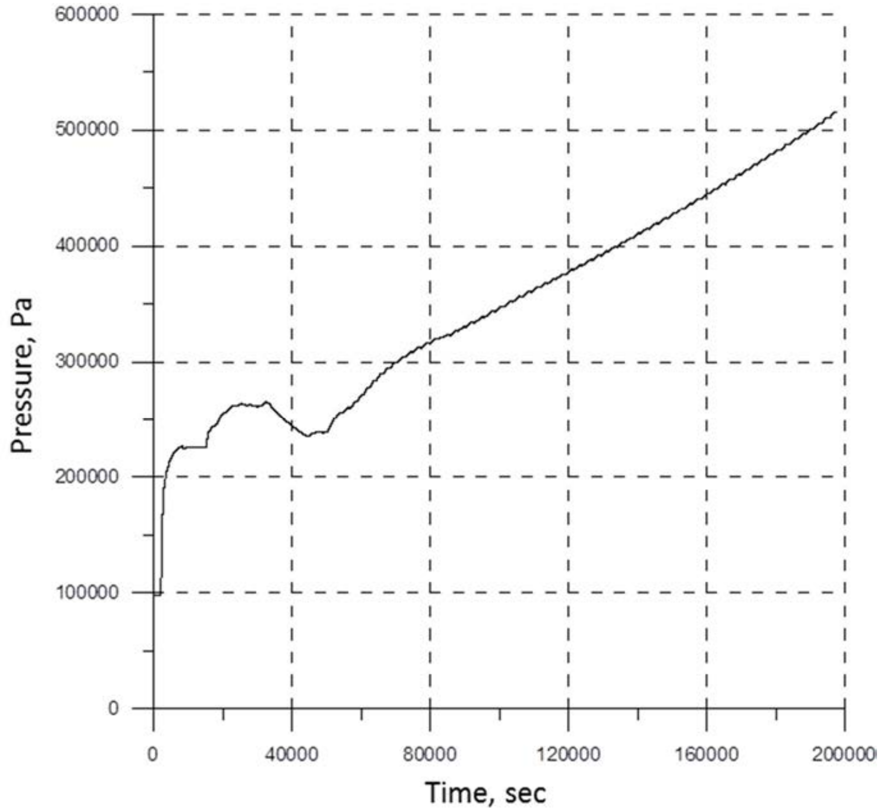

FIG. 1. WWER-1000 (mod. V-320) containment pressure vs time, under station blackout at power operation.

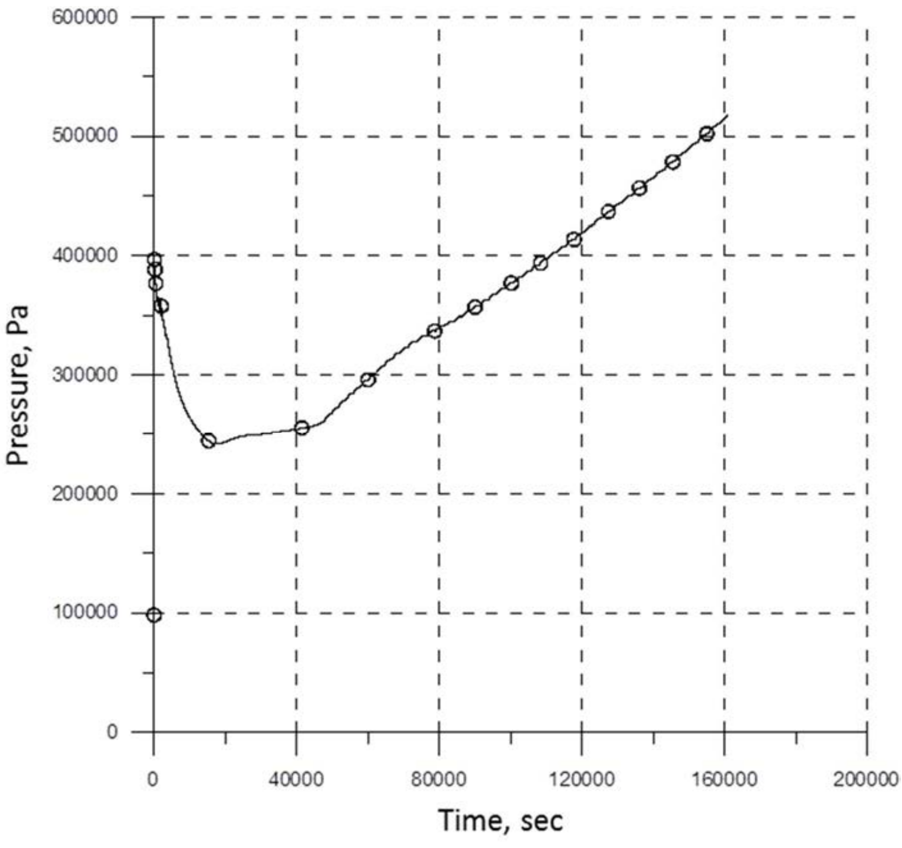

FIG. 2. WWER-1000 (mod. V-320) containment pressure vs time, under LB LOCA.

Depending on the purpose of MPU, the sources of water are: emergency sump or water storage tanks, refueling water storage tanks, demineralized water storage tanks, service water intake channels and open natural sources.

All additional mobile equipment maintains operability after SSE. At present, the number of additional mobile equipment already delivered to the NPP sites is:

- 29 Mobile 2.0 MW DG (6 kV; 0.4 kV; 220V DC);
- 37 Mobile 0.2 MW DG (0.4 kV);
- 36 Mobile high pressure pump units with different capacity and pump head;
- 1 High capacity separate pump unit;
- 80 Monoblock pumps with different capacity and pump head.

MPU 150/900 are stationary placed near the main reactor building, MPU 500/50, MPU 150/120 and MPU 40/50 are stored on-site in hangars or light sheds and in case of necessity are brought promptly to the connection points. The Mobile diesel-generator units MDGU-2.0 MW are placed at specially equipped grounds near the main reactor building.

4. RESULTS OF BDBA ANALYSES FOR OPERATING WWER-1000 WITH ACCOUNT FOR ADDITIONAL TECHNICAL MEANS

The results of analyses have demonstrated that in case of station blackout, loss of ultimate heat sink or LOCA the use of additional (post-Fukushima) technical means prevents reaching the containment limiting pressure. An example showing the effectiveness of additional mobile equipment use for prevention of containment pressure growth over the limiting design level is provided at the Figure 3, presenting the results of BDBA analyses for the worst case LB LOCA with account for additional technical means.

Table 1 provides the summary of the performed analyses results with comparison of the available time reserve for actuation of additional equipment and the real time needed for its actuation as it is shown by on-site emergency exercises.

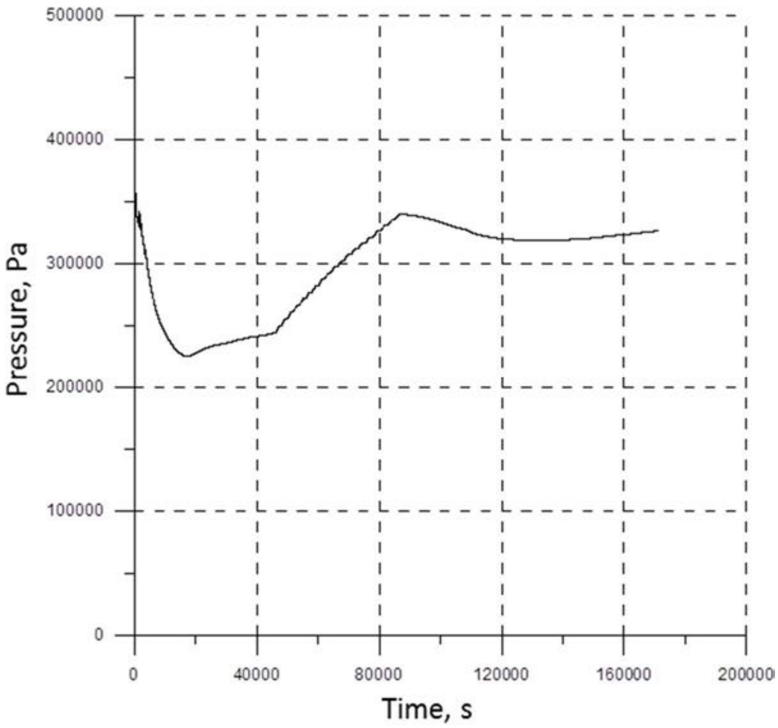

FIG. 3. WWER-1000 (mod. 320) containment pressure vs time, under LB LOCA with account for water supply to SFP from mobile pumps.

TABLE 1. SUMMARY OF THE PERFORMED ANALYSES

Accident condition	**Required additional equipment**	**Time reserve for actuation (requirements based on TH calculations), h**	**Real time needed for actuation (resulting from on-site emergency exercises), h**	**Comment**
LOCA+Blackout	MDGU-2,0 MW MPU 40/50	24 24	<1 1 - 1.5	Prevention of containment failure
Blackout with full core unloaded to SFP	MDGU-2,0 MW MPU 40/50	24 5-10	<1 1 - 1.5	Prevention of fuel damage and containment failure
Blackout during power operation	MDGU-2,0 MW or	2	<1	
	MPU 150/900 MPU 150/120 MPU 40/50	2 2 12	1.5 - 2 1 - 1.5 1 - 1.5	
Blackout during refueling	MDGU-2,0 MW or	4 - 4.5	<1	
	MPU 150/900 MPU 500/50 MPU 40/50	4 - 4.5 4 - 4.5 12	1.5 - 2 1.5 - 2 1 - 1.5	

Grounding on these results, the NPP General designer made the following conclusions on the expediency of FCVS implementation at operating NPPs:

- Due to specific design of WWER-1000 containment and spent fuel pool, without account for management actions the available time reserve to possible containment failure is in the range from 2 to 4 days;
- The comparison of the estimated times required for deployment of the additional technical means necessary for BDBA management and the real times required for bringing them into action (basing on the results of emergency exercises at NPPs) demonstrates the capability of their successful implementation in case of accidents;
- The BDBA management using additional technical means makes it possible to exclude containment failure due to pressure increase and ensures integrity of the core and fuel in the spent fuel pool. The availability of several ways of accident management utilizing additional high-reliable technical means gives confidence in high probability of successful performance of the required safety functions;
- Taking into account the fact that usage of the additional technical means stops the containment pressure build-up, the installation of containment venting system is deemed a supplementary protection measure that does not contribute much to NPP safety.

Nevertheless, the Russian operating organization, Rosenergoatom has issued a directive to investigate the possibility of FCVS implementation at operating WWER NPPs. At present, the following activities are already carried out:

- The initial input data for designing of FCVS are defined in assumption that the additional technical means are unavailable;
- An Industry's Decision on the way of implementation of FCVS is issued;
- The initial technical requirements to FCVS for operating WWER NPPs are developed;
- The mode of operation of FCVS is determined (periodical short-run operation with succeeding recharging);
- The decision is taken that first FCVS will be installed at Rostov NPP unit 1 and after that FCVSs will be implemented at all other operating WWER NPPs;

- The Technical Assignment for delivery of non-standard process equipment for Rostov NPP unit 1 is developed;
- The FCVS implementation schedules are developed for all NPPs;
- The engineering design documentation for upgrading of standard systems that will be used for connecting FCVS is developed.

5. POSSIBLE FCVS DESIGNS FOR WWER NPPS

Two filtration methods are considered for implementation at WWER NPPs:

- «Wet» filtration;
- «Dry» filtration.

For 'wet' filtration (AREVA) there are difficulties in placement of the equipment due to its weight and dimensioning specifications:

- At WWER-1000/VB-320 NPPs two layout variants are possible: in auxiliary building compartments or near the RB wall from the outside;
- At 'small series' WWER-1000 NPPs the equipment can be mounted in SWT building.

Table 2 provides the comparative analysis of the two variants of placing FCVS.
If the «dry» filtration is selected, then the equipment can be placed inside the containment and in compartments of auxiliary building as well.

TABLE 2. COMPARATIVE ANALYSIS OF THE TWO VARIANTS OF PLACING FCVS

Placement in the auxiliary building	**Placement outside the RB**
advantages: – Constant readiness to operation in case of BDBA (the system is already filled); – Absence on necessity to construct additional building for placing of equipment	*advantages:* — No need in relocation of existing pipes and communications; – Possibility to perform erection works in periods between planned maintenance.
disadvantages: — The main scope of work is possible only during planned maintenance outage; – Difficulties of installing large-size equipment requiring removal of the roof, provision of additional sanitary zone, relocation of existing pipes and communications, inconvenient conditions of works.	*disadvantages:* – A less availability for operation (a need to fill the Venturi tank); – Confirmation of feasibility to design the system as of 1st seismicity class accounting for connections with other systems is needed; – Additional justification of feasibility to design pipelines and equipment containing highly radioactive substances outside the RB is required; – Possible problems related to start-up of the system at negative temperatures.

The issues to be solved when implementing FCVS:

- Ready FCVS designs that fully meet the laid requirements do not exist;
- Decisions regarding the design and type of FCVS for WWER NPPs are not made yet;
- The layout solutions require additional elaboration (due to difficulties of placing and mounting);

- Difficulties in purchasing non-standard imported equipment, and long manufacturing period;
- Licensing of equipment and erecting works.

6. RESULTS OF BDBA ANALYSES FOR NEW WWER DESIGNS

The issue of necessity of implementation of FCVS in the new WWER NPP designs such as AES-2006 (WWER 1200) or WWER-TOI (WWER.1300TOI) was investigated as well. All these designs are based on AES-92 design and their distinctive feature is a wider application of passive safety systems capable to perform safety functions in the full scope.

The results of ‘stress test’ analyses for accident Fukushima NPP conditions showed that they do not result in the core damage and the time reserve for bringing either active safety systems or additional (post-Fukushima) technical means into operation is at least about 10 days.

Safety analysis of new WWER NPP designs for more burdened conditions as compared to Fukushima NPP such as Loss of all AC electric power sources combined with LOCA of primary side main circulation pipeline showed that under the stated emergency conditions the residual heat is reliably removed from the reactor core by joint operation of passive heat removal system (PHRS) and the second-stage and third-stage hydroaccumulators.

The NPP autonomy (no damage to the core) in this mode of operation is determined by water inventory in the HA-2 and HA-3 hydroaccumulators and is not less than 72 h.
The ultimate containment pressure level could be reached after ~ 5 days.

7. ACTIVITIES ON SAFETY IMPROVEMENT OF NEW WWER NPP DESIGNS

In order to improve the new WWER NPP robustness against low probable and hypothetic events and increase the autonomy period in case of BDBA, the designs are provided with a number of additional technical means for heat removal from the core, spent fuel pool and containment, such as:

- Mobile diesel-generator unit;
- Alternate intermediate circuit pump;
- Mobile fan cooling tower,

and their connection to the regular equipment provided for design-basis accident management is foreseen.

8. CONCLUSIONS

For operating WWER NPPs it was demonstrated that in presence of additional technical means the installation of containment venting system is deemed a supplementary protection measure that does not contribute much to NPP safety. But taking into account that in certain accident modes the time needed for connection of additional technical means is comparable with the required time reserve for their bringing into operation, the possibility of FCSV implementation was analyzed. If a final “political” decision on equipping NPPs with FCVS will be taken, then the first FCVS will be installed at Rostov NPP unit 1 and after that FCVSs will be implemented at other operating WWER NPPs.

For new NPP designs, taking into account large time reserves (ensured by operation of passive safety systems) for enabling additional technical means under accidents resulting in containment pressure increase, the decision on inexpediency of FCVS implementation has been taken.

SESSION II

LONG TERM CONTAINMENT RESPONSE TO SEVERE ACCIDENTS IN LIGHT OF THE FUKUSHIMA DAIICHI ACCIDENT

BEHAVIOUR OF CONTAINMENT SYSTEMS UNDER LONG TERM STATION BLACK OUT CONDITIONS

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Abstract. Nuclear reactor containment constitutes the most crucial barrier to protect surroundings of a nuclear reactor from a reactor accident as also to protect the reactor from external events. The prime objective of the nuclear containment is to limit the radioactivity release into the environment in case of an accident and in no case more than the prescribed limits. Containment performance analysis for the TAPS BWR has been carried out for Source term estimation under long term SBO for with and without availability of the Emergency Condensers (EC). In this assessment, performance of proposed hard vent system for TAPS BWR has also been assessed for containment over pressure protection. Prior to this, containment modelling capability with ASTEC code has been assessed for Fukushima accident for Unit-I, CANDU6 reactor under IAEA-CRP and validation against in-house experiments conducted at Containment Studies Facility. In general predictions are found to be in good agreement.

1. INTRODUCTION

Nuclear reactor containment is the last and most crucial barrier to protect surroundings of a nuclear reactor from a reactor accident as also to protect the reactor from external events. More numbers of Nuclear Reactors are required for energy need for its cleaner and greener energy output. As land availability for new reactors is limited hence they may be placed close to the public area and this can only be fulfilled by the philosophy of minimum activity release and related dose to the public at any conditions. This can be achieved by targeting at all front like reduction of the containment pressure and fission product from the containment atmosphere as quickly as possible besides managing hydrogen and minimization of the leakage from the containment. The prime objective of the nuclear containment is to limit the radioactivity release into the environment in case of an accident and in no case more than the prescribed limits.

A source term analysis involves various codes for evaluation of Fission Product (FP) from core to the containment. The analyses involves core thermal hydraulics evaluation for pressure and temperature and Fission Products (FP) release from the core, assessment of containment thermal hydraulics along with FP transport, Molten Corium Concrete Interaction (MCCI) in case of vessel failure and performance of containment engineered safety systems. System code ASTEC is well in use for containment thermal hydraulics, FP transport, hydrogen transport and MCCI in the containment predictions for quite some time. On a similar line an analytical model has been developed on ASTEC code for Tarapur Atomic Power Station (TAPS) BWRs for evaluation of containment behaviour and source term under extended SBO without severe accident measures. As a confidence building the user's modelling capability with ASTEC code has been carried out for Fukushima accident for Unit-I. Predicted containment pressure and temperature transient were in good agreement with actual measured plant data. As a part of further confidence building measures, validation against in-house experiments conducted at Containment Studies Facility, BARC has been carried out. The participation in the IAEA CRP on "Benchmarking severe accident computer codes for PHWR applications" for CANDU6 reactor for prolonged SBO conditions has further improved our understanding of the models used for assessing different containment related phenomena as mentioned above.

The paper presents the containment performance analyses for the TAPS BWR for Source term estimation under long term SBO. In this assessment, performance of proposed hard vent system for TAPS BWR has also been assessed for containment over pressure protection. The paper also presents validation exercise carried out for ASTEC with CSF data as well as a brief on for containment behaviour under extended SBO condition for CANDU 6 and Fukushima Unit-1.

2. BRIEF DESCRIPTION OF THE CONTAINMENT SYSTEMS

For severe accident (other than DBA), substantial fraction of core inventory of fission product (FP) may be released into the containment. Mostly FPs are released into the containment in the form of aerosol excluding nobles gases like Xe and Kr. Fission product release to the environment depends upon airborne concentration of FP in the containment, containment pressure and the containment leakage rate with time. Containment leakage rate is directly proportional to the containment pressure. In order to minimize the leakage rate, containment pressure has to be reduced as quickly as possible. The Containment engineered Safety Features (ESFs) are provided to perform energy and radioactive management in containment building during and after an accident. Containment safety systems of a NPP are provided to limit/mitigate the consequences of postulated accidents in order to protect the plant personnel, the public and the environment. Containment systems consist of various sub systems and are designed to ensure or contribute to the achievement of the following safety functions:

(1) Confinement of radioactive substances in operational states and in accident conditions;
(2) Protection of the plant against external natural and human induced events;
(3) Radiation shielding in operational states and in DBA & BDBA accident conditions.

Major objective of the containment system is to provide envelope and thus isolate systems and components whose failure could lead to unacceptable amount of the activity into the environment. The reactor core, as well as all piping and components of the reactor coolant system should be totally enclosed within the containment structure. The envelope should also include those components connected to the reactor coolant pressure boundary that cannot be isolated from the reactor core in the event of an accident. The structural integrity of the containment is required to be maintained and the specified maximum leak rate should not be allowed to exceed design leak rate in any condition pertaining to design basis accidents including severe accidents condition. The containment structure should be able to withstand the maximum peak pressures arising from postulated loss of coolant accident (LOCA)/main steam line break (MSLB) events considered in the design, in conjunction with other loads such as internal and external events. Considerations such as potential for generation and behaviour of flammable gases like hydrogen, assessment of containment pressure build-up in the event of selected beyond design basis accidents should be given in the design for containment systems for postulated severe accidents. The release of radioactivity from the containment can be controlled by a combination of the following:

(1) Containment isolation system;
(2) Leaktightness of containment;
(3) Limiting the containment pressure by containment cool down/ depressurization features/systems;
(4) Reduction of radionuclide concentration by containment clean-up/confinement features/systems.

3. METHODOLOGY FOR ESTIMATION OF SOURCE TERM

Source term is defined as the quantity, timing and characteristics of radioactive material released to the environment, following a reactor accident. The estimation of source term is carried out with the integration of several computational tools available as shown in Figure 1.

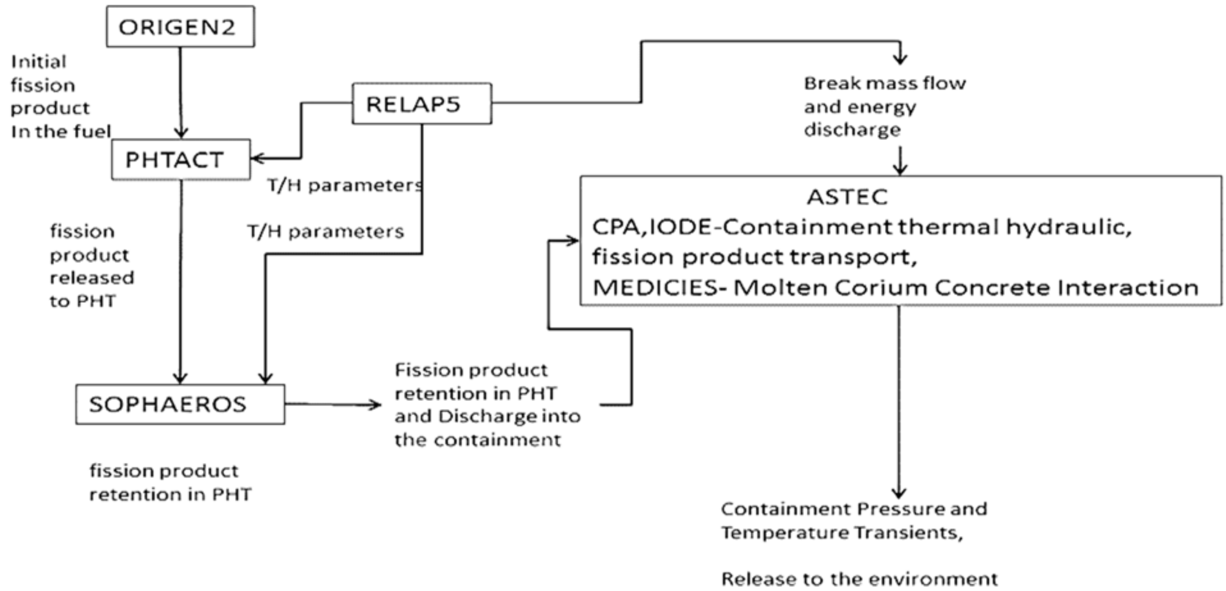

FIG. 1. Methodology for estimation of source term.

The core thermal hydraulic calculations are carried out using best estimate code i.e. RELAP5/MOD3.2 [1,2]. The initial fission product inventory in the core is calculated by ORIGEN2 code [3]. The initial inventory data is input into the PHTACT along with RELAP5 fuel temperature transient to get the fuel release data under various postulated accidents. The PHTACT code [4] is well-validated code with various data available in the literature. The release data along with other relevant thermal hydraulic parameters are input in to the SOPHAEROS module of ASTEC code [5] to find out the retention of fission species in the PHT and further release into containment along with break discharge. Containment thermal hydraulic, fission product transport and molten corium concrete interaction have been simulated using CPA, IODE and MCCI modules of ASTEC code.

4. BRIEF DESCRIPTION OF CONTAINMENT RELATED MODELS OF COMPUTER CODE ASTEC

The code ASTEC [5] is capable to simulate core degradation, fission product release and transportation and containment behavior under severe accident conditions in PWRs, Different modules of this code are being used to simulate some of the phenomena of heavy water reactor severe accident. Different modules like CPA, IODE, MEDICIS are being used for the containment and SOPHAEROS for fission product transport in PHT.

The CPA module of the code is used for containment thermal hydraulics and fission product dispersion and is composed of two main sub-modules, respectively in-charge of the computation of thermal–hydraulics and aerosol behavior in the reactor containment. The containment discretization uses a lumped-parameter approach in which volumes represented by nodes connected by junctions simulating simple or multi-compartment containments such as tunnels, pits, dome etc., with possible leakage to the environment or to other buildings. The module is well validated against series of experiment like BNWL, DEMONA, PHEBUS-FPD, PANDA etc. The SOPHAEROS module of code ASTEC has been used to assess and analyses the transport of fission products in the PHT system of HWRs in the event of a postulated accident scenario. The module is validated against PHEBUS experiments.

The MEDICIS module of code is used for MCCI. It simulates MCCI using a lumped parameter approach with averaged melt/crust layers. Corium remaining in the cavity interacts with concrete walls. This module assumes either a well-mixed oxide/metal pool configuration or pool stratification into separate oxide and metal layers. It describes concrete ablation, corium oxidation and release of non- condensable gases (H_2, CO, CO_2) into the containment. Most convective heat transfer correlations from the literature are implemented in the MEDICIS module. MEDICIS consist of models of corium coolability in case of water injection upon the corium pool surface, including water ingression through the upper crust and corium eruption through the upper crust towards the overlying water pool. Different models of evolution of corium pool configurations with homogeneous and stratified pools are available. Shapiro diagram is used to indicate the hydrogen and CO concentrations in various compartments of the containment and the possibility of hydrogen or CO burning and hydrogen detonation.

5. CONTAINMENT STUDIES FACILITY

Containment Studies Facility (CSF) is designed to study containment performance under DBA and beyond DBA in Indian PHWR. Blowdown experiment up to 100 kg/cm2(g) have been conducted in the CSF. The CSF consists of a PHTM, a Containment System model (CM) and a Control and Instrumentation room. The containment model is approximately 1:250 volumetrically scaled down model of the prototype 220 MW(e) PHWR containment of Kaiga Atomic Power Plant. A schematic of the facility is given in Fig. 2. which shows inner details of the containment Model. Like Indian PHWR containment system, the containment model is also divided into V1 volume (dry well) and V2 volume (wet well). The V1 volume is further divided into many compartments to simulate the various rooms such as pump room, fuelling machine vault etc. present in actual reactor containment. The V2 volume is connected to V1 volume through vent pipes and suppression pool. The containment model is a cylindrical structure with an ellipsoidal dome, made out of RCC with epoxy painting on inner surface. The outer diameter of the model is 6.9 m and its height is 10.95 m. Pressure vessel (PHTM) is meant for simulating the primary heat transport system of the prototype reactor. The pressure vessel is connected to the containment model by a blowdown pipe. A Double Rupture Disc (DRD) with suitable operating mechanism is mounted on this pipe. The blow down could be initiated by causing rupture of these rupture discs. The process instrumentation for the CSF mainly consists of pressure, temperature and level measurements at various locations on the PHTM and Containment Model for the requirement of transient data recording. In vapor suppression pool, the level variation and associated loads generated during the flow of air-steam mixtures through pool water are measured using submersible pressure transducers. Several blowdown tests have been conducted [6] for different pressure and corresponding saturation temperature in the PHTM. Table-1 shows comparison among experimental and analytical results for peak containment pressure and peak temperature.

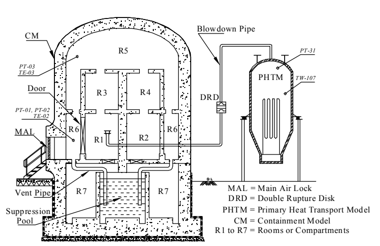

FIG. 2. Schematic of containment studies facility.

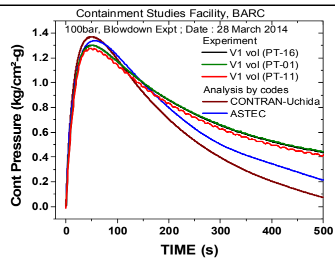

FIG. 3. Containment pressure (V1volume) variation (100bar Expt.).

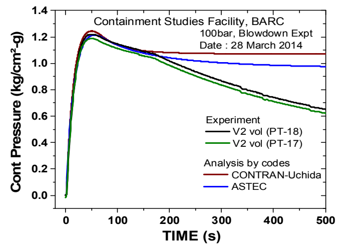

FIG. 4. Containment pressure (V2volume) variation (100bar Expt).

5.1. Comparison with posttest calculation

Post-test analysis was carried out in two steps. In first step, blowdown discharge and energy rate was obtained indirectly [6] using thermal hydraulic code RELAP5/Mod3.2 code. This rate was independently cross verified with strain based flow measurement device on the principle of impact load on the plate which was developed and installed in the containment model. The obtained blowdown mass and energy data were then used as input for the second step, in which, the ASTEC and in-house containment code, CONTRAN were used for calculating containment model transient parameters like compartment pressure, temperature, wall temperature etc. Pressure variation in the PHTM vessel has been compared with the RELAP code results. Figures 3 and 4 show containment pressure variation in V1 (Dry well) and V2 (Wet well) with analytical results using CONTRAN [7] and ASTEC code. From Post-test analysis it has been found that Uchida condensation model gives containment pressure close to the experimental observed value. ASTEC results were also in good agreement with experimental data. Wall temperatures and compartment temperatures were in good agreement with experimental data.

TABLE 1. PEAK CONTAINMENT PRESSURE AND TEMPERATURE

S.N.	Blowdown Test Pressure ($kg/cm^2(g)$)	Peak Containment Pressure ($kg/cm^2(g)$)			Peak Containment Temperature (^oC)		
		Exp.	CONTRAN	ASTEC	Exp.	CONTRAN	ASTEC
1	30	0.57	0.6		-NA-	90	
2	50	0.82	0.83	0.83	116.4	119.0	117.2
3	75	0.99	1.01	1.01	120.2	122.3	120.1
4	100	1.3	1.36	1.33	125.0	127.8	125.0

6. IAEA COORDINATED RESEARCH PROJECT - CANDU REACTOR

Containment analysis for severe accident conditions was carried out for 700 MW(e) CANDU reactor under IAEA Coordinated Research Project (CRP) [8]. Objective of this CRP was benchmarking of severe accident computer codes for PHWR applications under prolonged Station Black Out (SBO) condition. Containment Pressure, Temperature transient, concrete ablation rate due to MCCI, hydrogen and CO distribution, fission product transport and aerosol transport in the containment has been carried out using ASTEC code. Results were compared among 7 International Participants from different countries. In general there was a good agreement in the initial phase of the transients. Analysis was carried out in four phases starting (i) SBO initiation to first channel uncovering (ii) Core disassembly (iii) calandria vessel failure (iv) calandria vault failure. Containment analysis was done using code ASTEC. The containment model consists of modelling of several compartments of containment, exhibiting thermal hydraulic volumes, heat structures representing walls, junctions, rupture discs. The model calculates thermal hydraulic behavior of volumes, fission gas release and deposition on the Calandria Vault wall. It also takes care of the MCCI. The Nodalization used for the containment modelling is the same as that of AECL. Figure 5 shows comparison of time predicted by participant for different chronological events during progression of the severe accident. Figure 6 shows total amount of hydrogen generated during accident by all the participant. Containment pressure variations predicted by all the participants are shown in Figure 7. Comparison is found to be good in agreement in the initial phase of transients. Various codes including RELAP, PHTACT, ASTEC modules (SOPHAEROS, CPA, MEDICIS) have been used for the analysis.

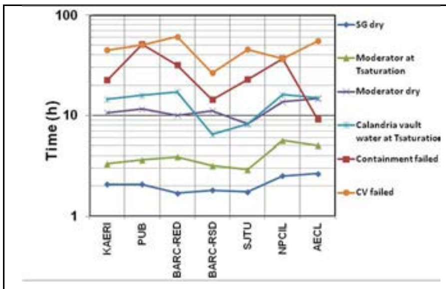

FIG.5. Comparison of time predicted by participant for different chronological events.

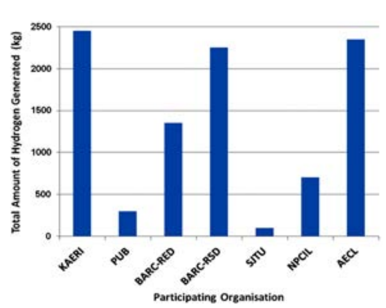

FIG. 6. Total amount of Hydrogen generated during accident.

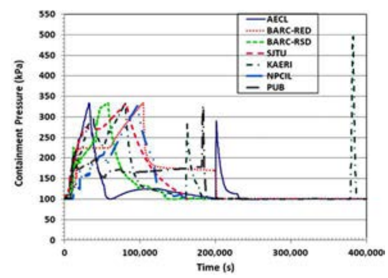

FIG. 7. Containment pressure variation calculated by the participants.

7. SEVERE ACCIDENT ANALYSIS OF FUKUSHIMA DAIICHI UNIT-1

Severe accident analysis for unit-1 has been carried out based on data available by analysing the thermal hydraulic behavior of containment, hydrogen transport, MCCI, fission product transport in the containment, and release to environmental using ASTEC code. For fission product transport in the containment, 18 species including Xe, Kr, Cs and Iodine and their cases of decay heat have been considered. Except Xe and Kr all other FPs and Iodine have been taken as aerosol form in the numerical model. Input required for Ex-vessel analysis has been obtained from In-vessel analysis carried out separately. The analysis has been carried out for one day transients with an accident scenario derived from the Japanese Government Report [9].

The scenario starts with reactor shut down with high magnitude earthquake signal followed by turbine trip, Main Steam Isolation valves (MSIV) closure and start-up of DGs to support Class-III systems. As the reactor pressure rises, repeated operations of the isolation condenser (IC) were made and attempted to control the reactor pressure. This is followed by water addition to IC secondary side and to the reactor pressure vessel (RPV). At the end of one day of the accident, the reactor building exploded from hydrogen detonation. For reactor core analysis, a plant specific model has been developed using safety analysis computer code RELAP5/SCDAP mod 3.4 [1, 2]. RPV and its internals, two recirculation loops, feed water system, and steam line with relief valves and MSIVs have been modelled. The RPV and its internals are nodalised with several control volumes, heat structures, flow paths and structural components. The transient pressure, RPV inventory, RPV level, recirculation flow, IC flow and inventory, SRV cumulative flow, temperature histories of fuel, RPV wall, generation of hydrogen etc. have been evaluated. The results of analysis compare well with the plant observations as well as calculation carried out by NISA with code MELCOR. The fission gas release at various stages of accident has been evaluated with PHTACT code [4].

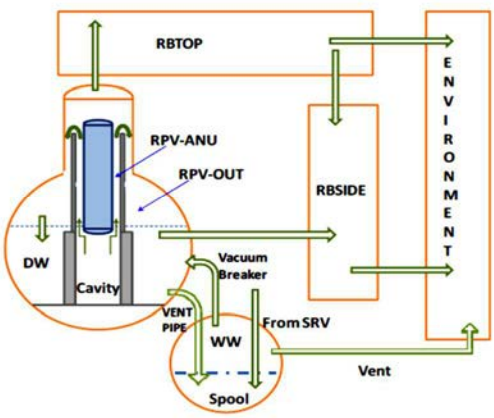

FIG. 8. Containment nodalization.

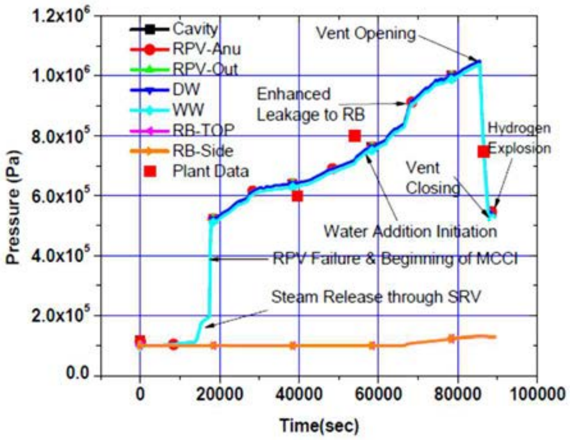

FIG. 9. Variation of containment pressure with time.

For the analysis, containment was divided into seven zones including drywell, suppression chamber and reactor building. Fig. 8 shows the Containment Nodalization. DW is connected to Suppression Chamber (SC) through vent pipes. The release from relief valves are also led to the SC as shown in Fig. 8. Deposited and suspended mass of the FP with time have been evaluated in all the zones. Inert atmosphere as initial condition has been considered in PCV and in suppression chamber. Continuous release of corium from RPV in case of vessel failure and external water injection were considered. A large relocation of the molten corium on the floor of reactor cavity and consequent Molten Corium Concrete Interaction (MCCI) was found to be releasing a large amount of hydrogen and carbon monoxide, making primary containment to further pressurize and the reactor building susceptible to explosion. At containment high pressure, enhanced leakage from the containment was modeled. For MCCI analysis, homogeneous corium configuration and Siliceous concrete for rector vault have been considered. CPA and MEDICIS module of ASTEC were used for containment thermal hydraulic and MCCI analysis. Predicted containment pressure and temperature transient were in good agreement with actual measured plant data. From the Shapiro diagram it was found that only the top region of the reactor building was entering in deflagration/detonation region.

The variation of containment pressure is shown in Fig. 9. The initial sharp pressure rise (at 17500 sec) was observed due to start of the MCCI phenomena. Production of large amount of combustible and non-combustible gases caused steady pressure to rise. At 55000 s fire water injection was started into the RPV. Due to the rupture of RPV bottom, the injected water flowed into PCV through the rupture. Interaction of injected water with the upper layer of MCCI, from 55000 s onwards, caused increase in steam production and containment pressurization rate as well as crust formation on the melt pool surface. At 67000 s, the pressure difference between PCV and RB reached 0.75 MPa, causing enhanced leakage from PCV to the RB, however, the PCV pressure continued to rise due to MCCI. PCV venting was done at 85440 s from wetwell to the environment reducing the pressure in the containment. Vent was closed based on pressure criteria (pressure <0.53 MPa).The estimation shows a large release of radioactive inert gases and volatile fission product release to atmosphere with the collapse of the reactor building after one day of the accident. Source term (ground and stack level release from containment to Environment) of first unit of Fukushima Daiichi was evaluated up to H_2 explosion in unit-1. By and large, the present analysis predictions agree with the event chronology, recorded plant parameters and measured activity and with the analysis prediction of NISA.

8. LONG TERM SBO ANALYSIS OF TAPS-BWR

Fukushima accident in March 2011 demonstrated the importance of the containment over protection system like hard vent. Subsequent to this in almost all the reactors, hard vent system has been proposed. Since containment size is very small in the BWR especially Mark-I type containment, hard vent system is the immediate requirement. India is having two BWRs (TAPS-1 and TAPS-2) at Tarapur with power production of 160 MW(e) by each. It is proposed to install hard vent system in the TAPS1&2. On a similar line an analytical model has been developed on ASTEC code for TAPS BWRs for evaluation of containment behavior and source term under extended SBO for with and without availability of the Emergency Condensers. In this assessment, performance of proposed hard vent system for TAPS BWR has also been assessed for containment over pressure protection. The analyses involves core thermal hydraulics evaluation for pressure and temperature and FP release from the core, assessment of containment thermal hydraulics along with FP transport, MCCI in case of vessel failure and performance of containment engineered safety systems. The TAPS primary containment consists of drywell, wetwell and common chamber volumes (Figure 10). A containment nodalization is shown in Figure 11. Drywell houses reactor pressure vessel, reactor recirculation loop etc. Drywell is a pear shaped containment pressure vessel which connects to the suppression pool through the vent pipes. Suppression pools of both the unit are connected to a common chamber which is normally not in communication with the pool. Large blowout diaphragms are provided between pool and the Common chamber which get open at the pressure differential of the 6" psig. Drywell, suppression pool and common chamber are parts of the primary containment. Primary containment is surrounded by secondary containment called as Reactor Building which houses various systems like refueling and reactor servicing equipment, spent fuel storage facilities and other equipment.

Proposed hard vent system will be linked to the common chamber and it can relive containment pressure when the differential pressure between common chamber and environment reaches to 6 psig. Source term analysis was performed for two cases for with and without availability of the Emergency condenser. Input considered for the analysis is given in the Table 2. About 31 number of fission products were considered for the FP transport. CPA, MEDICIS and IODE module of the ASTEC has been used for the containment analysis. Analysis has been carried out for homogenous corium configuration. Siliceous as concrete aggregate type has been considered. As melt falls into the cavity MCCI starts and crust formation takes place. Thickness of crust layer depends on the solidus-liquidus temperature of the corium melt and pool and ambient temperature. During MCCI, Crust formation takes place at the interface, which may break due to continuous formation of the gas bubbles. Temperature of the melt which is falling into the cavity floor from RPV has been taken as 2600 °C. Core thermal hydraulics analysis has been carried out for both the cases (TAPS SBO with and without EC) using RELAP code. Figures 12 and 13 show steam mass flow rate from SRV to the wetwell and pump seal leakage of steam to the drywell respectively for TAPS SBO without EC. This input was used for containment thermal hydraulics calculations.

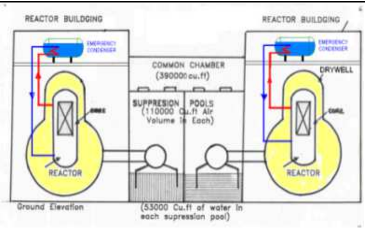

FIG. 10. Schematic arrangement of TAPS 1 & 2 Containment and other systems.

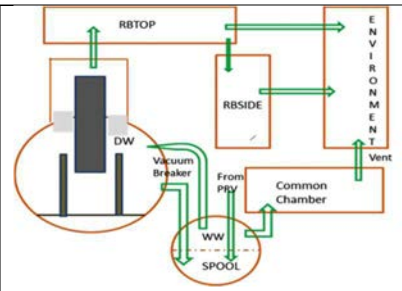

FIG. 11. Containment nodalization.

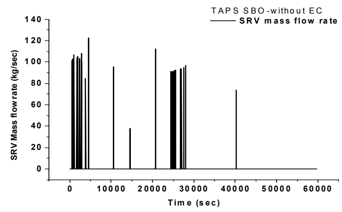

FIG. 12. Steam mass flow rate from SRV to wetwell.

8.1. Containment pressure

The variation of containment pressure is shown in Figures14 and 15 respectively for with and without EC availability. Which show that initial containment pressure start rising due to steam and hydrogen release into the containment. With availability of Emergency Condensor the peak containment pressure delays by 3 hours. Pressure reduction due to opening of blowout panel between suppression pool and common chamber can be seen. Due to hydrogen release, pressure again starts increasing and common chamber pressure reaches to the hard vent system operating pressure. On pressure differential criteria, hard vent system starts reliving pressure by discharging gases into the environment. In actual condition, water vapours will be condensing in the scrubber tank and fission products gets removed due to venturi action in the water pool where fission products containing gas passes with very high velocity through venturi tube. Here Decontamination factor due to the venturi scrubber has not been accounted in the modelling. From Figures 14 and 15 it can be seen that due to operation of hard vent system, pressure in the primary containment remains below the design pressure and it does not rises further.

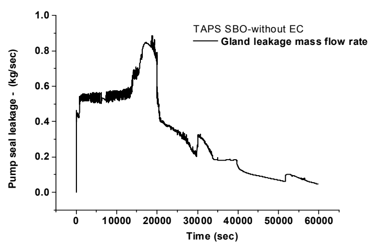

FIG.13. Steam flow rate due to pump seal leakage.

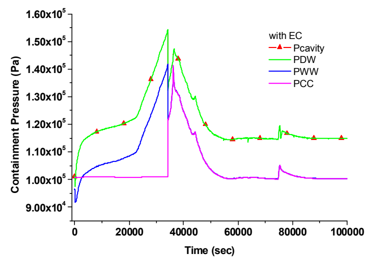

FIG.14. Containment pressure transients (for SBO with EC).

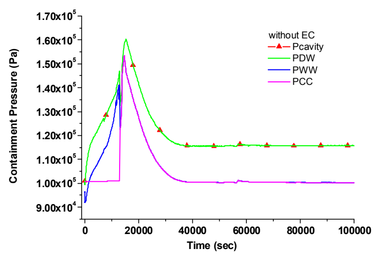

FIG.15. Containment pressure transients (for SBO without EC).

8.2. Containment and corium temperature

Figure 16 shows the temperature variation of various compartments of the containment. An increase in the cavity temperature was observed due to RPV failure and start of MCCI at 56500 s. Later on rise in temperature is observed due to release of hot gases from the corium due to MCCI and increase in cavity wall temperatures. Corium temperature variation with time is shown in Figure 17, which shows that corium temperature remains constant after initial reduction. It is because crust thickness increases with time results in reduction of the heat transfer from the upper surface to the containment atmosphere. Almost all the decay heat is getting used for ablation of the concrete.

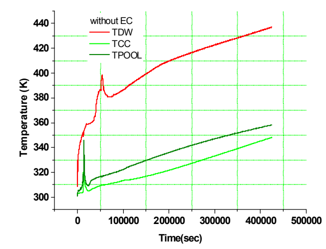

FIG. 16. Containment temperature transient.

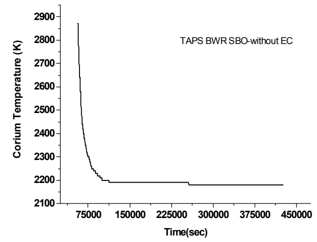

FIG. 17. Corium temperature.

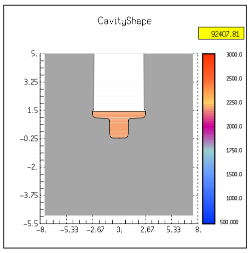

FIG. 18. Erosion of cavity at 92407 sec.

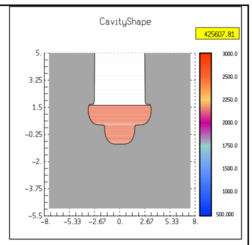

FIG. 19. Erosion of cavity at 425607 sec (~ 5days).

8.3. Cavity shape

Figures 18 and 19 show the cavity erosion due to MCCI at the end of 1st day and 5th day of start of the accident. Cavity erosion starts on vessel failure at 56500 sec due to heat transfer from melt to concrete. The core melt pool is stirred by the rising gas bubbles, which enhances the heat transfer. On the other hand, the possible corium crust at the interface inhibits heat transfer. Since thermal conductivity of the concrete is very poor, almost all of the heat is getting used for melting of the concrete wall. The heat transfer is also affected by change in the viscosity of the melt pool and change in melt composition and its physical properties as concrete is added in the melt continuously. Analysis has been carried out for homogenous corium configuration. Siliceous as concrete type has been considered. As melt falls into the cavity MCCI starts and crust formation takes place. Thickness of crust layer depends up on the solidus-liquidus temperature of the pool and pool and ambient temperature. During MCCI, Crust formation takes place at the interface, which may break due to continuous formation of the gas bubbles.

8.4. Hydrogen mass

Figures 20 and 21 show the variation of hydrogen mass present in different compartments of the containment with and without EC. A sharp rise in the hydrogen mass was observed initially due to steam zirconium reaction within the vessel. Second peak is due to start of MCCI due to RPV failure. Cumulative hydrogen generation for with and without EC case up to 5 days is shown in Table-4, which show that In-vessel hydrogen generation is higher for without EC case and Ex-Vessel hydrogen generation is maximum for without EC. Total amount of H_2 generation up to 5 days is maximum for the without EC.

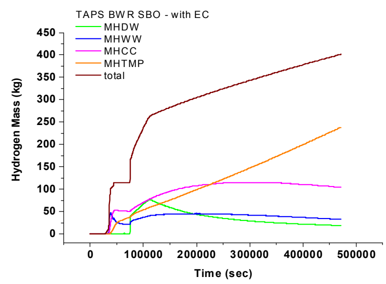

FIG. 20. Variation of Hydrogen Mass Inventory (with EC).

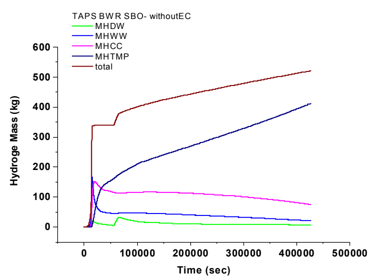

FIG. 21. Variation of Hydrogen Mass Inventory (without EC).

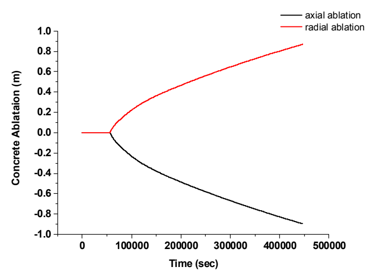

FIG. 22. Erosion thickness in axial and radial direction.

8.5. Concrete erosion

Figure 22 shows the variation of mass of eroded concrete and erosion thickness in axial and radial direction respectively. A total 120 tonnes of concrete is found to have reacted with molten corium leading to erosion of cavity in five days. In the axial and radial direction the erosion thickness is found to be 0.9 m and 0.8 m respectively. At the end of the 5th day calculation, the cavity is not found to be breached with corium interaction.

8.6. Release to the environment

With the clad oxidation and subsequent core melting the FPs starts releasing into the Suppression pool from the SRV of the reactor pressure vessel along with hydrogen and steam. Steam get condenses and non-condensable gases releases to the gas space of the suppression pool after scrubbing of the associated fission products in the water pool. Almost all the fission products are removed in the pool. After vessel failure FP also starts releasing into the Dry Well. A decontamination factor of 80 has been assumed for all the fission products for all sizes of aerosols in the suppression pool. Major amount of FPs are deposited in the wetwell in comparison to the drywell. Transport of 31 FPs have been considered and release into the containment from the core and some of the main Fps are given in the Table-3. Decay heat of fission product with transport has been accounted in the modelling. Release from containment to the environment is also shown the Table 3. Figures 23 and 24 show Cs and I release to the environment, these FPs routed to the environment though gas space of the wetwell, common chamber and then release to the environment. Here DF due to the hard vent system have not been accounted which is generally 1000 for FP in Aerosol form, 100 for Iodine and 90 for organic iodide .

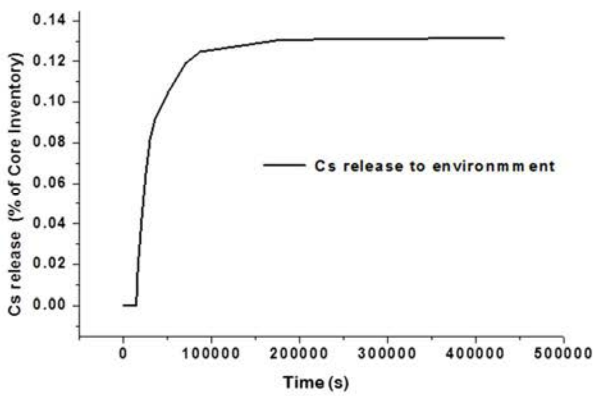

FIG. 23. Cs release to the environment.

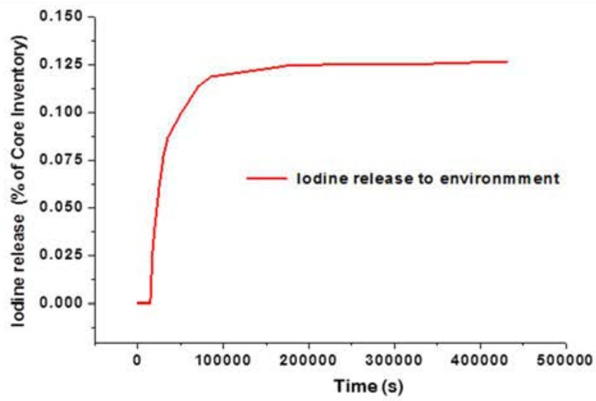

FIG. 24. Iodine release to the environment.

TABLE 2. INPUT FOR THE ANALYSIS

Hydrogen and steam injection into the Suppression pool due to opening of SRV (for without EC)	339.8 kg (hydrogen) 57929 kg (steam)
Pump seal leakage to the DW	19438 kg
Melt release into the cavity (for without EC)	53.1 tonne

TABLE 3. RELEASE FRACTION OF FISSION PRODUCT

	SBO without availability of Emergency Condenser	
Species	Release to the containment (% of core inventory)	Release to the environment (% of the core inventory)
I	100	0.1256
Cs	100	0.1228
Xe	100	100
Kr	100	100
Ba	98	0.08869
Mo	100	0.115
La	44.1	0.021

TABLE 4. CUMULATIVE HYDROGEN GENERATION

	SBO with EC	SBO without EC
In-Vessel	120 kg	340 kg
Ex-Vessel	280 kg	170 kg
Total H_2 generation (in five days)	400 kg	510 kg

9. CONCLUSION

Containment performance analysis for the TAPS BWR has been carried out for Source term estimation under long term SBO for with and without availability of the Emergency Condensers (EC). In this assessment, performance of proposed hard vent system for TAPS BWR has also been assessed for containment over pressure protection. Prior to this, containment modelling capability with ASTEC code has been assessed for Fukushima accident for Unit-I, CANDU6 reactor under IAEA-CRP and validation against in-house experiments conducted at Containment Studies Facility. In general predictions are found to be good in agreement.

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COMPARATIVE ANALYSES ON CONTAINMENT OVER-PRESSURIZATION MITIGATION STRATEGY IN PWR USING MAAP CODES

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Abstract. After the Fukushima Daiichi nuclear power plant accident, re-examination of containment depressurization capability and counter-measures are performed internationally. The representative means to depressurize the containment includes external cooling of the containment, internal decay heat removal and containment venting. In this paper, the comparative analyses between internal decay heat removal using portable equipment and containment filtered venting system are performed using MAAP code. Results show that internal decay heat removal using portable equipment, if successfully prepared, can prevent radiological release and maintain containment integrity as an active way. However, optimal operating strategy is needed to prevent flooding inside the containment and hydrogen flammability. Filtered venting can mitigate radiological release below 10-3 of initial inventory as a passive way. But the molten core concrete interaction cannot be mitigated sufficiently. Overall, each option has its own pros. and cons. So, the final design option should be chosen considering the plant specific overall safety effect and design characteristics and bases of containment structure.

1. INTRODUCTION

After the Fukushima Daiichi nuclear power plant accident, the Government of the Republic of Korea and industry performed comprehensive special safety inspections on all domestic nuclear power plants against beyond design bases external events. As a result, a total of 50 recommendations were defined as safety improvement action items. These were classified into 5 categories, as shown in Table I, and should be implemented. One of the major action items is to maintain containment integrity against over-pressurization scenarios. Containment may lose its functional capability as last barrier of radioactive material release to the environment due to over-pressurization [1]. The applicable means to prevent and/or mitigate the over-pressurization are external cooling, internal decay heat removal and containment venting [2]. In the Republic of Korea, four PHWR will install the filtered venting system because their containment design are vulnerable compared to that of a PWR.

2. CONTAINMENT OVER-PRESSURE PROTECTION ALTERNATIVES

External cooling of the containment requires sufficient heat transfer component or facility, such as steel containment or large scale air cooled heat exchanger and containment penetrations. Also, internal decay heat removal using existing system is not available in station blackout scenarios. So, in this paper, the feasibility of internal decay heat removal using portable equipment system and containment filtered venting system on existing PWR with large dry containment are examined.

2.1. ECSBS

Emergency Containment Spray Backup System (ECSBS) is designed to be used as an active means to control the containment pressure within its ultimate pressure by proper operator action using portable equipment. Similar concepts are used in FLEX strategy against spent fuel pool and BWR design [3]. The schematics of ECSBS are shown in Fig. 1 and are already installed in APR1400, Gen-III+ type reactor.

This system includes:

— independent spray nozzle to the existing safety grade spray;

— independent piping connection to the exterior of the containment building;
— portable pumping source that is independent of site AC power sources;
— external connections for temporary hookup of an external source of water.

TABLE 1. SAFETY IMPROVEMENT ACTION ITEMS IN THE REPUBLIC OF KOREA AFTER THE FUKUSHIMA DAIICHI ACCIDENTS

Categories	Major Action Item
1. Design of structures and equipment against earthquakes	- Installing an automatic seismic trip system - Improving seismic capability of the safe shutdown systems and MCR - Reassessment of maximum potential earthquakes for nuclear plant sites
2. Design of structures and equipment against coastal flooding	- Extension of the height of sea wall - Installation of waterproof gates and discharge pump - Investigation of the design basis sea water level of nuclear plant sites
3. Integrity of electric power, cooling and fire protection system upon inundation	- Securing availability of portable power generator vehicles and batteries - Upgrading design basis of Alternative AC diesel generators - Ensuring counter-measure against loss of spent fuel pool cooling function - Improving fire protection facilities and fire protection plan
4. Countermeasures against severe accidents	- Installation of passive hydrogen removal equipment - Installation of filtered vent system or backup spray system - Revision of SAMG to enhance effectiveness - Installation of reactor injection flow path from external cooling source
5. Emergency response	- Amending emergency plans to address concurrent events at multiple units - Development of extensive damage mitigation guidelines

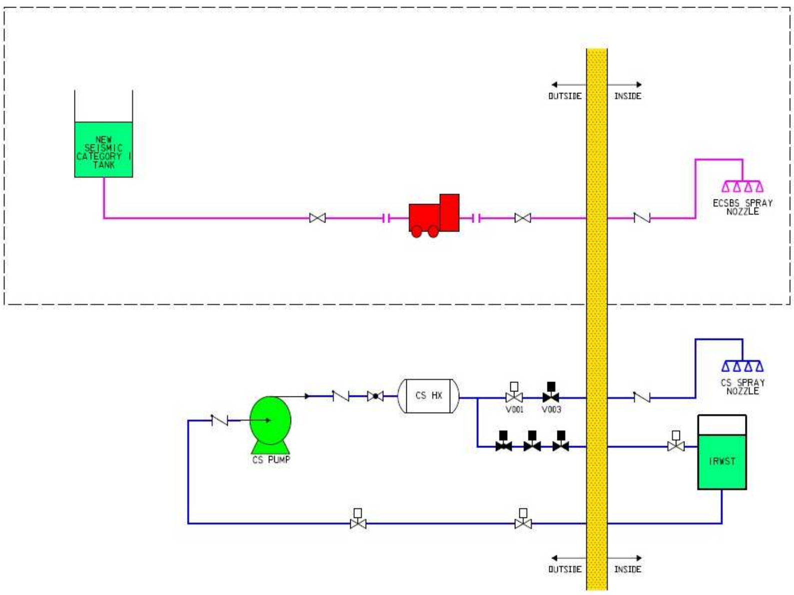

FIG. 1. Schematics of ECSBS configuration.

2.2. Containment filtered venting system

The basic idea of filtered venting system is to provide a controlled flow path to the external environment to relieve the steam and non-condensable gases that are generated inside the containment with a passive manner. By doing this, it is possible to prevent structural failure of the containment. Also it provides additional time to mitigate the accident and reduces the off-site consequences compared to those produced by containment failure.

Typically, efficiency of external filter on aerosol is required above 99.9 % to prevent long term ground contamination and 99 % for elemental iodine to limit thyroid doses and short term evacuation.

Different kinds of filtered venting design were investigated in 1980s. After the Chernobyl accident, several European countries (France, Sweden and Germany) installed Containment Filtered Venting System (CFVS). After Fukushima Daiichi nuclear power plant accident, many countries re-examined the applicability of filtered venting as a mitigation measures for containment integrity [4].

3. SEVERE ACCIDENT ANALYSES

Analyses on containment over-pressurization scenarios are performed using MAAP4 computer code [5]. The MAAP code can analyse severe accidents following Loss of Coolant Accidents (LOCAs), Station Black Out (SBO) and various general transients. Developed for the Electric Power Research Institute (EPRI) by Fauske et al., MAAP4.0.4 is used here. The reference plant is OPR1000, 1000 MW(e) PWR in the Republic of Korea. ECSBS is modeled as a spray component and injection flow path is connected to the containment dome. Injection flow rate is assumed based on existing commercial portable pump performance data. To activate the ECSBS, portable equipment and water source should be aligned prior to the containment penetration valve opening by operator.

Also, CFVS is modeled as a junction connecting containment upper compartment to the environment. Venting area is assumed to 23 cm (9 inch) diameter and flow resistance through CFVS is considered. To activate the CFVS, the containment isolation valves should be opened either by operator as an active means or rupture disc as a passive means when set-point reached. Decontamination factor of 1000 for aerosol is assumed in this calculation. Initiating event is extended SBO. Initial and boundary conditions are summarized in Table 2.

TABLE 2. INITIAL AND BOUNDARY CONDITIONS

Parameter	Value	Remark
Rx power	2815 MW(th)	
CTMT net free volume	7.73×10^4 m^3	Large Dry Containment
CTMT design pressure	0.5 MPa	ASME Level II
CTMT ultimate pressure	1.01 MPa	ASME Level III
ECSBS flowrate	750 gpm	Analysis Assumption
CFVS opening/closing set-point	0.9 / 0.6 MPa	
Other Safety System	Containment Spray Fan Cooler	Fail to Start

3.1. SBO scenarios with ECSBS

When an SBO occurs, all the active systems become inoperable. The RCS inventory is discharged into the containment through pressurizer safety valve cycling and core start to uncover at 1.6 Hrs. Eventually the RCS fails at 3.3 Hrs. Then, a large amount of superheated steam and hydrogen is discharged into containment. Also interaction between molten core material and cavity water that is discharged from accumulator lead large amount of steam generation in the cavity. So, the containment pressure rapidly increased. After cavity water is depleted, hydrogen and non-condensable gas are steadily generated by molten core concrete interaction in the base-mat.

When ECSBS actuates at 18 Hrs, containment pressure initially decreases. Then, pressure instantaneously increases by 0.2 MPa due to the steam spike, interaction between molten corium and ECSBS water collected in reactor cavity. After molten corium quenching, containment pressure decrease by condensation and the containment integrity is maintained as shown in Fig. 2. However, continuous condensation reduces only partial pressure of steam. So, the relative hydrogen concentration increases. To prevent hydrogen flammability, additional hydrogen mitigation facility is needed. Also, water inventory buildup within the containment lead additional risk such as, hydrostatic load on the containment structure and un-availability of instrumentation and control system in floor level as shown in Fig. 3. So, the ECSBS is required to be used intermittently to prevent adverse effects.

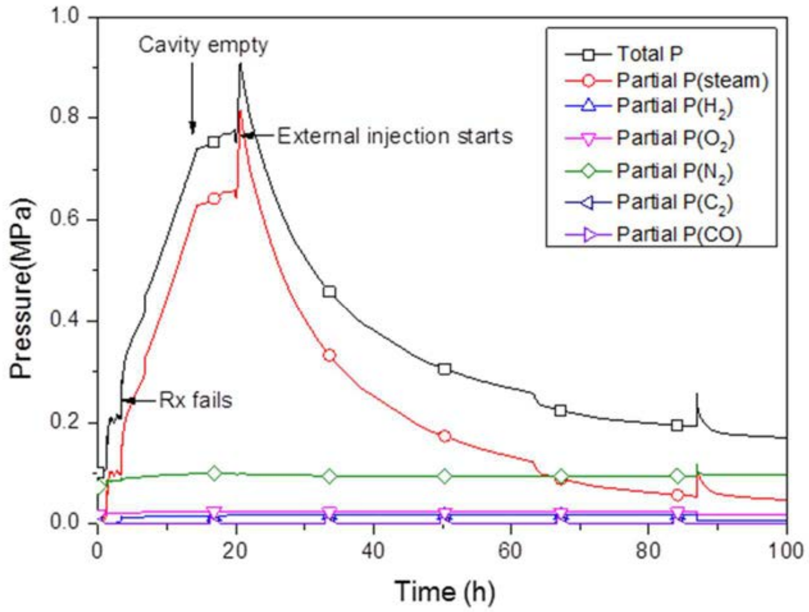

FIG. 2. Containment pressure time history in ECSBS case.

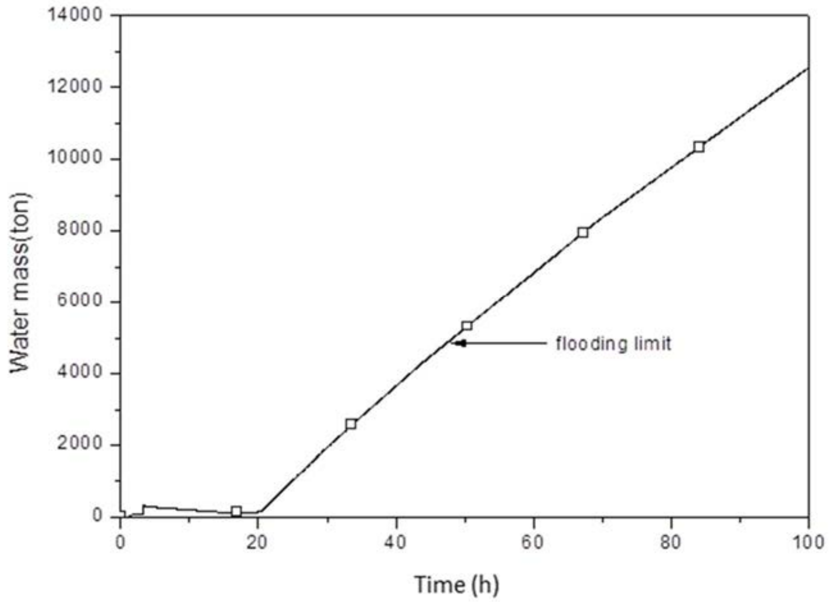

FIG. 3. Integrated containment water mass time history in ECSBS case.

3.2. SBO scenarios with containment filtered venting system

The containment behavior before CFVS actuation is identical to that of ECSBS case. The CFVS start to operate at 36.8 hours when containment pressure reaches 0.9 MPa as shown in Fig. 4. Containment pressure gradually decreases to closing set-point of 0.6 MPa within 2.2 Hrs. Discharge flow is homogeneous mixture of steam and non-condensable gas. So, no additional negative effect on hydrogen risk inside containment exists during CFVS operation. However, no coolability function for molten corium is provided. So, MCCI phenomena could not be mitigated.

Fractional releases of radioactive material are compared in Fig. 5. Representative radioactive material (CsI) release to the environment is reduced by 10^{-5} due to the effect of de-contamination factor of 1000 and the effect of prevention of large release of radioactive material by ultimate containment failure by 10-2. In addition, cyclic venting strategy reduces the noble gas release by 10-2 due to the limited operational period of filtered venting even though CFVS cannot filter the noble gas.

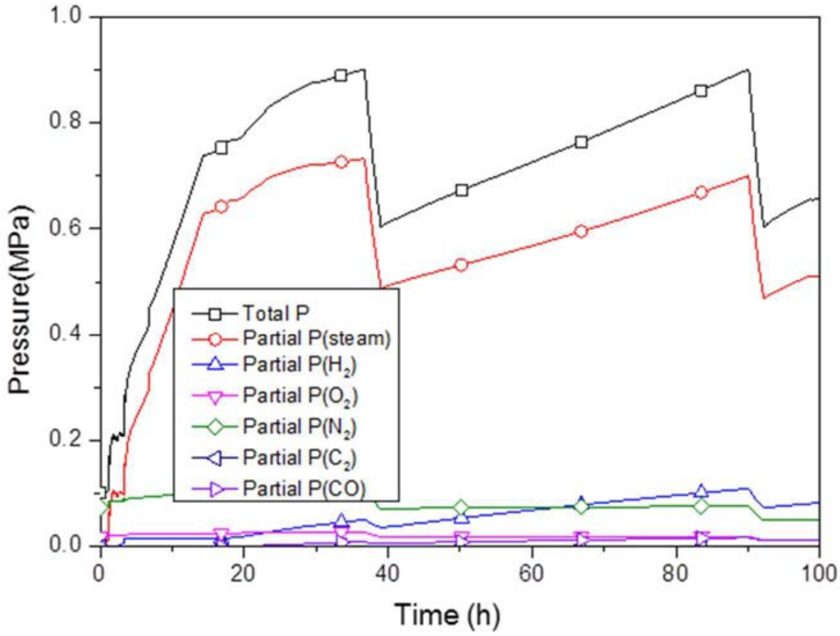

FIG. 4. Containment pressure time history in CFVS case.

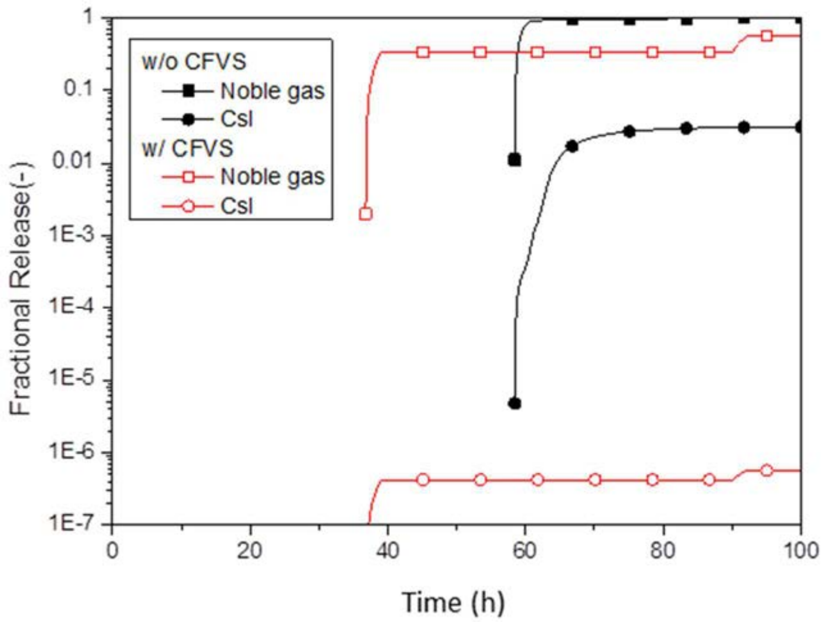

FIG. 5. Fractional release of noble gas and CsI time history in CFVS case.

4. CONCLUSIONS

The applicability and effectiveness of containment backup spray system and containment filtered venting system are examined on PWR with large dry containment. Thermo-hydraulic analysis results show that both options can effectively de-pressurize the containment against representative containment over-pressurization scenarios. However, each option has its own advantages and disadvantages. The characteristics of each option are summarized in Table 3.

The final design option should be chosen considering the overall safety effect based on thermo-hydraulic result, risk reduction factor, incorporation into design concept (diversity .vs redundancy), long term operability, etc.

TABLE 3. COMPARISON OF CONTAINMENT DEPRESSURIZATION COUNTERMEASURES

Parameter	ECSBS	CFVS
Design Concept	Redundancy	Diversity
Operating Mechanism	Active	Passive
Radiological Release	Prevention (No release before rupture)	Mitigation (Limited release 10^{-3} ~10^{-5})
Adverse Effect	Hydrogen flammability Hydrostatic Load I&C system availability	Radiological release No direct corium cooling function
Pressure behavior	Initial rise (steam spike) and decrease	Decrease/Increase (Strategy dependent)
Depressurization Mechanism	Condensation	Mass transfer
Limitation on Operability	Containment water level	Aerosol clogging
Key design factor	Uniform spray	DF > 1000
Sub-component	Pump Water source Electric Power	Replenish system (water, buffer agent)

ACKNOWLEDGEMENTS

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AN INVESTIGATION OF POOL STRATIFICATION EFFECTS ON BWR WETWELL EXTERNAL VENTING USING GOTHIC

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Abstract. Strategies for coping with severe accidents and long term loss of AC power at BWR nuclear plants may include the use of wetwell and drywell vents to the atmosphere. In some forensic analyses for the Fukushima events, stratification of the wetwell has been postulated as a contributing factor to the higher than expected pressurization rate of the containment. The GOTHICh code has been used to investigate the containment response during the Fukushima events, including the possibility of suppression pool stratification [1 – 5]. Thermal stratification, if it develops, would diminish the heat absorbing capacity of the pool, leading to earlier and more frequent vent valve cycling and earlier steam bypass through the saturated pool. This paper describes the use of the GOTHIC computer code to investigate the containment response during a long term loss of AC power with periodic wetwell venting, including the possibility of suppression pool stratification. The applicability of GOTHIC to pool stratification is discussed and modeling strategies are described for dealing with the long transients that must be considered for vent performance. GOTHIC results are presented for one selected loss of AC power scenario with and without stratification effects included.

1. INTRODUCTION

In the event of a long term loss of AC power (ELAP) at a BWR Mark I or Mark II plant, with or without core damage and possible primary system breach, hardened vents connecting the Wetwell (WW) to the atmosphere can be utilized to prevent over pressurization of the containment. The performance of the containment (Drywell (DW) and wetwell), during such an event will influence venting system design requirements, including valve and DC power requirements to accommodate the number and duration of valve cycles for venting, water accumulation in the vent lines, and vent capacity.

During the postulated event, mass and energy may be transferred to the WW via the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) turbine pump exhaust lines, Safety Relief Valve (SRV) discharge lines, DW to WW vent system and direct leakage from the DW to the WW. The pressurization of the containment is largely controlled by amount of energy that can be retained in the WW pool and the temperature of the WW gas space. The temperature of the gas space is controlled by the pool surface temperature and the heat transfer through the walls of the DW to WW vent system. If the energy is added to the WW in a way that promotes or allows thermal stratification of the pool, the amount of energy retained in the WW will be less than if the pool is well mixed. This could result in earlier and more frequent cycling of the hardened vent valves and possibly earlier steam bypass due to saturation of the pool water above the steam release point.

The GOTHIC computer program has been used to simulate stratification and mixing in pools due to steam release at various locations [4]. It has also been used to investigate the effects of heat transfer through the walls of the DW to WW vent system as well as the potential for steam bypass of the pools as the local pool temperature approaches the saturation temperatures. Much of this analysis was done as part of the post Fukushima investigative studies carried out by EPRI [1–3].

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[1] GOTHIC incorporates technology developed for the electric power industry under the sponsorship of EPRI.

GOTHIC is a general purpose thermal hydraulics code that is used extensively in nuclear power industry [6]. It bridges the gap between the lumped parameter codes frequently used for containment analysis (e.g. MAAP, COCOSYS, ASTEC and MELCOR), and CFD codes. While GOTHIC lacks the capability to model the “in-vessel” degraded core and vessel breach phenomena available in MAAP or MELCOR, it can utilize the output from such codes as input for more detailed containment response modeling to provide greater insight into the subtle phenomena that may significantly impact ELAP-type events. Within a single model, GOTHIC can include regions treated in conventional lumped parameter mode and regions with three-dimensional flows in complex geometries. Although it does not include the capability to model the details of the boundary layers as in most CFD codes, through the use of standard wall functions for heat and momentum transfer, it can give good estimates of the 3 dimensional flows and distributions with significantly less computational time than typical CFD codes. Additionally, it includes the capability to model multiphase flows situations including drop wise and wall condensation, pool surfaces and sprays without special user supplied models.

2. SUPPRESSION POOL STRATIFICATION – GOTHIC VALIDATION

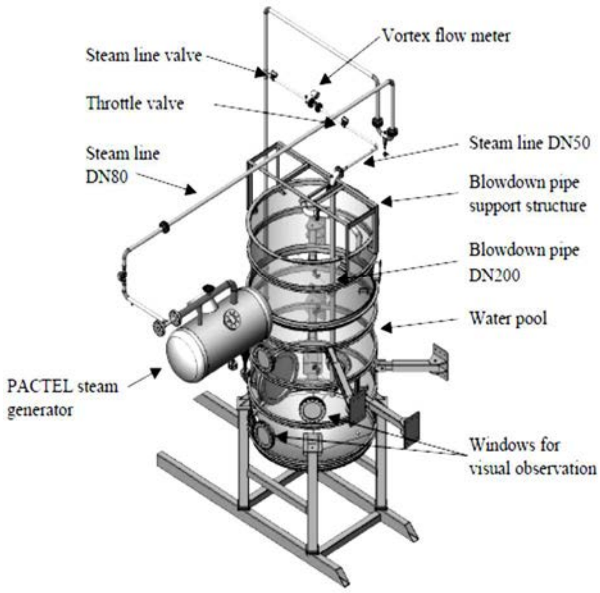
The possibility of Suppression Pool (SP) stratification under some circumstances is well established [7–10]. GOTHIC’s capability to model pool stratification has been demonstrated with various benchmark efforts including the following benchmarks specific to SP applications.

2.1. POOLEX test

The POOLEX (POOL Experiment) and PPOOLEX (Pressurized Pool Experiments) were performed to investigate the behavior of the pressure SP during a possible steam line break accident in Olkiluoto type BWRs [7] with geometry similar to GE Mark II containments. The POOLEX test facility has been extensively used for benchmarking studies on condensation, stratification and mixing phenomena in a pool of water [8, 9] The POOLEX test facility (Figure 1(a)) consists of a cylindrical stainless steel tank; 5 m in height with 2.4 m in diameter, open to the lab atmosphere with a water pool that is 2.95 m deep. The vertical discharge pipe is 200 mm in diameter and submerged by 1.81 m into the pool. The pool temperature transient was measured at evenly spaced elevations. There was no change in the pool temperature at the five measurement locations below the injection level. At the end of the 4 hour transient, the data shows a large change in the vertical temperature profile just above the injection level and then a more gradual temperature rise to the maximum pool temperature at the top of the pool (Figure 1(b)).

The measured and GOTHIC calculated vertical temperature profiles at the end of the 4 hour POOLEX STB20 transient are shown in Figure 1(b). The dashed line indicates the level of the bottom of the injection pipe. Except for some minor departure from the data in the strong stratification zone between 1 and 1.5 meters, the GOTHIC results agree well with the data. Results are shown for two vertical grid spacings. Mesh I used a nominal 10 cm vertical spacing. For Mesh II, the vertical space was reduced to 5 cm in the stratification zone resulting in a small improvement in the comparison with the data.

The GOTHIC model for the test facility is for axisymmetric geometry. However, in the test the discharge pipe is off center by approximately 410 mm. There could be additional circulation in the horizontal plane that cannot be captured in the 2-dimension model. Nevertheless, the overall top to bottom stratification is well predicted.



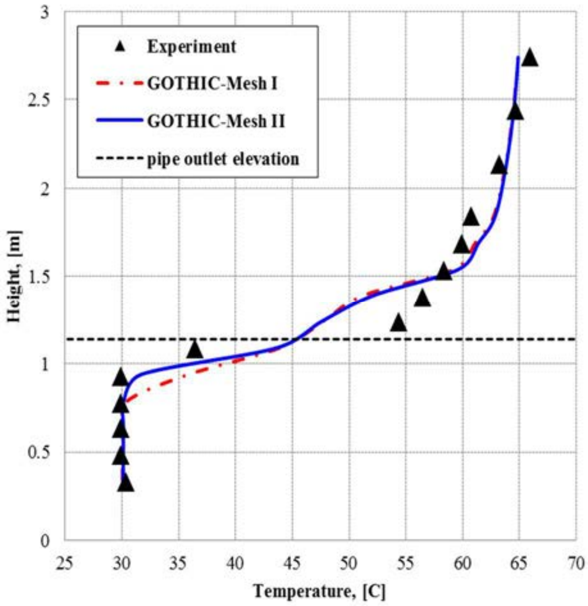


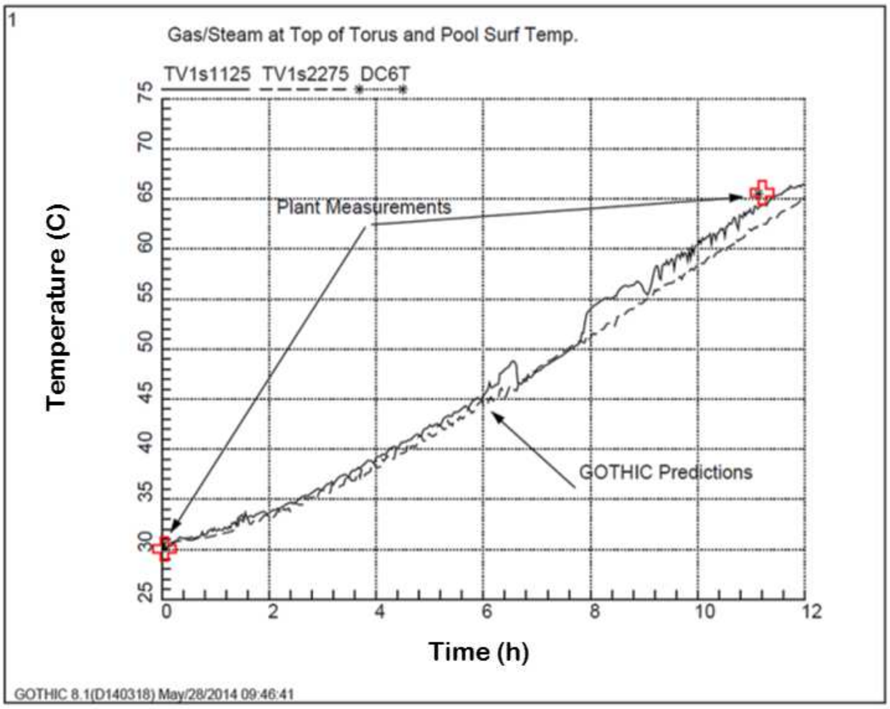
FIG. 1. (a) POOLEX test vessel [7] *(b) GOTHIC results and test data.*

2.2. Browns Ferry nuclear plant test

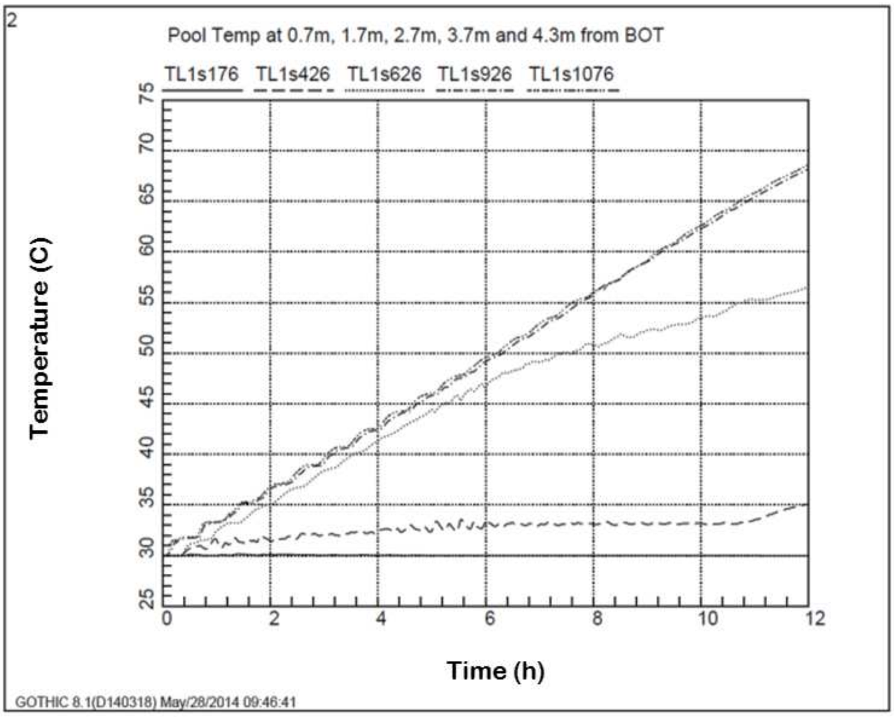
Stratification in a full scale Mark I containment was observed in a 1991 test for primary system pressure control by RCIC at the Brown's Ferry Nuclear Plant [10]. The RCIC system was operated continuously for approximately 11 hours. The outboard Main Steam Isolation Valves were closed and the Reactor Pressure Vessel pressure was about 5840 kPa. During the test, neither the pool temperature indicator nor the gas space temperature monitor showed the expected increase in temperature. However, it was necessary to vent the DW periodically by manual control during the test to maintain the containment pressure below the test specification (118 kPa). When the Residual Heat Removal (RHR) pump was started at the end of the test, the suppression pool temperature indicator rose rapidly from 30.6 °C to 48.9 C. Also, when the SP air temperature recorder was tapped, the indicated temperature jumped from 34.5 to 65.6 °C. The rise in the mixed pool temperature is consistent with the amount of energy deposited in the pool by the RCIC. Since the pool temperature indicator was located low in the pool, these results strongly suggest that the pool was thermally stratified during the tests.

The GOTHIC calculated WW gas space and pool temperature predictions were compared with the plant measurements from the Brown's Ferry test. Figure 2 (a) shows the predicted gas space temperature just above the pool surface (solid line) and at the top of the torus (dashed line) along with the single end of test measurement. The results indicate that the gas space is fairly well mixed and the calculated temperature is in close agreement with the measured temperature.

Figure 2 (b) shows the pool temperature at 0.7, 1.7, 2.7, 3.7 and 4.3 m from the bottom of the torus. The RCIC sparger was modeled and GOTHIC predicted that the steam was release from only the upper part of sparger (computation cells at 3.7 m and 3.5 m above from the pool bottom). During the early part of the transient, the heated region of the pool extends down to the 2.7 m elevation. But as the pool heated up and the buoyancy of the source decreased the heating at the 2.7 m level decreased. At the end of the test the GOTHIC predicted pool surface temperature was about 66 °C, about 4 °C hotter than the predicted gas space temperature. These predictions are in line with the indicated gas space temperature at the end of the test. The predicted liquid temperatures at the lower elevations of the pool prior to the RHR pump mixing agree with the measured temperature at the 2 m elevation (30.6 °C).



(a)

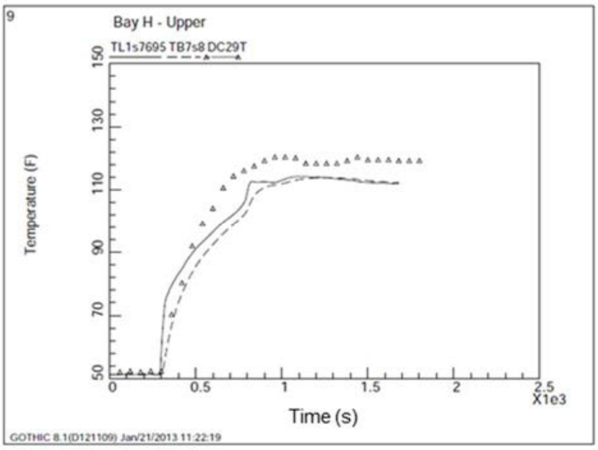


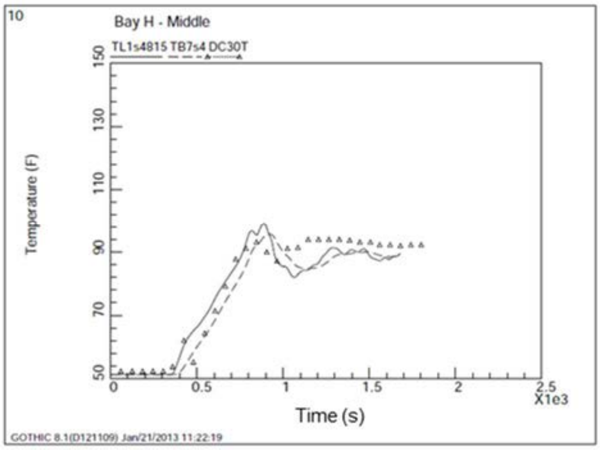
(b)

FIG. 2. GOTHIC Browns Ferry benchmark results; WW gas temperature (a), WW pool temperature (b).

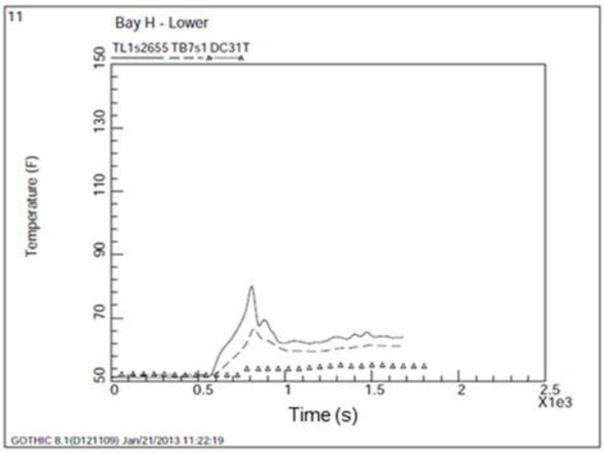
2.3. Monticello nuclear plant SRV discharge SP heat-up test

Tests were conducted at the Monticello nuclear plant in 1978 to evaluate the performance of the T-quenchers under extended SRV operation [11]. Two tests were conducted. In the first test steam was released to the Mark I pressure suppression pool from a single SRV. The transient temperature response at various locations in the pool was recorded. The first test had a continuous steam release for about 11 minutes. The test results indicate that the heated water around the discharging T-quencher rose to the top of the surface and spread across the upper region of the pool resulting in significant thermal stratification in the upper part of the pool. However, the thermal plume spreading across the pool surface and around the circumference of the torus just reached the opposite side of the pool when the test was terminated. It is suspected that a longer test would tend to gradually bring the stratification zone down to the level of the SRV T-Quencher near the bottom of the torus. Nevertheless, the test gives valuable data for the transient development of the temperature distribution in a pool with SRV steam release.





(a) *(b)*



(c)

FIG. 3. GOTHIC Monticello benchmark results; upper (a); middle (b) and lower pool elevation (c).

The GOTHIC predicted temperature transient in torus sector H (~180° from the injection SRV) for the Monticello test are shown in Figure 3. GOTHIC predicts slightly less top to bottom temperature difference but the transient response variation in the prediction versus the data are within the uncertainty in the test configuration and thermal couple placement. The results for sector H are representative of the code versus data at other locations around the torus.

3. GOTHIC ANALYTICAL MODEL

With the ability of GOTHIC to predict pool mixing and stratification established, the code can be used to investigate possible stratification and its ramifications on the containment response with wetwell venting.

The transients for extended loss of power may extend over several days and the 3 dimensional models used for the validation described above run at several times real time. For parametric studies a faster solution is desired. As part of the post Fukushima investigations [4], a pool mixing correlation was developed that can be used to simulate the effects of pool stratification in a lumped parameter model. The model noding is shown Figure 4.

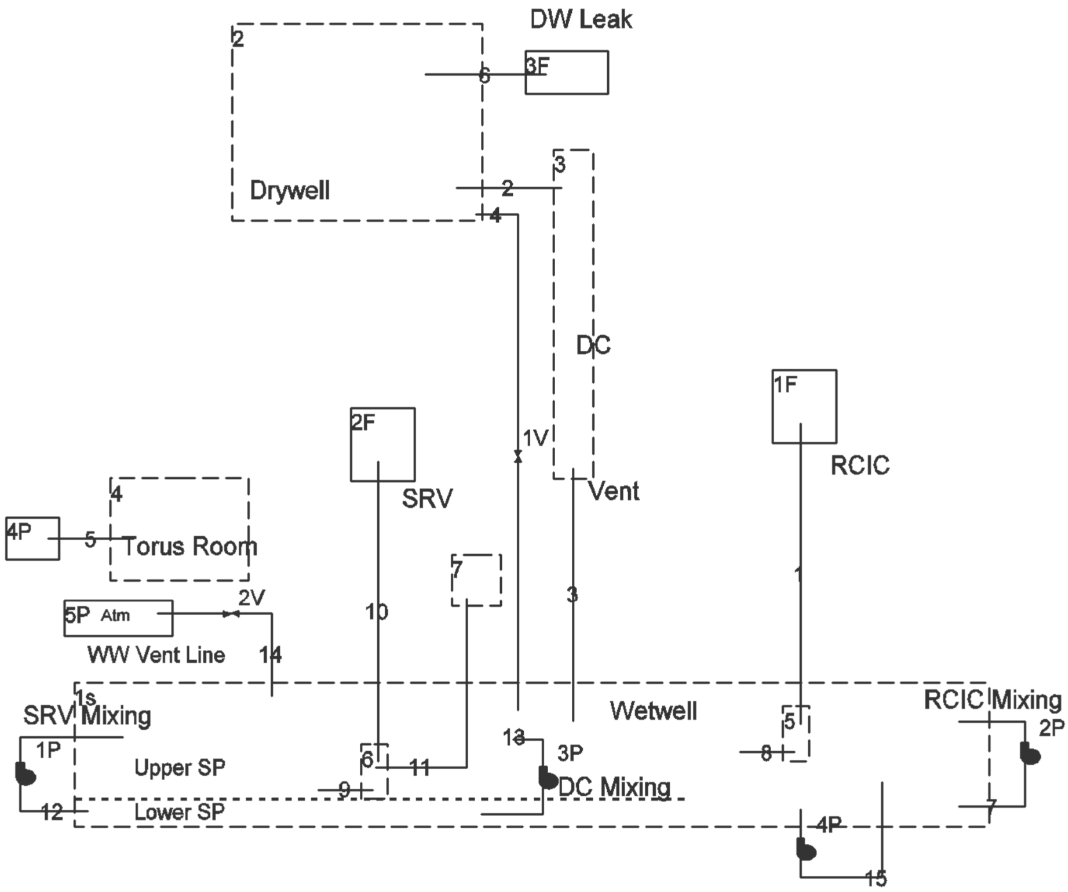


FIG. 4. Lumped models with pool stratification effects.

In this model the pressure suppression chamber is split into two parts (SC Pool and SC Gas Space in the noding diagram). The SC Gas Space volume includes the gas space and the upper 20 % of the pool. The model is designed to allow user specification of various sources to the wetwell. Regardless of the actual location of the source in the actual pool, in the lumped model all of the pool sources are added to the water in the SC Gas Space volume. This approach gives maximum stratification effects. To account for actual stratification, a mixing pump is used to circulate the water between the SC Pool volume and the SC Gas Space volume. The mixing rate is given by a correlation depends on the steam and gas injection rate, submergence of the injection point and the difference between the bulk pool temperature and the saturated water temperature at the WW pressure. Since there is limited experimental data available to cover all of the possible injection possibilities, a pseudo experimental data base was developed using a 3 dimensional GOTHIC model similar to that used for the Brown's Ferry test. The simulated data base included cases at various steam flow rates through the RCIC sparger and from a single SRV T-quencher. For a RCIC or HPCI sparger, the injection depth depends on the steam injection rate and the WW pressure. An analytic formula for the injection depth was developed for the necessary submergence parameter in the mixing formula. The mixing formula was extended to include the effects of non-condensing gas injection with or without steam.

The mixing correlation can be used for steam injection from a RCIC/HPCI turbine exhaust sparger, SRV tail pipe or DW to WW vent. In a long term pressurization event, discharge to the pressure SP may include any combination of these sources. For application of the mixing model with multiple, simultaneous sources, it is assumed that the mixing from each source is independent and that the total mixing is simply the sum of the mixing rates from the individual sources.

4. CONTAINMENT RESPONSE FOR EXTENDED LOSS OF POWER WITH VENTING

To demonstrate the potential impact of SP stratification during containment venting scenarios, the above simplified pool mixing model was use to predict pool performance during a simplified scenario representative of extended ELAP-type events where active pool mixing via RHR or other means is not available. The GOTHIC model was used to predict the containment response during a simulated cyclic venting operation to demonstrate the potential impact of SP stratification during venting operation. There are obviously a great number of potentially complicated scenarios that may be encountered that could lead to containment venting. For the purpose of investigating the potential impact of SP stratification, a highly simplified scenario has been constructed specifically selected to promote stratification for demonstration purposes. Based upon experimental test data and GOTHIC predictions, it is expected that scenarios with significant SRV steam discharge via the quenchers near the bottom of the SP will have a fairly well mixed pool. Therefore, for the purpose of this demonstration, the analysis simulates some period during the event where selective failures or operating conditions lead to minimal SRV flow and the majority of the steam from the reactor being discharged to the upper portion of the pool. This may be the result of some combination of RCIC and HPCI turbine steam exhaust spargers and the venting of Reactor Coolant System (RCS) liquid or steam leakage to the DW through the downcomers of the DW vent system.

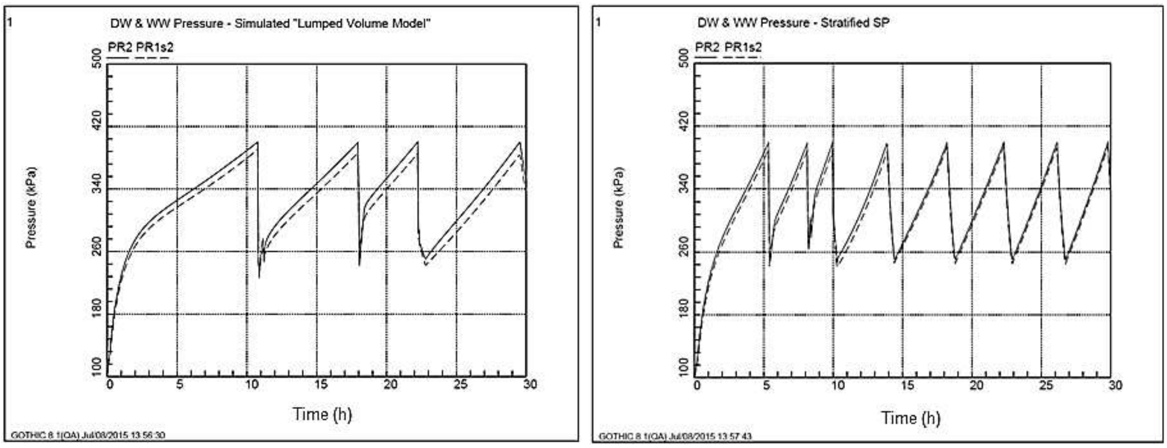
The GOTHIC mixing correlation [4] used to simulate SP stratification in the model can also be adjusted to force complete mixing of the SP and thereby simulate the results that would be expected from a typical single volume suppression chamber model with no capability for modeling pool stratification. This investigation utilized the above model to simulate both a stratified and fully mixed SP to demonstrate the potential impact of stratification during venting operations and the potential limitations of "Lumped Volume models" in this application.

The scenario was constructed with a number of simplifying assumptions such as elevated initial SP temperatures and constant decay heat and steam flow to the SP to approximate possible conditions during venting operations that may lead to stratification. There is no attempt to accurately simulate the numerous variables that would accompany an actual event. The scenario was intended solely to provide a basis for this comparison of the stratified versus lumped volume modeling during a simulated venting operation and is not intended to be representative of the progression or timing of venting during a realistic scenario.

5. RESULTS & CONCLUSIONS

The plots of the DW and WW pressure response for the simulated venting scenario demonstrating the potential impact of SP stratification are shown in Figure 5.

As demonstrated by the comparison in this scoping study, a potentially stratified SP could result in achievement of the initial vent pressure earlier in the event and require more frequent vent cycling operation (if a vent cycling strategy is utilized) than may be predicted by a lumped parameter code with a will mixed SP. As previously noted, these plots are for comparison of effect of stratification only and not intended to be representative of the timing and progression of an actual scenario. If the timing for initiation of venting or the frequency of cyclic venting is of importance to the event coping strategy the potential for SP stratification should be considered.



(a) *(b)*

FIG. 5. Vent cycling comparison; fully mixed (a) and stratified (b) SP.

This study demonstrates that stratification may have a significant impact on the containment performance during venting. The correlation used in the mixing model was developed using conditions with a single source to the pool and where the pool was essentially subcooled. It has not been fully validated under conditions where the pool is at or near saturation temperature or with simultaneous multiple sources as may occur during extended venting scenarios. Additional investigation is warranted to validate and extend the mixing correlation for typical venting scenarios using experimental data and/or 3D GOTHIC simulations. Based on GOTHIC's demonstrated ability to predict multidimensional pool performance, 3D GOTHIC modeling can also be used to directly evaluate venting scenarios with potential for stratification.

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SESSION III

STRATEGIES TO ENSURE CONTAINMENT INTEGRITY FOR EXISTING PLANTS DURING SEVERE ACCIDENTS

EXPERIENCE WITH CONTAINMENT VENTING AT MALAYSIAN RESEARCH REACTOR

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Abstract. This paper briefly describes the Malaysian 1-MW PUSPATI TRIGA Reactor, its building and the active ventilation system required to provide partial containment for safe operation of the reactor. It also presents some results of the Safety Performance Indicators related to the radioactive effluent due to operation of the reactor.

1. INTRODUCTION

Malaysia initiated the nuclear research reactor project as a step towards development of nuclear power in the country. The presence of the research reactor has succeeded in enhancing understanding the various issues related to nuclear power. It also identified the gaps in nuclear engineering in addition to highlighting the training needs for human resources in all fields relevant for nuclear power operations.

2. THE MALAYSIAN PUSPATI TRIGA REACTOR

The Malaysian PUSPATI TRIGA Reactor (RTP) was constructed in 1981 and achieved first criticality on 28 June 1982. The maximum reactor power is 1.0 MW. Reactor PUSPATI TRIGA (RTP) is an open pool type reactor. Reactor coolant system consists of a tank, water cooling system and water purification system. The water cooling system consists of primary cooling system and secondary cooling system. The primary cooling system equipped with pumps, heat exchanger, probes, a nitrogen-16 diffuser and associated valve is connected to the tank by aluminium pipes. The 6.5m high with 2m diameter aluminium tank is filled by demineralised light water (H2O), serves as coolant, moderator as well as shielding.

Figure 1 shows the cutaway view of the RTP.

3. RTP BUILDING AND CONTAINMENT

3.1 RTP building description

Reactor building is a four-storey reinforced concrete structure housing the reactor in the reactor hall. It also contains the supporting rooms and facilities namely control room, plant room, basement, electronic room, counting room, stores, staff rooms, etc.

The reactor floor slab is 45.7 cm thick, designed to take a maximum load of 3 tonne/m^2. The floor is furnished welded continuous PVC tiles. A trench 91.4 × 38.1 cm, is provided on the floor running from reactor shielding block to the reactor basement. This is normally used to run the tubes, pipes, and conduits for reactor coolant system, transmitting pipe, argon vent tubes, service water pipes etc. Other smaller trenches are provided to run cable conduits to reactor shielding wall faces for experimental purposes. The trench also acts as emergency liquid effluent channel from the reactor hall floor to a pit in the basement.

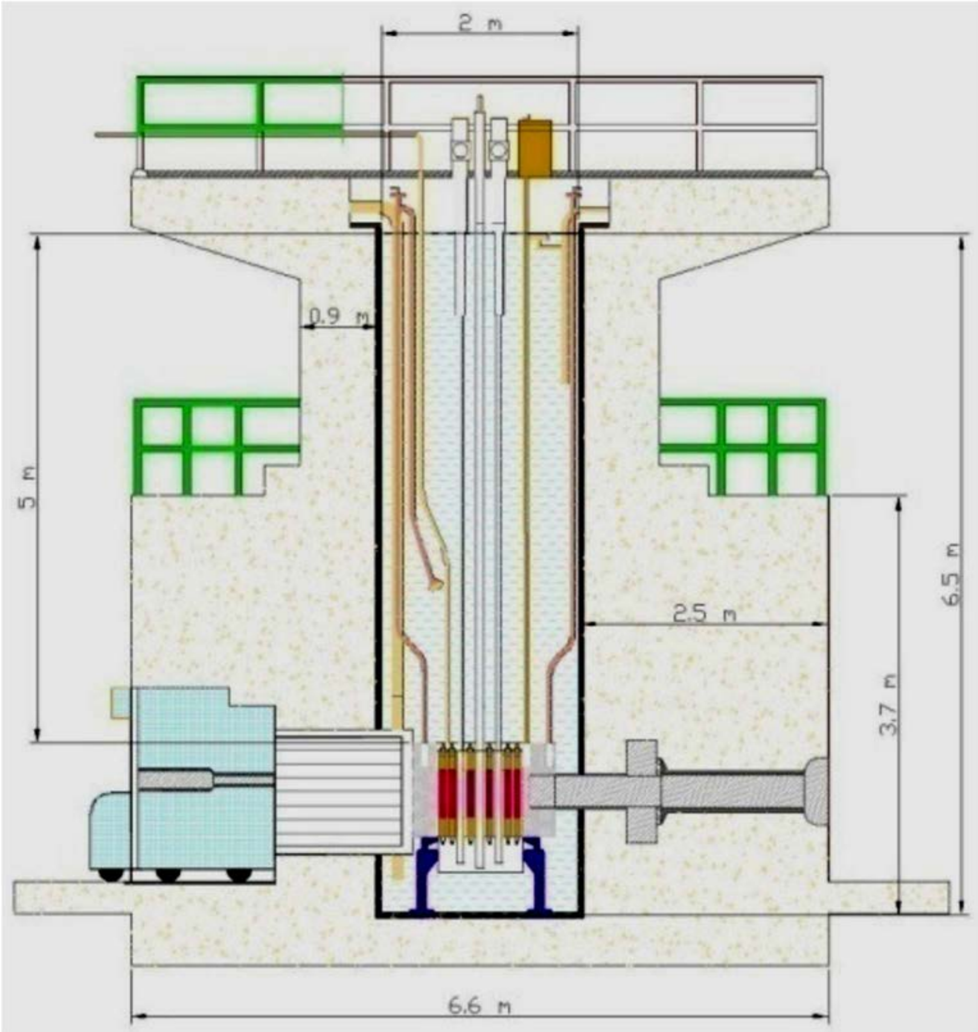

FIG. 1. Cutaway view of RTP.

Reactor hall walls are made of 22 cm thick reinforced concrete throughout the entire height except between the equally spaced columns or 'fins'. In between these columns or fins, the concrete walls extend up to 3 m, and then the rest of the heights are fixed with 91 cm wide toughened glass panels with steel mesh. The concrete walls are painted with epoxy-based paint.

The reactor roof consists of reinforced concrete with an additional galvanised steel decking laid to fall. On the roof, 3.66 m exhaust stack with a stack gas monitor is installed.

In the design, the maximum column loading for the Reactor Block is 390 tonnes. The Reactor block was built on shallow pile cap foundation of allowable bearing pressure of 2 tonnes per square foot at nominal 1.5 m depth from the formation level. The TRIGA Mark II shielding structure is capable of withstanding seismic acceleration level of at least 0.6 g horizontal and 0.3 g vertical which is exceeding IX Mercalli scale.

3.2 RTP ventilation system

The design principle of the air-conditioning and ventilation system of the reactor building is to control the potentially contaminated air to flow through the designated flow path before being safely discharged to the environment. Besides this, conditioned air is also needed for comfort and for protection of equipment against severe heat and humidity. The systems basically comprise of the supply (i.e. conditioned air) to reactor hall and the extract systems.

The main supply to the reactor hall is at 42800 m^3/h or approximately 5.4 air changes/hr in order to maintain the desired room condition of 24 ± 1 ºC and 50 ± 5 % R.H. Supply air is 44% recirculated and the remainder is fresh charge, which is in turn being pre-cooled in the energy wheel by the extracted air. Discharge of the supply air into the reactor hall is through the ceiling mounted grilles. Two air handling units, namely AHU 1 and 2, each of 21400 m^3/h capacity installed to meet this requirement.

Other rooms in the reactor building such as control room, counting room, electronic room etc., air supplied by a separate air handling unit, namely AHU 3 with control room, computer and electronic rooms having fan coil-package unit as standby system. Sampling Room is also provided with a separate ventilation system. Ventilation for Counting Room is provided with two window units for back up purposes.

The main reactor hall extract system is designed to maintain a negative pressure of -6mm H_2O (-¼" H_2O) which is needed to ensure any infiltration is always inwards. Extraction rate varies from 42800 m^3/h to 51300 m^3/h to allow for pressure differential control. Since Argon is heavier than air, the air is exhausted from the reactor hall through four extraction openings situated at the four corners near the floor level of the reactor hall.

The negativity is controlled and maintained by the use of variable flow axial fans whose regulation is annunciate by the pneumatically-controlled butterfly damper installed immediately downstream to the fan. The damper in turn receives the signal from the pressure differential transducer fixed in the reactor hall. Two variable flow axial fans installed for this purpose, one shall be running during normal reactor operation and the other as standby.

Isolation dampers are installed in both supply and extract line. They are pneumatically controlled and the manual override switches are in the reactor control room. The extracted air is discharged through the stack on the roof of the reactor.

The emergency and argon ventilation systems are separated from the main extract system, having total extraction rate of 420 m^3/h working during normal operation. The argon ventilation system extracts air in the thermal column beamports, and reactor pool top at a total rate of 17 m^3/h. The emergency extract points are at the ground floor level and the other at the ceiling level, each having extraction rate of 200 m^3/h rated during normal operation. The air from this system is filtered by Pre-filters (85 % ASHRAE, Gravimetric) and HEPA (99.97 %, 0.3 μm DOP smoke) prior to joining the reactor stack.

Two centrifugal fans, 420 m^3/h capacity each, installed with either one of them running at any one time. Both shall be running on receiving the signal indicating emergency situation.

Fire dampers are installed before the filters to protect them in case of fire. Pneumatically control isolation valves are installed at the fan upstream. It is normally shutdown or close valve for building isolation during emergency.

The other potentially contaminated room i.e. the pneumatic room is kept at negative pressure of 3 mm H_2O (1/8" H_2O). Counting room is slightly positive (+2 mm H_2O) to exclude the ingress of possible contaminated air. Other rooms are kept at atmospheric pressure.

Extracted air from pneumatic room (1220 m^3/h) is filtered with separate pre-filters and HEPA filters (85 % ASHRAE gravimetric and 99.9 % 0.3 μm DOP smoke respectively) prior to discharge through the reactor stack. Two centrifugal fans, each of 1,220 m^3/hr capacity installed for system back-up.

The plant room, and the basement are kept ventilated each having extract rate of 4,200 m^3/h with two variable flow axial fan provided. All the exhaust air mentioned above join the reactor stack and discharge at approximately 20 m/s velocity.

The stack monitor is installed to monitor the activity level of the discharge air and sounds the alarm if a present level of activity is exceeded. In this case, the fans shall be 'off' and the pneumatically operated isolated valve for both the exhaust and the supply system are closed by manually operated isolated valve for both the exhaust and the supply system are closed by manually operated switches in the control room - thus isolating the reactor hall from the environment. In emergency case the argon and emergency system can be used to slowly purge all reactor room air through the filters and out the discharge plenum.

Preventive maintenance of the ventilation system is carried out every month to ensure reliability and availability. This is managed by the Engineering Division.

Figure 2 shows a schematic of the RTP Active Ventilation System.

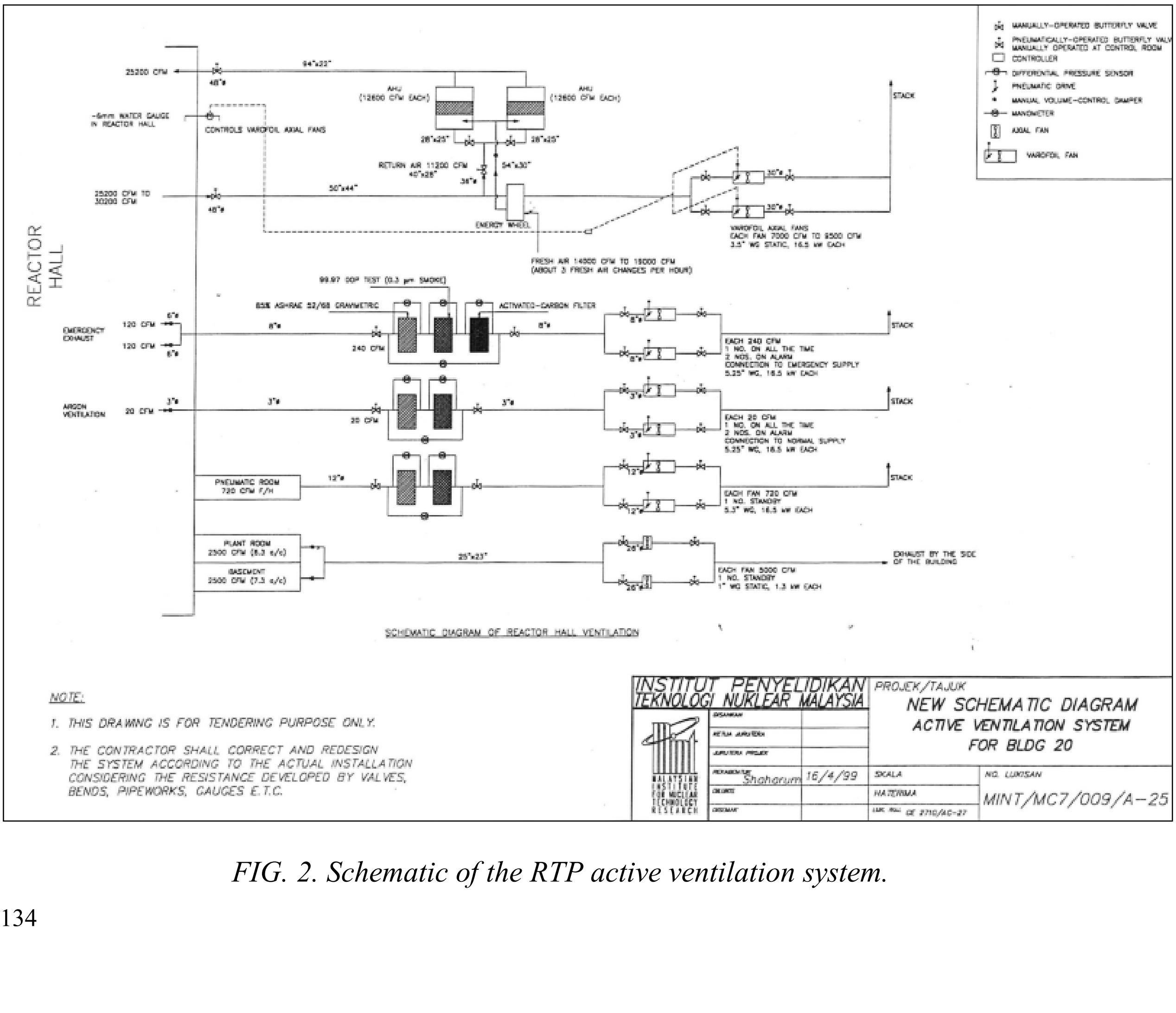

FIG. 2. *Schematic of the RTP active ventilation system.*

4. SPI RESULTS

The results of the Safety Performance Indicators related to the RTP radioactive effluent release are tabulated in Table 1.

TABLE 1. SPI – RTP Radioactive Release (2009–2014)

	2009	2010	2011	2012	2013	2014
Radioactive Release:						
Activity of Noble Gas Release (GBq)	1082	2170	1403	1854	866	1653
Activity of Iodine Released to Atmosphere (MBq)	62	538	21	70	30.0	24.8
Liquid Radioactive Waste Released (m^2)	158*	8.91	3.9	12.16	24.8	20.3

5. ISSUES AND CHALLENGES

Among the issues and challenges are:-

- Confinement, not Containment: RTP is only equipped to confine any radioactive effluent within the building structure;
- Safety assessment of modification to reactor hall confinement.

6. CONCLUSION

In conclusion, the current RTP confinement is adequate to provide sufficient and effective control of effluents during normal operational conditions. Further assessment of expanded confinement boundary is required.

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STRATEGIES TO ENSURE CONTAINMENT INTEGRITY FOR ABWRS IN KASHIWAZAKI-KARIWA NUCLEAR POWER STATION

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Abstract. This paper describes strategies to ensure containment integrity for ABWRs in Kashiwazaki-Kariwa Nuclear Power Station reflecting the lessons learned from the Fukushima Daiichi accident. The accident and response actions were analysed to extract lessons. Based on the lessons basic policies to enhance safety and containment integrity were derived. Firstly with considerations of multiple failures of the Fukushima Daiichi accident, defense in depth (DID) was enhanced by applying more diverse safety measures. For this purpose, in addition to refurbishing safety measures for beyond design base events, safety measures for each DID layer was enhanced not only by strengthen robustness to single failure but also by strengthen diversity and by physical separation. Secondly phased approach was introduced in choosing mitigation measures considering timing of the response actions and required reliabilities. Lastly performance requirements was clarified for containment vessel and its auxiliary systems after core damage. Various safety measures were implemented based on these policies and applied at ABWRs in Kashiwazaki-Kariwa Nuclear Power Station to ensure containment integrity.

1. BRIEF OVERVIEW OF THE FUKUSHIMA DAIICHI ACCIDENT

On 11 March 2011, Fukushima-Daiichi's units 1 to 3 were in operation and units 4 to 6 were in outage, and all units (units 1 to 4) were in operation in the Fukushima-Daini Nuclear Power Station. In response to the Tohoku Pacific Ocean Earthquake, all units in operation were shutdown. All off-site power were lost in Fukushima-Daiichi after the earthquake, but emergency diesel generators (EDG) started to keep AC power that ensured the reactor safety.

However, many switchgears got inundated and also all EDGs except one EDG for unit 6 were failed because of the tsunami that hit about 50 minutes after the earthquake. As a result, units 1 to 4 experienced prolonged station blackout (SBO), and unit 1 and unit 2 experienced not only loss of AC power but also loss of DC power. Although the efforts of water injection were done as flexible responses, units 1 to 3 experienced a certain duration time of no water injection into the reactor core and ended up in the core failures and the production of a lot of hydrogen gas. Upper parts of the reactor buildings of units 1 and 3 were destroyed because of the explosions that were caused by the hydrogen gas leaked from each primary containment vessel (PCV), and unit 4 reactor building upper part was also destroyed because of the explosion that was caused by the hydrogen gas leaked in from unit 3 PCV through the common exhaust line.

2. LESSONS LEARNED FROM THE FUKUSHIMA DAIICHI ACCIDENT AND BASIC POLICY FOR REACTOR SAFETY ENHANCEMENT

2.1. Enhancement of DID

Safety functions including water injection into the core using safety and non-safety systems were lost concurrently except reactor shutdown function because of the beyond design tsunami in the Fukushima Daiichi accident. Protection against tsunami that is beyond design basis had not been implemented based on the principle of DID since the probability of beyond design tsunami had been judged as infinitesimal in spite of insufficient knowledge about tsunami. As a result, plant staff had to respond with no experience or knowledge and faced with many challenges immediately after the tsunami. Although safety measures had been implemented based on the concept of DID even before the Fukushima Daiichi accident, the events that were considered had been actually limited internal events that were initiated by random failure inside the plant. Therefore the consideration of DID had been implemented by preparing high reliable equipment redundantly before the Fukushima Daiichi accident, which was based on the precondition that the less possibilities of random failure are, the less possibilities of concurrent failure become.

Based on the lesson learned from the accident that the DID that had been implemented for internal event did not function for external one, features of external event was well considered, which influences multiple systems at the same time. The basic policy of enhancing DID therefore was to prepare safety measures focusing on diversity and physical separation based on the assumption that multiple failure could happen, and to adopt and deploy the policy to each layer of DID, from prevention of abnormality (Level 1) to control of severe plant conditions (Level 4). Regarding external events that were considered, they were not only limited to earthquake and tsunami, but also expanded to 40 natural events and 20 human induced external events that are shown in the US NUREG and the IAEA Safety Guide. From those external events TEPCO selected the typical external events based on their cliff edge effects and their probabilities, and then established safety measures against them. When enhancing DID and implementing safety measures, insights from probabilistic risk assessment (PRA) were also utilized in order to select accident sequences to be evaluated and to evaluate effectiveness of the established safety measures.

2.2. Preparation for SBO and adoption of phased approach

In the Fukushima Daiichi accident, the Isolation Condenser (IC) did not function as expected and the High Pressure Coolant Injection system (HPCI) function was also failed in unit 1 after the tsunami. Unit 2 relied on the Reactor Core Isolation Cooling system (RCIC) and unit 3 relied on RCIC and HPCI for a prolonged time, and those units experienced big struggles for RPV depressurization that were needed to lead them to cold shutdown. Those kinds of situations could happen in the case of SBO, and before the Fukushima Daiichi accident the probability of SBO had been deemed small because the reliabilities of the off-site power and the EDGs were high. Therefore the safety measures for SBO had been done by installing equipment that could accommodate high voltage and low voltage AC powers from neighboring units and by preparing procedures for them taking advantage of multi-unit site. So if a unit experienced SBO, it was supposed to be kept safe using Safety Relief Valves (SRVs) and RCIC.

Considering the lessons that those safety measures did not function in the Fukushima Daiichi accident, AC power got enhanced by diversification to prevent SBO, and additional safety measures was installed to prevent function failures of safety significant systems such as reactor cooling function even if extended SBO happened. In addition, considering that selection of safety measures and their required reliabilities are different depending on time to spare, the concept of phased approach was adopted. The concept of phased approach is shown in Figure 1.

For example portable equipment such as fire engines and power supply cars are useful if they are well prepared because they can enable plant staff to use them flexibly depending on the development of accidents. On the other hand, installed equipment does not need setting time and they can be started automatically when there is little time to respond. Based on those characteristics, phased approach is going to be developed as follows. The initial response phase is accomplished using installed equipment because initiating events have to be responded quickly. The second response phase is accomplished using portable equipment stored on-site because a certain amount of time is available. Then the final response phase is accomplished using the additional portable equipment stored on-site and also consumables obtained from off-site.

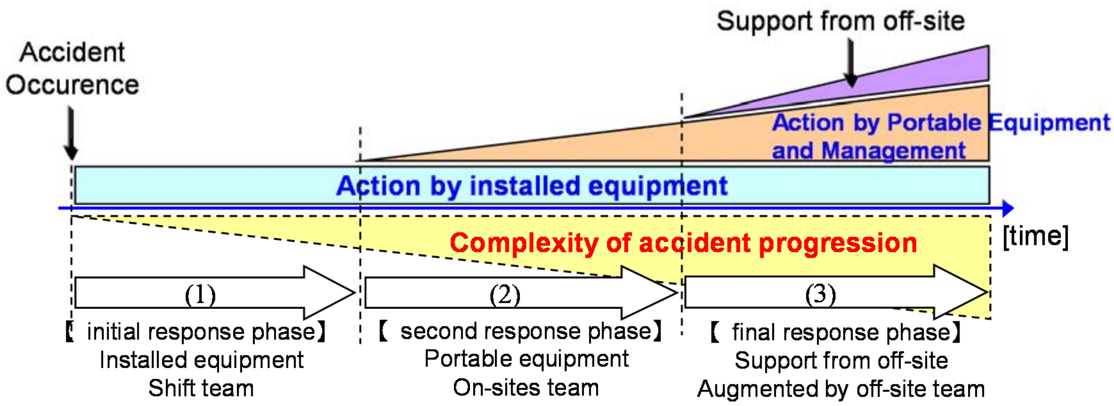

FIG. 1. Phased approach for DID enhancement.

2.3. Ensuring containment integrity against severe accident

In the Fukushima Daiichi accident fission products were released in an uncontrolled manner as a result of PCV failures that experienced over pressure and over temperature after the core damages. Originally a PCV and its auxiliary systems had been designed based on the LOCA requirement, and as accident management measures considering core damage, alternative spray function, water injection into pedestal area, and hardened PCV vent path had been added before the accident. However those measures had been limited to be utilizing and expanding the capacity of existing equipment, and concrete requirement for mitigating severe accidents that include core damage had not been developed.

Based on the experience that fission products were released uncontrollably in the Fukushima Daiichi accident, performance requirements after the core damage were clarified for a PCV and its auxiliary systems. The basic policy was to define clearly twice design pressure of PCV as its upper limit pressure and to define 200 °C as PCV upper limit temperature, and then to define function requirements for other auxiliary systems in order to contain fission products in the PCV and to make them decay as long as possible not only by enhancing PCV capacity but by utilizing auxiliary systems to ensure PCV integrity after core damage.

3. SAFETY MEASURES TAKEN AT ABWRS IN KASHIWAZAKI KARIWA NPS

3.1. Enhancement of the first and the second layer of DID

Based on the basic policies for enhancement of nuclear safety that were described in the previous chapter, various safety measures were implemented for ABWRs in Kashiwazaki-Kariwa NPS. The goals of the first and the second layer of DID are to prevent abnormalities from occurring and to prevent abnormalities from developing to accidents. Regarding the protection measures against tsunami, tidal embankment was built in order to keep the yard around reactor buildings and other buildings dry so all safety-related equipment could be protected against tsunami. With respect to the height of the tidal embankment, uncertainties in tsunami wave height evaluation were considered, which resulted in a conservative height of 15 m above the sea level. In addition to the tidal embankment, measures such as flood barriers, watertight doors, waterproof treatment for piping and cable penetrations, and drain pumps to protect safety significant equipment were also installed for inundation beyond postulated damage. These measures are useful to respond internal flooding as well as tsunami.

Not only for tsunami and earthquake, 40 natural events and 20 human induced external events were evaluated, which were shown in the US NUREG [1] and the IAEA Safety Guide [2] . From those external events, typical external events were selected based on their cliff edge effects and their probabilities, and then established safety measures. For example, new tornado proof gasoil tanks are being constructed against tornado, and fire protection zones around the yard are being settled against external forest fire.

3.2. Enhancement of the third layer of DID and supporting systems

The principal goal of the third layer of DID is to prevent core damage. When a reactor goes to shutdown state from operation, keeping reactor water level high is needed using high pressure water injection. At Fukushima Daiichi unit 1, DC power was lost because of the inundation caused by the tsunami and then there was no water injection into the core. With respect to units 2 and 3, steam driven water injection systems such as RCIC and HPCI were able to be used as high pressure water injection function after the SBO. However, restart of these systems was impossible after the DC power loss and after the systems failed for some reasons, and then high pressure water injection functions were failed.

From these lessons, new battery was installed on the higher floor of the reactor building to prevent inundation as well as reinforcing the capacity of existing batteries. In addition, installation of small generator and measures to provide DC powers from power supply vehicles through battery chargers were also prepared. In case of the DC power failure regardless of those improvements, procedures to start RCIC without DC power manually in the field (RCIC black start) were prepared and those trainings have also been executed. Furthermore, stream driven High Pressure Alternate Cooling System (HPAC) has been installed as a further safety enhancement measure, which can enhance the reliability of high pressure water injection in a prolonged SBO condition.

Then RPV has to be depressurized, low pressure injection systems and heat removal systems have to be in operation in order to lead the reactor to cold shutdown. SRVs are used to depressurize RPV and DC power and pneumatic supply (nitrogen gas) are needed for their operations. Spare batteries and spare nitrogen gas cylinders were therefore deployed to keep the function of RPV depressurization.

With respect to the low pressure water injection after the depressurization, in addition to Low Pressure Core Flooder (LPFL) as existing ECCS, diesel driven fire pumps (DDFP), make up water condensate system (MUWC) and fire engines were also available as alternative water injection measures. Only fire engines were available in the Fukushima Daiichi accident because of SBO, and switching from high pressure injection to low pressure injection were also delayed because there had been no preparations for such extensive damage conditions. The lesson learned from this experience is that ensuring power and preparing diverse measures to respond to extended damage conditions are very important to keep high reliability for low pressure water injection. Therefore the measure of reviving MUWC with alternative power supply was implemented, and procedures for water injecting using fire engines in addition to the procedures for alternative water injection using MUWC and DDFP were also prepared, which enabled plant staff to select diverse measures for low pressure water injection. These low pressure injection measures are also available for injection into the spent fuel pools (SFP).

Regarding decay heat removal, alternative heat exchanger vehicles were deployed on the high ground in the yard as the alternative heat removal measure. It has become possible to lead a reactor to cold shutdown within 48 hours in case of loss of ultimate heat removal (LUHS) by connecting the vehicle to the intermediate loop of RHR system.

As supporting systems, gas turbine generators, power supply vehicles, emergency high voltage switchgears, and storage facilities of diesel fuel were installed for the enhancement of power supply. Water reservoir was newly installed for the enhancement of water supply, which has about 20000 m^3 capacity.

The whole picture of safety measures for third layer of DID and supporting systems is shown in Figure 2.

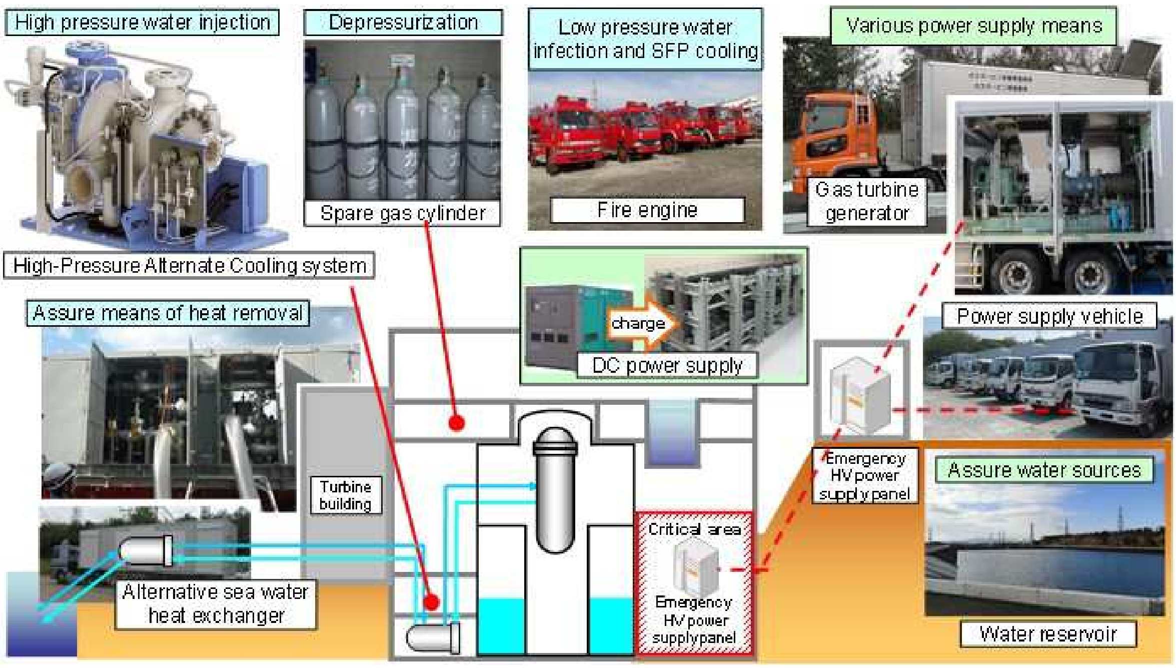

FIG. 2. Measures of DID the 3rd layer and supporting systems.

3.3. Enhancement of the fourth layer of DID

The goal of the fourth layer's measures is to protect PCV function and to prevent fission products from releasing into the environment in an uncontrolled manner. Although high reliabilities of the safety measures to prevent core damage were accomplished by the measures and the procedures for the third layer of DID as described in the previous chapter, safety measures for enhancing the fourth layer of DID were prepared independently.

Figure 3 is the outline of the safety measures of the fourth layer of DID. Typical safety measures are described in the following sections.

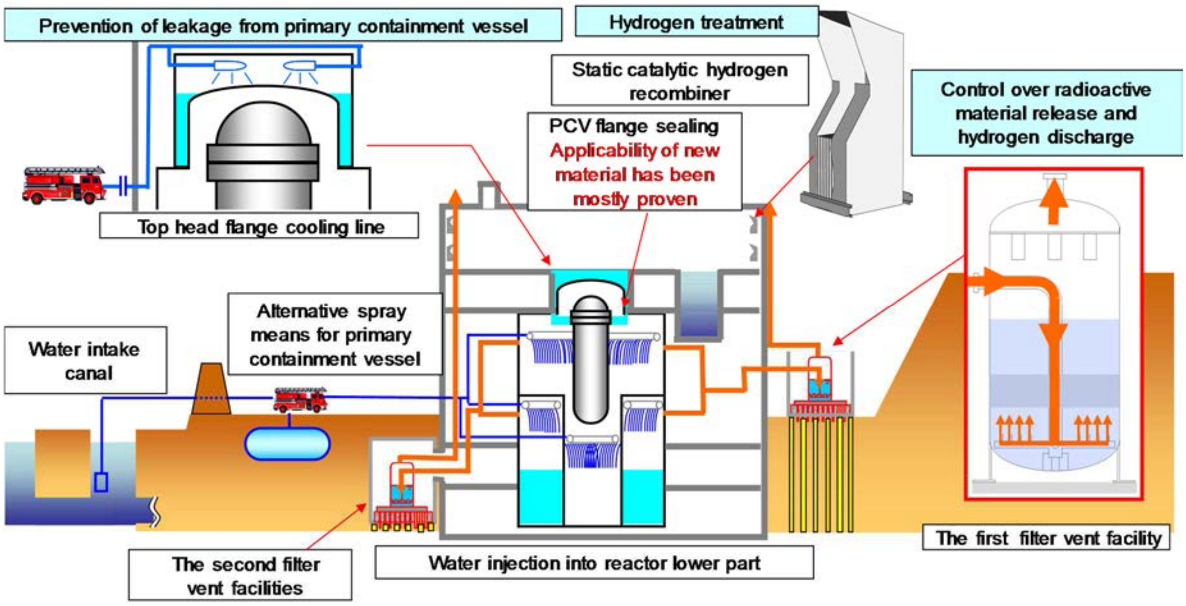

FIG. 3. Outline of measures for DID the 4th layer.

3.3.1. Enhancement of PCV capability against leakage

Firstly the PCV capability of containing fission products was enhanced. In the Fukushima Daiichi accident, the PCVs were exposed with high temperature and high pressure for a prolonged time, and then fission products were released from the PCVs. The basic approach based on the lessons was to define clearly twice design pressure of PCV (2 Pd, 620 kPa[g] for ABWRs) as its upper limit pressure and to define 200 °C as PCV upper limit temperature, and then to define function requirements for other auxiliary systems in order to contain fission products in the PCV, make them decay as long as possible not only by enhancing PCV capacity itself but by utilizing auxiliary systems to ensure PCV integrity after the core damage.

For this purpose, the measures for suppression of PCV pressure were implemented by making it possible to utilize alternative pumps as the alternative PCV spray in addition to the existing PCV spray using RHRs, which made the reliability of the function to suppress PCV pressure high. Installed MUWC pumps and fire engines are available as the alternative pumps. On the other hand, the seals of the top head flange and the component hatches need to be improved from the viewpoint that they have to keep their seal capabilities under a prolonged high temperature and high pressure, although the PCV structure and the seals of PCV penetrations for electric cables have already had enough seal capabilities under the condition of 2 Pd and 200 °C. Once PCV pressure goes up beyond its design pressure, opening area of these flanges is going to increase. The seal materials however were able to keep their seal capabilities by following the flanges' opening if their compression sets are small because they are located on those flanges with compressed state. However, existing silicon rubber has the weak point that when it is exposed to high temperature steam caused by the flange opening, its compression set gets increased as time passes. Therefore, improved EPDM seal materials that improved the capability of heat tolerance and radiation tolerance are adopted and heat tolerant back-up seal material is also put outside of the EPDM seals to improve capacity against leakage as shown in Figure 4.

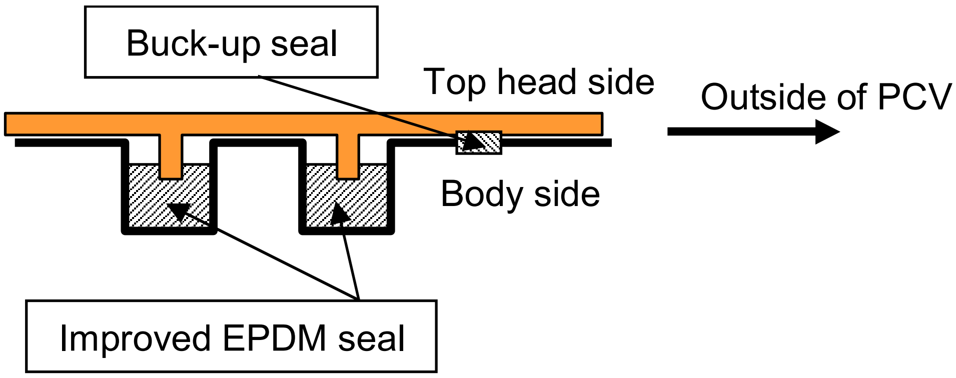

FIG. 4. Packing arrangement for PCV.

3.3.2. *Alternative coolant circulation to prevent PCV venting*

When a plant experienced core damage with all ECCS failures, alternative pumps such as MUWC pumps and fire engines would work to cool the core and the containment. However if all ECCS pumps and main condenser fail in a severe accident, the plant is in the loss of ultimate heat sink (LUHS) condition because all RHRs are unavailable and PCV venting is inevitable in the end.

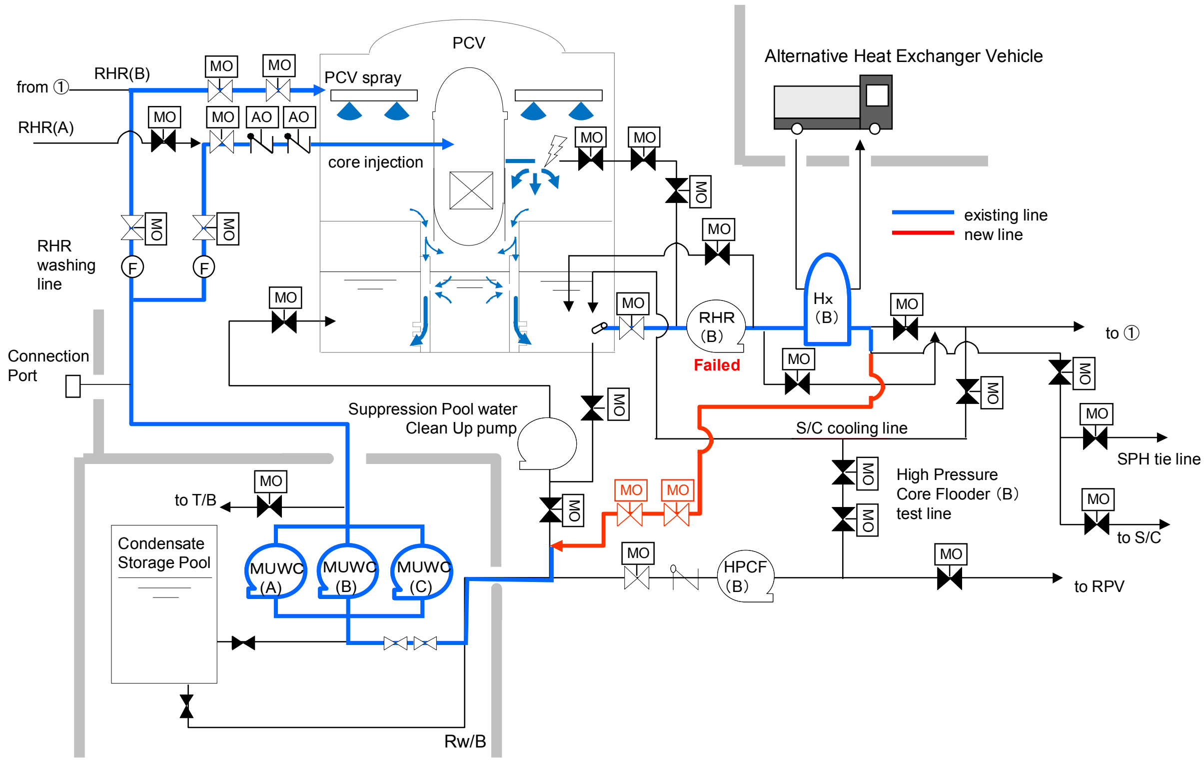

FIG. 5. Alternative coolant circulation system outline.

Alternative coolant circulation system (ACCS) is going to be installed to function in the above severe accident condition in order to prevent containment venting using MUWC pumps. The water source of ACCS is suppression pool and the suction water passes through RHR piping, RHR heat exchanger, HPCF piping, and is provided to MUWC pumps. Then MUWC pumps send the water to core injection line and PCV spray line through RHR piping.

The water that is injected into the RPV and the containment flows over from broken pipe (in LOCA case) or from SRVs (in non LOCA case), passes through the diaphragm floor and the lower drywell, and goes back to the suppression pool to make coolant circulation line (See Figure 5). The AC power for MUWC pumps and related valves operation is provided by the gas turbine generators that are located on the high ground of the yard, so ACCS is able to function in SBO scenario. ACCS makes it possible to prevent PCV venting even in the severe accident condition with all ECCS function failures and SBO, so it greatly improves the safety level to tackle severe accidents.

3.3.3. Filtered vent

If ACCS did not work in the severe accident conditions, PCV would need to be vented through the filtered containment venting system (FCVS). FCVS is composed of the aerosol filter and the iodine filter.

The aerosol filter is composed from 3 basic components (See Figure 6). The first component is scrubber nozzles, through which the vent gas is jetted into the scrubber water uniformly. The second component is mixing element, which can mix the gas and scrubber water and atomize bubbles to enhance decontamination factor. The third component is metal filter, which captures aerosol in the vent gas and separate droplet from the vent gas. Thanks to the aerosol filter, fission products can be decontaminated more than 1000 times (DF>1000).

The iodine filter is located in the downstream of the aerosol filter to reduce organic iodine release into the environment. The outline structure of the iodine filter is shown in Figure 7. The 19 candle units filled with silver zeolite are located in the iodine filter, and DF for organic iodine is more than 50.

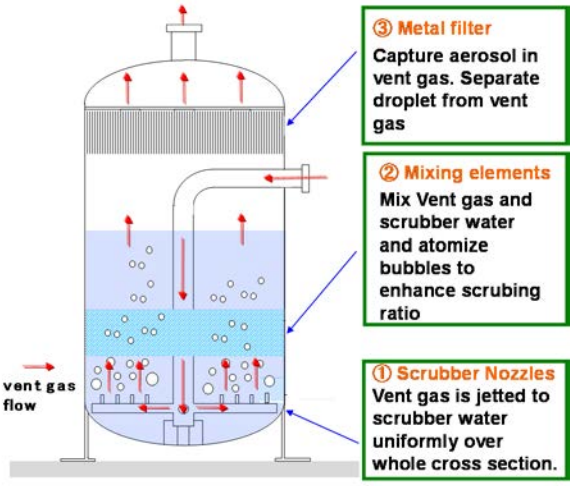

FIG. 6. The aerosol filter for PCV venting.

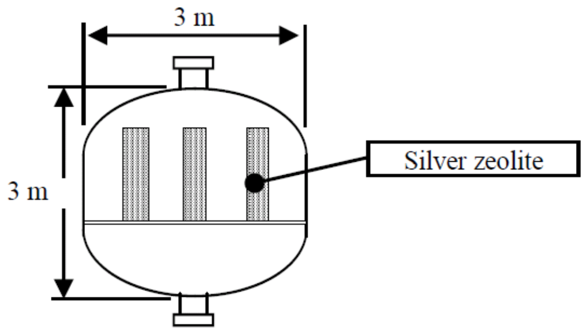

FIG. 7. The iodine filter for PCV venting.

3.3.4. pH control to suppress the production of organic iodine

According to NUREG-1465 referred NUREG/CR-5732, iodine entering the containment is at least 95 % CsI with 5 % as I plus HI, and where the pH is controlled at values of 7 or greater within the containment, iodine in organic form may be taken as comprising no greater than 0.15 % of the total iodine released. However, NUREG-1465 indicates that without any pH control, large fraction of the dissolved iodine will be converted to elemental iodine and be released to the containment atmosphere.

Therefore, it is important to keep pH 7 or more in the PCV to reduce iodine release into the environment when the PCV has to be vented. In a severe accident condition the acidic materials are iodine as HI, atmospheric species such as nitric acid, hydrochloric acid from cable insulation, etc. Chloroprene rubber used for cable insulation in PCV produces a lot amount of hydrochloric acid if it was degraded by radiation or heat. To neutralize this and to keep PCV water pool alkali side, aqueous sodium hydroxide injection system is newly installed. Aqueous sodium hydroxide is mixed with water from the condensate storage pool and injected into the containment by MUWC pumps through drywell spray line and wetwell spray line. The outline of the PCV pH control system is shown in Figure 8.

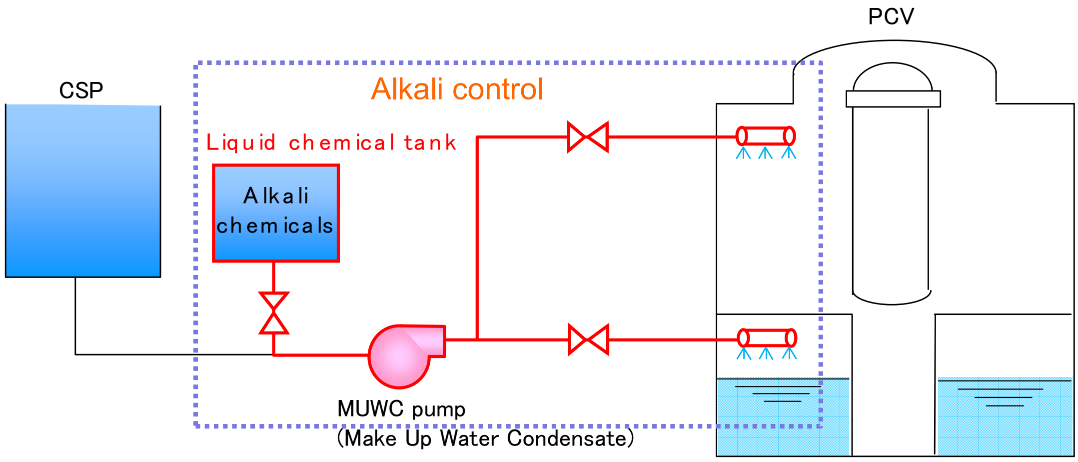

FIG. 8. Outline of the PCV pH control system.

3.3.5. Hydrogen gas control in the reactor building

In the Fukushima Daiichi accident, upper parts of the reactor buildings of units 1, 3 and 4 were destroyed because of the explosions that were caused by the hydrogen gas leaked from the PCVs of units 1 and 3. The risk of such hydrogen explosion in the reactor building has been improved by various safety measures described above, though, Passive Autocatalytic Recombiners (PARs) and a system for water injection into outer region of the PCV top head flange are installed independently to suppress the influence of hydrogen gas into the reactor building. PARs are settled in the top floor of the reactor building to recombine hydrogen gas leaked from the PCV with oxygen gas by catalytic reaction to prevent hydrogen gas explosion without any operator action. Temperature indicators are settled around the inlet and outlet of PARs to make sure their operation, and hydrogen gas concentration meters are also settled on the top floor of the reactor building. Water injection system to cool the PCV top head flange prevents PCV from failure by overheating and from leaking hydrogen gas into the reactor building.

4. CONCLUSIONS

Safety level of ABWRs in Kashiwazaki Kariwa Nuclear Power Station has been improved by the safety measures that has been installed based on the basic policies of DID enhancement, phased approach, and ensuring containment integrity against severe accident. Especially capacities for ensuring containment integrity have been drastically improved by installing diverse safety measures for the third layer and the fourth layer of DID including the alternative coolant circulation system. TEPCO will continue best efforts to improve our nuclear safety level based on these basic policies without slipping into self-complacency.

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[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-3, IAEA, Vienna (2010).

[3] U.S. NUCLEAR REGULATORY COMMISSION, Accident Source Terms for Light-Water Nuclear Power Plants, NUREG-1465 (1995).

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SESSION IV

STRATEGIES IN CONTAINMENT COOLING AND ENERGY MANAGEMENT FOR ADVANCED REACTOR DESIGNS

STRATEGIES TO ENSURE CONTAINMENT INTEGRITY FOR EC6®

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Abstract. The Enhanced CANDU♦6® (EC6®◊) is a 740 MW(e) reactor, which has evolved from the well-established CANDU line of reactors, which are heavy-water moderated, and heavy-water cooled horizontal pressure tube reactors, using natural uranium fuel. The EC6 design consists of multiple lines of defense of preventative and mitigative features which prevents uncontrolled radioactive releases during a severe accident, including a severe core damage (SCD) accident. The EC6 design addresses accident prevention, accident mitigation, severe accident resistance and recovery, and post-accident control and monitoring to meet the international standards in accident management. This includes international standards for safety goals for new plants in terms of Core Damage Frequency and Large Release Frequency with good margin. There are always at least two heat sinks, passive or active, available at any stage of accident progression that will halt or delay progression of the accident, allowing time for other mitigating measures to be employed. In the extreme event that all engineered features are unavailable, the large water inventories in the steam generators, the reserve water tank, the calandria vessel, and the calandria vault will slow accident progression. At any stage of the event, heat removal using on-site or off-site water supplies may be established using provided connections to halt event progression. Containment atmospheric hydrogen control is achieved by passive autocatalytic recombiners (PARS) and active hydrogen igniters that limit the concentration of hydrogen in the Reactor Building atmosphere to below the threshold limit at which rapid deflagration or detonation could occur. The severe accident recovery and heat removal system (SARHRS), emergency containment filtered venting system together with the calandria vessel and calandria vault make-up system prevents the containment failure and uncontrolled release. Extended coping strategies are in place to provide core and spent fuel bay (SFB) cooling, and heat transport system∇ (HTS) and containment integrity. These strategies rely on portable equipment which is safely stored onsite and protected from external events. This extended coping time will be sufficient for use of pre-planned and pre-staged offsite resources for maintaining core and HTS, SFB cooling, and containment integrity.

1. INTRODUCTION

The EC6® reactor is an evolution of the proven CANDU 6 design most recently built at Qinshan site in China. It retains the fundamental core design of the CANDU 6 reactor, a heavy-water- moderated, heavy-water-cooled horizontal pressure tube reactor using natural uranium fuel. While retaining the basic features of the CANDU 6 reactor design, the EC6 reactor incorporates innovative features and state of the art technologies that enhance safety, operation and performance. After the Fukushima event, nuclear operators, regulators and nuclear power plant designers performed systematic reviews of nuclear power plant designs to confirm the plant robustness to cope with severe accidents, and to identify improvements to further reduce the risk of radioactive releases to the environment. The EC6 design takes into account operating experience feedback as well as the recent international trends in design of modern nuclear power plants. The Enhanced CANDU 6 (EC6) is a Generation III, 740 MW(e) class heavy water moderated and cooled pressure tube reactor (see Figure 1). Heavy water (D_2O) is a natural occurring isotope of water that is used as a moderator to slow down the neutrons in the reactor, enabling the use of natural uranium as fuel. This feature is unique to CANDU reactors. The choice of D_2O as the moderator also allows other fuel cycles to be used in CANDU reactors. The use of natural uranium fuel in EC6 reactors permits fuel cycle independence and avoids having to deal with complex issues such as fuel reprocessing and enrichment. Technology transfer for localizing fuel manufacture is simple and has been achieved very successfully in a number of countries, such as Argentina, China, India, Pakistan, Republic of Korea, and Romania.

♦ CANDU® (CANada Deuterium Uranium®) is a registered trademark of Atomic Energy of Canada Limited.

◊ Enhanced CANDU 6® (EC6®) is a registered trademark of Atomic Energy of Canada Limited.

∇ Heat transport system is equivalent to reactor coolant system in PWR.

In general, the main advantages of the EC6 reactor design are the use of natural uranium, on-line refueling, excellent safety performance, high capacity factor, and it is perfectly suited for small and medium sized electrical grids. EC6 can rely on either cooling towers or lake or river or sea.

This paper will discuss the CANDU 6 existing features as well as the new design changes incorporated in EC6 for the prevention and mitigation of severe accidents with emphasis on maintaining containment integrity. These design features ensure that the EC6 meets international safety standards.

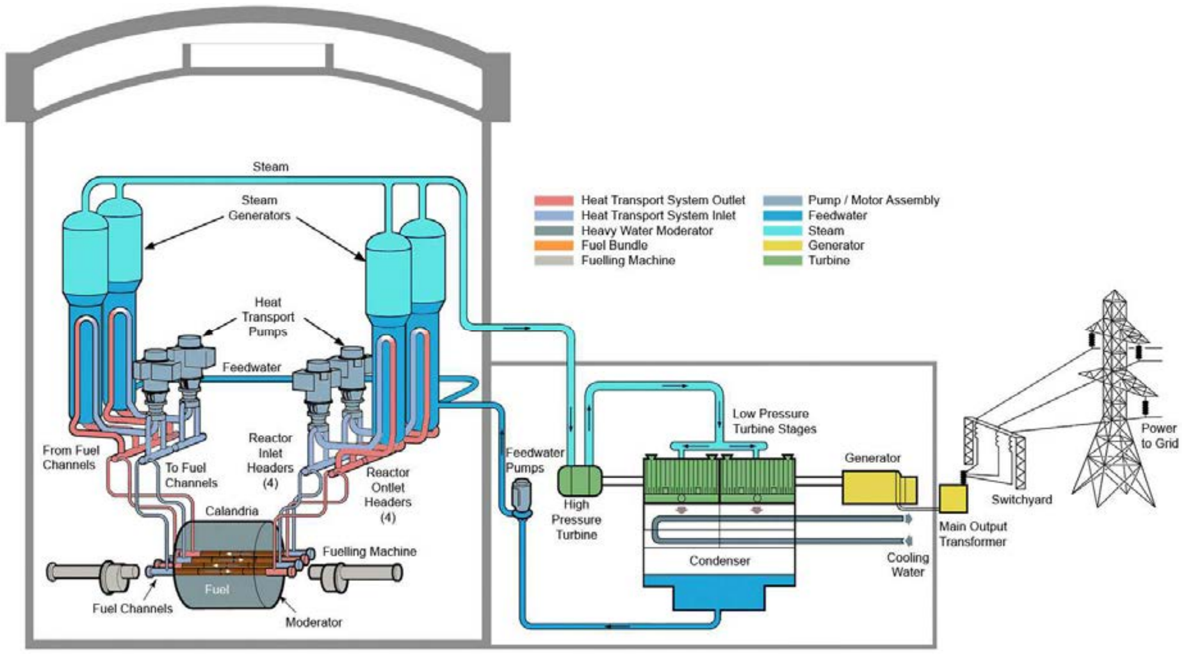

FIG.1. EC6 schematic.

2. INHERENT CANDU 6 DESIGN FEATURES

The CANDU 6 reactor has a reliable reactor regulating system (RRS) and two independent, fast-acting, diverse and separate shutdown systems, each capable of rapid shutdown of the reactor after an initiating event.

Following any design basis accident, reactor shutdown occurs rapidly by dropping shutoff rods into the reactor core, or by injecting liquid poison into the heavy water moderator inside the calandria. The two shutdown systems (SDS1 and SDS2) are independent from each other and the reactor regulating systems, including use of different sensors, logic, and end-devices, and each shutdown system has sufficient negative reactivity to quickly achieve and maintain the reactor core sub-critical.

Re-criticality during severe accidents is not possible as the reactor remains sub-critical as a result of loss of reactor core geometry. This is due to the small excess reactivity in the CANDU core, natural uranium (fast fission rate is relatively low), loss of moderation (degradation of the D_2O purity) and the presence of H_2O in the fluid surrounding the fuel will contribute to absorbing neutrons.

The CANDU severe core damage accident progression is considerably different from that in LWRs, resulting in different challenges caused by severe accident phenomena. The unique and inherent robust design features, are the low pressure and temperature moderator in the calandria vessel surrounding the fuel channels, and the large volume of water in the calandria vault which, in turn, surrounds the calandria. Both provide a passive heat sink which prevents or significantly delays the severe accident progression. The approximate inventory of heavy and light water, available for heat removal, is as follows: 190 Mg heavy water in heat transport system (HTS), 240 Mg heavy water in calandria and 520 Mg of light water in calandria vault (see Figure 2) and as well as 210 tonnes water in the high pressure emergency core cooling tanks and over 2000 Mg water in the dousing tank. These significant quantities of water inventories surrounding the fuel and the entire core act as a heat sink to remove the decay heat after reactor shutdown, even if all engineered heat removal systems fail. There is no need for any operation of any valves or pumps as heat removal is through passive boil off. The large water volumes mean that such boil off times are long (many hours), allowing sufficient time for the implementation of severe accident management actions.

The CANDU 6 design is inherently well-suited for debris retention in the calandria vessel as summarized below:

- The moderator system has ample heat removal capacity when available. To help in the prevention of an unlikely severe accident, the moderator serves as an additional barrier to further progression of the accident and helps maintain core integrity. As long as water can be maintained in the calandria vessel, fuel channel collapse is precluded.
- The calandria vessel is strong enough and its steam relief capacity is large enough to withstand the collapse of the channels, should the moderator heat removal system fail.
- Steam relief from the calandria is provided through rupture disks on the calandria vessel.
- The rupture discs are designed to operate at a pressure well above the operating pressure of the relief valves and will burst under severe accident conditions.

If the moderator system is not available, the decay heat is removed by boiling off the moderator inventory to the reactor building atmosphere. Since the fuel channels are horizontally oriented in the calandria vessel, the pressure tubes will sag/strain into contact with their calandria tubes when heating up, so that a heat rejection pathway to the moderator is established. Decay heat can be transferred from the overheated fuel through the pressure tube and calandria tube contact to the moderator. If the moderator cooling system is not available or subsequently lost, it will take more than 5 hours for the moderator inventory to heat up and boil off. This allows sufficient time for operator action to establish means of arresting core degradation in the calandria vessel (within 5 hours by providing inventory to calandria vessel), and preventing further accident progression.

If the moderator cooling system and both gravity-fed and active moderator make-up are not available or are subsequently lost, the water level in the calandria vessel will start to drop and uncover the top row of fuel channels. Uncovered fuel channels heat up, eventually lose their mechanical strength and deform by sagging.

Deformed fuel channels will break off eventually and slump down onto the bottom of the calandria vessel forming a coarse debris bed. This is the process of core disassembly or core collapse which will take several hours to develop. Typically core collapse occurs at about 8 hours after the start of a severe accident in a CANDU 6 reactor. During this process, the significant volume of water inside calandria vault cools the outer calandria vessel wall, maintaining the external cooling of the vessel. It will take many hours for the calandria vault inventory to heat up and boil off (between 10 to 20 hours). This long process would provide time for operators to take appropriate actions to mitigate the severe accident progression by providing make-up to calandria vault. As long as calandria vessel is mostly submerged in water, and the calandria vault water inventory can be maintained, it is expected that corium will be retained in the calandria vessel and accident progression arrested in-vessel. The externally cooled calandria vessel acts as a "core catcher" containing the core debris.

It should be noted that core disassembly and relocation takes place only at low HTS pressures and that melting of core materials is avoided until after the debris has relocated to the bottom of the calandria vessel.

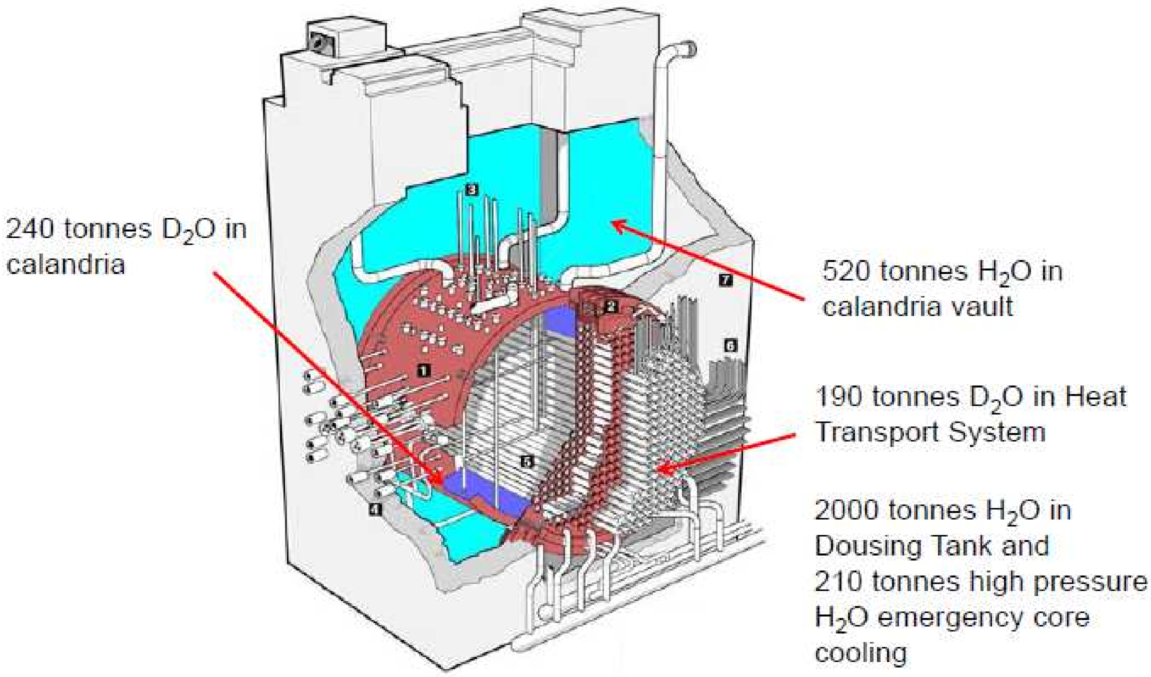

FIG. 2. CANDU 6 reactor core and water inventories — passive features.

The containment building provides the fundamental barrier protecting the public in the unlikely event of a severe accident by limiting the radioactive releases to the environment. Its effectiveness requires limiting the interior temperature and pressure following such an event.

The CANDU 6 containment is a leak tight envelope around the reactor and associated nuclear systems, and consists of the following structures and systems:

(1) Reactor building;
(2) Containment isolation system (including valves and dampers);
(3) Airlocks;
(4) Dousing system;
(5) Reactor Building Local Air Coolers (LACs);
(6) Hydrogen Control System with PARs and/or H_2 Igniters;

Containment isolation is a fail-safe design[i].

The dousing system and reactor building LACs are used to reduce containment pressure and containment heat removal if available.

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[i] Containment isolation valves are designed to fail closed on loss of instrument air or loss of control power supply, therefore creating a fail-safe condition.

3. NEW REGULATORY SAFETY STANDARDS

The IAEA Safety Standard for the design of new nuclear power plants, SSR-2/1, [1] is being revised and will be issued. This standard will lead to subsequent revision of IAEA guidance documents. The standard has several new clauses dealing with the prevention and mitigation of severe accidents. These clauses related to EC6 design for design extension conditions are on the following items:

(1) Combinations of events leading to early or large radioactive release are practically eliminated (clause 2.13 item (4));
(2) Safety features for design extension conditions are practically independent of safety system (clause 4.1.3a);
(3) There is an adequate margin in design for items ultimately necessary to prevent early or large radioactive release exceeding those considered in the design (clause 5.21a);
(4) For multi-unit stations, each unit shall have its own safety systems and safety features for design extension conditions (requirement 33);
(5) Provisions for alternate power supply for design extension conditions (requirement 68);
(6) Clauses on spent fuel bays with respect to design and monitoring (clause 6.68).

In terms of containment the following clauses are applicable [1]:

"6.28 The capability to remove heat from the containment shall be ensured, in order to reduce the pressure and temperature in the containment, and to maintain them at acceptably low levels after any accidental release of high energy fluids. The systems performing the function of removal of heat from the containment shall have sufficient reliability and redundancy to ensure that this function can be fulfilled.
6.28a. Design provision shall be made to prevent the loss of the containment structural integrity in all plant states. The use of this provision shall not lead to early or to large radioactive releases.
6.28b. The design shall also include features to enable the safe use of nonpermanent equipment for restoring the capability to remove heat from the containment."

4. ENHANCED CANDU 6®ADVANCED DESIGN FEATURES

Enhanced CANDU 6 design incorporates features intended to improve plant safety, thus reducing risk when compared to current generation nuclear power plants. Some of these advances design features are preventive in nature, while others are mitigative.

Preventive features aim to prevent severe accidents while the mitigative features aim to arrest the progression of core damage and prevent a breach of the containment pressure boundary. The mitigative features and the effect of each of these features on severe accident mitigation are presented. Some of these features address station blackout scenarios (SBO). The design includes seismically qualified battery backup for 24 hours. This feature will ensure that operation of safety critical process loads and a subset of instrumentation monitoring parameters, emergency lighting and emergency communication will be available continuously during an extended SBO. This will allow the operator to start the portable diesels or restore some other electrical systems to ensure power to the critical components.

5. DESIGN PROVISIONS TO MAINTAIN IN-VESSEL COOLING

To prolong the duration of moderator as a viable heat sink, and keep the core flooded (submerged in water), a new moderator make-up and recovery is proposed as part of safety improvement for design. This provides:

- A heat sink: by adding water to calandria vessel and releasing steam into containment. A connection to the elevated reserve water tank was added, that provides gravity-fed make-up water to the calandria vessel (see Figure 3) to ensure a short-term moderator make-up supply. This ensures that water is available in the calandria vessel for heat removal from the fuel channels through the overheated pressure tubes, and thereby limits the core damage progression at the fuel channel boundary. This heat sink:
  - provides sufficient flow to compensate for boil off of the moderator at decay power levels;
  - extends core cooling, delays severe accident progression and facilitates implementation of accident management measures.
- A heat sink: by adding a new moderator make-up and recovery system designed to provide a long term heat sink when the gravity-fed makeup to the calandria vessel is exhausted. This system is part of the Severe Accident Recovery and Heat Removal System♦ (SARHRS) which provides means to recover and cool water from the Reactor Building (RB) sumps and return it to the calandria vessel (see Figure 3).
- Instrumentation allows the operator to monitor the water level in calandria vessel and the status of the make-up water source, and remotely monitor the supply of water into the calandria vessel.

There are ~ 2056 tonnes H_2O in reserve water tank available to the steam generators, emergency core cooling (ECC), low flow spray system, calandria vessel, and calandria vault.

6. DESIGN PROVISIONS TO MAINTAIN EXTERNAL COOLING OF CALANDRIA VESSEL

The severe accident mitigation provisions have been designed with the focus on systems and design features that will ensure external cooling of the calandria vessel and in-vessel retention of the corium. The list below provides a number of design features, some of which are design changes proposed for EC6 , to increase the robust safety features already existing in CANDU plants. These are:

- Passive thermal capacity of calandria vault water. The calandria vault provides the third line of defence (after the ECC and the moderator) in cooling the reactor core during a severe accident. The large volume of water in the calandria vault has adequate thermal capacity to passively prevent calandria vessel failure (see Figure 2). Water in the calandria vault can provide continued external cooling of the core debris relocated at the bottom of the calandria.
- EC6 will have adequate pressure relief capacity of the calandria vault to ensure sufficient pressure relief by decay heat removal such that calandria vault integrity is maintained during severe accidents. This design feature protects the vault from

♦ The SARHRS is a new system for severe accidents mitigation, which includes: the moderator make-up & recovery and the calandria vault make-up.

over pressurization due to the increased heat load to the calandria vault as a result of core disassembly and debris heat-up within the calandria vessel. If there is no makeup water to the calandria vessel, then the decay heat will be transferred through the calandria wall to the surrounding calandria vault water. The calandria vault water will act as the heat sink, discharging steam through the pressure relief ducts. Under severe accident conditions additional relief is required.

— Active emergency make-up to calandria vault from the reserve water tank to prolong the duration of shield cooling water as a viable heat sink, to keep the vault flooded.
— Active emergency make-up supply system to the calandria vault, designed to ensure continuous supply of cooling water to the calandria vault for decay heat removal, once the gravity-fed makeup from reserve water tank is exhausted. This design feature is part of the seismically qualified containment heat removal system (SARHRS), which recovers water from the RB sumps, cools it via a heat exchanger and returns it to the calandria vault (see Figure 3). The SARHRS is supplied by a separate dedicated diesel generator. The calandria vault make-up water system will maintain external cooling of the calandria vessel providing long term heat removal.
— External calandria vault make-up provided by connection to a fire truck is also possible.

Without any make-up supplies available, the water level in the vault decreases gradually, the calandria vessel bottom heats up and fails due to creep at ~42 h, when the debris relocates into the calandria vault. This allows sufficient time for the operators to restore failed mitigating systems or for accident management actions to be implemented.

7. SPENT FUEL BAY

The spent fuel bay cooling must be maintained following accidents. The CANDU 6 design includes Class III power to the re-circulating cooling water system (RCW) pumps. In the event that cooling is lost to the spent fuel bay following an accident, boiling in the spent fuel bay occurs after about 60 hours. The radiation fields become significant due to loss of water shielding after approximately 9 days. The fuel is first uncovered after about 2 weeks. These heats up times provide a large margin for adequate recovery following any type of accident. A makeup of ~1 kg/s is sufficient to maintain water level and this can be provided by fire trucks or mobile pumps. In the EC6 design, provisions are made for radiation, water level and temperature monitoring for the spent fuel bay. It should be noted that re-criticality is not an issue and cannot occur for CANDU plants using natural uranium.

8. DESIGN FEATURES TO MITIGATE EX-CALANDRIA MOLTEN FUEL CONSEQUENCES

The defence in depth approach of the CANDU design ensures that as long as provisions of each fission product barrier are operable, the failure of the next level of defence is highly unlikely. The inherent safety features of the CANDU design provide high confidence that severe accidents will not progress beyond in-vessel core damage state. In the unlikely event of ex-vessel failure there is provision to provide water to the calandria vault to keep the corium cool and remove decay heat. The reactor floor has a large area to facilitate the corium spread and cooling. Calandria vault water make-up as part of the moderator recovery system or connected to a fire truck will ensure continued supply of cooling water to the calandria vault for decay heat removal, if makeup water directly to the calandria cannot be provided. All of these mitigating provisions will minimize or even prevent the Molten Core Concrete Interaction (MCCI), thus retaining the core materials within the calandria vault. MCCI may occur when molten core debris breaches the calandria vessel and contacts concrete surfaces at the bottom of the calandria vault, whereby the thermal and chemical properties of the melt contribute to the potential degradation of the concrete and generation of non-condensable gases. Analysis has shown that, if the ex-vessel corium is not submerged in water and cooled, MCCI does not occur until at least two (2) days after the accident initiation, with complete concrete ablation at least four (4) days after accident initiation. The EC6 has a refractory layer on top of the calandria vault concrete to delay molten core debris interaction with structural concrete. This provides sufficient time for operators to mitigate consequences in the containment and bring the accident into a controlled and stable state. These are conservative timeframes, since during the calandria vessel failure, there is calandria vault water surrounding lower part of the calandria vessel.

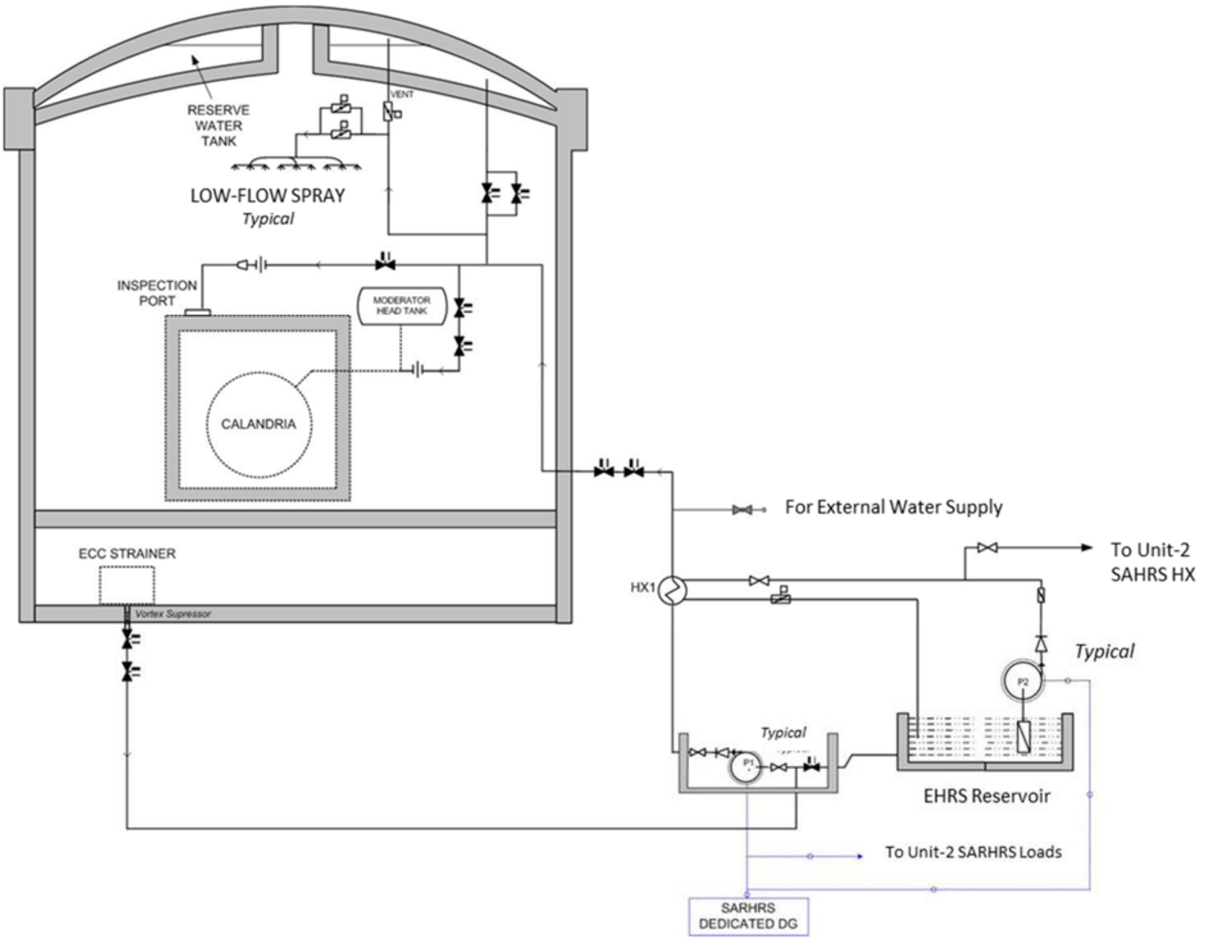

FIG. 3. EC6 Severe Accident Recovery and Heat Removal System (SARHRS).

9. DESIGN FEATURES TO MAINTAIN CONTAINMENT INTEGRITY AND MINIMIZE OFF-SITE RELEASES

The challenge to containment integrity in the event of a severe accident is slow over-pressurization due to steam generation by decay heating as a result of a loss of the heat sinks. Multiple features are available to avoid steaming into the containment; highly reliable, active, post-accident heat sinks are provided for the HTS, the calandria vessel, and the shield cooling, which would stop the steaming into the containment atmosphere when available. The reactor building has one active heat removal system dedicated to cool the containment atmosphere, the reactor building cooling system. The low flow reserve water sprays provide containment heat removal for design extension conditions including severe accidents. They are initiated automatically upon detection of high pressures that may challenge containment integrity.

The list below provides a number of design features, some of which are design changes proposed for EC6, to increase the robust safety features already existing in CANDU plants. These are:

- The containment has a large volume that reduces the potential for developing the detonable concentrations of hydrogen under severe accident conditions and the potential for containment overpressure from noncondensable gas buildup. The containment pressure capacity has been increased to be sufficiently large that the pressure loads associated with the early challenges (e.g. moderator boiling and calandria relief valve burst) will not pose a threat to containment integrity.
- PARs and Hydrogen Igniter System to ensure either recombination of hydrogen or combustion at lean hydrogen concentrations and to minimize the potential for large deflagrations or detonations. PARs are complete passive, i.e. do not require power. Hydrogen igniters are powered from one electrical power supply and backed up by another one which is seismically qualified.
- Emergency Containment Filtered Venting System (ECFVS) that can be used to control containment pressure in the unlikely event of long term over pressurization of the containment. The proposed ECFVS is an AREVA's combined Venturi Scrubber FCVS. The Combined Venturi Scrubber Unit is connected at one end to the containment via vent piping and isolation valves and at the other end to the exhaust stack via a line equipped with the throttling orifice. The ECFVS is activated when the containment vent pressure has been reached by opening of the containment isolation valves. Closing of one of the containment isolation valves when the predetermined pressure has been reached shuts off the venting process. The system is passive.

10. CONCLUSIONS

As described above the EC6 design relies on both passive and active safety related systems for accident preventions and mitigation. The passive systems rely on natural forces, such as gravity, to perform their function. Because of the multiple levels of defense in depth and duration of the severe accident progression that allows operator to mitigate consequences in the containment and bring the accident into a controlled and stable state, the probability of reaching a long term over pressurization of the containment is very unlikely. Also, even for this very unlikely, situation EC6 design includes multiple features both active and passive that can be used to reduce the containment pressure and successfully mitigate the situation and ensure containment integrity is maintained.

The design and safety features for the EC6 shows that the new IAEA Safety Standards [1] are met.

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SESSION V

FILTERED VENTING TECHNOLOGY

KAERI ACTIVITIES IN THE DEVELOPMENT OF FILTERED CONTAINMENT VENTING SYSTEMS

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Abstract. A Filtered Containment Venting System (FCVS) is one of the strategies maintaining the integrity of the containment by releasing the high-temperature and pressurized gas from inside the containment to the outside the pipe line by capturing and scrubbing radioactive gases and aerosols during a severe accident in a nuclear power plant. During the releasing gas through the FCVS, fission products such as radioactive aerosol and iodine are filtered simultaneously by the filtration system of the FCVS to prevent leakage of the radioactive material into the environment. An integral test facility was prepared to verify the performance of the developed FCVS. The FCVS test facility is a scaled test facility using scaling analyses with all of the following components included: pool venture scrubber, droplet separator, particulate filter, and molecular sieve filter. The test facility consists of a test vessel, thermal-hydraulics, and aerosol/iodine generation and measurement parts. Tests to quantify the performance of the FCVS will be performed such as a thermal-hydraulic test, aerosol removal test, elemental iodine removal test, organic iodine removal test, re-suspension of aerosol test, re-volatilization of iodine test, and dynamic test under a high-pressure gas ejection from the containment. The effects of the thermal hydraulic conditions on the aerosol scrubbing and hydrogen concentration in the postulated FCVS were also estimated using the MELCOR computer code under a station blackout in an OPR1000 as the target nuclear power plant. The decontamination factor of metal iodide aerosol, especially, cesium iodide (CsI), on the scrubbing solution in the FCVS was calculated when the pressure in the containment building approached 5 bars. The possibility of a hydrogen combustion in the initial operation of the FCVS was also evaluated.

1. INTRODUCTION

In the case of a severe accident such as the Fukushima accident [1], the melted core material (corium) can be relocated to the lower plenum of the reactor pressure vessel. If the reactor vessel is breached, the molten corium will be discharged into the containment environment. The defense in depth approach is used to prevent significant release of radioactive materials outside the containment. In the case of a LWR (Light Water Reactor), there are four physical barriers. The first barrier is a fuel pellet that can contain fission gas and all radioactive materials. The second barrier is fuel cladding enclosing the fuel pellets. During a severe accident, the first and second barriers are breached, and the reactor vessel and containment can then prevent the release of radioactive materials as the third and fourth barriers.

The discharge of corium will lead to substantial releases of fission products in the form of aerosols and gases, which are simultaneously released with steam and non-condensable gases such as hydrogen. Hydrogen is generated during the oxidation of zircaloy in the core material, while non-condensable gases are generated due to corium concrete interaction.

During severe accident, pressure inside the containment can increase substantially, and the integrity of the containment may be threatened. During the Fukushima accident, the last barrier was breached, which led to a significant release of radioactive materials into the environment and the contamination of huge area including the sea, soil, and rivers. A prolonged station black out (SBO) due to an earthquake and tsunami caused a loss of the four barriers discussed above. Therefore, the use of an additional system like a FCVS is considered in many countries to provide an additional barrier [2] to maintain the containment integrity even during a prolonged SBO.

FCVS (Filtered Containment Venting System) is employed to maintain the integrity of the containment by releasing the high-temperature and pressure gas from the containment to the outside through a pipe line while capturing and scrubbing radioactive gases and aerosols. The FCVS should decrease the pressure inside the containment by releasing steam and non-condensable gases to maintain the integrity of the containment during a severe accident. In addition, the performance of the filtration should be considered in the FCVS design so that radioactive materials such as cesium and radioactive aerosol are not released outside the environment. In addition, the FCVS was designed to operate passively. Accordingly, the FCVS immediately copes with increasing the pressure inside the containment and prevents the failure of the containment. At this time, the flow rate of the discharge gas is changed according to the sliding pressure inside the containment to make a continuous operation in a wide pressure range. In addition, the FCVS does not need off-site power because of the passive operation. For this reason, the FCVS can operate in perfect working order to maintain the integrity of the containment during a long SBO such as the accident in the Fukushima nuclear power plant. Furthermore, the FCVS was designed to prepare for an earthquake. However, still there is a risk of early containment failure due to a steam explosion, direct containment heating, or hydrogen explosion. In addition, the bypass sequence such as a steam generator tube rupture or interfacing system LOCA cannot be mitigated by an installation of the FCVS. Additional provisions have to be made to mitigate the consequences of bypass sequences.

The Republic of Korea has 24 reactors (19 PWR, 4 PHWR) at four sites (Hanbit (6), Hanul (6), Kori (6), Wolsong (2 PWR, 4 PHWR)) providing about 30% of its electricity. The Fukushima accidents triggered a discussion on the need to protect from a containment failure by an over pressurization and mitigate an uncontrolled release of activity into the environment. In the Wolsong-1 PHWR unit, the installation of the first FCVS (High Speed Sliding Pressure Venturi (HSSPV) type) was completed at the end of 2012 (during the preparation to acquire approval for a continued operation after a 30 year operation). The remaining 23 Korean NPP units (20 PWR, 3 PHWR) are planned to be equipped with the FCVS.

The Republic of Korea is now developing a wet-type FCVS for light water reactors as a depressurization system of the containments under a severe accident. To verify the performance of the FCVS, tests [2] were performed in the scaled test facilities such as ACE and JAVA. The FCVS test facility was designed and is under construction in the KAERI site. The FCVS test facility consists of a test vessel, thermal-hydraulics, and aerosol/iodine generation and measurement parts. In the same manner, an integral test facility is designed to verify the performance of a proposed FCVS. The proposed FCVS system consists of a pool venturi scrubber, a droplet separator, a particulate filter, and a molecular sieve filter.

KAERI is now preparing an integral test facility to verify the performance of the developed FCVS. The FCVS test facility is a scaled test facility through scaling analyses with all of the following components including the pool venture scrubber, droplet separator, particulate filter, and molecular sieve filter. KAERI has been also performing MELCOR analyses to evaluate the effects of the thermal hydraulic conditions on aerosol scrubbing and hydrogen concentration in the “postulated” FCVS in an OPR1000 (Optimized Power Reactor 1000) as the target nuclear power plant. The OPR1000, a pressurized water reactor (PWR), has been developed by incorporating the latest technologies and the experiences in the construction and operation gained from previous nuclear power plants, including the EPRI Advanced Light Water Reactor (ALWR) requirements. The design features of the OPR1000 are a two-loop RCS design and a 2815 MW core thermal power.

2. PERFORMANCE TEST FOR NEW DESIGNED FCVS

2.1. Design features of FCVS

The Republic of Korea is now developing a domestic FCVS for light water reactors as a depressurization system of the containments. As shown in Fig. 1, the Korean FCVS consists of four main components. The venturi scrubber including the pool, droplet or particle separator, metallic fiber filter (MFF), and molecular sieve are the main components of the FCVS.

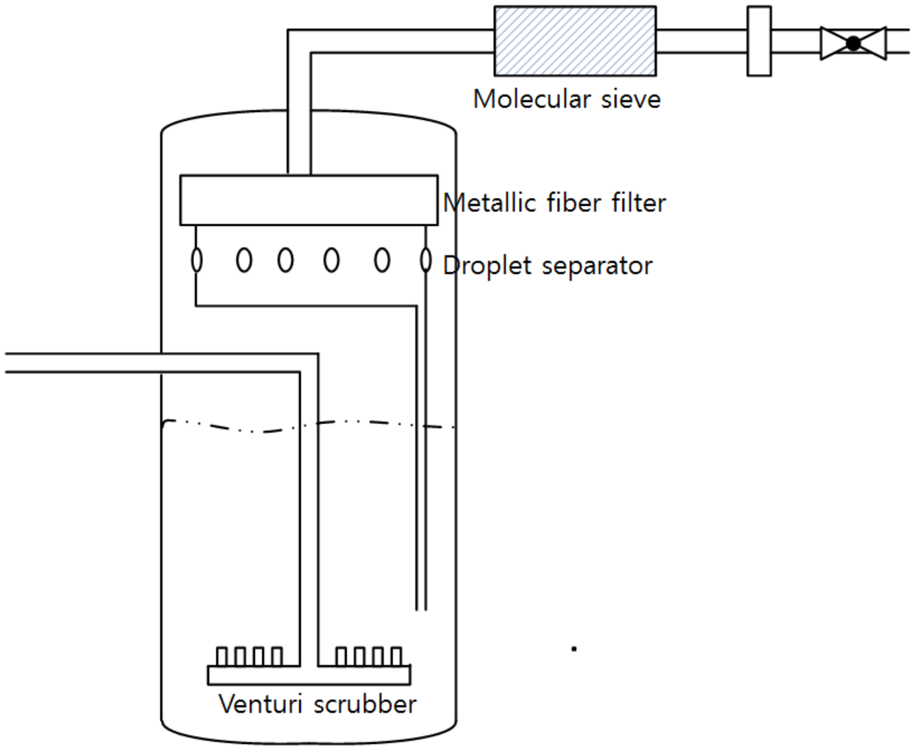

FIG. 1. Conceptual design of proposed CFVS system.

During a severe accident, if the pressure inside the containment reaches the desired set point, the vent valve pipe will be opened by the operator or through the operation of the rupture disk. Then, the radioactive gas mixture which consists of steam, gases, and aerosols are discharged into the filtration tank filled with water through the vent pipe. The species for the fission product, sizes, and event sequences leading to the release of radioactive gases are discussed in reference 3. While the mixture gas passes the venturi nozzle, a swarm of droplets coming through the hole in the throat of the nozzle is supplied to the mixture gas. During this process, impaction occurs between the gas and the droplet due to a velocity difference, and therefore the radioactive materials are removed from the gas. The mixture gas including the aerosols will pass through a pool of water. While the gases pass through the pool in the form of bubbles, scrubbing process occurs. It will result in a scrubbing of gas such as elemental iodine and aerosols. The scrubbing of gas is by a mass diffusion process, while the scrubbing of aerosols is driven by the inertial impaction and settling in the gas bubbles. The gas bubble dynamics and thermal hydraulic condition of the pool are dominant factors affecting the decontamination [4, 5]. In addition to the bubble behavior, the pool chemistry is highly important for the decontamination.

Most of the radioactive aerosol and iodine are removed in the venturi scrubber however the gas passing the pool still includes tiny aerosols and entrained water droplets. In the droplet separator, the heavy particles and hygroscopic moisture of the gas are removed by centrifugal force. As shown in Fig. 1, a number of droplet separators are contained in a closed space like a box while the inlet of each droplet separator is exposed to the gas space such that the mixture gas containing the water droplets and aerosols can pass through the droplet separators. The box has several holes for the droplet separator inlet. The exit of the droplet separator is contained in the closed space, at the bottom of which a drain line is connected deep into the pool such that the droplets will drain into the lower part of the water pool. The addition of chemicals is known to augment the scrubbing process for the elemental and organic iodine gases.

In the next stage, the metallic fiber filter removes very tiny aerosols and droplets whose sizes are less than 2 μm. The metallic fiber filter can be a pre-filter and an aerosol filter. The pre-filter consisting of metallic fiber with a thickness of 6 – 10μm is installed in front of the aerosol filter. The aerosol filter installed after the pre-filter consists of metallic fiber with a thickness of 2 – 8 μm in order to remove the tiny particles. The thickness of the metallic fiber is a parameter affecting the efficiency of the capturing performance; the performance is better as the thickness decreases.

The molecular sieve box is located outside the FCVS vessel to remove the organic iodine. The gas passing the metallic fiber filter goes into the molecular sieve in order to remove the iodine. The molecular sieve plays a role in filtering the organic iodine. The molecular sieve consists of synthetic zeolite, and silver zeolite is used so that the radioactive iodine is removed during the high-temperature process. There are a lot of uniform tiny holes, and therefore the gas coming from the molecular sieve will pass the holes to remove the iodine. Finally an orifice and a valve are installed after the exit of the molecular sieve. The location of the orifice has to be optimized because it will change the volumetric flow into the molecular sieve, which would affect the size of the molecular sieve, which is quite expensive as it contains Ag.

The target performance of the filtering efficiency would be as follows:

- Decontamination factor on aerosol : 10000;
- Decontamination factor on elemental iodine : 1000;
- Decontamination factor on organic iodine: 50.

2.2. Test category for performance test

The following test category need to be performed to verify the performance of the FCVS. A brief summary of test and measurement requirement is described in Table 1.

Thermal hydraulic test: An investigation into the thermal hydraulic phenomena in the FCVS will be performed without iodine/aerosol feeding. During tests, the measuring parameters are the change of the scrubber pool height, the absolute, gauge pressures along the test section, and the temperature of the pool, the gas, and the wall. To measure these parameters within the operating pressure range of the FCVS, at least two cases, which are high/low inlet pressure conditions with, will be performed. In addition, a test to observe the non-condensable gas effect in the scrubber pool will be performed, using the steam/nitrogen or steam/air gas. This result will be used to finalize detailed test planning.

Dynamic test under high pressure gas discharge from containment: During a severe accident, if the pressure inside the containment is reached at the specific opening value, the vent pipe will be opened and high-pressure gas will release into the FCVS. Through a dynamic test, it will be confirmed that the components in the FCVS are functioning normally during the transient process. It also needs some time to reach normal operation from a start-up. Thus, the measuring parameters are the change of the scrubber pool height, the pressure at each component, and the time to reach the normal operation condition from a start-up. The non-condensable gas effect will be investigated.

TABLE 1. SUMMARY TEST AND MEASUREMENT REQUIREMENT

	Design Requirement and Measurement requirement
Extended operation	– Since the pool has to be heated up to the saturated condition, the duration of the test has to be estimated. – During the heat up process, the steam has to be discharged either to the ambient or to the suppression pool – The aerosols at the exit of the FCVS should be collected and should not be discharged to the ambient. – The duration of the test at saturated pool condition should be estimated to determine the size of the suppression tank. It could be in the range of hours. – Extended operation for the re-suspension test would require more hours for the operation – The test with iodine would require closed loop operation for the safety reason. However, closed loop system would result in a very big facility like cooling tower
Re-suspension/ Re-entrainment of aerosols from the pool	– Extended operation of aerosol test approaching the maximum particle density in the pool – Measure DF just before MFF – Monitor Liquid droplet entrainment and removal by droplet separator
Retention of Iodine	– Iodine gas concentration at inlet and exit of FCVS at given pool height, temperature, pH, and mixture velocity – Contaminated pool water should be treated carefully for the safety – Iodine concentration at the exit of the FCVS should be monitored, if it is above the safety level the test has to be terminated – The pipe should be coated to minimize iodine deposit on the piping.

Aerosol removal test: In a high inlet pressure condition, the collection efficiency of the scrubber and droplet separator increases while the efficiency of the metal fiber filter and molecular sieve decreases. In contrast, the collection efficiency of the scrubber and droplet separator decreases while the efficiency of the metal fiber filter and molecular sieve increases in the low inlet pressure condition. Thus, an aerosol removal test will be performed under the same conditions as a thermal hydraulic test. The decontamination factor (DF) of the FCVS is measured. All tests are planned lasting 4-5 hours to collect an appropriate number of aerosol samples at the inlet and especially at the outlet, respectively. To simulate the most severe case, a test will be conducted at a maximum aerosol concentration condition (less than $10 g/m^3$). Non-soluble particles such as SiO_2 which is the size of AMMD 3 μm will be used.

Furthermore, because it is hard to remove a 0.7 μm particle, one aerosol removal test using small particles (near 0.7 μm) will be conducted.

Re-suspension of aerosol test: As a change of the vessel temperature and pressure, the particles are re-suspended from the scrubber pool and surface of the component. In this test, it will be confirmed that the re-suspension of aerosol is removed in the FCVS. Most of this test will be conducted after an aerosol removal test.

Elemental/organic iodine removal test: The iodine removal test will be performed in the high inlet pressure with high and low steam mass fraction using the steam/nitrogen to observe the non-condensable gas effect in DF of the test facility. In order to simulate the most severe case, the test will be performed at the maximum iodine concentration condition. All tests are planned lasting 4-5 hours to collect an appropriate number of I_2 samples at the inlet and especially at the outlet, respectively. Although the remove efficiency of iodine in scrubber pool is dependent on pH, this effect would not be observed in this performance test. However, the pH scale of scrubber solution is maintained higher than 10 using the chemical additive. Furthermore, the pH of the scrubber solution will be monitored online during the test by a pH electrode device. Also, the temperature of pool, the mixture level swell, and the pressure, will be is measured.

Re-volatilization of iodine test: As the change of the vessel temperature and the pressure, the iodine is re-volatilized from the scrubber pool and a surface of each component. In this test, it will be confirmed that the re-volatilization of elemental/organic iodine is removed in FCVS. Most of this test will be performed after elemental/organic iodine removal test.

3. PERFORMANCE EVALUATION OF FCVS

3.1. Target plant and modeling

To simulate a severe accident in a nuclear power plant, we choose the OPR 1000 with a thermal power of 2815 MWt under a Station Blackout (SBO). The OPR 1000 is a pressurized water reactor including two steam generators and four reactor coolant pumps, where the reactor and turbine are tripped, and the pumps of the Main Feed Water (MFW) and Auxiliary Feed Water (AFW) are stopped, and Main Steam Isolation Valve (MSIV) is closed as soon as an SBO occurs [6]. A design pressure of the containment building of OPR1000 is 500 kPa, and the containment building could fail at 1010 kPa [7].

The reliability of the input file of the MELCOR computer code was confirmed by comparing the thermal-hydraulic behaviors in a normal operation and a Design Based Accident (DBA) with the values presented in a Final Safety Analysis Report (FSAR) [8]. Until now, the MELCOR computer code has not included a package to assess the FCVS in a severe accident. We are trying to simulate the thermal conditions in the "postulated" FCVS using the previous models in the MELCOR computer code.

The modeling of the postulated FCVS is described below [6]: A vessel for the FCVS was connected with the containment building and the environment through a venting pipe and an exhaust pipe, respectively, as shown in Fig. 2. In the MELCOR computer code, a Control Volume (CV) presents the containment dome, a vessel for the FCVS, and the environment, and the Flow Path (FL) simulated the venting and exhaust pipes, where the Elevation (EL) was based on a center line of a hot leg connected with a reactor vessel in the containment building. A cylindrical vessel having 3 m in diameter and 6.5 m height contains 21 tonnes of water as a scrubbing solution in the FCVS. The venting pipe at the inlet of the FCVS has the pool scrubbing model in the Radionuclide (RN) package, and the exhaust pipe at the outlet involves a filter model to simulate the decontamination. Both pipes are 250 mm in diameter and 6 m length. In this study, the severe accident progress over time was assessed by this modeling mentioned above, where we did not consider the burning of a gas mixture and the pool chemistry model in the control volumes.

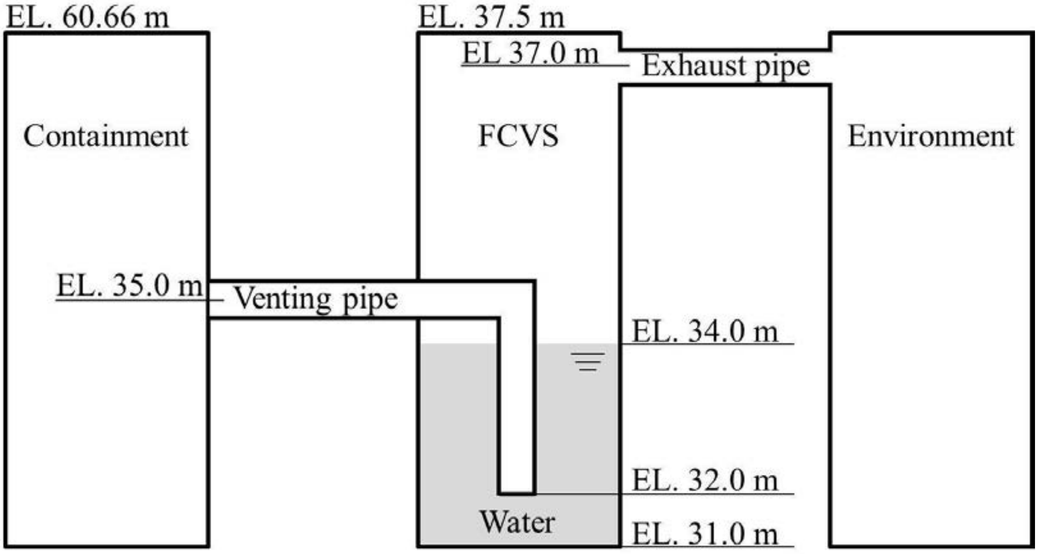

FIG. 2. Modeling of the postulated FCVS in the MELCOR computer code [6].

3.2. Thermal condition

This study calculated the pressure and temperature in the containment building of an OPR-1000 under an SBO using the MELCOR computer code as shown in Fig. 3. The pressure and temperature were increased continuously after the start of the SBO, because a large amount of steam and gases generates from the chemical reaction between the high-temperature molten core material and coolant or structures. The pressure in the containment building was reduced as soon as it approached 500 kPa as a set value of the operation of the FCVS. This means that steam and a gas mixture generated in the containment building are released into the FCVS. The gas temperature was also decreased at about 33 hours, i.e., the start of the FCVS operation, and the reduction rate was then slower than that of the pressure owing to the remaining decay heat in the containment building. After the operation of the FCVS, the decrement rate of the partial pressure of steam is slower than that of the pressure in the containment building. Although steam condensation can occur in the containment building owing to the pressure drop, the amount of steam generated from the pools in the containment building would be bigger than that of the steam condensation. As shown in Fig. 3, the water saturation temperature is lower than the atmosphere temperature when the enclosed containment building becomes open to the FCVS.

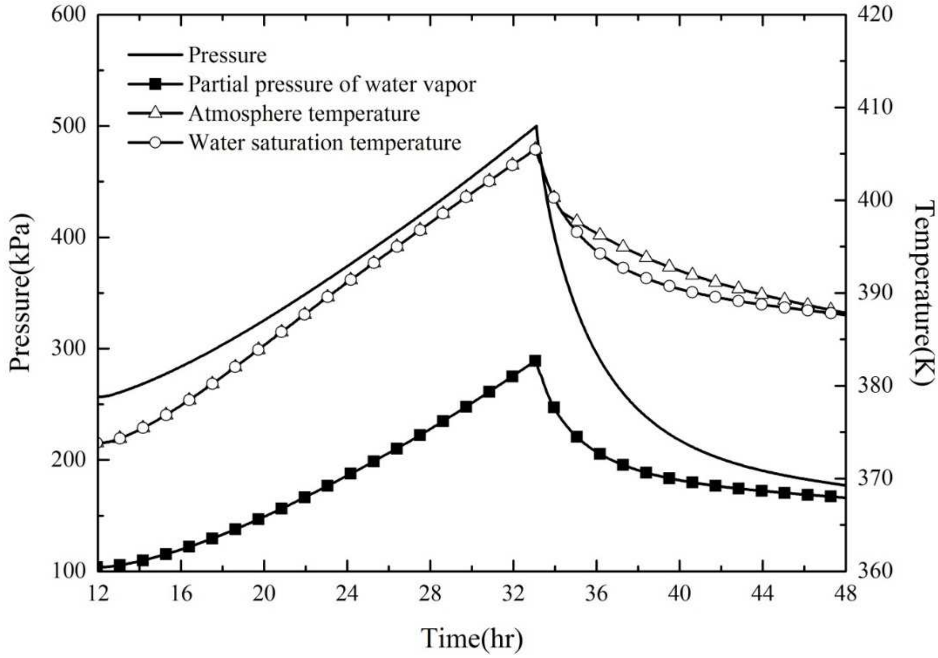

FIG. 3. Pressure and gas temperature in the containment building [9].

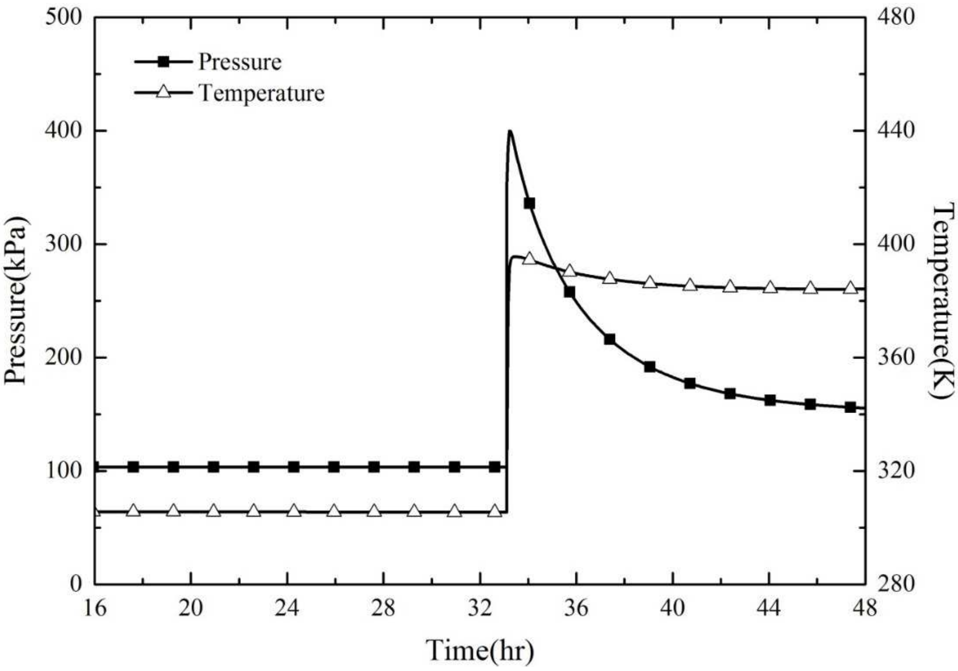

FIG. 4. Pressure and gas temperature in the postulated FCVS [9].

Figure 4 shows the pressure and temperature in the postulated vessel of the FCVS. Before the operation of the FCVS, the initial conditions of pressure and temperature were 103 kPa and 305 K, respectively, in the FCVS. The pressure jumped suddenly at about 33 hours when the pressure in the containment building approached 500 kPa as shown in Fig. 3, because steam and gases generated in the containment building were discharged into the FCVS. The injection of steam and gases is induced by the pressure difference between the containment building and the FCVS. The pressure peaked at 400 kPa, which is the difference between the set pressure of the containment building of 500 kPa and the initial pressure of the FCVS of 103 kPa. After the pressure approached the peak value, it was reduced in time. The decontaminated materials that sequentially pass through a scrubbing solution and filters in the FCVS are continuously discharged out of the FCVS. The gas temperature also jumped up to 395 K in the same manner as the pressure increment. After the operation of the FCVS, a relatively low reduction rate of the atmosphere temperature indicates that the high-temperature steam and gases pass through the FCVS with little heat transfer. As a result, steam and a gas mixture releasing from the containment building to the FCVS experience a big change in the thermal-hydraulic condition, which can affect the composition ratio of a flammable mixture.

3.3. Hydrogen issues

The partial pressures of steam and non-condensable gases in the postulated vessel of the FCVS are shown in Fig. 5. The volume concentrations of steam, hydrogen, and air were calculated as 3 %, 14 %, and 62 %, respectively, at 119200 seconds, i.e., as soon as the FCVS operates. The volumetric concentration of hydrogen was increased from 6 % in the containment dome to 14 % in the postulated vessel of the FCVS at the initial operation of the FCVS, while the concentration of steam was reduced from 58 % in the containment dome to 3% in the FCVS, because the steam condensation can occur at a relatively low-temperature scrubbing solution in the FCVS. Our previous study observed the steam condensation at the initial operation of the FCVS. The initial mass of the scrubbing solution was sharply increased. This composition ratio change can cause a gas mixture to enter the combustion zone in the Shapiro diagram, which indicates the possibility of hydrogen burn, as shown in Fig. 6.

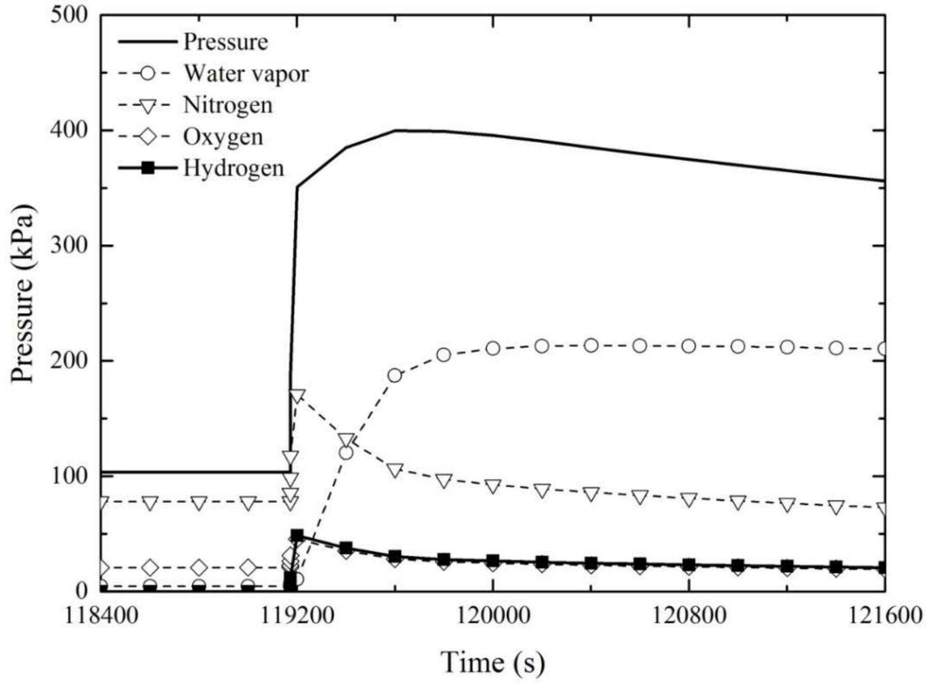

FIG. 5. Partial pressure of vapor and gases in the postulated FCVS [9].

There are uncertainties in the burn limit, as shown in Fig. 6, because it does not consider the effect of the size of a vessel, pressure, and temperature. Hydrogen injected with high pressure will not mix well with the ambient air in the FCVS, and hydrogen combustion can then occur locally. This can narrow the region of the burn limit. In addition, the burn limit depends on the gas temperature around the flame propagation when hydrogen combustion occurs. It is necessary to consider the heat transfer between the gas mixture and vessel wall of the FCVS. Although there are uncertainties to defining exactly the burn limit on our postulated FCVS, the increment of hydrogen concentration assures the high potential for combustion, which can threaten the integrity of the FCVS.

Figure 5 shows that the volume concentration of hydrogen was reduced in the FCVS after approaching the peak value at the initial operation of the FCVS. While the volume concentration of hydrogen decreased relatively, that of steam increased because the amount of steam released continuously into the FCVS cannot be fully condensed in the FCVS consisting of a vessel having a limited volume. The volume concentrations of steam, hydrogen, and air are 47 %, 8 %, and 34 %, respectively, in the postulated vessel of the FCVS at 400 seconds after the operation of the FCVS.

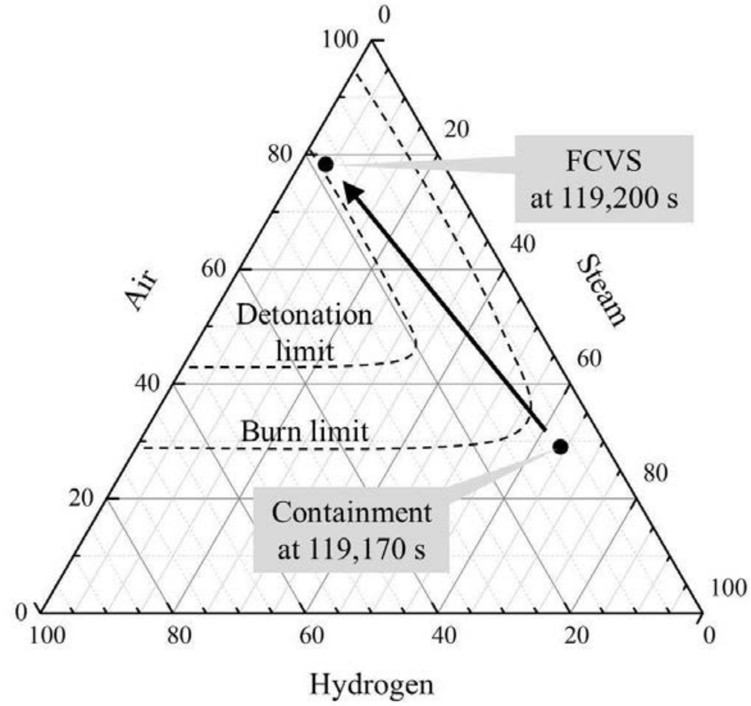

FIG. 6. Possibility of hydrogen combustion at the initial operation of the postulated FCVS.

3.4 Aerosol pool scrubbing

The decontamination factor of a cesium iodide (CsI) aerosol on a scrubbing solution in the FCVS is defined by the ratio of the input mass of CsI aerosol to its output mass. The MELCOR computer code can calculate the mass of the CsI aerosol in the atmosphere and a pool in the postulated FCVS vessel and the outside environment, as shown in Fig. 2. The input mass of CsI aerosol into a scrubbing solution in the FCVS is the sum of CsI aerosol mass in a pool in the FCVS vessel and the output mass of CsI aerosol from a scrubbing solution, where the output mass can be estimated by CsI aerosol mass in the FCVS vessel atmosphere plus that in the outside environment. A filter model applied on an exhaust pipe was removed to estimate the decontamination factor on a scrubbing solution. To assess the thermal hydraulic effect on cesium iodide (CsI) aerosol scrubbing in the FCVS (Filtered Containment Venting System), the variation of aerosol mass during the accident progress process was calculated by the MELCOR computer code.

Figure 7 shows the input and output of the accumulated CsI aerosol mass on a scrubbing solution in the FCVS in the left axis, where the input indicates the accumulated mass of CsI aerosol released from the containment building to a scrubbing solution, and the output presents the accumulated mass of CsI aerosol discharged from a scrubbing solution in the FCVS. In addition, the right axis of Fig. 7 shows the decontamination factor of CsI aerosol on a scrubbing solution in the FCVS. The output mass is the sum of accumulated mass of CsI aerosol in the FCVS vessel atmosphere, and that in the outside environment. The input mass can be calculated by the accumulated mass of CsI aerosol in a scrubbing solution plus the output mass. To estimate the decontamination factor of CsI aerosol on a scrubbing solution in the FCVS, the input mass was divided by the output mass. The input mass increases continuously from 33 to 64 hours, where the FCVS operation started at 33 hours, and a scrubbing solution was completely evaporated at 64 hours. As soon as the pool level approaches the bottom of the FCVS vessel at 64 hours, the inlet mass falls sharply, and its variation then follows the accumulated mass of CsI aerosol in the outside environment. The output mass in Fig. 7 also depends on the accumulated mass of CsI aerosol in the outside environment, where the amount of the accumulated mass of CsI aerosol in the atmosphere in the FCVS vessel is relatively too small. The output mass in Fig. 7 starts to increase when the pool level is reduced by the level of a venting pipe exit initially submerged in a scrubbing solution in the FCVS vessel. The accumulated output mass is unvaried from about 90 hours because the amount of CsI aerosol mass generation in the containment building under a severe accident is constantly decreasing from 72 hours.

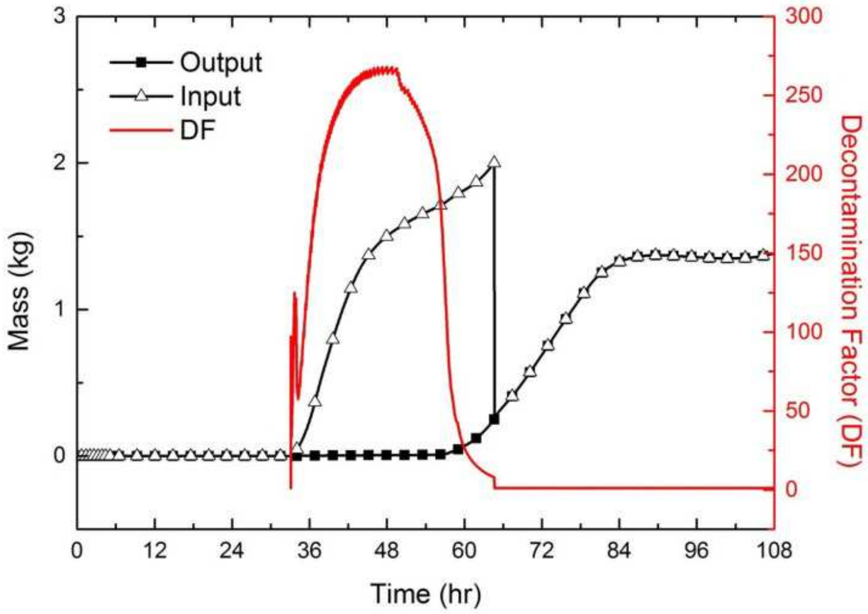

FIG. 7. Decontamination factor calculated by input and output of CsI aerosol mass on scrubbing solution [10].

The decontamination factor of CsI aerosol on a scrubbing solution in the FCVS presented in the right axis of Fig. 7 starts to dramatically increase at 33 hours as soon as the FCVS operates. It peaks at 50 hours, where the pool temperature in the FCVS vessel reached its saturation temperature at that time, and it then decreases sharply by unity, i.e., no scrubbing of CsI aerosol in a pool in the FCVS, at 64 hours, where a scrubbing solution was completely evaporated at 64 hours. In the early FCVS operation, the condensation of high-temperature steam released from the containment building to a room-temperature scrubbing solution in the FCVS vessel occurs due to the temperature difference. Steam condensation can enhance the ability to capture particles in a scrubbing solution. In addition, aerosol particles can stay longer in a scrubbing solution because steam condensation increased the pool level. Opposite the steam condensation in the early FCVS operation, the evaporation of a scrubbing solution took place owing to a consecutive supply of high-temperature steam from the containment building. The evaporation negatively affects the particle scrubbing in a FCVS pool, where the pool level was continuously decreased from 36 to 64 hours. In addition, the scrubbing depth also decreases during the pool evaporation, i.e. the time for particles staying in a scrubbing solution becomes short. The decontamination factor of CsI aerosol on a scrubbing solution can be finally dependent of the thermal hydraulic conditions in the FCVS vessel.

4. SUMMARY AND FUTURE WORKS

A Filtered Containment Venting System (FCVS) is one of the strategies maintaining the integrity of the containment by releasing the high-temperature and pressurized gas from inside the containment to outside the pipe line by capturing and scrubbing radioactive gases and aerosols during a severe accident in a nuclear power plant. During the releasing gas through the FCVS, fission products such as radioactive aerosol and iodine are filtered simultaneously by the filtration system of the FCVS to prevent leakage of the radioactive material into the environment.

An integral test facility is prepared to verify the performance of the developed FCVS. The FCVS test facility is a scaled test facility by scaling analyses with all of the following components included: pool venture scrubber, droplet separator, particulate filter, and molecular sieve filter. The test facility consists of a test vessel, thermal-hydraulic, and aerosol/iodine generation and measurement parts. Tests to quantify the performance of the FCVS will be performed such as a thermal-hydraulic test, aerosol removal test, elemental iodine removal test, organic iodine removal test, re-suspension of aerosol test, re-volatilization of iodine test, and dynamic test under a high-pressure gas ejection from the containment.

The effects of thermal hydraulic conditions on aerosol scrubbing and hydrogen concentration in the postulated FCVS were also estimated by the MELCOR computer code under a station blackout in an OPR1000, as a target nuclear power plant. The decontamination factor of metal iodide aerosols, especially, cesium iodide (CsI), on a scrubbing solution in the FCVS was calculated when the pressure in the containment building approaches 5 bars. The possibility of hydrogen combustion in the initial operation of the FCVS was also evaluated. Further studies on the modeling of the FCVS are required to enhance the reliability of the calculated results using the MELCOR computer code. We used a multi-hole sparger to simulate nozzles immersed in a scrubbing solution, and applied a pool scrubbing model and filter model for the decontamination features, which can affect the thermal-hydraulic behavior in the FCVS. The effect of the vessel size on the hydrogen behavior and the flame propagation in the FCVS should be considered to reduce the uncertainties of the burn limit. Furthermore, the assessment of the inert gas mixture or preheating in the FCVS vessel is necessary to mitigate hydrogen risk.

ACKNOWLEDGEMENTS

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THE EUROPEAN PASSAM PROJECT: EXPERIMENTAL STUDIES FOR ATMOSPHERIC SOURCE TERM MITIGATION WITH FOCUS ON FILTERED CONTAINMENT VENTING SYSTEMS

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Abstract. The PASSAM (Passive and Active Systems on Severe Accident source term Mitigation) project has been launched in the frame of the 7th framework programme of the European Commission. Coordinated by IRSN, this four year project (2013 – 2016) involves nine partners from six countries having a strong experience on severe accidents: IRSN, EDF and university of Lorraine (France); CIEMAT and CSIC (Spain); PSI (Switzerland); RSE (Italy); VTT (Finland) and AREVA GmbH (Germany). It is mainly of an R&D experimental nature, aiming at studying phenomena that, under severe accident conditions, might have the potential for reducing radioactive atmospheric releases to the environment. Improvements of existing source term mitigation devices (pool scrubbing systems; sand bed filters plus metallic pre-filters), as well as innovative systems which may achieve larger source term attenuation (Acoustic agglomeration systems; High pressure spray agglomeration systems; Electric filtration systems; Improved zeolite filtration systems; Combined Filtration Systems) are studied. After two years and a half, the current status of the project is as follows. A state of the art report has been completed, allowing a precise definition of the experimental test matrices in direct link with the identified gaps of knowledge. The experimental work is in progress and in some domains (e.g. pool scrubbing, use of zeolites for iodine trapping, combined wet and dry filtrations systems, etc…) major results have already been obtained. An open workshop mainly targeted to R&D organizations, to National Safety Authorities and their Technical Support Organizations, to utilities and to vendors, was held on February 26th, 2014 in Madrid. A second workshop will be organized at the end of the project, in December 2016 or early 2017. Simple models and/or correlations should result from in-depth analysis of PASSAM experimental results. Once implemented in severe accident analysis codes, these models should allow enhancing the capability of modelling Severe Accident Management measures and developing improved guidelines. The project's final outcomes will constitute a valuable database which may be strategic for helping the utilities and regulators in assessing the performance of the existing source term mitigation systems, evaluating potential improvements of the systems and developing severe accident management measures.

1. INTRODUCTION

After the Fukushima-1 accident one of the main concerns of the nuclear industry has been the search for improved atmospheric source term mitigation systems. The motivation underneath stems from two major evidences: venting of the containment building might be an essential accident management measure in order to prevent the loss of its mechanical integrity, but this containment venting might result in substantial radioactive releases if no efficient source term mitigation system is implemented. Many countries have recently considered the implementation of Filtered Containment Venting Systems (FCVS) as a means to further enhance their Nuclear Power Plants (NPPs) safety; while some others, like Sweden, Switzerland, Finland, Germany and France, had already adopted that measure in their NPPs.

The renewed interest for FCVS has pushed new international activities, like writing a status report by the working group on the Analysis and Management of Accidents of the CSNI (Committee on the Safety of Nuclear Installations) of the OECD-NEA [1], or launching the European Commission (EC) project on “Passive and Active Systems on Severe Accident source term Mitigation” (PASSAM) [2].

This paper gives an overview of the objectives, work structure, current status and main results of the PASSAM project two years and a half after its starting date.

2. PASSAM OBJECTIVES

The industrial FCVS are essentially based on two approaches: “dry” and “wet” systems. In dry systems, large trapping surfaces are provided either by gravel or sand beds, or by metal fibers or by molecular sieves. In wet systems, trapping occurs in a liquid (water plus additives) pool as a consequence of several removal mechanisms the efficiency of which depends mainly on thermal-hydraulic conditions. These systems can be enhanced by including venturi scrubbers: water droplets injected in the gas stream capture aerosol particles and make them more easily trapped by solid filters or liquid pools.

These systems, some of which are installed on NPPs, have been well characterized as regards aerosol retention efficiency, but to a far lesser extent as regards volatile iodine retention. In this context the Advanced Containments Experiment (ACE) can be considered as the largest international test programme [3]. Since then, the potential high contribution of gaseous organic iodine to the radioactive release in case of a loss of the containment building leak-tightness has been evidenced. Additional tests have been performed on FCVS, and improvements of the systems have been proposed for organic iodide retention in water pools. Nevertheless the research on this specific aspect needs to be complemented. A lack of knowledge also appears clearly for organic iodine retention in dry systems. Finally, gaps may also remain in the investigation of systems performance that have not been properly addressed, either because evolution of anticipated working conditions and/or because of new insights coming from recent research on fission product behaviour. Hence, some specific experiments could provide meaningful insights into the systems performance that could result in an optimization of their working conditions.

In parallel, several alternative and innovative approaches are appearing in the literature or are even already proposed by some vendors. Electric filtration systems are already widely used in industry, out of the nuclear field, for separation of particles, for instance, from flue gases of coal-fired power plants or from product gases in iron manufacturing or in cleaning indoor air applications. There are also a number of proposed solutions based on molecular sieves (improved zeolites, metal organic frameworks, etc.) that are widely used as filter in many industrial processes. Many kinds of zeolite exist; the key point would be to find the most efficient zeolite for iodine species in severe accident relevant conditions. Another promising innovative solution consists of the combination of wet and dry filter systems. Besides, as it is well known that, generally speaking, the filtration systems are less efficient for aerosol particles of some tenths of microns, some systems are being developed, aiming at agglomerating aerosol particles in order to get bigger particles which will be better filtered. Implementing such a device upstream of the filtration system would overcome the decrease of filtration efficiency of sub-micron particles.

Another subject which has poorly been studied is the long term behaviour of the trapped fission products (i.e., potential for re-vaporisation and/or re-entrainment of radioactive material) due to surrounding conditions relevant to an accident and more especially, due to continuous irradiation, coupled to continuous flow-rate (if the venting system remains open), temperature, humidity, etc… Indeed the re-entrainment of trapped aerosols was tested in different small scale facilities and, at larger scale, in the international

ACE programme where the aerosol loaded filters (dry and wet filters) were operated with clean gas and the aerosol concentrations of the gas downstream of the filters were measured. Nevertheless, no data on long term re-vaporisation under irradiation, both for aerosols and for gaseous iodine forms, could be found in the open literature.

In this context, the PASSAM project has been built to explore potential enhancements of existing source term mitigation devices and check the ability of innovative systems to achieve even larger source term attenuation. Heavily relying on experiments, the PASSAM project aims at providing new data on the capability and reliability of a number of systems related to FCVS: pool scrubbing systems, sand bed filters plus metallic prefilters, acoustic agglomerators, high pressure sprays, electrostatic precipitators, improved zeolites and combination of wet and dry systems. Nonetheless, the scope of some of the ongoing research — as fission products and aerosol retention in water ponds — goes beyond FCVS and might be applied for accident situation other than containment venting.

The main outputs of the project will be:

- a relevant extension of the current database on these existing or innovative mitigation systems,
- a deeper understanding of the phenomena underlying their performance and models/ correlations that allow modelling[j] of the systems in accident analysis codes, like ASTEC [4],
- an estimation of the order of magnitude for source term reduction for each filtration system and suggestions for improved filtration systems[k].

3. PASSAM WORK STRUCTURE

The PASSAM project launched under the 7th framework programme of EURATOM has started in January 2013. Participated by 9 organizations (IRSN, EdF and University of Lorraine, France; CIEMAT and CSIC, Spain; PSI, Switzerland; RSE, Italy; VTT, Finland; and AREVA GmbH, Germany) and coordinated by IRSN, the project involves nearly 400 person-months and the associated cost is estimated to be more than 5 M€.

The PASSAM project is organized into 5 Work Packages (WP) that are described next:

- WP1: (MANAG) Project Management (leader: IRSN). Project scientific, administrative and financial coordination, quality management included.
- WP2: (THEOR) State of the art and modelling (leader: CIEMAT). Analytical work supporting both test matrices definition and development of models and/or correlations from the experimental results.
- WP3: (EXIST) Existing filtration systems (leader: PSI). Two types of devices are experimentally studied: pool scrubbers (WP3.1) and sand bed filters - plus metallic pre-filters(WP3.2).
- WP4: (INNOV) Innovative filtration systems (leader: VTT). Experimental campaigns are conducted on devices that might enhance mitigation of source term, either through pre-agglomeration of the radioactive load that would reach the filter

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[j] The implementation of models in Severe Accident codes is out of the scope of PASSAM.
[k] Note that the hydrogen risk within the FCVS, which has to be taken into account, is not studied in PASSAM which focuses more on the retention of aerosols and gaseous iodine and on their potential release at mid or long term.

stage (acoustic agglomerators (WP4.1) and high pressure sprays (WP4.2)) or alternative filter systems (electric precipitators (WP4.3), improved zeolites (WP4.4) and combination of wet and dry filters(WP4.5)).
— WP5: (DKS) Dissemination of knowledge and synthesis (leader: IRSN). Matters concerning training of young researchers, PASSAM visibility, the communication of its results and the sharing of the experience gained.

4. PASSAM CURRENT STATUS

The experimental nature of the project and the final outcomes foreseen, make its planning to be rather linear as a whole, with three conceptual project phases:

— set-up of technical and organizational bases (2013);
— execution of experimental campaigns (mid-2013 to mid-2016);
— in depth analysis of the experimental results and project wrap-up (mid 2015 to end 2016).

The first phase included a literature survey on the existing and innovative systems to be experimentally studied in the project [5]. This survey confirmed, with more details, the anticipated gaps of knowledge and so it allowed optimizing the test matrices for each experimental work-package [6, 7]. It ended up with an open workshop held in Madrid (Spain) in February 2014 [8].

The experimental phase, which constitutes the largest part of the PASSAM activities, is presently ongoing and will last up to mid-2016. It will result with the development of robust and sound databases for each system tested. The main results already obtained are presented in the next section of this paper.

The last phase will focus on managing the project output in the best way so that nuclear community can easily benefit from the research conducted. In particular a final PASSAM workshop will be organized by end 2016 or early 2017 and the final synthesis report of the project will be available in open literature.

5. PASSAM MAIN EXPERIMENTAL RESULTS

5.1. WP3.1: Experimental studies of pool scrubbing systems (leader PSI)

This work package is, by far, the biggest one of the PASSAM Project. Its context, its status and its current outcomes are not reported in this paper but in a specific paper presented at this IAEA Technical Meeting [9].

5.2. WP3.2: Experimental studies of sand bed filters plus metallic pre-filters (leader IRSN)

Sand bed filters were decided to be implemented on the French PWRs in July, 1986 and all French NPPs in operation had been fitted with FCVS since mid-90's. Their initial design was based on a set of boundary conditions: containment pressure and temperature of 5 bar and 140 ºC, respectively; gas flow rate through the FCVS line of 3.5 kg/s; gas mixture composed of air (33 wt%), steam (29 wt%), CO_2 (33 wt%) and CO (5 wt%) (H_2 risk was addressed later on); aerosol concentration of 0.1 g/m^3 with an Aerodynamic Mass median Diameter (AMMD) of 5 µm. The initial objective consisted of obtaining a Decontamination

Factor (DF) of 10 for aerosol particles. The system was tested at two different scales: PITEAS and FUCHIA [10]. The PITEAS-filtration program enabled to determine the characteristics of the filter material as well as its domain of use, and check its ability to meet the filtering efficiency criterion under accident conditions. The full-scale FUCHIA program validated the thermo-hydraulic and filtering behaviour of the filtered venting line (without pre-filtration).

Then, it was decided to add a metallic pre-filter upstream of the sand filter, in the containment building.

According to the experiments performed, the minimal DF attributed to the sand bed plus metallic pre-filter system is:

- 1000 for aerosol particles (metallic pre-filter minimal DF assumed to be 10);
- 10 for inorganic gaseous iodine (due to deposition on internal surface of piping);
- 1 for organic gaseous iodine (no test done).

A synthesis of all theoretical and experimental studies performed for French FCVS design is presented in [11].

Despite the qualification of the system, two domains appeared not (or not enough) to be covered and are presently investigated in the PASSAM project:

- Filtration efficiency of gaseous molecular and organic iodines (IRSN). First laboratory scale experiments recently performed showed no trapping of gaseous molecular iodine (I_2) or of iodomethane (CH_3I) in the tested conditions on pure sand. On the contrary, when the sand is pre-loaded with compounds having chemical affinity with iodine like caesium or silver, a significant retention of gaseous molecular iodine (I_2) is observed. In parallel, a dedicated iodomethane Gas Chromatography analysis method was largely improved and allows now to reach extremely low detection limit for CH_3I (10 ppt = 10×10^{-12}) without pre-concentration. Next experiments planned in 2015 and 2016 will focus on I_2 and CH_3I retention by metallic prefilters.
- Mid/Long term stability of filtered fission products, in particular iodine, under severe accident conditions (temperature, flow-rate, pressure, humidity, irradiation) (IRSN). A first test the IRSN EPICUR facility allowing to work under irradiation (^{60}Co source) with labelled iodine (^{131}I) was performed very recently in order to check the long term stability of caesium iodide (CsI) aerosols trapped on a sand bed filter. The analysis is under progress. Other tests are foreseen by end 2015 and early 2016.

5.3. WP4.1: Experimental studies of acoustic agglomeration systems (leader CIEMAT – main contributor CSIC)

Enhancement of particle agglomeration would have the potential of improving mitigation for two reasons. Agglomeration of sub-micron particles in the containment building would result in larger particles that would be much faster depleted by sedimentation (settling velocity is proportional to aerodynamic diameter squared). The performance of the filtration devices would be improved by increasing particle sizes, since the efficiency of those systems show a minimum in the range of 0.1 to 0.3 microns.

Acoustic agglomeration (AA) was first experimentally investigated in the 1930´s and has been extensively studied over many years to understand its basic mechanisms. Its practical applications came a little later in the late 1940's with the development of powerful air-jet sound generators. Since then a broad research of potential applications have been investigated [12–15]. The development of acoustic high-intense chambers as pre-conditioners of filters was specifically tested in some of these works. These studies found out optimum working conditions as regards exposure times; particle concentrations and distributions in the agglomeration chamber. As for the ultrasonic field optimization in nuclear severe accident conditions, it is known to be largely dependent on particle size (micron/sub-micron) and humidity while no degradation should be expected under high pressure and temperature environments.

Therefore, the main challenge for a potential application of AA in nuclear severe accident conditions will consist of checking the system performance under the anticipated working conditions during a severe accident. In the PASSAM frame, following a series of qualification tests with an Acoustic Agglomerator (21 KHz, 300W/unit) developed by CSIC and implemented in the CIEMAT PECA facility, the experimental phase lasted from November 2014 up to May 2015. Working with SiO_2 and TiO_2 aerosols, the effects of number concentration, treatment time and aerosol polydispersion were studied. The SiO_2 aerosol concentration decreased up to a factor of 10 by the action of ultrasonic field for several initial concentrations, flow conditions and particle sizes. An increase up to 0.4 µm in size of the aerodynamic Mass Median Diameter (AMMD) was observed.

5.4. WP4.2: Experimental studies of high pressure spray agglomeration systems (leader RSE)

High pressure sprays have been used in multiple industrial applications [16], including NPPs (High Pressure Core Spray System of BWR/5 and BWR/6). This type of sprays works between 50 and 200 bar and can generate droplets smaller than 100 µm at velocities near 100 m/s. Under these conditions, in addition to clean-up and cool atmospheres, they can also foster growth of submicron particles [17]. However, few studies have addressed this potential capability and most investigation on sprays performance has been conducted for low pressure sprays [18]. Therefore, high pressure sprays application as an agglomerator in the pre-filter stage of a FCVS would require addressing generic studies for determining the agglomeration efficiency as a function of water pressure, initial aerosol concentration and aerosol size (monodisperse aerosol and mix of monodisperse aerosol).

These studies are currently on going in the frame of PASSAM using the RSE SCRUPOS facility and are planned up to mid-2016. The first results, to be confirmed, show that the aerosol depletion rate is a linear function of the water pressure and that it depends on the aerosol size. A model has been developed which provides a good agreement with experimental data, except for 0.5 µm aerosol needing further investigation.

5.5. WP4.3: Experimental studies of electrostatic precipitators (leader VTT)

Potential of electric filtration systems were discovered early in the 19th century, when corona discharge was found to remove particles from gas streams. Today many types of these filters exist including electrostatic precipitators (ESP), air ionizers and ion wind devices. Typically they are very efficient filtration systems with minimal resistance to the gas flow. They are applied for reduction of many industrial emissions, including coal and oil fired power plants, salt cake collection from black liquor boilers in pulp mills, and catalyst collection from fluidized bed catalytic cracker units in oil refineries

An ESP removes aerosols from gas flow due to forces induced by strong electric fields [19]. In case particles do not get high enough charge for their collection, a pre-charging water mist can be sprayed to enhance particle growth in diameter and charging: This system is called Wet ESP (WESP). The operation temperature of a typical WESP at atmospheric pressure is limited to 90 ºC and the pressure drop through the system is small. The reported energy requirement for the filtration of 1 m^3 of gas by ESP varies from 10^{-5} to 10^{-3} kW.h. The atomization of water is probably the main power drain of WESP. In case of the atomization of small droplets, it is approximately 400-800 W for a gas flow-rate of 1 m^3/s.

Typical collection efficiency of a WESP for particles is between 99 to 99.9 %. Nonetheless, collection efficiency can be increased by maximizing the strength of the electric field [20] and the residence time within the precipitator. Other factors affecting efficiency are dust resistivity, gas temperature, chemical composition (of the dust and of the gas flow), and particle size distribution. WESP performance has been hardly tested for gaseous iodine, although preliminary tests were conducted on gaseous iodine filtration by VTT in 2011. In this case higher efficiency might be reached by adding an oxidation stage (ozone generator) that turns gaseous iodine species into iodine oxide particles.

Application of WESP under the foreseen conditions of a nuclear severe accident poses two major challenges:

- Conditions still not explored, like radiation and high temperatures (>90ºC), and pressures (>1 bar);
- Filtration of iodine volatile species including organic iodides.

The status of the PASSAM experiments on this domain is as follows. The facility has been set-up and the experimental phase started in April 2014 using WESP at room conditions, then in more representative conditions. Using TiO_2 aerosols, a strong decrease of the trapping efficiency for voltages below 15 kV (negative) was evidenced. Some retention tests for gaseous and particulate iodine have also been conducted and the optimum parameters related to WESP operation have been determined (number of corona needles, effect of the flow-rate, etc.) and a preliminary fitting of curves to experimental data has been done. New tests are currently under way at higher temperature (70 °C) and in the presence of steam. In these experiments, the focus is on the retention of gaseous molecular iodine (I_2) and organic iodine (CH_3I).

5.6. WP4.4: Experimental studies of improved zeolites (leader IRSN – main contributor UniLor)

Zeolites are crystalline materials containing pores of molecular size (5–12 Å). They are composed of $[SiO_4]_4^-$ and $[AlO_4]_5^-$ tetrahedra linked by oxygen atoms. The substitution of the Si^{4+} by Al^{3+} leads to a negative charge, which is compensated by the presence of a metallic (M) cation of n valence ($M_{x/n}$ $[AlO_2]_x$ $[SiO_2]_y$. zH_2O, where y/x should be higher than 1; i.e. more Si than Al tetrahedral). Several zeolite families are presently being used for different industrial processes: Mordenite, Ferrierite, Faujasite and others. A number of studies have been conducted to assess the zeolites capability for iodine retention [21–24] and among all the metallic cations tested (Cd, Pb, Na, Cu …) for iodine trapping, silver seems to give the best results for potential nuclear applications. Two types of silver-containing solids have been successfully tested for iodine trapping: silver mordenite (AgZ) and silver faujasite (AgX). The trapping mechanism seems to involve chemical redox reactions with some embedded Ag species, which results in the formation of a stable insoluble iodine AgI precipitate.

Investigation has given insights into the iodine capture by zeolites: structural and chemical features of zeolites heavily affect trapping; the Faujasite Ag-X adsorbents showed high decontamination factors (DF) for both molecular iodine and organic iodides, but their efficiency depends on steam and nitrous oxides presence; in Mordenite I_2 chemisorption is enhanced if metallic silver is present [25]; high humidity affects negatively organic iodide trapping; silver zeolites are robust and withstand harsh temperature, humidity and radiation conditions.

In spite of all the knowledge gathered concerning iodine trapping in zeolites, little information is available concerning impacts of zeolite structure and cation loading as well as the persistence of high iodine DF under oxidizing conditions resulting from radiation.

More precisely, the experiments in the frame of the PASSAM project are focused on:

- the optimization of the nature of the zeolite in order to avoid poisoning effects and to obtain high iodine DFs;
- the definition of optimal active sites for the trapping of iodine and organic iodides (deeper understanding of the trapping mechanisms);
- providing extensive and sound data on the effect of expected severe accident working conditions including influence of chemical effects (steam and gas contaminants) and potential irradiation effects on iodine trapping capacity.

The experimental programme started in 2013 and is foreseen up to 2016. The main results acquired up to now by the French University of Lorraine are as follows:

- Up to 20 metal-exchanged zeolites were prepared and characterized.
- Adsorption tests of CH_3I in gaseous phase under different conditions were performed. About the trapping efficiency, silver zeolites lead to the best adsorption performances in the absence of contaminants. In order to be in more realistic conditions the effect of some “contaminants” was studied: the effect of H_2O and CO appeared to be weak on the adsorption performances of the zeolites, while NO_2 lead to a significant alteration of the performance. So the experiments on retention capability of the zeolites in severe conditions are still going on. About the trapping stability, the formation of a highly stable AgI phase was evidenced as well as less stable products. The global behaviour of trapped iodine under irradiation and steam conditions will be checked in 2016 by IRSN.

5.7. WP4.5: Experimental studies of “combined” filtration systems (leader AREVA)

The AREVA FCVS is based on an extensive series of laboratory tests and further large-scale investigations. As a result, a combined retention process was selected, consisting of complementary wet scrubbing and particulate filtering. The AREVA FCVS standard comprises a venturi scrubber unit consisting of a venturi section (wet section), a combined droplet separator and metal fibre filter section (dry section) combined with a throttling orifice for a sliding pressure process. By the entrainment and dispersion of the scrubbing liquid, large reaction surfaces are created in the venturi nozzles which result in an additional effective sorption of gaseous iodine. For optimum iodine retention, the scrubbing liquid is conditioned with caustic soda and other additives.

The AREVA FCVS provides already retention efficiencies higher than 99.99 % for aerosols (even those smaller than 0.5 μm) and higher than 99.5 % for I_2. Despite this performance, as it is reminded in the PASSAM objectives section (Section II), there are studies [26] which indicate that a significant generation of organic iodine in the containment atmosphere might occur during a severe accident. Consequently the AREVA FCVS Plus was developed by supplementing the existing AREVA FCVS Standard with a sorbent (zeolite) section as a third retention stage with combined passive superheating [27].

In link with the PASSAM project, the AREVA experimental programme on this "FCVS PLUS" combined system was completed in 2013. The main results on organic iodine retention as stated by AREVA are:

Retention efficiency of zeolite depends upon many parameters: sorbent material, geometry / flow dynamics, superheating / relative humidity, heat repartition, residence time, gas composition, flow velocities, etc.

Extensive large scale test campaigns were conducted at the JAVA PLUS test facility. The performed JAVA PLUS tests verified that this combined filter is able to reach the target DF, e.g. superior to 50 for organic iodine (more than 98% retention).

The studied parameters allowed to develop a model in order to customize the DF for organic iodine by appropriate design of the sorbent stage for the requested operating range.

The performed test campaign results in a fully qualified combined filter design of this "FCVS PLUS".

6. CONCLUSION

The four year (2013–2016) European PASSAM project devoted to mitigation of atmospheric source term is being conducted as initially planned. After two years and a half of operation, the experimental work is in progress and in some domains major results have already been obtained

Thanks to the PASSAM project, databases on existing or innovative mitigation systems will be extended. Besides the studies on pool scrubbing [9], among the various topics studied, two of them may be quoted for their potential direct impact on severe accident management: gaseous iodine retention (molecular and organic iodine), and long term stability of trapped compounds in the various FCVS.

It is expected to get a deeper understanding of the phenomena underlying the performance of the mitigation systems studied, and to be able to propose simple models or correlations which should be easy to implement in accident analysis codes, like ASTEC.

Orders of magnitude for source term reduction for each mitigation system studied, including on the long term, in accident conditions, as well as general advantages and drawbacks of all these systems (efficiencies, passive behaviour, robustness, long term retention, etc.) is being assessed. It should allow giving hints for improved filtration systems.

The main technical outcomes of the project will be documented in a final synthesis report and presented in a final workshop to be held in Paris area by end 2016 or early 2017. Finally, all the public documents of the project are uploaded on line on a dedicated web site at https://gforge.irsn.fr/gf/project/passam/ .

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ADVANCES ON UNDERSTANDING OF POOL SCRUBBING FOR FCVS BASED ON THE PASSAM PROJECT

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Abstract. The PASSAM project investigates several topics related to both existing and innovative source term mitigation systems to be employed during severe accidents. One of the work packages concentrates on experimental investigation of existing mitigation systems, and specifically, on pool scrubbing phenomena. The phenomena to be investigated are the characterization of the two-phase flow in the water pools under churn-turbulent flow regime, aerosol retention under jet injection regime, the effect of impurities and surfactants on aerosol retention in the water pool, the effect of additives and submerged structures on organic iodide retention, and the long term retention / release of iodine species from water pools. This paper describes the experimental facilities which are used for the investigation of pool scrubbing, presents the phenomena which are addressed in each of the facilities, and gives the first results of some of the experiments. The PASSAM project is presented elsewhere in this publication [1].

1. INTRODUCTION

Some BWR and PWR severe accident scenarios involve transport of fission products (aerosols and gas phase species) through pools of water where the fission products can be retained. This phenomenon, known as pool scrubbing, has the potential to reduce the source term significantly.

Pool scrubbing was extensively investigated in the 1980s' and 1990s' in several large research programmes [2–9]. These investigations provided valuable insights into the effects of variables like particle type and size, gas flow rate, submergence and carrier gas composition, on the decontamination factor. Most of these investigations were integral in nature and were carried out under bubbly flow and globule regime. The most common pool scrubbing codes are based on the data from investigations from this time [10–12] even though there has been further development of the codes based on data from later investigations [13]. It is worth noting that more data are available on the retention of aerosols in the water pools than the retention of gas phase iodine species. A more thorough description of the current state of the art has been recently reported in Refs. [14, 15].

The use of the results of the investigations for model improvement is limited by the fact that the hydrodynamic characteristics of the water pools, aerosol size distributions, and other critical parameters were not always determined in detail. Furthermore, the accident in Fukushima drew attention to topics which had not been the focus of pool scrubbing research: containment venting could cause saturation and boiling of heavily contaminated suppression pools, and accidents can last much longer than previously assumed. This results in a need to extend the boundary conditions of the investigations to pool boiling and long term fission product behavior.

To address the mitigation of source term during severe accident, EU is funding a project PASSAM [1] in which different experimental facilities are used to investigate several topics related to source term transport and mitigation. One of the work packages concentrates on issues related to pool scrubbing, and specifically, the aspects which have not been investigated in detail earlier. Among those is the characterization of the two-phase flow in the water pools under churn-turbulent flow regime, aerosol retention under jet injection regime, the effect of impurities and surfactants on aerosol retention in the water pool, the effect of additives and submerged structures on organic iodide retention, and the long term retention / release of iodine species from water pools. The results of the experiments will be used for understanding of source term behavior in water pool, and to develop improved models which can be implemented in integral severe accident analysis codes, such as ASTEC.

2. HYDRODYNAMICS OF THE TWO-PHASE FLOW

2.1. TRISTAN facility

The test facility TRISTAN was built at PSI, Switzerland, for the experimental investigation of the hydrodynamics of the pool scrubbing conditions as well as conditions corresponding to the steam generator tube rupture (SGTR) meltdown accidents. In the international ARTIST project, it was observed that the aerosol retention was much higher in the presence of submerged structures (SGTR) than in bare pools [13, 16]. The effect was attributed to the inertial effects close to the injection point, and the interaction of the gas jet and the structures affecting the hydrodynamic behavior of the water pool in the presence of the tube bundle in the ARTIST facility. However, at the time few data existed, if any, on the effect of submerged surfaces on pool scrubbing.

As a consequence of the results of the ARTIST project, the TRISTAN facility was designed to study the hydrodynamics of two phase flows in tube bundle and large channel geometries [17]. TRISTAN is a hydrodynamic facility which can be used for characterization of two-phase flows at ambient temperature and pressure. The height of the facility is 6.2 m, and the side length of the square cross section is 0.5 m. The gas flow rate can be varied in the range 10–600 kg/h. The used gas is dry nitrogen or air. The void fraction, bubble size, average velocity and interfacial area are determined using a wire mesh sensor [18].

The liquid inventory of the TRISTAN facility is demineralized water in which Na_2SO_4 can be added to increase the electrical conductivity of the liquid phase. The geometry can be varied to simulate different pool scrubbing sequences. For this investigation, experiments were conducted in the bare pool configuration with only one injection tube in the middle of the facility, and with a tube bundle configuration with 221 steam generator tubes to simulate SGTR sequences. This enabled the assessment of the effect of the tube bundle on the pool hydrodynamic characteristics.

2.2. Results for the pool hydrodynamics

Experiments were carried out both in the bare pool configuration and in the SGTR set-up. The gas flow rate was varied between 50–450 kg/h, and the gas was dry nitrogen. The measurement with the wire-mesh sensor were in both cases conducted at 100 mm, 625 mm, 1250 mm and 2500 mm from the injection point in vertical direction. By measuring at different distances between the injection point and the sensor, the development of the flow in the facility could be determined. At all the flow rates and measurements distances, the void fraction, bubble size, velocity, and locational distribution, as well as the interfacial area could be measured. Some examples of the results are presented here.

The flow in both geometries was characterized by a large initial gas globule adjacent to the injection tube. The size of the globule depended on the gas flow rate, as well as the injection and the pool geometry. This large globule broke into smaller gas bubbles, but very large gas volumes stayed close to the injection tube all the way up to the top of the facility. These structures had varying shapes with very large shape factors. The flow was in general very turbulent (i.e. churn-turbulent flow) with some recirculation close to the facility walls. In practically all the tests, most of the flow development took place within the first about 1 meter from the injection point shown by almost identical bubble sizes and void fraction distributions measured 1250 and 2500 mm from the injection point.

The measurements showed clearly the effect of the tube bundle on the pool hydrodynamics. The tube bundle has a clear confining effect on the flow, as seen in Fig. 1 for the average void fraction at different flow rates with the distance of 625 mm between the injection and the measurement. In addition to confining the flow into a narrower area, the tube bundle also induces a directional gas distribution following the tube rows.

The bubble size distribution was also affected by the tube bundle. The bubble size distributions in the bare pool geometry had a large peak at the small bubble sizes and a very broad tail of the distribution, whereas the distribution in the tube bundle was clearly bimodal with a large peak at the small bubble size and a second peak at the larger size. The size of the bubbles of the secondary mode was dependent on the gas flow rate. This was mainly due to the breakup processes of the primary jet which were slower in the tube bundle geometry. In both geometries, large bubbles had a tendency to stay close to the center of the facility where the gas was injected into the facility, and the smaller bubbles were present further away from the facility centerline [17].

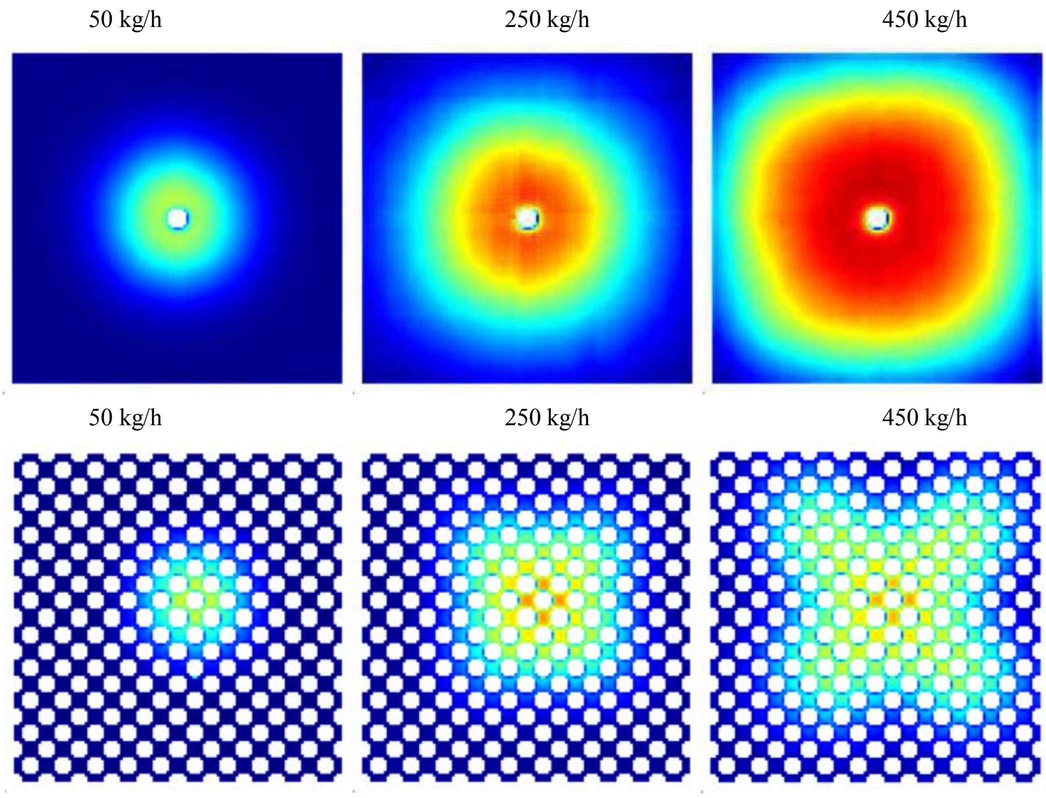

FIG. 1. The average void fraction in the bare pool (above) and tube bundle (below) set-ups at 50 kg/h, 250 kg/h and 450 kg/h gas injection flow rates with 625 mm distance between the gas injection and the measurement.

3. AN IMPROVED BUBBLE BREAK-UP MODELLING FOR POOL SCRUBBING

3.1. SCRUPOS facility

SCRUPOS facility was constructed at RSE, Italy, to investigate the effect of the hydrodynamics, and specifically, the bubble break-up on the aerosol retention. In addition, the effect of potential surfactants, the use of sea / dirty water, and other impurities on the hydrodynamics, and consequently, on the aerosol retention is addressed in the SCRUPOS facility.

The SCRUPOS is a small-scale facility of 1.5 m height, and a cross section of 0.5 m × 1.0 m. The walls are made of glass to allow for optical access to the facility. One gas injection orifice is installed in the center of the facility close to the facility bottom. The size of the orifice can be varied in the range 10–50 mm, and the gas flow rate 5–25 kg/h. The orientation of the gas injection nozzle is vertical with the opening upwards. The injection gas is dry air. In the tests, the water level in the facility was about 1.1 m.

The void fraction, as well as the bubble size and velocity are measured with an optical two-tip probe, and the bubble size and shape in addition with a high-speed camera. The first tests were used to compare the data given by the two methods. In these tests, the bubble size data from the two methods was found to be in agreement.

The aerosols can be injected into the water pool along with the carrier gas from the bottom of the facility. Monodisperse, spherical SiO_2 particles of three different sizes in the size range Dp= 0.5–1.5 μm will be used for the aerosol retention tests. The aerosol retention will be investigated with pure demineralized water, demineralized water with added salts (representative of sea water) and with demineralized water with added surfactants.

3.2. Results for the pool hydrodynamics

In the first set of tests, bubble chord distributions, Sauter mean diameters, rising velocities and the local void fractions were measured by a double fiber optical probe while a photo camera was used to evaluate mainly bubble shape and swarm evolution inside the pool, Figure 2.

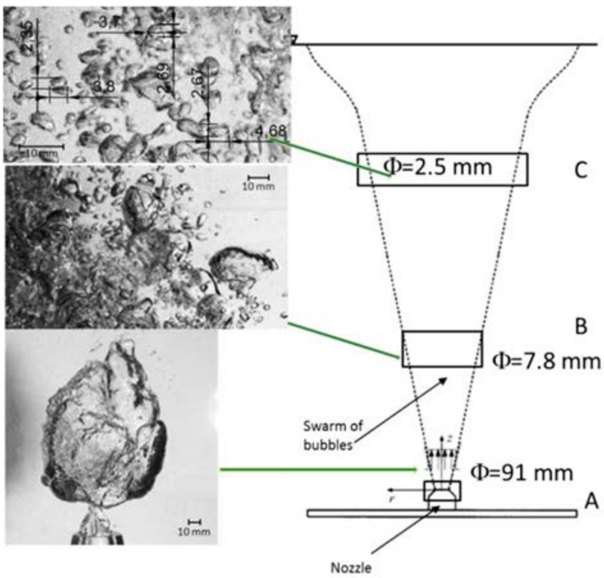

FIG. 2. Bubble images and sizes at different height position (orifice 10 mm, gas flow 18 kg/h).

The experimental results and their analysis shows that the first bubble injected in the pool has both the length and width related to nozzle diameter and Weber number measured at the injection point.

Bubble shapes change radically during the rise inside the pool, the first injected bubbles are cone-shaped due to the high bubble velocities, then, when the first bubbles collapse, the generated swarm is made mainly by spherical cap bubbles and oblate ellipsoids. In the upper part of the pool all detected bubbles have an ellipsoidal shape. The swarm of bubbles in the upper part of the pool consists mainly of small bubbles with the size between 2 mm and 3 mm.

Swarm expansion is linear from the injection point almost up to the pool surface with a more rapid expansion in the last part of the pool. The radial profiles of the void fraction for each measured point are well described by Gaussian curves since more bubbles are located in the central part of the swarm.

The analysis of the data obtained during experimental tests allowed validating a simplified model to describe the evolution of the bubbles inside the pool.

4. AEROSOL RETENTION IN THE JET INJECTION REGIME

4.1. PECA facility

PECA facility at CIEMAT, Spain, has been used for a wide variety of aerosol investigations in the area of nuclear safety and severe accidents, and particularly in the field of pool scrubbing in the 1990s. In the PASSAM project, the PECA facility is used to investigate the pool scrubbing of aerosols at the inlet region concentrating on high gas injection velocities. The facility is about 5 m high and 1.5 m in diameter steel vessel with a volume of 8.4 m^3, Figure 3. It consists of a gas supply system that provides the gases to carry particles, an injection line through which particle carrier gas is led to the injection point, an aerosol generator, control and measurement devices and instrumentation, and the data acquisition system that records on-line the main thermal and hydraulic variables. It can be operated up to pressures of 3.5 bar and temperature of 140 °C, with the water temperature varying from sub-cooled to saturation. The injection gas will be a mixture of air and steam, or a mixture of helium and steam, and spherical, monodisperse SiO_2 particles with a diameter of 1.0 μm will be used as the aerosol.

4.2. Test for aerosol retention in the jet injection regime

As said above, the tests at the PECA facility are focused on retention at the nearby of the injection point under high gas velocities (i.e., the jet injection regime). The test matrix is shown in Table 1 and it considers two major boundary conditions of retention at the inlet of the pool: the gas mass flow rates (i.e. gas injection velocity) and the saturation state of the pool. The former will be controlled through the non-dimensional Weber number (i.e. ratio between inertia and surface tension forces) and the latter through the gas saturation ratio (i.e. ratio between steam pressure in the carrier gas and saturation pressure at the pool temperature). Through this approach the role that thermal-dependent mechanisms (i.e. diffusiophoresis) and "entrained-droplet mechanisms" (i.e. inertial impaction and interception) play in the "near-field" retention is foreseen to be assessed. In addition, the investigation will allow characterizing the size distribution of particles entering what is called the swarm region of the pool, in those scenarios with enough submergence. A minimum pool depth is to be set in all experiments, so that the any contribution of the swarm region to the pool decontamination efficiency will be negligible. The specific values to be set will be determined based on some preliminary tests. In principle, one-micron SiO_2 will be injected, but other sizes might be tested if feasible.

FIG. 3. The PECA facility for investigation of aerosol scrubbing at high injection velocities.

TABLE 1. THE TEST MATRIX FOR THE PECA FACILITY FOR POOL SCRUBBING OF AEROSOLS

Test	Regime (We)				Saturation ratio			
	$<10^5$	$>10^5$	$>>10^5$	$>>>10^5$	Very low	Low	Medium	High
0	X						X	
1		X					X	
2			X				X	
3				X			X	
4			X		X			
5			X			X		
6			X					X

5. RETENTION OF ORGANIC IODIDE IN THE WATER POOL

5.1. VESPER34 facility

The retention of organic iodides in water pools was investigated in the test facility VESPER34 at AREVA, Germany. The focus of the investigation was the possible effects of submerged structures, scrubbing liquid temperature, and the use of additives on the retention of organic iodides in a generic test facility of a relatively small scale. The volume of the facility is approximately 20 l and it was operated at conditions up to pressure of 6 bar abs, and temperature of 130 °C. As an organic iodide, CH_3I was used due to its high volatility and assumed abundance relative to other organic iodides under severe accident conditions. The test facility is located in a control area enabling the use of labelled iodine for the measurements, and thereby the use of realistic iodine concentrations in the tests.

The VESPER34 test facility has three feed streams: i) steam supply, ii) non-condensable gas (e.g. air / nitrogen) supply, and iii) dosing gas (e.g. gaseous CH_3I in nitrogen). The feeds are joined to a common gas stream and injected into the scrubber test vessel that is located in a heated cabinet. Inside the scrubber test vessel there is a submerged gas injection nozzle (miniaturized Venturi nozzle) where the gas stream is injected into the scrubbing liquid.

In general, the scrubbing liquid of FCVS is composed of deionized water with scrubbing additives (NaOH $Na_2S_2O_3$). Downstream of the test vessel the steam is routed through several beds of molecular sieve sorbent. Finally the steam is condensed and the non-condensable gas is directed over charcoal filter in order to establish an activity balance so that the retention performance of the scrubber can be calculated.

5.2. The influence of the scrubbing liquid temperature

For the standard scrubbing liquid composition of 0.5 wt% NaOH and 0.2 wt% $Na_2S_2O_3$, the temperature was varied between 85 and 130°C. The results presented in Figure 4 show that for temperatures higher than 85°C, the absorption efficiency of the scrubbing liquid is moderate (from 18 to 33 %, roughly). Under 110 °C the absorption efficiency is practically constant, whereas between 110 °C and 130 °C a noticeable increase was observed.

5.3. The influence of the submerged structures

To increase the gas residence time and the gas/liquid interfacial area in the pool, internal structures may be introduced in the wet scrubber. To investigate the potential effects of such structures, tests were performed with a scrubber filled with so-called mixing elements randomly filling the bottom of the facility. All the tests with the submerged structures were performed at 110 °C. The structures were generic in nature, and did not represent any particular wet scrubber geometry.

The results of the tests showed, that if the standard scrubbing liquid composition was used, the addition of mixing elements to the VESPER34 facility led to increased retention of CH_3I. Under the conditions and geometry of the tests, the retention of CH_3I increased approximately by a factor of two, Fig. 4.

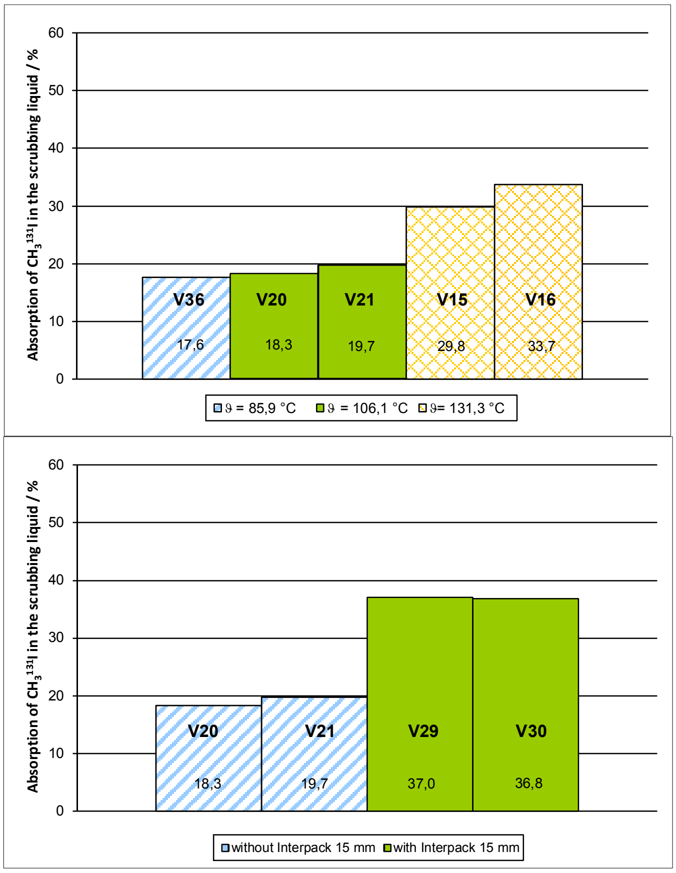

FIG. 4. The effect of the scrubbing liquid temperature and submerged structures on the retention of CH_3I using the standard scrubbing liquid composition.

6. MEDIUM AND LONG TERM RETENTION OF IODINE IN WATER POOLS

6.1. EPICUR facility

The long term stability of retained iodine in wet scrubbers is investigated in the EPICUR facility at IRSN, Cadarache, France. The EPICUR is an irradiation facility composed of i) a panoramic irradiator containing ^{60}Co sources designed to deliver an average dose of several kGy/h to represent the effect of radiation related to the presence of radioactive products in the containment during an accident, and ii) a test loop equipped with an irradiation vessel (4.8 l) subjected to the irradiation from the irradiator, and a Maypack filter system composed of four stages of filters designed to quantitatively measure various iodine species, namely, aerosols, molecular iodine, and organic iodides.

For the PASSAM tests, the irradiation vessel is filled with demineralized water with typical FCVS additives ($Na_2S_2O_3$, NaOH). Other additives, such as HCl and HNO_3, may be used as well. Iodine is added in the water in different concentrations. Gas flow is sparged through the water for 10-30 hours and the released iodine is measured at the vessel outlet with the Maypack filter system, Figure 5. The iodine is labelled with ^{131}I to allow precise measurements of iodine partitioning during and after the tests. In the tests, the pressure in the vessel is approximately 2.6-2.8 bar abs, and the temperature close to saturation in order to simulate the potential loss of coolant system during a nuclear severe accident.

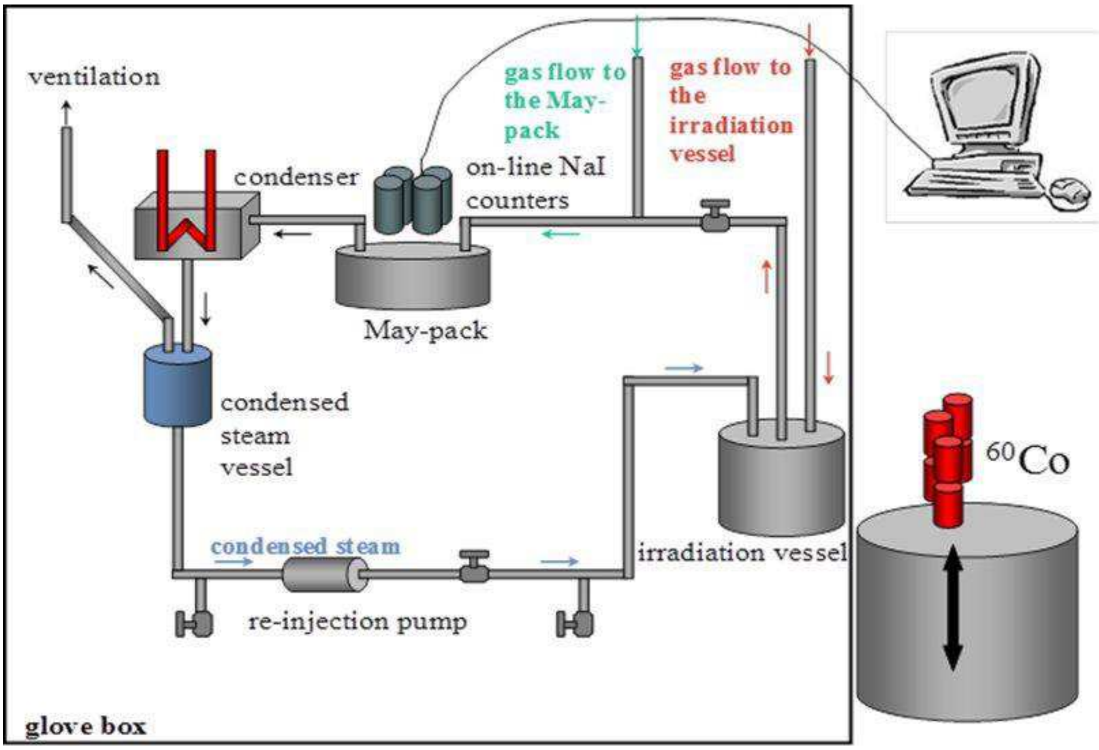

FIG. 5. The EPICUR facility for investigation of long term stability of iodine in water under FCVS conditions.

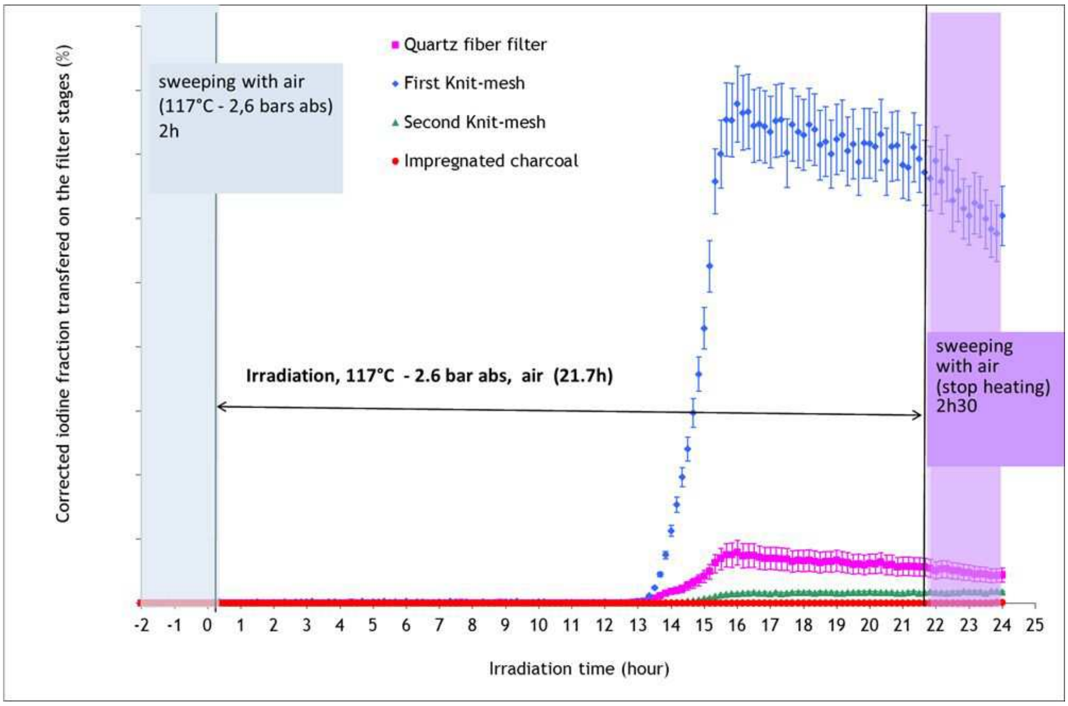

FIG. 6. Iodine release form the water pool with decreasing pH in the water during the test.

6.2. First results for the iodine stability

Three tests have been carried out in the EPICUR facility in the project. In the first two tests, the initial pH in the water was 8.3 and 10.5, respectively, in order to simulate the acidification processes which might occur during a severe accident sequence (mainly due to CO_2 dissolution in the scrubber). The preliminary results show that during the tests without sufficient buffer, the pH decreased significantly below the neutrality point due to the oxidation and decomposition of $Na_2S_2O_3$ under irradiation leading to highly acidic compounds. This resulted in modifying the iodine speciation dissolved in the aqueous solution and thus significant release of iodine from the water could be observed, Figure 6, as measured by the Maypack filters at the test section outlet.

7. FINAL REMARKS

The PASSAM project investigates several topics related to pool scrubbing and FCVS. The project is ongoing, and many experiments will still be carried out. However, already now the project has provided new and important experimental data on the areas of the effect of submerged structures on the pool hydrodynamics and the organic iodine retention in the water, for pool hydrodynamics in the jet injection regime, and for the long term stability of iodine under FCVS conditions. These data will be complemented by further experiments for aerosol retention in the jet injection regime, and the effect of surfactants on the pool hydrodynamics and the aerosol retention.

Models are being developed to describe the investigated phenomena, and this work continues. It is foreseen that in the form of models and correlations, the data from the project will be implemented in severe accident codes, such as ASTEC.

ACKNOWLEDGEMENTS

The European Atomic Energy Community (Euratom) is acknowledged for funding this work in the PASSAM Project in the frame of the 7th framework programme FP7/2007-2013 under grant agreement n° 323217. We are grateful for the staff involved in the experiments in all the facilities for their invaluable contribution.

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DESIGN OF CONTAINMENT FILTERED VENTING SYSTEM (CFVS) FOR TAPS-1&2 TO LIMIT THE CONTAINMENT PRESSURE BELOW DESIGN PRESSURE DURING DESIGN EXTENSION CONDITIONS

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Abstract. Under postulated design extension conditions leading to core melt in a nuclear power plant; the radio-nuclides generated in fuel may be released into the containment atmosphere in gaseous and aerosol form, imposing potential threat to the containment integrity due to overpressure. The containment being the last physical barrier; it is necessary to prevent its over-pressurization by releasing the containment atmosphere in a controlled manner in order to maintain the containment integrity. The containment function is achieved through provision of Containment Filtered Venting System (CFVS), as an additional defense for the removal of particulate matter and gaseous pollutants from the gas stream prior to its release to the outside environment; to keep the containment pressure below the design pressure for extended period of time during design extension conditions as well as to reduce on-site and off-site radiological consequences. A scale down experimental test facility of the proposed design of CFVS for TAPS-1&2 was erected and number of tests were conducted to assess its performance of achieving the desired decontamination factor as well as to demonstrate its capability of meeting the design objective. The assessment of decontamination factors in the test facility was carried out using injection of non-active chemicals such as Elemental Iodine, Cesium Iodide and Methyl Iodide mixed with compressed air and steam as a carrier medium. The experimentally obtained decontamination factors were higher than the target values of DF 1000, 100 and 10 for CsI, Elemental Iodine and Organic Iodide respectively. This paper depicts the methodology adopted for design of experimental facility and its outcome.

1. INTRODUCTION

Tarapur Atomic Power Station is a two-Unit Boiling Water Reactor (BWR). It is a generation-1 BWR (BWR-1), designed by General electric (GE) of USA. Its containment is of pre-Mark-I design. It was made operational in year 1969. Original power rating of each Unit was 210 MW(e) (660.9 MW(th)), which was de-rated to 160 MW(e) (530 MW(th)) in year 1984 due to operational problem. Based on in-depth review post Fukushima accident, various modifications were suggested by regulatory body to enhance capability to handle extreme external event that can cause prolonged station black out condition. One of the recommendations was to install reliable containment filtered vent system to assure the integrity of primary containment under design extension condition to mitigate the consequence.

Under postulated severe accident scenario leading to core melt in a nuclear power plant; the radio-nuclides generated in fuel may be released into the containment atmosphere in gaseous and aerosol form, imposing potential threat to the containment integrity due to overpressure. The containment being the last physical barrier; it is necessary to reduce its overpressure by releasing the containment atmosphere in a controlled manner in order to maintain the containment integrity. The containment function is achieved through provision of Containment Filtered Venting System (CFVS) as an additional defense for the removal of particulate matter and gaseous pollutants from the gas stream prior to its release to the outside environment; to keep the containment pressure below the design pressure for extended period of time during severe accident as well as to reduce on-site and off-site radiological consequences. A scale down experimental test facility of the proposed design of CFVS for TAPS-1&2 is erected and tests are conducted to assess its performance of achieving the desired decontamination factor as well as to demonstrate its capability meeting the design objective. The assessment of decontamination factors in the test facility was carried out using injection of non-active chemicals such as elemental Iodine, Cesium Iodide and Methyl Iodide mixed with compressed air flow and steam as carrier medium. Around forty tests were carried out including the characterization tests to establish the repeatability.

2. OBJECTIVE

To demonstrate the capability of the experimental set-up of containment filtered venting system to achieve the minimum target decontamination factor with air as carrier medium is decided by referring various literature available world-wide[1]. Target decontamination factors with as carrier medium are as follows:

- Particulates 1000 (Retention efficiency 99.9%);
- Elemental iodine 100 (Retention efficiency 99.0%);
- Organic iodide 10 (Retention efficiency 90.0%).

Additionally, the objective is to generate data base to facilitate finalization of the design of CFVS for TAPS-1&2, based on the observed performance of:

(1) CFVS in terms of observed pressure drops across various parts of the system for a given inlet flow rate using different venturi-nozzle;
(2) The three stage filter systems for eliminating mist of droplet mean diameter exceeding 0.5 µm;
(3) The cumulative decontamination factors achievable for removal of aerosol, elemental Iodine and organic form of Iodine and Cesium Iodide.

3. SCOPE

The scope of this experiment is to assimilate adequate performance data in the mock-up test facility to finalize the design of CFVS for TAPS-1&2. For meeting the above mentioned objectives, establishment of a test facility, which comprised of: compressed air system simulating air carrier medium, electrical steam boiler, mixing tank, needle venturi injection for liquid injection, heater system for aerosol and vapor injection, a suitable scrubber tank, alkaline water, bubble breaker, proprietary venturi-nozzle and mist eliminator and necessary instrumentation to record temperature, pressure drops, pH and flow rates at appropriate locations.

4. EXPERIMENTAL SETUP

A CFVS Test Facility was erected to conduct the experiments at TAPS-1&2. Figure 1 depicts the 3-D model of the experiment facility at TAPS-1&2. The schematic of the test facility is shown in Figure 2. The facility consists of two tanks, namely, mixing tank and scrubber tank respectively, which are connected in series to form once through circuit. The mixing tank represents the reactor containment in which air and steam mixture ratio of appropriate composition is injected, to simulate the accidental conditions. An air compressor and a steam generator of adequate capacity are connected to the inlet line of the mixing tank to supply steam-air mixture, prior to its injection into the scrubber system. A diffuser is provided within the mixing tank which will help to mix the chemicals homogeneously with air-steam mixture to simulate the containment conditions. The various chemicals representing fission products and corrosion products are mixed with air-steam mixture, then, are made to individually pass through the scrubber tank with the help of venturi injector, one at a time separately. The chemicals selected for the tests have similar chemical and physical forms as of radioactive radio-nuclides, likely to be released from the reactor molten core during the severe accident conditions.

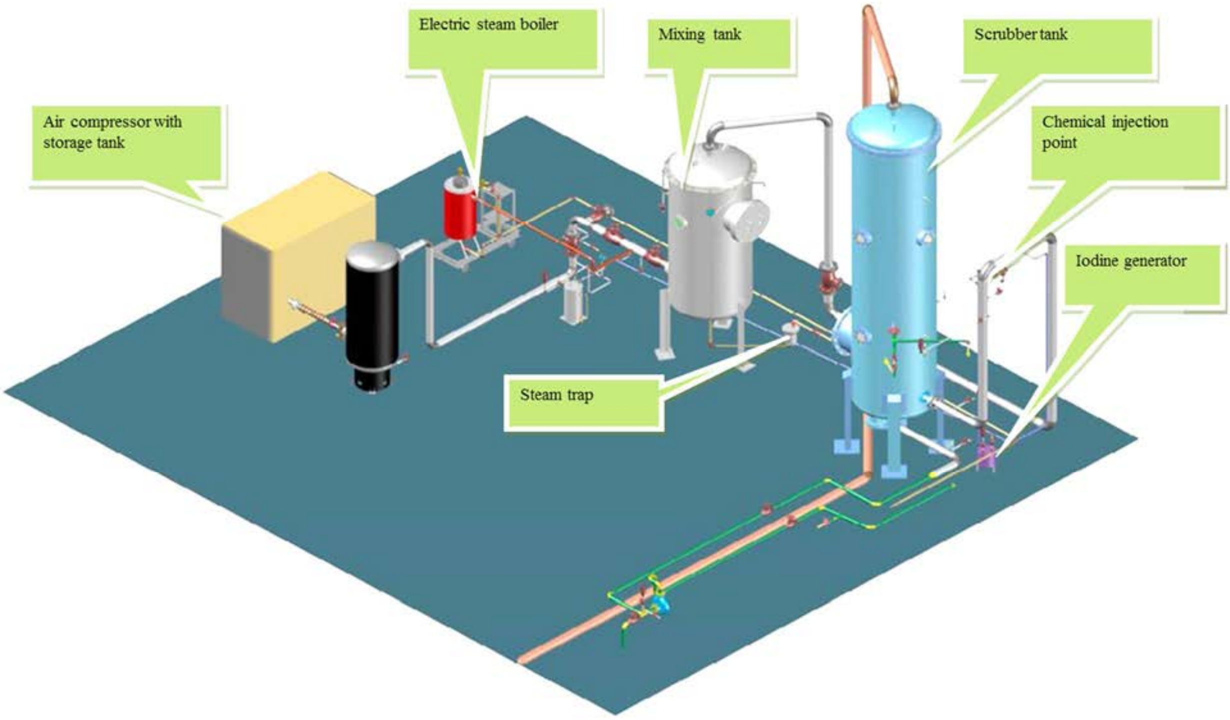

FIG. 1. 3D model of CFVS experimental facility a TAPS 1&2.

The scrubber tank (Figure 3 and Figure 4) houses the venturi scrubber, bubble breaker and three stages of mist eliminator. The scrubber tank is partially filled with water so that venturi scrubber and bubble breaker are completely submerged in water for self-priming action. The mixture of air-steam-chemicals is injected into inlet line of the scrubber tank, which is supported by carrier steam-air mixture in venturi-nozzle of scrubber tank, the mixture interacts (mix) with water filled in the scrubber tank. The injected air-chemical mixture would break the water into small droplets by impaction, replenishing fresh stock of water due to venturi action. The carrier air-chemical mixture is sent upward, exiting the nozzle and fresh incoming water is replenishes earlier water, which becomes available to be broken into small droplets. The nozzle design permits vigorous recirculation of water, increasing the recirculation time that allows time for mass exchange between the entrapped gas in the water vapour to liquid phase of water and good liquid to gas ratio. This process helps in promoting turbulence as well as provides good scrubbing action permitting a good decontamination factor of the order of 100 during this process. Should steam-air-chemical mixture enter the Scrubber Tank; water in scrubber would allow condensation of the steam and also remove the added chemicals by dissolution process. The air along with remaining chemicals would escape the water surface in the form of bubbles and these will further break into smaller bubbles by use of bubble breaker which is placed just below the water surface in the scrubber tank. This bubble breaker is made up of fine mesh, which in addition to breaking bigger bubble into smaller bubbles will also facilitate further mixing of air with water, enhancing mass transfer. Further, air steam mixture will contain some chemical dissolved in water droplets i.e. mist. Large sized mist coming out from water surface is settled due to gravitational settling. Subsequently remaining water droplets are removed by various stages of bulk and fine mist eliminators before discharging into the outside environment. The chemicals are injected into the scrubber tank through injection system (Figure 5) located close to the scrubber tank. The differential pressures across mist eliminator-filters and scrubber tank are measured by separate DP transmitters of Rosemount make and differential pressure recorded in digital recorder (Figure 6).

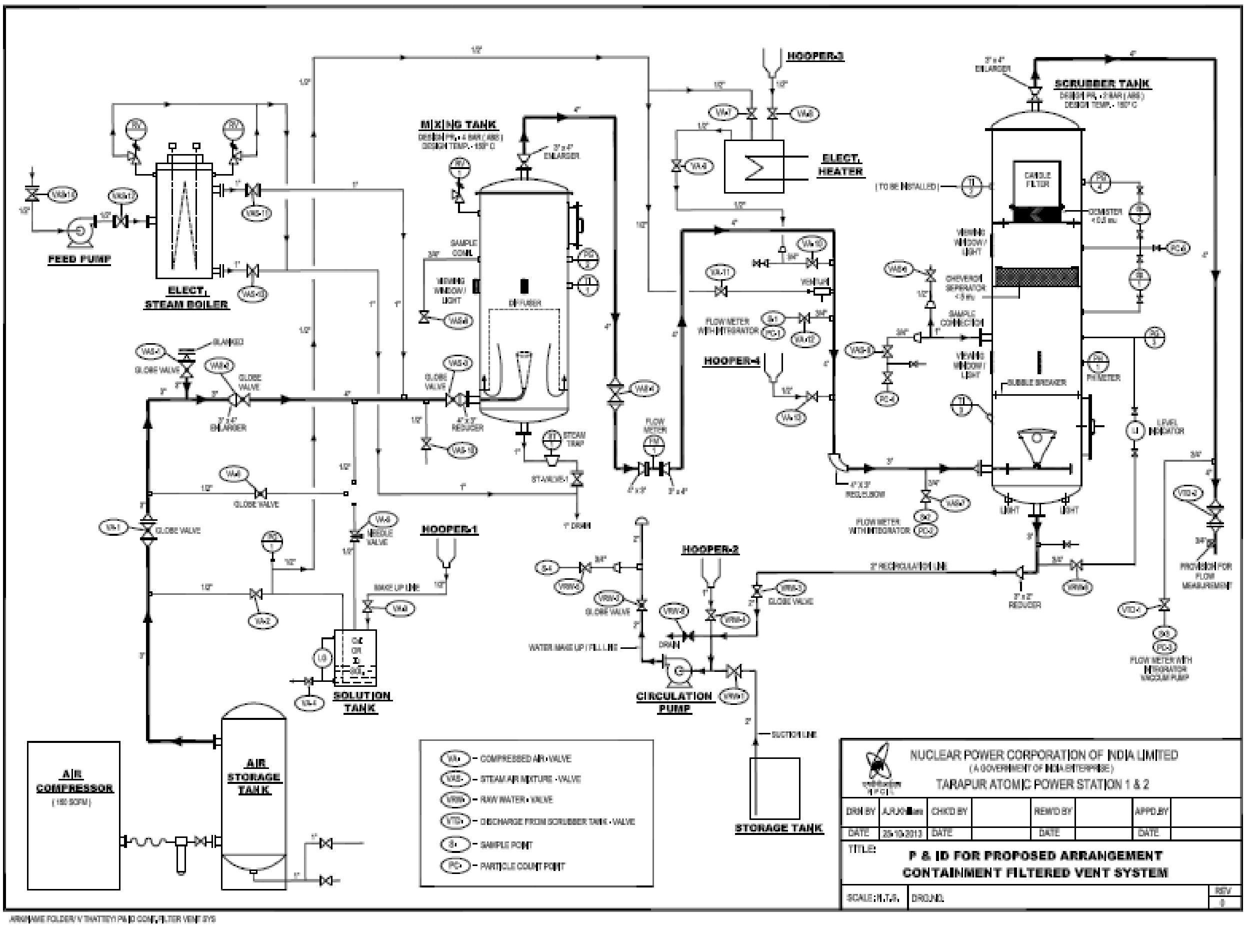

FIG. 2. Schematic of CFVS test facility at TAPS-1&2.

The test facility is equipped to measure the pressure of both the tanks, differential pressure across the various components such as, venturi-nozzle, mist eliminators and the temperature of gas phase as well as water residing inside the tanks, water level in the scrubber tank, pH of the scrubber tank water and flow of air-steam mixture at the inlet of scrubber tank. Adequate sampling points are provided to collect the relevant data required for analysis purpose and estimation of decontamination factors.

FIG. 3. Scrubber tank housing venturi nozzle and three-stage filter system.

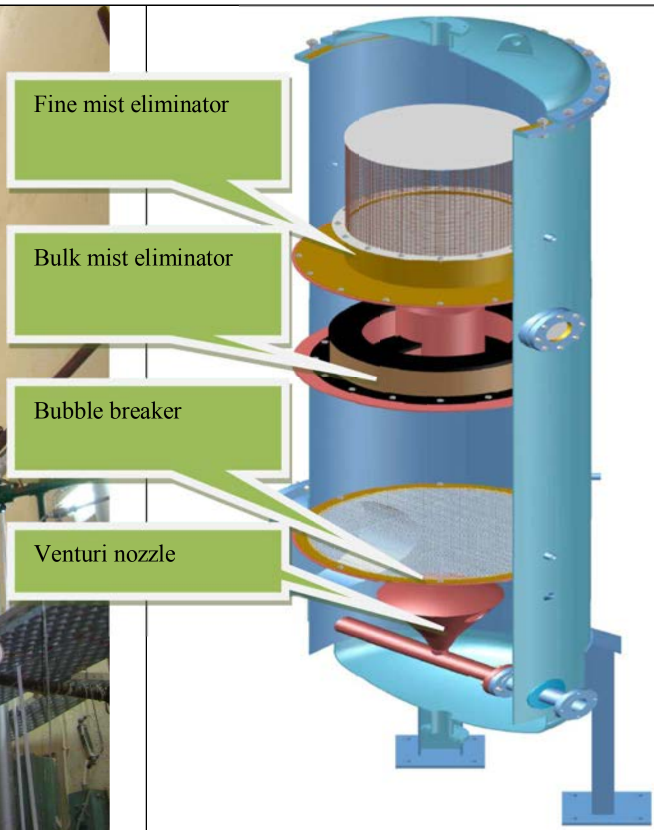

FIG. 4. Sectional view of scrubber tank housing venturi nozzle and three-stage filter system.

FIG. 5. Injection of chemical in the scrubber tank.

FIG. 6. Pressure drops across filters and scrubber tank.

5. DESIGN BASIS

The design basis of the test facility considers upper bound of the expected discharge flow rates exiting the containment, and expected loading of fission products in the scrubber tank. The design of the venture-nozzle is expected to provide 90–99.0 % scrubbing efficiency and the three stage filters are expected to provide 99.9 % efficiency for retention of mist particles for diameter beyond 0.5 μm. The overall decontamination factor is expected to exceed 1000 for particulates. The technical specification of the test facility is given in Table 1.

TABLE 1. TECHNICAL SPECIFICATIONS OF THE EXPERIMENTAL FACILITY

Sr. No.	Item Description	Value
1	Scrubber Tank diameter, m	1.0
2	Height of the Scrubber Tank, m	4.6
3	Mixer Tank diameter, m	1.0
4	Mixer Tank height, m	2.0
5	Air compressor , kg/hr	200.0
6	Steam Generator, kg/hr	200.0
7	Inlet/Outlet Piping, mm	100.0
8	Scrubber Venturi nozzle diameter, mm	19.0
9	Pressure Gauge, kg/cm^2	0-5.0
10	Temperature Gauge, ºC	0-150.0
11	Level Transmitter, cm	0-200.0
12	Differential Pressure across Filter, mm	0-100.0
12	Flow meter at exit, m^3/hr	0-500
13	Maximum steam-air mixture mass flow rate, kg/hr	324.0
14	Maximum steam flow rate, kg/hr	108.0
15	Input Power to steam generator, kW	65.0
16	Maximum throat velocity, m/s	300.0
17.	Maximum velocity in piping, m/s	23.0
18.	Non-active Chemicals used - I_2, CsI, CH_3I and NaOH	

6. PROCEDURE OF EXPERIMENTS

The known amount of chemical solution was injected through injection station for certain duration keeping inlet air-steam flow constant. The samples were drawn at various locations such as near inlet to scrubber tank and at downstream of the filters. These samples were analyzed in the chemistry laboratory of TAPS-1&2 using Chemito AAS-203 and UV-2450 Spectrophotometer by following standard ASTM methods of detection. Colorimetric analysis was performed on UV-2450 Spectrophotometer and light path of approximately 10 mm was used to measure absorbance. The measurement range variation was from 0.60 ppm to 2000 ppm. The sensitivity of detection expressed as limit of detection (LOD) and limit of quantification (LOQ), which were 0.60 ppm and 2.0 ppm respectively. Analysis of cesium was done on Chemito AAS-203 on emission mode. LOD and LOQ values for cesium were 0.079 ppm and 0.39 ppm respectively. The samples, below detectable limits, were analyzed on An Inductively Coupled Plasma mass spectrometer (ICP-MS).

7. EXPERIMENTAL RESULTS

The experiments were conducted to estimate the decontamination factors for elemental Iodine, Cesium iodide and Methyl iodide. The data collected for various tests conducted on the experimental facility for assessment of the decontamination factor is presented in the following Table 2.

TABLE 2. LIST OF EXPERIMENTS PERFORMED AT TAPS-1&2 CFVS TEST FACILITY

S.No.	Chemical Injected	Amount Injected(gm)	Air flow rate in m^3/hr	Steam flow rate m^3/hr	DF
1.	Methyl iodide	25	160	70	40.20223
2.	Methyl iodide	25	160	70	23.90899
3.	Methyl iodide	25	160	70	37.1246
4.	Methyl iodide	25	160	70	39.96981
5.	Cesium iodide	250	160	70	3327.3
6.	Cesium iodide	250	160	70	3722.602
7.	Iodine	100	156	0	130.1416
8.	Iodine	100	156	0	103.4246
9.	Iodine	100	54.72	0	400.2377
10.	Iodine	100	54.72	0	370.7469
11.	Iodine	100	101.2	0	241.2221
12.	Iodine	100	101.2	0	238.0722
13.	Iodine	50	109.4	0	126.2956
14.	Iodine	50	109.4	0	124.6967

8. CONCLUSION

The experimental facility was set up to verify the design of the containment filtered venting system including its components such as venturi-nozzle and filters. The decontamination factors were experimentally obtained for the injected particulates, elemental Iodine and Organic Iodide using either dry air or steam & air mixture as carrier medium. The experimentally obtained decontamination factors were higher than the target 1000, 100 and 10 i.e. for Particulate, Elemental Iodine & Organic iodide respectively. It validates the design.

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FILTERED CONTAINMENT VENTING FOR PURGING VAPOR-GAS DISCHARGES OF WATER COOLED REACTOR AT SEVERE ACCIDENT

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«Rosenergoatom» Corporate Group OJSC regards safety provision at operation of NPP's power units as the basis of its engineering policy [1]. One of the lines of increasing safety of existing and newly designed NPPs with WWER-440 (RU V-213) and NPPs with WWER-1000 (RU V-187, RU V-320, RU V-338) is equipping power units with systems of filtered pressure vent and gas filtration (FPV) for preventing collapse of containment and purging of discharges from radioactive products to allowable values in conditions of progressing BDBA with core melting

There are a number of well-known NPPs outside the Russian Federation where such methods are used as discharge dry filtration, plus combination of liquid and dry purging steps, which differ from each other with structural peculiarities of their elements, and also with the composition of applied purging liquids: systems by SIEMENS, AREVA, Westinghouse [2] and others. Practically all the mentioned systems can provide necessary purging of emergency discharges from aerosols (with purging factor of 103 and over). Technical characteristics of foreign FPV systems leave us doubtful about their ability to catch volatile forms of iodine, both in the molecular and (especially) in organic form. When this process is realized by means of liquids the composition of the latter requires continuous correction during an accident, as the liquid absorbs a very wide range of products of the accident; in practice, such a correction occurs to be unrealizable. Moreover, in case of vapour condensing from the medium, potentially explosive hydrogen concentrations can involuntarily occur. The proposed sorbent for the dry version of foreign FPVs, zeolite with silver coating, in reserve mode can drastically lose its activity due to oxidation of silver and continuous absorption of moisture from air by zeolite. It is also worth to note that during the initial period of an accident, despite overheating of the medium in the throttling device, it is cooled very quickly on the zeolite cold surface with moisture condensed and efficiency of the sorbent lost.

JSC «SVERD E & F» and SF NIKIET (Sverdlovsk branch of OJSC Research and Development Institute of Power Engineering) conducted, as long ago as in 1990s, R&D aimed at creation of FPV system for NPP with WWER reactors [3]. The principal feature of the homeland FPV is the use of a specially developed heat-resistant inorganic sorbent «Thermoxid-58» on basis of titanium dioxide possessing high thermal and radiation resistance and high purging efficiency, especially from iodine volatile forms, organic iodine included. Lab tests of the sorbent were made in SF NIKIET. Experiments with «Thermoxid-58» on basis of titanium dioxide were conducted for such degrees of filling or charges for iodine, which were multiple times bigger than a standard expected inflow of iodine volatile forms according to the maximum design accident scenario.

Tables 1-5 show the sorbent characteristics and its efficiency for various parameters of the environment. Fig. 1 presents the dependence of TF (transmittance factor of CH_3J) on thickness of «Thermoxid-58» layer at various temperatures and constant air humidity of 80 %.

In the reserve mode, the sorbent can stay in up-heated state in nitrogen or air environment. As temperature is rising, the efficiency constant will increase. Water vapours being in the air render significant influence on efficiency of the sorbent. At air moisture of 60–90 % it is stipulated by capillary condensing of water vapours in pores. The farther parameters of vapour-gas environment are from saturation state, the more efficiently sorbent will act. Increase of efficiency of the sorbent together with increase of the temperature at the same absolute moisture is connected not only with the temperature effect, but also with decrease of the relative moisture. Overheating of the vapour or vapour-gas environment from the saturation state can be achieved by up-heating by internal environment, throttling and other methods.

TABLE 1. PHYSIC-CHEMICAL CHARACTERISTICS OF «THERMOXID-58» SORBENT ON BASIS OF TITANIUM DIOXIDE

Description of characteristics	Unit of measurement	"Thermoxid-58»
Appearance	-	Granules of white or yellowish colour, of spherical or elliptic shape
Granulometric composition	mm	0,1-0,2 0,4-1,0 1,0-2,0
Mechanical strength	MPa (kgf/cm^2)	20-40 (200-400)
Bulk weight	g/cm^3	1,0-1,3
Specific surface	m^2/g	90-130
Specific volume of pores	cm^3/g	0,27-0,35
Absorptive capacity per CH_3J, not less than*	mg/g	10
Allowable working temperature	°C	100-450
Radiation strength, not less than	M Gr	100

*The characteristics determined at the following parameters of vapour-air medium: temperature 150°C; moisture 80%; CH_3J concentration: 100 mg/m^3.

TABLE 2. RESULTS OF STATIC EXPERIMENTS CONFIRMING HIGH THERMAL STRENGTH OF «THERMOXID-58» IN THE VAPOUR-AIR ENVIRONMENT AT ELEVATED TEMPERATURES AND PRESSURES

	Initial sample	Temperature of tests					
		150^{O}C			200^{O}C		
Duration of tests, hours	0	48	96	168	48	96	168
Specific surface, m^2/g	95±5	91±5	95±5	93±5	97±5	94±5	95±5
Mechanical strength, kgf/cm^2	335±70	355±70	445±80	460±80	355±70	390±70	470±80

TABLE 3. CONSTANT OF EFFICIENCY OF PURGING FROM IODINE VOLATILE FORMS FOR «THERMOXID-58» AS A FUNCTION OF TEMPERATURE

Temperature, °C	Efficiency constant (K), c^{-1}
60	13,2
80	17,3
100	20,2
120	24,6
140	34,3
160	57,5

TABLE 4. CONSTANT OF EFFICIENCY OF PURGING FROM IODINE VOLATILE FORMS FOR «THERMOXID-58» DEPENDING ON MOISTURE AT 100°C

Absolute moisture, %	Efficiency constant (K), c^{-1}
0	15,3
20	5,1
60	3,9
80	2,9
90	1,7

TABLE 5. CONSTANT OF EFFICIENCY OF PURGING FROM IODINE VOLATILE FORMS FOR «THERMOXID-58» DEPENDING ON TEMPERATURE AT ABSOLUTE AIR MOISTURE 80%

Temperature	Relative moisture	Efficiency constant
120	0,40	5,7
130	0,30	8,9
150	0,17	9,2
200	0,05	31,0
250	0,02	115,0
300	0,01	230,0

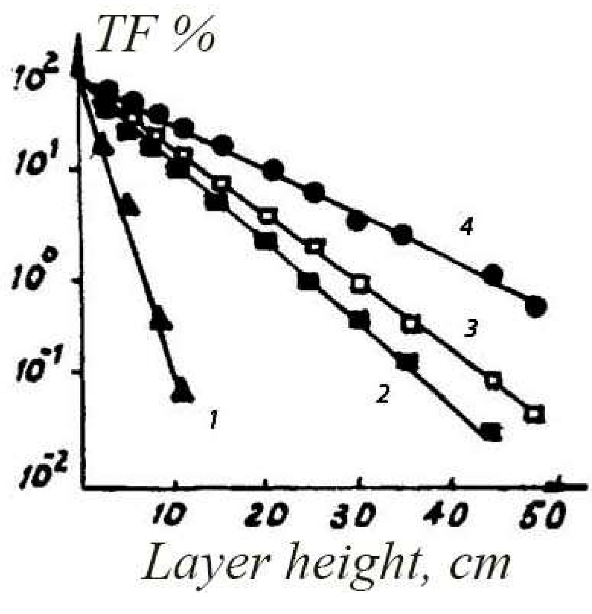

FIG. 1. Dependence of TF (transmittance factor of CH_3J) on thickness of «Thermoxid-58» layer at various temperatures and constant air humidity of 80%. Gases speed: 0.5 m/s; CH_3J concentration: 25 mcg/l, load on sorbent: 65 mcg/l: 1 – 200 °C; 2 – 150 °C; 3 – 130 °C; 4 – 120 °C.

On basis of the results of laboratory researches in JSC «SVERD E & F», models of jet filter and bed filter were designed and made for purging gases. These models of filtering equipment were tested for their efficiency of catching aerosols Cs and MnO_2 in Hanford (USA), within ACE international programme, and in Karlsruhe Nuclear Research Centre, on Typhoon test bench at catching barium sulfate aerosols. The filters demonstrated high efficiency of gas purging from aerosols and iodine volatile forms.

In course of conducting experiments of catching solid aerosols Cs, Mn, J on working conditions of a plasma generator, the average aerosol size of particles in the medium was larger than the preset size being equal to 1,8-4,2 μm.

Concentrations of aerosols were measured at inlet and outlet with use of filtered samples. Analysis was performed and factors of purging defined for cesium, manganese and iodine. They occurred to be very high, over 105, which speaks for excellent removal of aerosols. Efficiency of the system was also measured for removal of dioctylphthalate aerosols (aerosol diameter 0.7 μm). The overall purging factor of the system was from 2000 to 14000. After tests in SF NIKIET, a layer-wise chemical analysis of the filter filling with «Thermoxid-58» packing was conducted, whereas the bed layer of the filter was being discharged layer after layer, with the pitch about 10 cm. It was shown that uniform distribution of absorbed elements in the sorbent layers was provided.

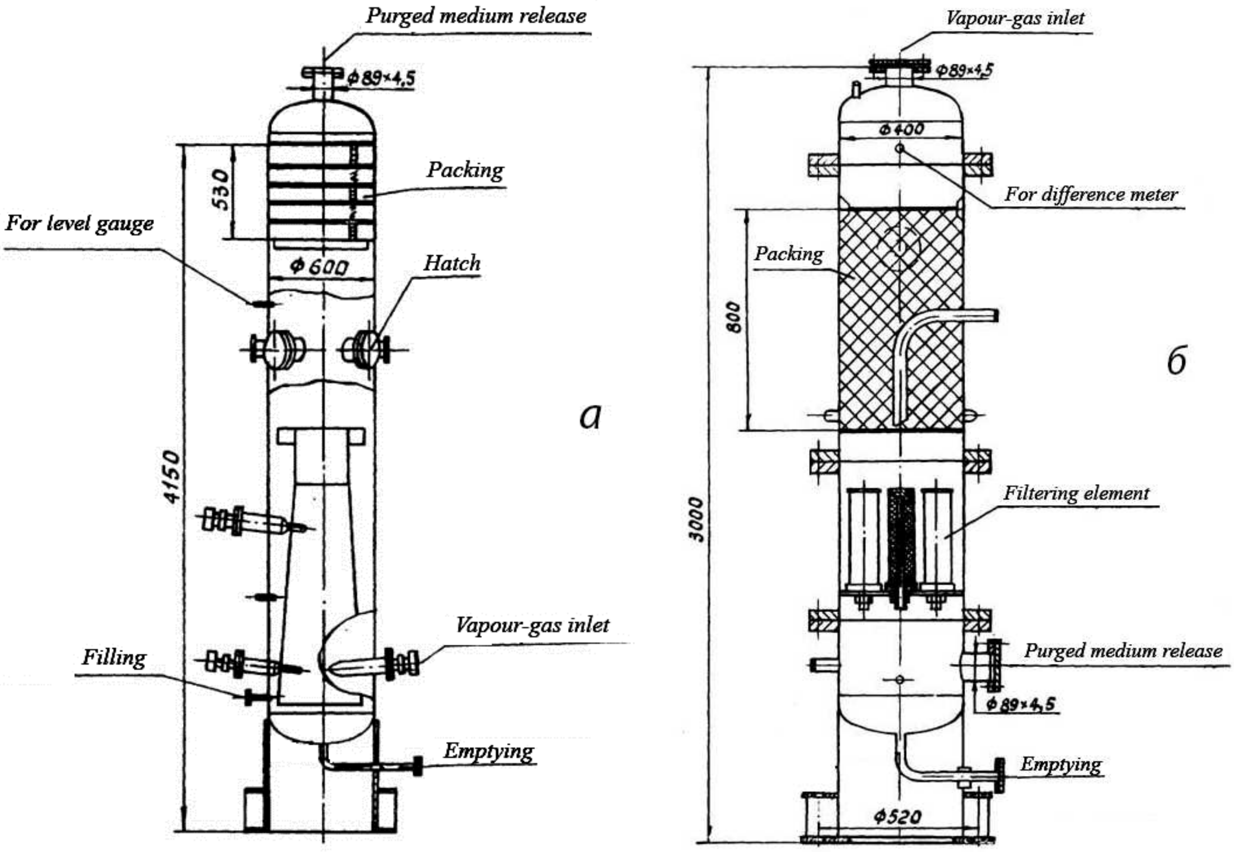

FIG. 2. Layout draw of jet filter and bed filter for gas purging
a – Jet filter; б – Bed filter («Thermoxid-58» packing).

In course of testing the model of the filtering system in Karlsruhe Nuclear Research Centre on Typhoon test bench, two series of experiments were conducted: testing at catching aerosols of uranine $C_{20}H_{10}O_5Na_2$ and basic experiments on barium sulfate $BaSO_4$ aerosols with the average particle diameter 0,7 μm, the distribution curve whereof is maximally close to the one expected at a severe accident calculated by NAUA code.

Figures 3 and 4 show the principal diagram and filter with «dry» purging of discharged gases, proposed for NPP with WWER-1000 reactor.

TABLE 6. CONDITIONS OF CONDUCTING EXPERIMENTS AND RESULTS OF CATCHING OF CS, MN, J SOLID AEROSOLS

Parameters	First series of experiments	Second series of experiments
Release of medium:		
Flow rate, m^3/s	*0,041*	*0,041*
Temperature, °C	*135*	*141*
Volume ratio of vapour	*0,012*	*0,42*
pressure (abs.), kPa	*268*	*258*
Saturation temperature, °C	*26*	*102*
Liquid:		
Initial temperature, °C	*14*	*21*
Initial level, m	*0,79*	*0,79*
pH initial value	*9*	*9*
Gas speed:		
Over liquid, m/s	*0,19*	*0,15*
Through packages, m/s	*0,44*	*0,33*
Duration of aerosols transmittance, min	30	30
Aerosols inlet (concentration), g/m^3:		
Cs	*13,7*	*12,9*
Mn	*4,9*	*4,7*
J	*1,54*	*1,02*
Full purging factor:		
Cs	*300000*	*370000*
Mn	*150000*	*800000*
J	*2000000*	*300000*

TABLE 7. RESULTS OF LAYER-WISE ANALYSIS OF FILTER FILLING WITH «THERMOXID-58» PACKING

The number of the layer	Distance from top of charging, cm	Layer height, cm	Mass of sorbent in layer, kg	Content of caught elements, mg/kg; Cs	Mn
1	12,7	12,7	32,27	2,8	1,08
2	22,4	9,7	24,56	2,4	0,78
3	33,0	10,6	27,11	1,8	0,70
4	44,0	11,0	27,86	1,8	0,70
5	52,0	8,0	20,31	1,2	0,64
6	61,4	9,4	23,96	1,0	0,40
7	69,5	8,1	20,66	1,0	0,40
8	80,0	10,5	26,56	0,8	0,22
9	Sorbent from cartridges		11,1	4,2	1,02
	Average content of elements in charging			1,8 mg/kg	0,63 mg/kg

TABLE 8 . CONDITIONS OF CONDUCTING EXPERIMENTS AND BASIC RESULTS OF BARIUM SULFATE AEROSOLS CATCHING ON TYPHOON TEST BENCH IN KARLSRUHE NUCLEAR RESEARCH CENTRE

Step	Medium purging factor
	Experiments on barium sulfate
Jet filter	10 – 100
Packing filter	500 – 5000
The entire system	5000 – 50 000

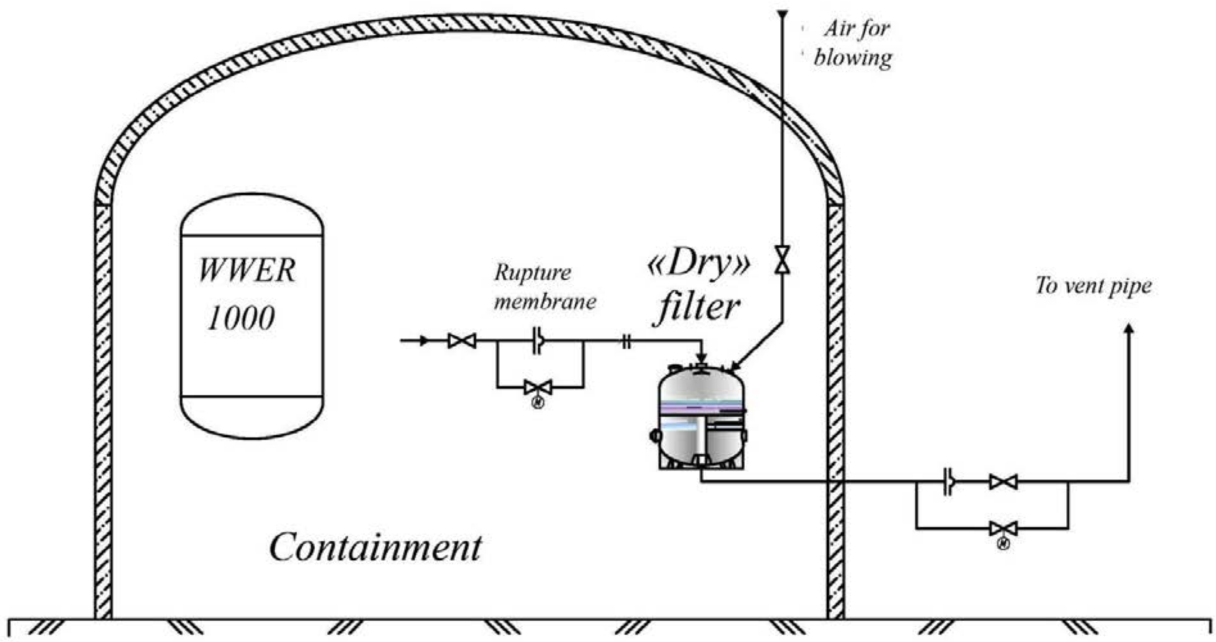

FIG. 3. Diagram of FPV filtering system with use of «dry» filter.

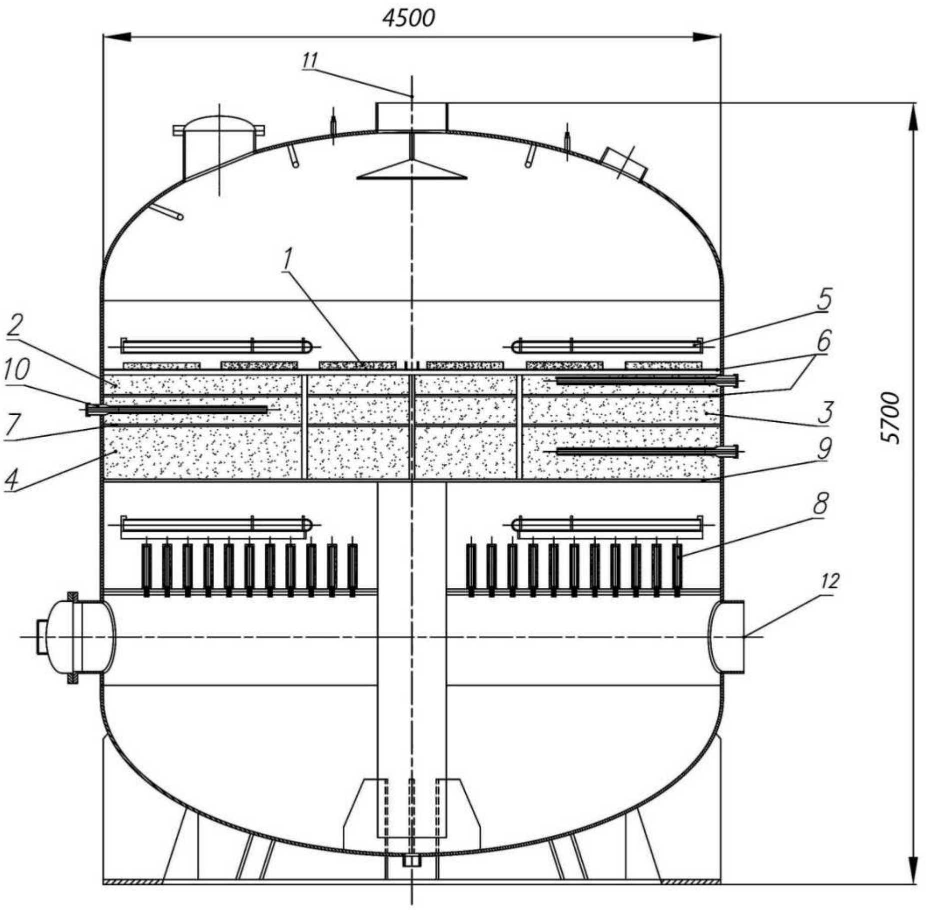

FIG. 4. Filter for purging of gaseous products of NPP accident: 1 Catalyst for hydrogen incineration; 2 Packing 2–4 mm; 3 Packing 1.6 – 2 mm; 4 Packing 0.4 – 1 mm ; 5 Tubular electric heater; 6 Mesh work 1.4 mm; 7 Mesh work 0.45 mm; 8 Filtering element; 9 Grating; 10 Sampler; 11 Working medium inlet; 12 Working medium outlet.

A filter with use of «dry» filtration method for purging gaseous products in an accident is a vertical vessel with a packing therein, which is a filtering material: heat-resistant, moisture-resistant and highly active to iodine gaseous forms sorbent on basis of titanium dioxide of various fractions. Coarse fractions of the sorbent are located as bed layers on a slit grating; fine fractions of the sorbent are located in filtering elements below the bed layer. Height of the packing, its fraction composition and layer-wise filling are chosen due to the most efficient purging of vapour-gas medium from aerosols and iodine volatile forms, iodine methyl included, and also due to sufficient dust holding capacity.

Hydrogen is present in the composition of vapour-gas medium which is removed from the reactor containment. Above the bed layer of the packing, there is a perforated sheet with pockets wherein catalyst is located for hydrogen incineration which provides explosion-safe conditions for operation of the filter. During incineration (re-combination) of hydrogen, temperature of the vapour-gas medium in the filter rises, and it improves efficiency of catching iodine gaseous forms and conditions of sediment of aerosols.

To improve efficiency of the filter and to exclude vapour condensation, the filter contains electric heaters and heat insulation which provide, in reserve mode, maintenance of temperature in the filter by 20–30 °C higher than the temperature of water vapour saturation at an accident.

At present, works are in progress aimed at further improvement of FPV in direction of modernization of design of the filter and applying a more efficient sorbent.

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CHARACTERIZATION OF THE PERFORMANCE OF WET SCRUBBERS USED IN FCVS

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Abstract. PSI has carried out research to develop and characterize wet scrubbers employed in filtered containment venting systems since early 1990s'. The work has included experimental and analytical characterization of the wet scrubbers as well as model development. The thermal-hydraulic characterization of the wet scrubbers has been extended to multiple venting cycles after 2011. Aerosol, iodine, and organic iodide retention in the wet scrubber has been determined in 1:1 height scale facility. The same facility has been used to determine the thermal-hydraulic characteristics relevant for FCVS, e.g. water inventory, swell level, and droplet release form the pool surface. An extensive research programme was dedicated for development of a method to improve the retention of organic iodides in the wet scrubbers, and to avoid the release of volatile iodine from the water pools in the medium and long term. Some examples of the results are presented in this paper.

1. INTRODUCTION

Many filtered containment venting systems (FCVS) employ wet scrubbers for removal of aerosols and gas phase iodine from the venting gas flow. The main principle of all the scrubbers is similar, i.e., they have an injection orifice or a venturi nozzle which injects the venting gas into a pool of water. The pool may contain internal structures to condition the gas flow and to break the gas globules to smaller bubbles to increase the interfacial area between the gas and liquid phase to enhance removal of aerosols and gas phase species. The liquid phase consists of de-mineralized water with added chemicals to improve the retention of gas phase iodine. Typical additives are sodium thiosulphate ($Na_2O_3S_2$) to bind iodine, and sodium hydroxide (NaOH) or sodium carbonate (Na_2CO_3) to keep the water pH high to avoid release of iodine from the liquid. In the upper parts of the scrubber vessel, droplet separators and / or additional aerosol filters are installed. In the recent years, new designs have been developed which incorporate an additional dry stage for removal of organic iodides downstream of the wet scrubber.

The retention of aerosols in the wet scrubbers is mainly based on physical behavior of the gas flow and the liquid, and their interactions in the injection orifice / venturi, and the hydrodynamics of the two-phase flow in the wet scrubber. The main removal mechanisms are the inertial forces of the aerosol particles at the injection and when they enter the liquid pool, as well as interception with the water droplets and bubbles. In general, applying high gas injection velocities, wet scrubbers have very high collection efficiency for aerosol particles. However, the collection efficiency depends on the particle size increasing with increasing particle sizes due to inertial forces, and having a minimum at around 0.1–0.5 μm [1].

The retention of gas phase species is a combination of mass transfer from the gas to the liquid phase, and the chemical effects determining the partitioning between the gas and the liquid. Here the chemical reactions with the radiolysis products in the liquid and air have to be accounted for.

Due to the very complex behavior of the two-phase flow in the injection, and in the water pool, and the complex chemical processes, the performance of wet scrubbers cannot be mechanistically modelled, and experiments are necessary to characterize their performance under different operating conditions. Models can and should then be developed based of the experimental data for further understanding and application of the results.

At Paul Scherrer Institut (PSI), research to characterize the performance of FCVS started in the early 1990s' with qualification of the FCVS by the Swiss company Sulzer for aerosol retention. Work continued with the measurement of elemental iodine (I_2) and organic iodide (CH_3I) retention in the same FCVS. After the renewed interest for FCVS following the accidents in Fukushima, research at PSI has continued by further experimental and modelling work. This work addresses issues which were not investigated earlier in detail, and which have proven to be necessary to investigate based on the recent research [2] as well as by the experience gained from Fukushima, such as the need for long term and / or repeated venting operation, and improved retention for organic iodides.

Some examples of the research carried out at PSI are given in this paper.

2. THERMAL-HYDRAULIC BEHAVIOR OF THE WET SCRUBBER

A 1:1 height scale test facility VEFITA, Figure 1, was built at PSI to investigate the thermal-hydraulic behavior of wet scrubbers during multiple venting cycles, including determination of temperature, water inventory and the level swell. In addition, the flow at the injection orifice can be visualized, and the droplet release from the water pool, and the collection efficiency of the end separators for the droplets can be studied.

FIG. 1. Height scale FCVS facility VEFITA.

2.1. Water inventory and level swell

For the present tests, the facility was equipped with an injection orifice, internal structures, and an end separator similar to the ones used in the FCVS designed by IMI Nuclear (former Sulzer). Water inventory during multiple venting cycles was measured, and compared with an analytical in-house model developed at PSI. The experimental results were found to be in good agreement with the model which is based on the first principles of thermal-hydraulic equations.

For the measurement of the swell level which is created in the pool due to the injection of gas, a guided radar probe was mounted inside the facility. The swell level was measured at different collapsed water levels, gas injection flow rates and gas compositions (nitrogen, steam, mixture of nitrogen and steam) in the pressure range 1–4 bar (abs). A RELAP5 model was developed to describe the FCVS including the internals, and the simulation results were compared with the experimental ones [3]. It was found that the code correctly predicted the trend of the swell level varying with respect to the gas flow and back pressure. Compared to the experimental data, RELAP5 model predicted the swell level well when the gas superficial velocity was low, but the prediction started diverging from the test data when gas superficial velocity became higher. At the ambient pressure, code model over predicted the swell level, probably due to the tendency of over-estimation of void fraction and lack of detailed internal multi-dimensional modeling. When the gas space was pressurized, code model underestimated the level swell, Figure 2.

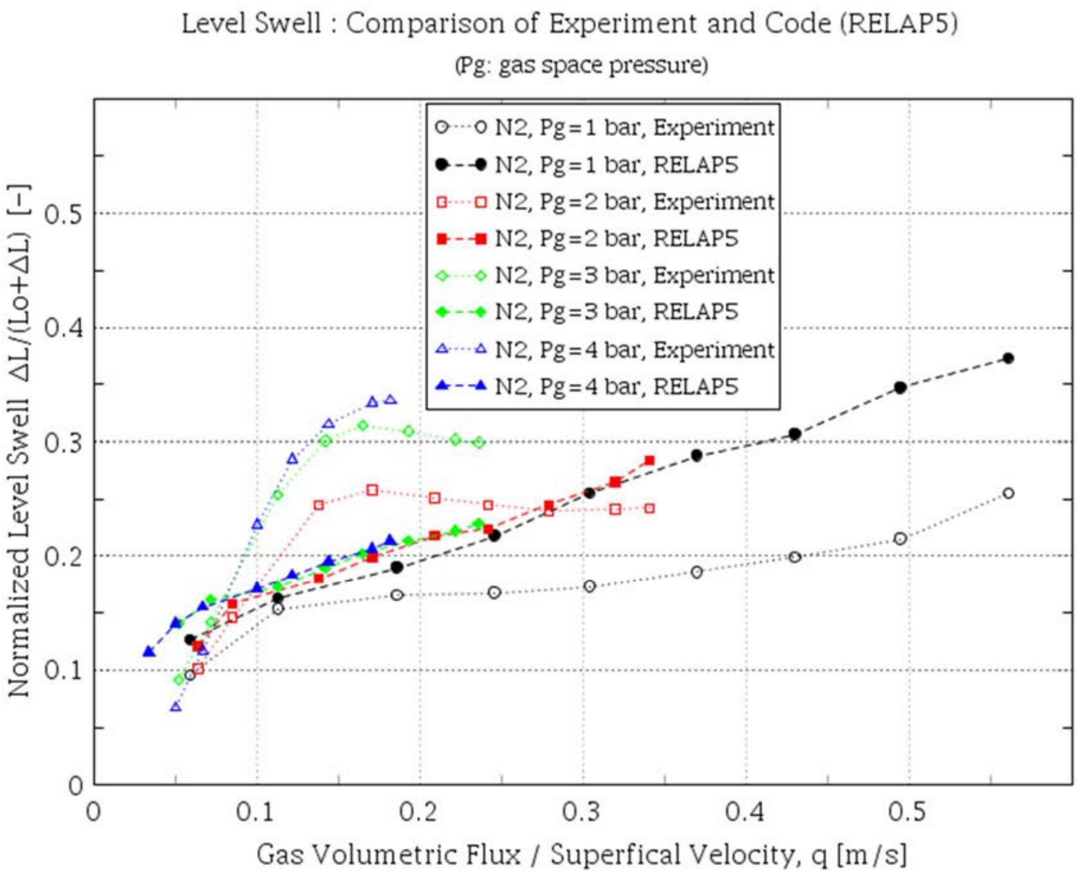

FIG. 2. Level swell simulation results by RELAP5 compared to the experimental data from VEFITA.

2.2. Droplet release from the pool

Droplets are generated on the liquid pool surface in the wet scrubbers when the gas bubbles reach the surface and burst. The concentration of soluble compounds in the droplets is equal to that in the bulk solution, but the concentration of insoluble aerosols may be significantly enriched [4]. Therefore, the droplets may carry activity out of the wet scrubber. To retain the droplets and thereby to avoid aerosol and iodine release from the FCVS, the wet scrubbers are equipped with droplet separators and / or aerosol filters above the pool. The efficiency of these devices combined with the generation rate of the droplets influences the release of aerosols and iodine from the FCVS. For this reason, it is important to measure the concentration of droplets generated on the pool surface, as well as the collection efficiency of the droplet separators / aerosol filters.

A Phase-Doppler Particle Analyser (PDPA) was installed in the VEFITA facility approximately 2 m above the water surface, and downstream of the first droplet separator stage to measure the size and concentration of droplets at these two locations. This way, the concentration of droplets which were transported out of the pool, and the collection efficiency of the first droplet separator stage could be determined. The measurements showed that a considerable amount of droplets were released from the pool surface, Figure 3, and the concentration was increasing with increasing water level in the pool. The major fraction of the droplets was collected in the first separator stage. The droplets above the pool were rather large with an average Sauter mean diameter (SMD) increasing with increasing gas flow rate. The first droplet separator stage removed efficiently the large droplets, and the droplet size was mainly less than 10 µm downstream of this droplet separation stage. In this configuration, the FCVS has two additional droplet separation stages to remove fine droplets.

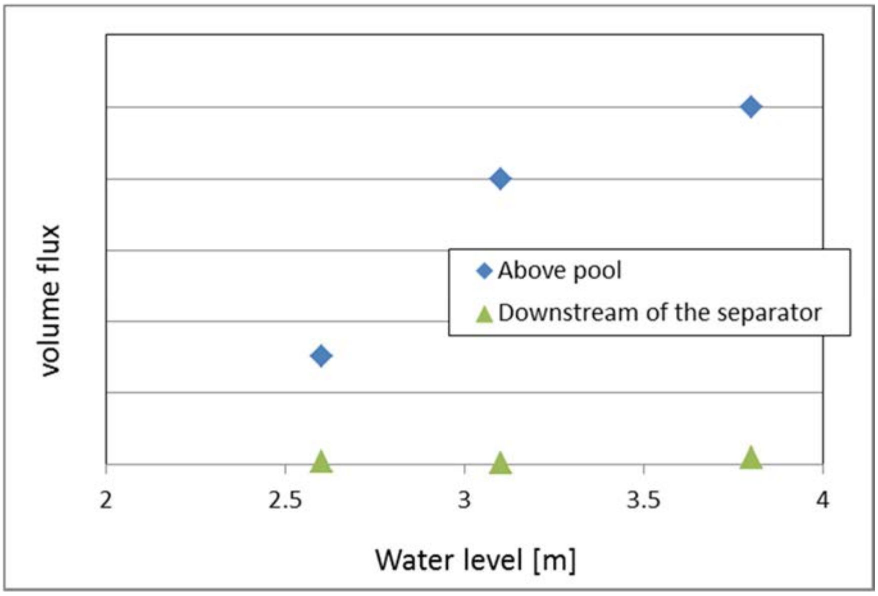

FIG. 3. Droplet volume flux above the pool and after the first separator stage.

3. RETENTION OF AEROSOLS

A series of tests was performed at PSI to determine the aerosol retention efficiency of the FCVS mock-up similar to the facility described in the previous section. The test section was a 1:1 height, reduced diameter facility with one layer of mixing elements and an end separator stage. The lower part of the facility which contained the water had a diameter of 0.5 m with an inner column with a diameter of 0.35 m.

All the tests were carried out with fine SnO_2 aerosol particles generated by an evaporation / condensation technique using a plasma torch system. The particles had an AMMD = 0.7-1.8 µm, Figure 4. The aerosol mass concentration at the test section inlet was 200-2000 mg/m^3 with about 20-30 % of the aerosol mass in particles smaller than 0.5 µm. The aerosol tests were conducted with the mixture of steam and nitrogen as the carrier gas with the gas flow rates 130-500 kg/h. The nitrogen mass fraction was approximately 50 %. The inlet pressure was 1.6-5 bar (abs) and the water temperature 90-120 °C. In the tests, the effect of reduced water level was investigated by starting the tests with a nominal water level, and then decreasing the water level during the test in one or two steps. The nominal water level was 1.0 m above the injection orifice.

The decontamination factor (DF) for aerosols was high, more than 30000, in almost all the tests at different pressures and different flow rates, Figure 5. At one test conditions, test S09 and the repetition test S10, the DF was lower than 10000 with the water level of 0.3 m above the inlet nozzle. With water level of 1.0 m above the nozzle, DF was always above 10000. With increasing water level, aerosol retention was found to increase significantly.

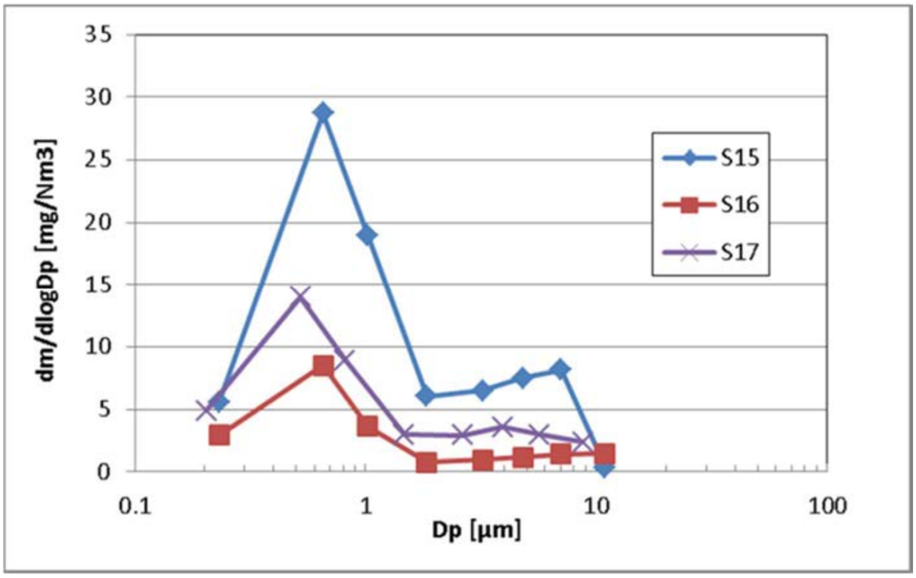

FIG. 4. The particle size distribution of the SnO_2 aerosol used for the aerosol retention tests. S15, S16 and S17 refer to different tests.

4. RETENTION OF IODINE AND ORGANIC IODIDE

4.1. Experiments in 1:1 height scale for iodine retention

Iodine (I_2) and organic iodide (CH_3I) retention in the FCVS was investigated in 1:1 height scale facility in five tests in which either I_2, CH_3I, or both were injected. CH_3I was used to represent the organic iodides as the presumably most volatile and abundant of them. The temperature in the tests was varied in the range 70–120 °C, and the FCVS inlet pressure was 4.4 bar (abs). The steam mass fraction in the gas flow was approximately 70-80 %, and the total flow rate was 325-350 kg/h. The additives relevant to FCVS were used in the water in the wet scrubber, i.e., $Na_2S_2O_3$ for retention of iodine, and Na_2CO_3 for pH control in the tests when high pH was used. No other additives were used in the water. The experimental conditions are given in Table 1.

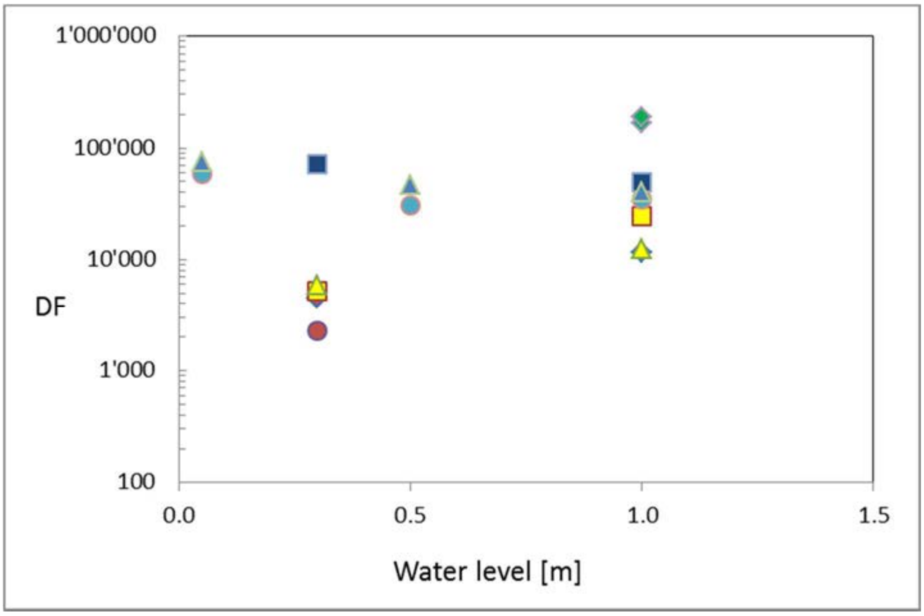

FIG. 5. Decontamination factor for aerosols in tests with maximum submergence of 1.0 m of the injection orifice.

TABLE 1. EXPERIMENTAL CONDITIONS IN THE I2 AND CH3I RETENTION TESTS IN THE FCVS

Test #	I_2	CH_3I	$Na_2S_2O_3$	Na_2CO_3	Water level [m]	Water T [°C]	pH at 20°C
B01	Yes	-	Yes	-	0.9	120	4.5-5.5
B02	-	Yes	Yes	Yes	0.9	120	9
B03	Yes	Yes	Yes	Yes	1.2	120	9
B04	-	Yes	Yes	Yes	0.75-1.6	70, 120	9
B05	Yes	-	Yes	-	2.1	120	4.5-5.5

The experiments showed the dependence of gaseous elemental iodine retention on the pH of the water in the wet scrubber, and the water level above the injection orifice, Figure 6 [5]. All the iodine retention tests were carried out at the water temperature of about 120 °C, see Table 1. The measured decontamination factors for iodine were around 300–400 with the water level of 1.2 m above the orifice at the initial pH of 11.25 (pH at 20 °C), Figure 6. The higher water level would increase the decontamination factor. Consequently, it can be seen that given a sufficient water level above the injection orifice and a high pH in the wet scrubber, high retention of elemental iodine in the wet scrubber can be achieved.

Organic iodide (CH_3I) retention was investigated in three experiments. The retention was found to depend on the water level and temperature. When the water temperature decreased to 70 °C and below, and the water level above the injection orifice was approximately 1 m or less, the CH_3I retention was low with the decontamination factors 2-10. With the water level of 1.6 m and more, and T = 120 °C, decontamination factor was more than 10. The effect of pH on the retention of CH_3I was not investigated in this experimental programme.

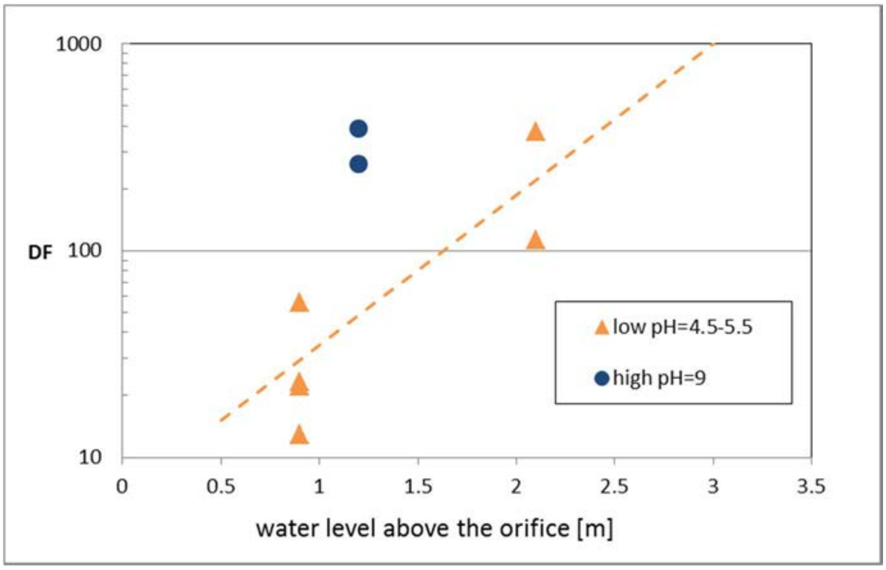

FIG. 6. Retention of elemental iodine in the 1:1 height scale facility (tests B01, B03, B05, Table 2). The dependence of the decontamination factor on the water level above the injection orifice and the pH.

Based on the tests, it was concluded that with the existing FCVSs, sufficient retention of gaseous elemental iodine can be achieved when the water level in the wet scrubber can be maintained at a high level, and the pH is high. However, the retention of CH_3I without any further actions is relatively modest. It has to be noted that other organic iodides than methyl iodide were not investigated.

4.2. Small-scale experiments for the decomposition of CH_3I

Due to relatively modest retention of organic iodide in the wet scrubber, research was carried out at PSI to improve the retention under FCVS conditions. The objective of the research and development programme was to establish a process for a fast and effective decomposition of gaseous methyl iodide and to reduce iodine volatility to a minimum or, ideally, to hinder it completely in the medium and long term also at relatively low pH of the water. In the development programme, PSI found a second additive to act as a co-additive to the widely used $Na_2S_2O_3$ to increase the CH_3I decomposition rate. The second additive is a quaternary long chain ammonium salt, Aliquat336®. It contains a mixture of C_8 and C_{10} chains with C_8 predominating. Aliquat336® is a commercial reagent [6], which is successfully applied in nuclear technological processes, such as spent fuel reprocessing and other metallurgical processes for metal extraction.

In the water, Aliquat336® has a dual function. It functions as a phase transfer catalyst by clustering both reactants, i.e. $Na_2S_2O_3$ ions and the CH_3I molecules together to accelerate their interaction. Thus the decomposition of CH_3I is enhanced. In addition, Aliquat336® contains an inorganic anion (chloride) which can act as an ion exchange for iodide ions (I-) thus preventing their radiolytic oxidation into I_2. Iodide ions are the decomposition product of both I_2 and organic iodides in the water, and this way, they can be bound in the water. This reduces the release of volatile iodine from the water in the medium and long term even with decreasing water pH as a result of e.g. the absorption of acidic gases. The reactions relevant to the behavior of iodine in water pools have been captured in the code PSIodine [7].

More details of the additive and its use can be found in Ref. [8].

5. FINAL REMARKS

PSI has carried out research to characterize and improve FCVSs since early 1990s'. Both experimental and model development work have been conducted to describe the thermal-hydraulic behavior of the wet scrubbers, and to determine aerosol and iodine / iodide retention in the FCVS. A method was developed to increase CH_3I retention in the water by using an additional additive. This additive also helps retain the iodide ions in the water in the medium and long term.

Investigation is ongoing to apply the improved CH_3I retention process in a large scale, and to characterize the retention of iodide in the water in the multiple venting cycles and in the long term. In addition, different FCVS geometries for improved performance will be investigated.

To improve the capabilities of the pool scrubbing codes to address FCVS, a better understanding of the gas-water two-phase flow in the wet scrubber is needed. For this, a hydrodynamic facility was built at PSI to be able to measure the two-phase flow characteristics with a wire-mesh sensor [9]. Some results of this work are given in another paper at this conference [10].

ACKNOWLEDGEMENTS

The work at PSI was carried out over many years with the support of numerous people from the severe accident group and the PSI Hot Laboratory. Even though it is not possible to list them all by name here, we are very grateful for the contribution of every one of them, and want to warmly thank all the staff involved in the work. A large fraction of the work on the novel iodine process was carried out by R. Cripps, H. Bruchertseifer and S. Güntay who have already retired. Their work is gratefully acknowledged.

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LESSONS LEARNED FROM ANALYTICAL RE-EVALUATION OF A VENTURI SCRUBBER VENTING SYSTEMS IMPLEMENTED IN GERMAN NPPS THROUGH COCOSYS ANALYSES

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1. INTRODUCTION

Already in December 1986, the German Reactor Safety Commission (RSK) specified the requirements for design, operation and construction of a filtered containment venting system (FCVS) for the German PWRs [1] and in June 1987 for the German BWRs [2]. Within the framework of the safety reviews of the NPPs performed in the late 80s and the concomitant final report as of 23 November 1988, the RSK recommended to give priority to the containment's global integrity and to the prevention of high pressure containment failure of German NPPs [3]. Thus the point of time for opening the corresponding relief valves of the venting system is determined by a critical pressure build-up in the containment. FCVS have been installed in all German BWR between 1988 and 1990 and in the PWR between 1990 and 2003 [4].

The most commonly used filtered venting system consists of a sliding pressure venturi scrubber together with a metal fiber fleece filter downstream of the venturi scrubber [5]. The RSK requested a retention rate of 99 % for aerosols and 90 % for elementary iodine. The retention capacity of the FCVS was validated in the late 80s by experiments mainly performed at the JAVA test facility [6].

After the Fukushima-accident the national [7] and ENSREG stress test [8] examinations showed that venting under SBO conditions is possible by manual operation of the valves etc. This was one main objective of stress test analyses to demonstrate that at least venting functions without power supply. The development of SAMG for German plants was another issue. There measures to re-establish the power have been described and as well measures to operate air ventilation systems during venting.

Recently a re-evaluation of the installed FCVS with a sliding pressure venturi scrubber and an internal metal fiber fleece filter has been done by GRS with regard to its retention capacity for aerosol and iodine and to its overall system behavior (e.g. thermodynamic conditions and decay power stored in the scrubber tank, hydrogen issue in the off-gas system). The re-evaluation is based on the application of current knowledge and modern computer codes especially the containment code system COCOSYS developed and applied at GRS [9]. The specific COCOSYS models developed for the FCVS components and validated using experimental results are described.

2. FILTERED CONTAINMENT VENTING SYSTEM (FCVS) WITH VENTURI SCRUBBER

Sliding pressure venturi scrubbers are installed in six of eight nuclear power plants being in operation in Germany today. The steam-gas mixture coming from the containment (containment-vent-flow) enters the venturi scrubber section (Fig. 1). The venturi nozzle(s) are covered with water. Near the throat of the nozzle there are slits through which water is sucked into the nozzle where it is atomized creating a large surface. This enables a strong transfer of gaseous I_2 from the containment-vent-flow into the water. Also aerosol particles are trapped by the water. NaOH is added to the water in order to achieve a high pH-value. This shifts the equilibrium of the I_2-hydrolysis-reaction (1) from the volatile I_2 to the non-volatile I- and IO_3- which stay dissolved in the water.

$$3\ I_2 + 3\ H_2O <=> IO_3^- + 5\ I^- + 6\ H^+ \qquad (1)$$

In addition $Na_2S_2O_3$ is added which transforms I_2 into I- also in case of lower pH-values. In the scrubber the main part of the aerosols and gaseous iodine (I_2) are retained. Then the containment-vent-flow goes through the metal fiber fleece filter which acts as a fine filtration of the remaining aerosols. The venting mass flow rate is limited by the throttling orifice located downstream the scrubber tank. The off-gas is directed through separate piping into the environment in case of BWR while it is mixed with off-gas of reactor building ventilation systems upstream the stack in a chamber in case of PWR.

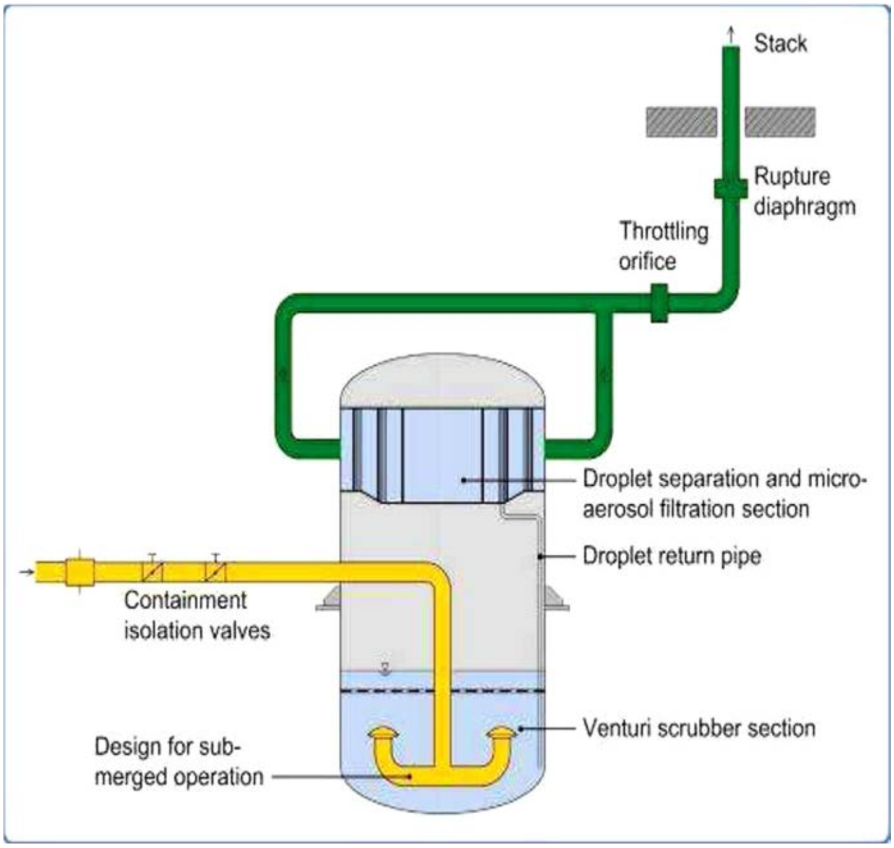

FIG. 1. FCVS with Venturi scrubber /8/.

3. EXPERIMENTAL VALIDATION OF THE VENTURI-SCRUBBER PERFORMANCE

The retention of aerosol and iodine in the venture-scrubber was mainly validated in the JAVA-experiments using the original venturi-nozzle [5, 6, 10]. The conditions in the scrubber-tank were accident-typical. In the iodine tests the pressure was mainly 4 to 6 bar, and the temperature of the scrubber water was 144 to 159 °C (saturated conditions). Some tests were done with 1.9 bar. The volume flow rate at the entrance of the scrubber was varied between 150 and 1200 m^3/h per venturi-nozzle. For pH-values of 12.7 an I_2-retention of more than 99 % was measured. Also tests with decreased pH-value caused by addition of CO_2 showed that high I_2-retention. Tests investigating the scrubber performance under high radiation in the water could not be performed. But for high pH-values the equilibrium of the

I_2-hydrolysis-reaction stays on the side of the non-volatile I- and IO_3-.-In the power plants it is possible to add NaOH into the scrubber-water to increase the pH-value. The release from scrubber water loaded with iodine was tested by blowing steam through the water. With pH of 9.2 less than 1 % of the inventory in the water was released within 21 h.

4. USED CODE-SYSTEM FOR RE-EVALUATION OF FCVS

The COntainment COde SYStem COCOSYS is developed by GRS. It provides a code system on the basis of mechanistic models for the comprehensive simulation of all relevant processes and plant states during severe accidents in the containments of light water reactors, also covering the design basis accidents. The inner part of the code system consists of three main modules:

- the thermal hydraulic main module (THY) for the simulation of the thermodynamic behavior and transient processes like hydrogen deflagration, simulation of safety systems;
- the aerosol fission product main module (AFP) for the simulation of the fission product behavior, decay heat release and chemical reactions;
- the core concrete interaction main module (CCI) for the simulation of the core melt behavior, concrete erosion, releases from core melt and chemical processes inside the core melt.

5. COCOSYS MODELLING OF THE FCVS OF A PWR KONVOI

The filtered venting system with the venturi scrubber, the metal fiber filter, the pipe lines and the throttling orifice are simulated with COCOSYS (Fig. 2). A special venturi scrubber model was developed and implemented into the COCOSYS code. The thermal-hydraulic part simulates all relevant processes: the atmospheric flow through the venturi nozzle(s), the pressure and the pressure drop in the venturi scrubber, the temperature (heat up) of the water, the atmosphere and the vessel walls, the condensation of steam in the water and on the vessel walls, the condensation and evaporation of steam, the increase and decrease of the water level. The aerosol and an iodine part simulates: the aerosol retention by the venturi nozzle(s) in the water and in the metal fiber filter, the transition of I_2 from the gas phase into the water phase in the venturi nozzle(s), the transformation of the gaseous I_2 into I- and IO_3- depending on the pH value and the retention of I- and IO_3- in the water. For development and validation of the scrubber model data gained from experiments performed at the JAVA facility [6] and the ACE facility [11] were used.

The flow resistances of the pipe lines of the FCVS were modelled based on the data provided by the reference PWR power plant. The analyses showed that critical flow conditions in the throttling orifice during the venting process exist as designed. The throttling orifice is necessary to limit the mass flow and to guarantee a volume flow rate through the venturi nozzle(s) which provides high aerosol and iodine retention as it was experimentally justified. In this case special attention has to be taken on the contraction coefficient, which describes a contraction of the flow area downstream of the orifice. The contraction coefficient for incompressible flow (Ci) is 0.62. This is also the case for a small difference between the pressures up- and downstream of the orifice (pressure ratio of 1). In the German PWR FCVS being analysed the pressure upstream of the orifice can be up to 5 bar, while the downstream pressure is about 1 bar (pressure ratio of 0.2). According to information provided in [12–15] the contraction coefficient increases to 0.85-0.9, respectively the effective flow area increases by a factor of 1.4. A diagram shown in Ref. [13] (Fig. 3) illustrates the dependence of the contraction coefficient on the pressure ratio.

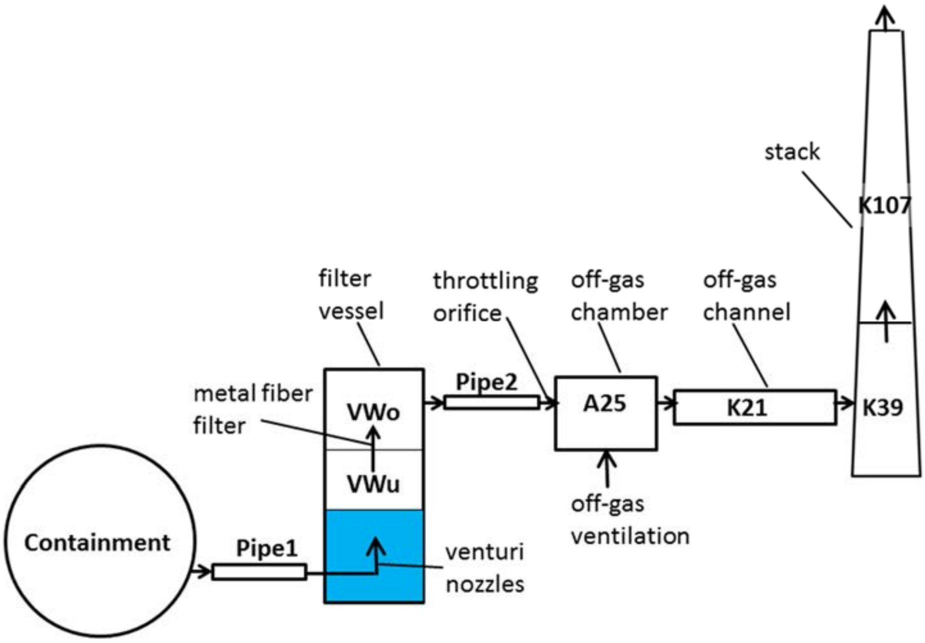

FIG. 2. COCOSYS nodalisation of the FCVS and the off-gas chamber, -channel and stack.

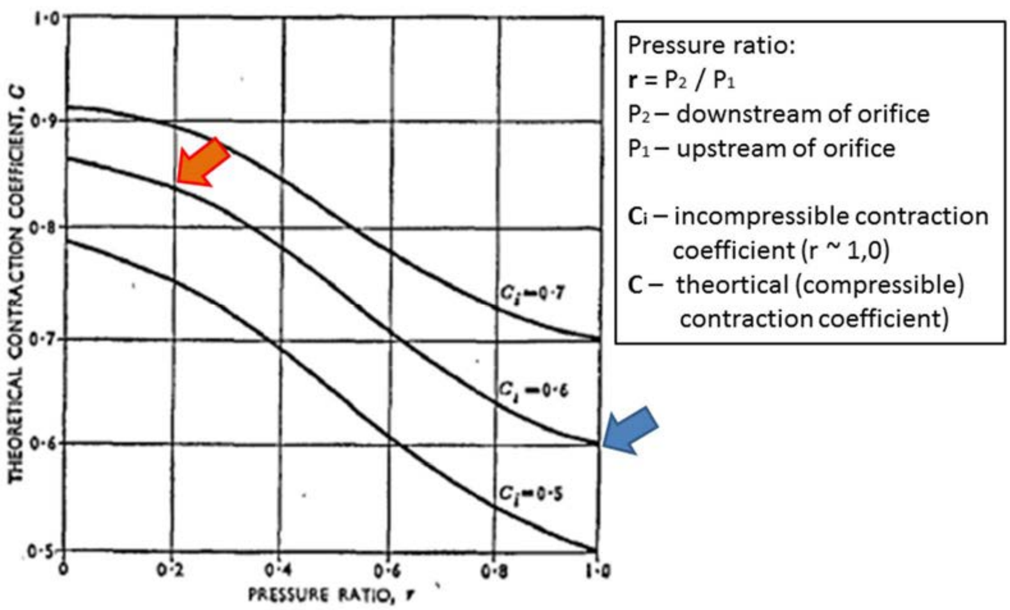

FIG. 3. Dependence of the contraction coefficient on the pressure ratio [13].

6. COCOSYS MODELLING OF THE CONSIDERED SEVERE ACCIDENT SCENARIOS

The containment of the PWR reference plant (Fig. 4) is formed by a steel shell with a sphere diameter of 56 m. Between the steel shell and the outer concrete shell there are the annular gap and the annular compartments located. The reactor pressure vessel (RPV) and the steam-generators (SGs) are placed in the equipment compartments. The radial boundary of these rooms is the missile protection cylinder. During normal operation doors and rupture foils at different locations separate the equipment compartments from the dome and compartments behind the missile protection cylinder (the periphery). The entrance of the venting line is in the lower periphery.

In the COCOSYS nodalisation the containment (Fig. 4) is subdivided into several zones in order to simulate difference in the atmospheric conditions and the local distribution of the fission products. The FCVS model (Fig. 2) is attached to the containment model. The zones representing the annular gap and the annular compartments are not shown in the figure.

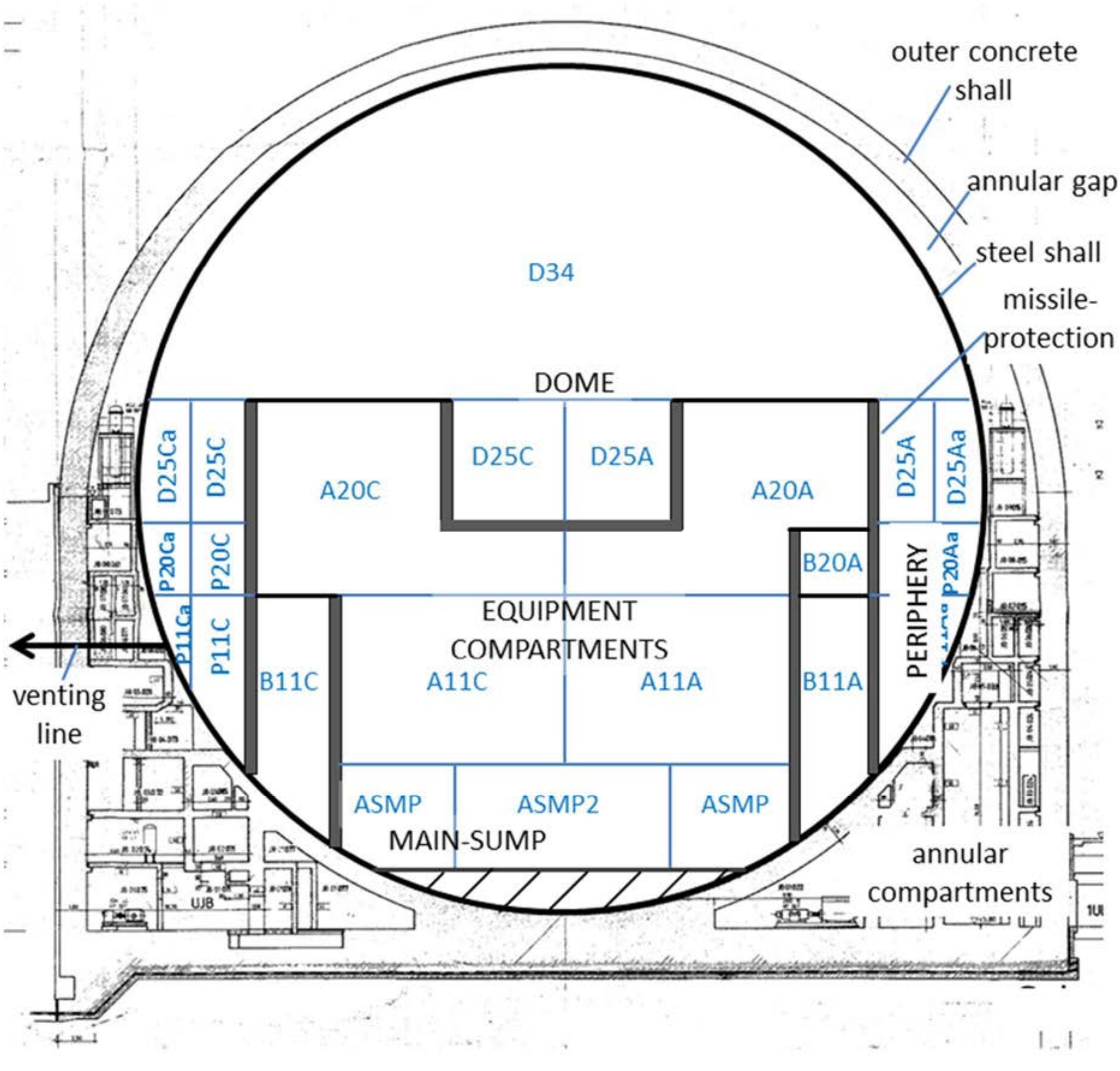

FIG. 4. COCOSYS nodalisation of the PWR containment and reactor building (annulus).

The releases from the primary circuit into the containment were taken from MELCOR analysis [16]. After failure of the reactor pressure vessel the gas and energy releases were calculated by the COCOSYS core concrete interaction main module. The pH-model of COCOSYS was not used. For the water in the sump and in puddles in the containment the pH-value is kept constant to 5.7. This shifts the equilibrium of the I_2-hydrolysis-reaction towards I_2 and is therefore conservative in respect to high I_2-concentrations in the atmosphere. For the scrubber water the pH-value is fixed to 12.7 assuming the venting would be stopped and NaOH would be injected in case the pH-value becomes too small.

Two core melt scenarios were investigated.

- The MBL case is a Medium Break LOCA (200 cm²) in a hot leg. It was assumed that the emergency core cooling system (ECCS) was working up to the switch over of the ECCS system into sump recirculation mode; than it failed and core degradation started.
- The ND* case is a low pressure transient caused by a total failure of steam-generator feed water supply. Here a complete failure of the ECCS (except accumulators) is assumed. In order to avoid a high pressure RPV failure primary depressurization was assumed prior to core damage as described in accident management procedures.

The two cases have been selected to account for main important differences with regard to the conditions inside the containment prior to containment venting. There are significant differences in the two cases with regard to the boundary conditions, the core degradation processes, the released amount of Ag aerosol, and the sump water mass being present in the containment, which dependents on the availability of the ECCS prior to core degradation. In the MBL case the sump mass is about 5 times larger than in the ND* case. In the analyses of the base cases for which main events are described in Table 1, cold water injection into the containment sump during venting as recommended in the SAMG has been neglected (maximum filter load).

Electric power supply is available in the MBL- and ND*-cases. SBO (station black out) was not analyzed, as it has a much lower probability in German plants as MBL- and ND* not only due to the multiple measures implemented to guarantee power supply. The SBO-transient would be similar to the ND*-transient with regard to the timing of the accident and the expected containment conditions.

TABLE 1. MAIN RESULTS OF THE TWO BASE CASE SCENARIOS CONSIDERED

Event	ND*-Scenario	MBL-Scenario
Initiating event	0,0 d	0,0 d
Begin core melting	0,14 d	0,24 d
RPV failure, melt release, begin MCCI	0,27 d	0,37 d
Sump water ingression into cavity	0,49 d	0,59 d
Begin 1. containment venting	3,2 d	4,0 d
Stop of containment venting or analysis	~4,5 d	10 d

7. SIMULATION RESULTS — CONTAINMENT PRESSURE

The containment pressure for the two base cases is shown in Fig. 5 in blue for the ND* case and in red for the MBL case. The additional case in green shows the effect of water addition to the sump during venting (as recommended by the SAMGs) in the MBL case.

The early containment pressure history is different and depends on the selected boundary conditions of the accident. When the core melt in the cavity gets in contact with sump water the containment pressure starts to increase continuously due to the strong evaporation in all cases. In the ND* case (blue) the pressure of 7 bar to start the venting is reached about 20 h earlier than in MBL case (red). The initial pressure decrease due to the venting process is similar. The differences in the long term pressure history are caused by the difference in the sump water mass. In the ND* case (blue) already at 4.2 days the sump is dried out leading to an accelerated pressure decrease and an assumed stop of the venting at about 4.5 days when the pressure reaches 3.4 bar. Afterwards the pressure would increases slowly caused by gas production of the ongoing MCCI and an increase of the atmosphere temperature. Under similar conditions the containment depressurization is not so efficient in the MBL scenario (green) and the pressure stabilizes at about 4.4 bar due to the continuous sump evaporation. The analysis was continued until 10 days without closing the venting line, what means that venting could go on until dry out of the sump.

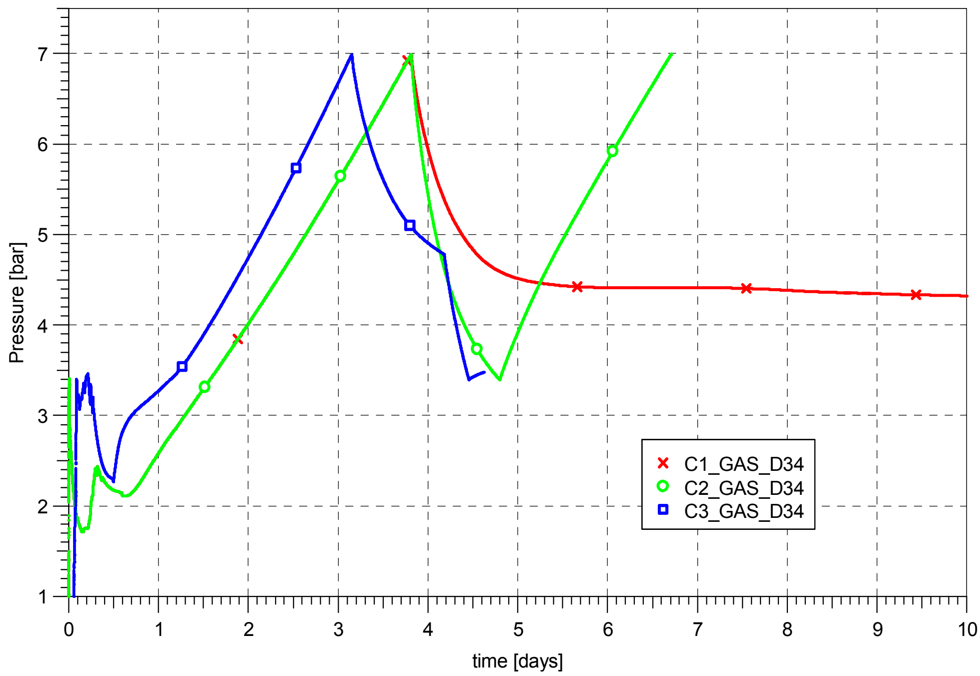

FIG. 5. Containment pressure courses of different scenarios, C1=MBL base case, C2=MBL with water injection, C3=ND base case.*

In case of an injection of 10 kg/s water during venting exemplarily analysed for the MBL case (green), the sump evaporation is reduced resulting in a faster pressure decrease. In this calculation it is assumed that venting and the water injection is stopped when 3.4 bar are reached. Then the pressure increases again and venting has to be repeated.

8. SIMULATION RESULTS — HYDROGEN BEHAVIOR

In all German PWR passive autocatalytic recombiners (PAR) are installed to recombine the hydrogen which is produced during core degradation and MCCI. The oxygen in the containment is used up by the recombination within about 24 h, than the recombination is finished. Hydrogen is however still produced by MCCI. In the MBL case at begin of venting there are ~3000 kg of hydrogen in the containment. Atmospheric flows create an unequal distribution. While the average hydrogen concentration is 12 vol%, 27.5 vol% are calculated in the lower periphery. This result is achieved under the assumption that atmosphere flow between periphery and equipment compartments is limited (no large open connections). In addition the MCCI calculation is conservative in respect to concrete erosion and hydrogen production; therefor 27.5 vol% is considered to be a conservative high value.

Containment-vent-flow originating in the lower periphery flows through the vent line and the scrubber into the common off-gas chamber and from there via the off-gas channel into the stack (Fig. 2). There it is mixed with air coming from the off-gas of other buildings. According to the SAMGs the off-gas ventilation system should be operated to prevent the build-up of combustible gas mixtures. The calculation of the MBL case with conservative gas concentration of 27.5 vol% hydrogen in the periphery shows that this measure keeps the gas mixture in the off-gas chamber, the channel and the stack below the flammability limit.

Although electric power supply is available in the analysed cases calculations have been performed assuming the off-gas ventilation system is not operating. Then combustible mixtures have been calculated in the off-gas chamber, the channel and the stack for 15 to 30 min in a period just after the venting has started. These results have been gained not only for the case with conservative high hydrogen concentration, but as well for the other case with 12 vol% hydrogen in the lower periphery. After this initial phase with air being present in the off-gas chamber and the stack the air has been replaced by inert atmosphere from the containment released through the venting process. As the average velocity at the exit of the stack is only 0.5 m/s additional investigations have been done. They show that there will be counter-currant flow. Therefor combustible mixtures in the upper part of the stack are probable for long time spans if the ventilation is not operating.

9. SIMULATION RESULTS — WATER LEVEL IN SCRUBBER TANK

The water level in the scrubber tank (Fig. 6) is shown for the most challenging conditions — the MBL base case without water injection into the containment. The level increases at start of venting due to condensation of steam from the containment-vent-flow in the cold scrubber water. The water is heated up to saturation. Then it starts evaporating because the enthalpy of the steam coming from the containment is higher than the saturation enthalpy in the scrubber and because of the decay heat of the fission products accumulated in the water. At the end of the calculation the water level is 0.8 m above the slits through which water is sucked into the venture nozzles. In JAVA experiments with a lower water level (venturi outlet already uncovered) iodine was still separated in the venture-nozzle. In the power plant it is described in the FCVS procedures respectively the SAMG how to inject water in order to increase the level.

10. SIMULATION RESULTS — IODINE BEHAVIOR

The iodine inventory of the core is assumed to be completely released during core melting, 99% in form of CsI aerosol and 1% as I_2. The aerosols are transported inside the containment by the atmospheric convection processes. They deposit on walls and floors due to different processes and especially soluble aerosols are transported by the water drainage into the sump, CsI in the sump is dissolved in form of I-. In the sump iodine reactions take place with surfaces and also with silver forming AgI. The radioactive radiation transforms I- into the gaseous I_2 which can get into the atmosphere above the sump. At the start of the containment venting an intensified sump boiling begins resulting in a strong phase transition of I_2 from the sump water to the atmosphere. Results are shown for the MBL base case in

Fig. 7. There an increase of the I_2 concentration in the equipment compartments can be seen after venting starts.

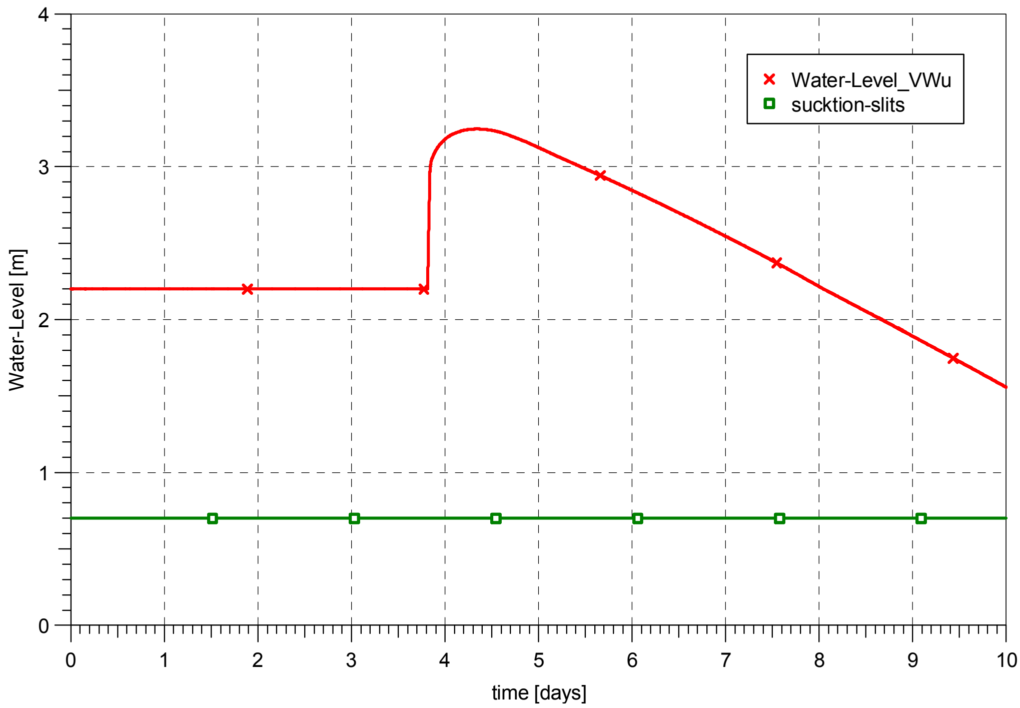

FIG. 6. Water level in scrubber tank, MBL base case.

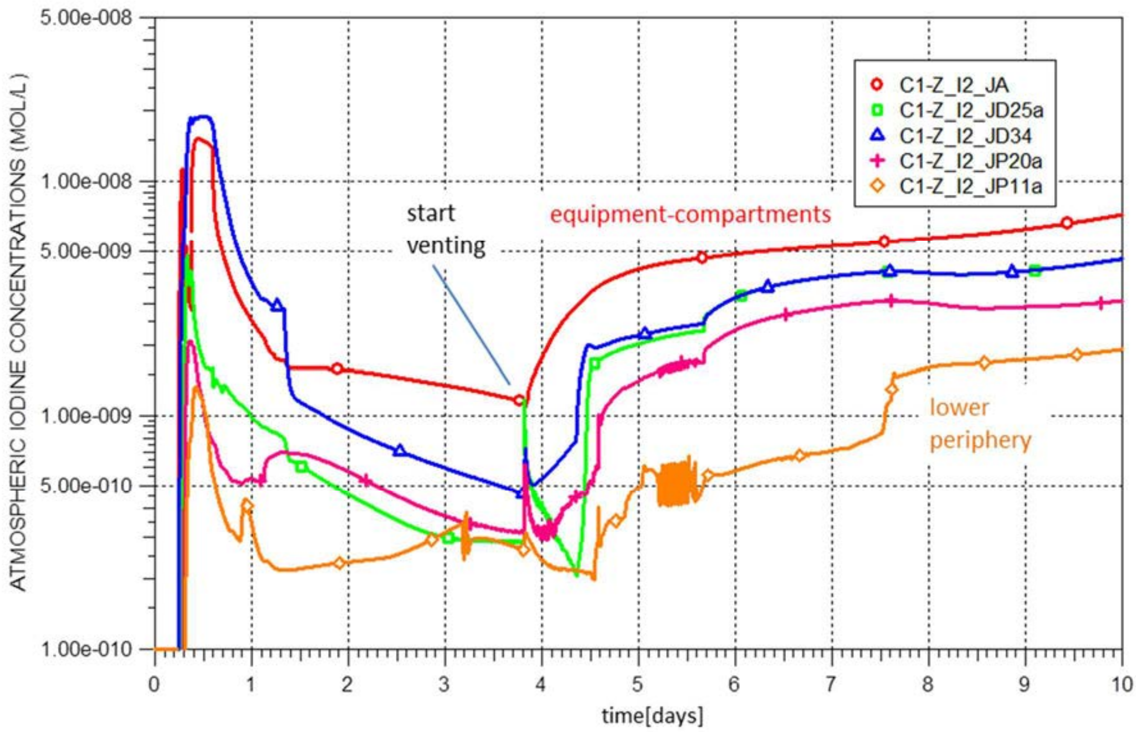

FIG. 7. Local iodine concentrations in the containment, MBL base case.

In the atmosphere region several iodine reactions are calculated. I_2 reacts with (deposits on) walls and floors, especially with painted surfaces. In case of strong enough wall condensation it can be washed from the walls and can be transported by the drainage back to the sump. Caused by temperature increase and radioactive radiation it can also be released into the atmosphere mainly in form of organic iodine. Radioactive radiation transforms part of the organic iodine back into I_2. Because of the iodine reactions the I_2 concentration decreases on its way from the source (equipment compartments) to the entrance of the venting line (lower periphery).

The calculation results showed that they are conservative in respect to high iodine concentrations in the atmosphere. The used model [17] overestimates the release of I_2 out of a boiling sump. In addition the formation of AgI is underestimated in the sump. Also the release of I_2 from the walls in the atmosphere region is tended to be overestimated. The consequences are discussed further.

11. SIMULATION RESULTS — FCVS FILTER LOADS

The filter loads are developed from the MBL base case calculation (without water injection) at 10 days. For the ND* calculation the results had to be taken already at 4.17 days, before the complete dry-out of the sump, because afterwards the iodine model works outside its validity range.

The calculated retention of aerosols and I_2 is higher than requested by RSK (Table 2). The aerosols reaching the filter originate from a small long term release by MCCI and resuspension from the sump surface, as the large aerosol masses released from the primary circuit and at begin of MCCI have already deposited when venting starts. The major part of the retained aerosol is found in the scrubber water, only a small part in metal fiber fleece filter as expected. The load with aerosols is much lower than the RSK request. However the load of decay heat is a factor of 2 higher, which is mainly caused by the higher iodine amount. The main effect of a higher decay heat is a stronger evaporation of the scrubber water during venting. An impairment of the filtration efficiency is not deduced. As the values for ND* are developed at an earlier time point of the accident, the decay heat of iodine is higher than for MBL, although the iodine mass is lower.

TABLE 2. COMPARISON OF THE COCOSYS RESULTS WITH THE RSK REQUESTS

	RSK	**COCOSYS**	
		MBL	**ND***
Aerosol mass in scrubber water (kg)	60	11.6	1.9
Decay heat (kW)			
Total	7	15.1	14.3
Iodine	5	11.6	13.1
Other fission products	2	3.7	1.2
Fission product retention (%)			
Aerosols	≥ 99,9	> 99,99	> 99,99
I_2	≥ 90,0	> 99[1]	> 99[1]
Organic iodine	0,0	0,0[2]	0,0[2]
Iodine in scrubber water (kg)		0.432	0.165

[1] mean value over the venting period, [2] no RSK requests, no simulation in the model

12. SUMMARY

Two representative severe accident scenarios for a German PWR KONVOI have been investigated. The results confirm first the necessity to operate the off-gas ventilation system of the reactor building in parallel to the containment venting process, what is caused by the specific systems design. It shall keep the hydrogen concentration in the off-gas system and the stack, where the venting gas is directed into, below the flammability limit. This measure is already recommended by the Severe Accident Management Guidelines (SAMGs) of the NPPs. The results show further that the calculated aerosol and iodine retention of the FCVS is higher than the one requested by the RSK [1–3]. The filter load with aerosols is lower than the request one; however the load due to the decay heat is a factor of 2 higher, which is mainly caused by iodine. An impairment of the filtration efficiency is not deduced.

ACKNOWLEDGEMENT

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CFD CALCULATIONS IN A MARK II VENTING SYSTEMS

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Abstract. In the Laguna Verde Nuclear Power Plant (BWR Mark II), it is anticipated the installation of Hardened Vent Systems in the wetwell and in the drywell to fulfill the requirements of the 2013 NRC Order EA-13-109. Models and calculations with computer programs such as MELCOR and RELAP are being developed in order to define the temperature and pressure conditions inside the Primary Containment under Severe Accident Conditions, but the inclusion of the venting pipes(and eventually the filters) in such models imposes a severe burden from the point of view of modeling and computing times due to the large differences in sizes. CFD calculations show that the acoustic transients inside the venting pipes are so fast in comparison with the containment depressurization rate that it is justified to use a quasi-static approach in which a pressure versus mass flow rate curve generated through CFD calculations can be used with the systems programs simulators. With this approach it is expected to perform in an efficient way the necessary parametric studies related to the venting strategies. As far as a drywell venting pipe is planned to be installed, it is expected to use the same approach to assess the effects of the filters in the venting line.

1. INTRODUCTION

It is anticipated that the Laguna Verde Nuclear Power Plant (BWR Mark II) will install two Hardened Venting Pipes, one to vent from the wetwell and other to vent from the drywell in order to fulfill the requirements of the 2013 NRC Order EA-13-109. To evaluate the venting effectiveness and conditions, computer codes such as RELAP and MELCOR ARE being used, but the venting pipes have to be represented by a set of parameters and the spatial resolution inside the pipe is lost.

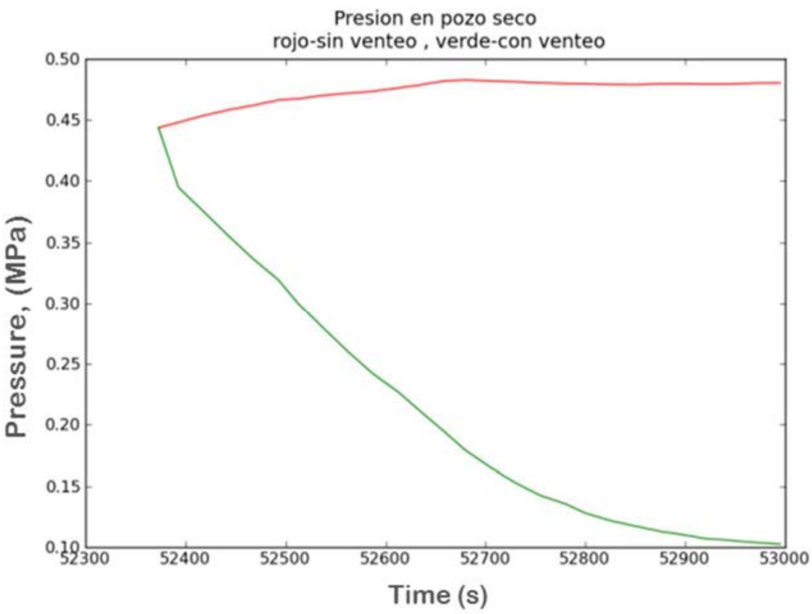
2. THERMODYNAMIC CONDITIONS INSIDE THE PRIMARY CONTAINMENT UNDER SEVERE ACCIDENT CONDITIONS

In a first approach the pressure, temperature, gas composition and mass and energy injections in the primary containment atmosphere are taken (or calculated) from the RELAP and MECOR outputs under severe accident conditions without venting. An important hypothesis is that venting does not affect the mass and energy injections coming from the primary cooling system or pressure vessel. In this way a simplified calculation procedure based on thermodynamical equilibrium in the containment volume and the friction factor formulas in piping systems can be used to estimate the venting mass fluxes, containment depressurization rates and thermodynamic conditions once the venting pressure is reached [1].

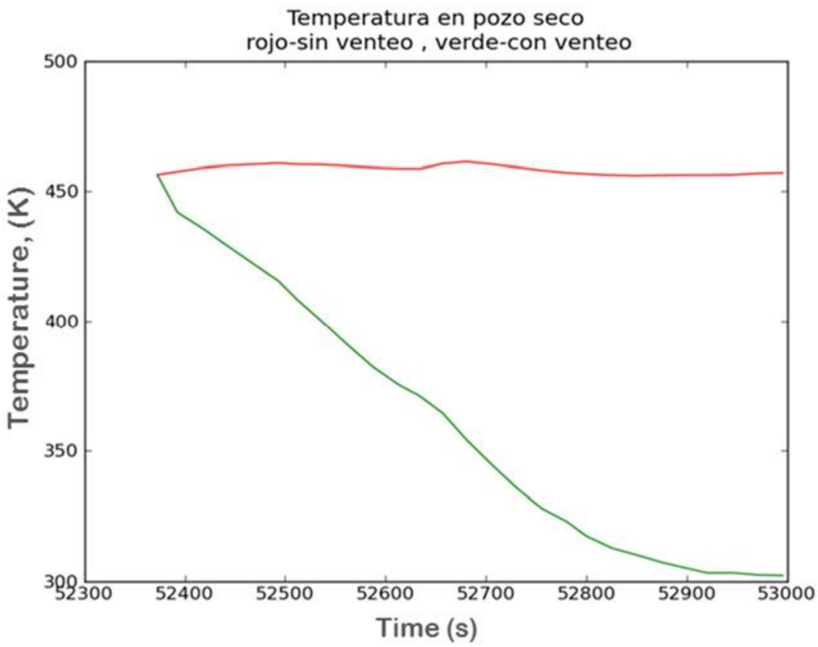
This simplified procedure provides time histories that can be used as input into the piping CFD models and Figure 1 shows the pressure and temperature curves inside the drywell without(red) and with venting(green) in a SBO scenario calculated with MELCOR.

3. ACOUSTIC WAVES PROPAGATION TIME

The propagation time is calculated with GASFLOW and OPENFOAM using a test pipe 46m long and 30cm in diameter with the boundary conditions of 4.4 x 106 dynes/cm^2 constant pressure and the containment gas composition and temperature at the input and 1.01 x 106 dynes/cm^2 and the atmospheric gas composition and temperature at the output, also the atmospheric conditions were taken as initial condition inside the pipe. The pipe was divided in 100 axial sections. The OPENFOAM mesh was generated with the SALOME-MECA program but a format translation has to be made before use in OPENFOAM.



Pressure VS time



Temperature VS time

FIG. 1. Pressure and temperature curve inside drywell without and with venting.

In GASFLOW a DUCT model was built with the same axial partition and boundary and initial conditions described above.

To generate 3D plots from the GASFLOW output a PYTHON script reads the CDF format and in the case of OPENFOAM, the tables generated in the graphic interface are read by a PYTHON script also.

The 3D plots shows the variables axial profiles from the initial state to the final steady state profile along the pipe. Figures 2 and 3 show the pressure and velocity profiles obtained with both programs.

Figure 3 shows the Hydrogen concentration profile (OPENFOAM) from which it can be seen that the pipe reaches a uniform profile with the same concentration value of the primary containment atmosphere in less than a second.

GASFLOW. [dynes/cm^2] | OPENFOAM. [Pa]

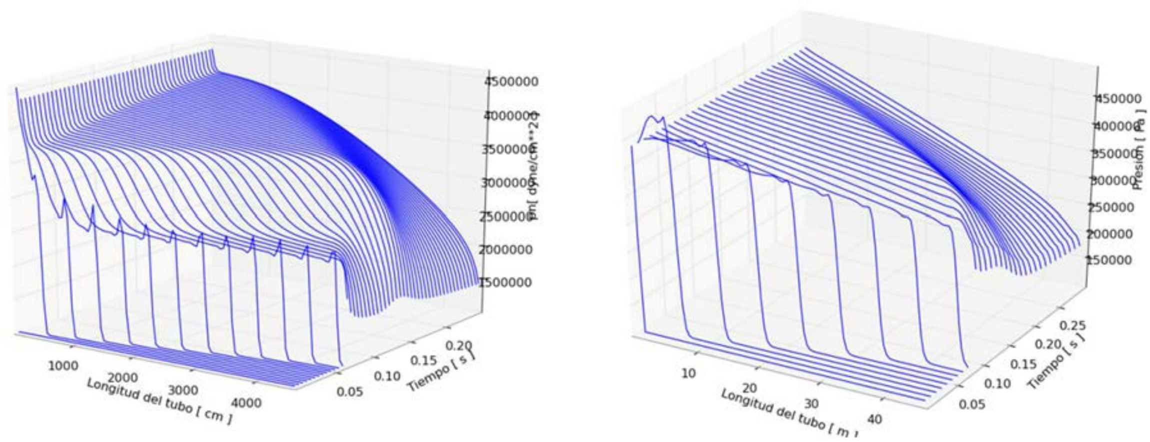


FIG. 2. Pressure profiles.

GASFLOW. [cm/s] | OPENFOAM. [m/s]

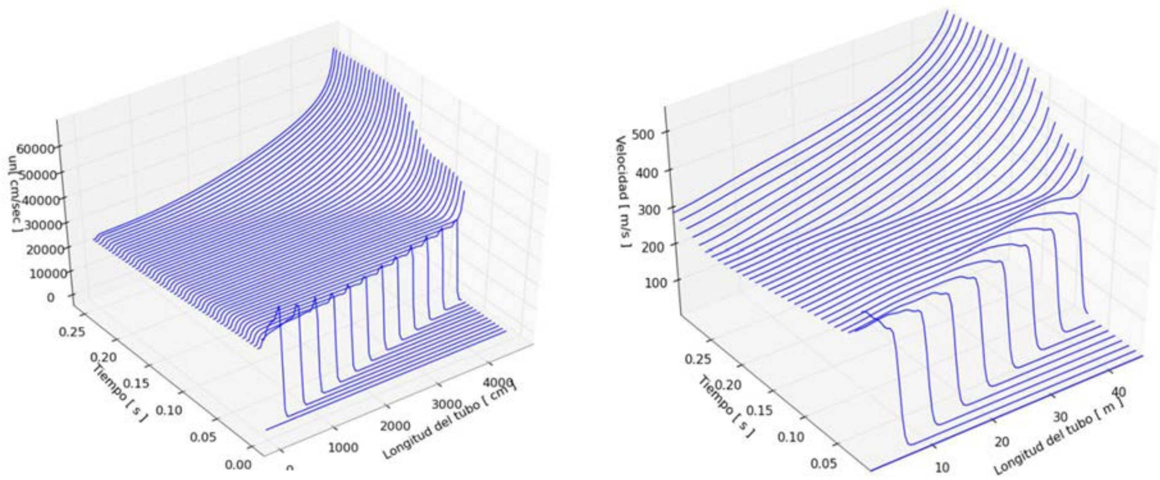


FIG. 3. Velocity profiles.

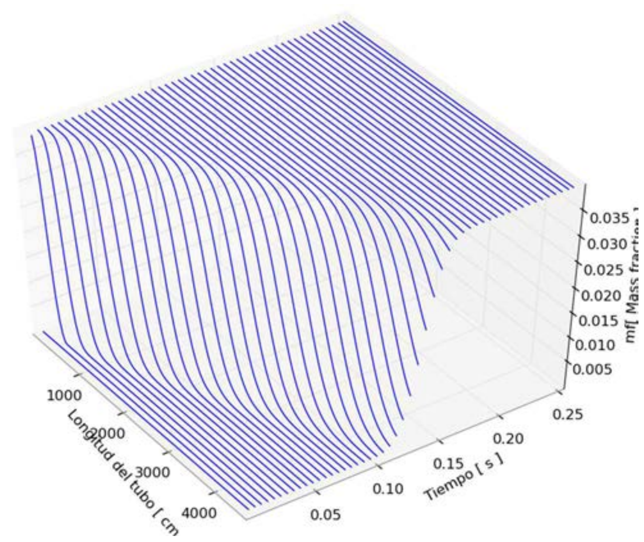


FIG. 4. Hydrogen concentration profile.

Both programs predict that any sudden change in the boundary conditions generates a transient state (pressure and mass wave) which is damped in less than a second when a stable steady state is reached. The transient conditions inside the primary containment last thousands of seconds and the variations in pressure temperature and gas concentrations are much slower than those occurring in the acoustic transients inside the venting pipe, so the quasi-static approach is justified.

4. CFD MODELS FOR THE DRYWELL VENT PIPE

The proposed drywell venting pipe consists of seven (7) straight sections and six (6) elbows and for OPENFOAM the geometry and mesh were generated with the open source code SALOME-MECA by a PYTHON script. This feature allows accommodating dimensional changes in a fast and efficient way. In the calculations that follow the same boundary and initial conditions as those of the test straight pipe are used.

4.1. OPENFOAM drywell venting pipe model

Pipe geometry was modeled with the PIPE element (GEOM module in SALOME-MECA) based in extruded sections along the piping axis, and the volume elements are triangular base prisms which are right prisms for the straight sections and oblique prisms for the elbows. The final mesh consists of approximately 65 triangles in the transversal sections and four (4) tangential sections in the elbows. Table 1 shows the axial partitions in the straight sections.

The OPENFOAM calculations predict that the steady state is reached in less than 700 ms and after that time the gas species concentrations are equal to those values inside the primary containment. Figure 5 shows the color maps for the hydrogen concentration in the transversal sections on the straight pipe section that runs vertically by the outside wall of the reactor building. It shows that the concentration is almost constant over the sections and only the axial variations are of importance. This justifies the use of the GASFLOW DUCT model in which only the axial dimension is taken into account.

TABLE 1. AXIAL OR TANGENTIAL SEGMENTS IN EACH PIPE ELEMENT (NUMBERING STARTS CLOSE TO THE CONTAINMENT WALL)

Section No.	1							2	3	4	5	6	7	8	9	10	11	12	13
No. Segments	6	4	19	4	10	4	85	4	5		4		232			4	3		

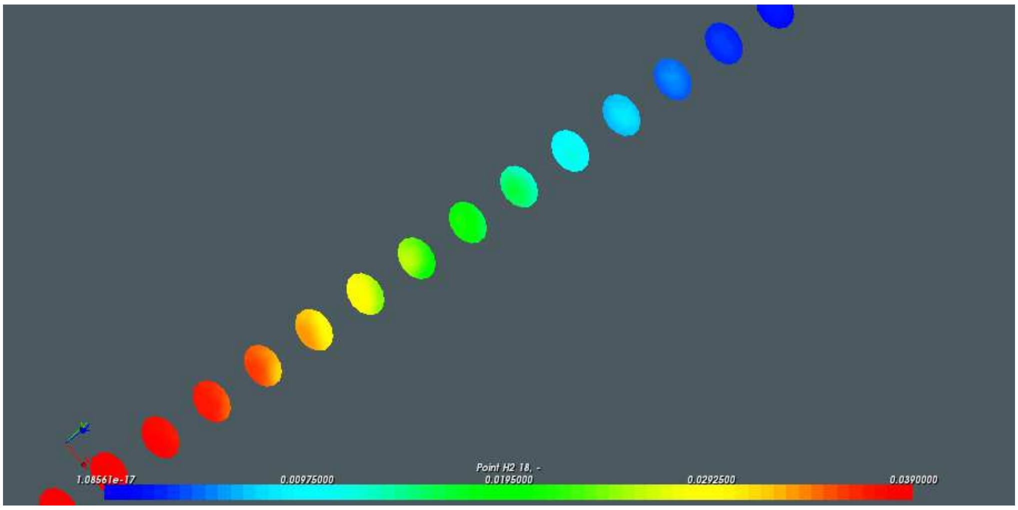


FIG. 5. Hydrogen concentration transversal color maps for pipe section 11.

4.2. GASFLOW drywell venting pipe model

In GASFLOW the geometric model consists of straight pipes of circular transversal sections joined by 90 degrees elbows. The axis of the elbows has a radius 1.05 the pipe diameter due to a GASFLOW restriction (curvature radius over pipe diameter greater than 1).

Figures 6, 7, 8 and 9 show the pressure, temperature, velocity and hydrogen concentration profiles for the same pipe section outside the reactor building. The figures also exhibit that all the variables reach the steady state condition in less than a second. On the other hand the DUCT model of GASFLOW delivers very fast execution times.

5. LONG TERM DEPRESSURIZATION ANALYSES

The results shown above suggest that the quasi-static approach in the venting system can be justified, and to take advantage of it, the CFD calculations in steady state conditions can provide the pressure loss versus mass flow relationships to be included in the systems analysis programs (RELAP and MELCOR). With this idea, for each venting pipe, a set of calculations with different input boundary pressures and with different input mass flows were made and both tabulations (pressure loss VS mass flow) were adjusted to a second degree polynomial. It was even possible to represent in just one polynomial the data for both venting pipes as shown in the plot of Figure 8.

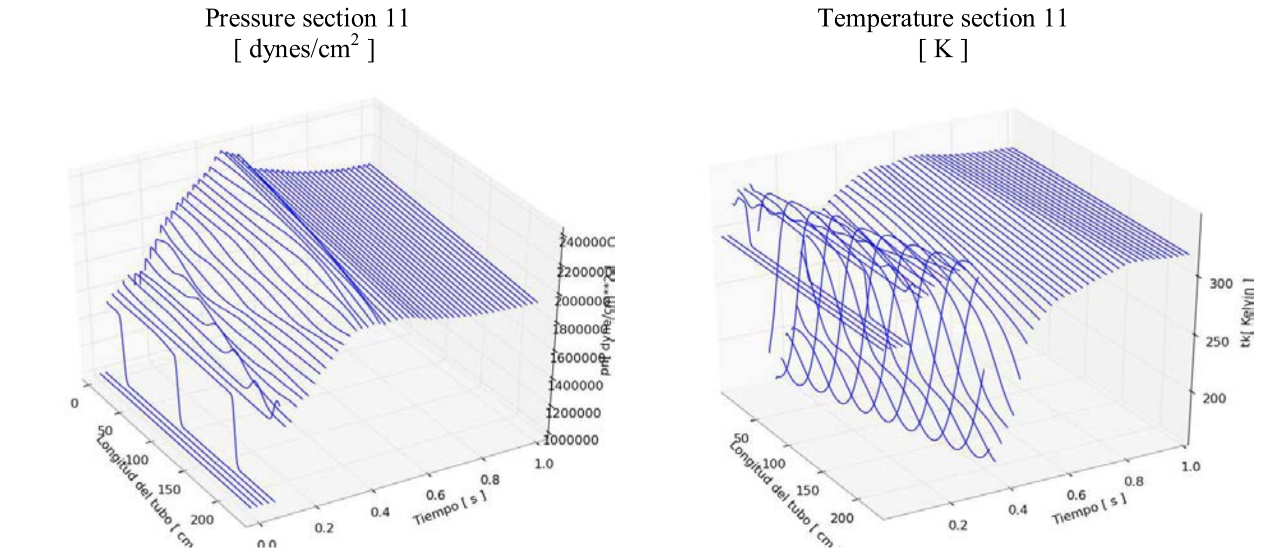


FIG. 6. Pressure and temperature profiles.

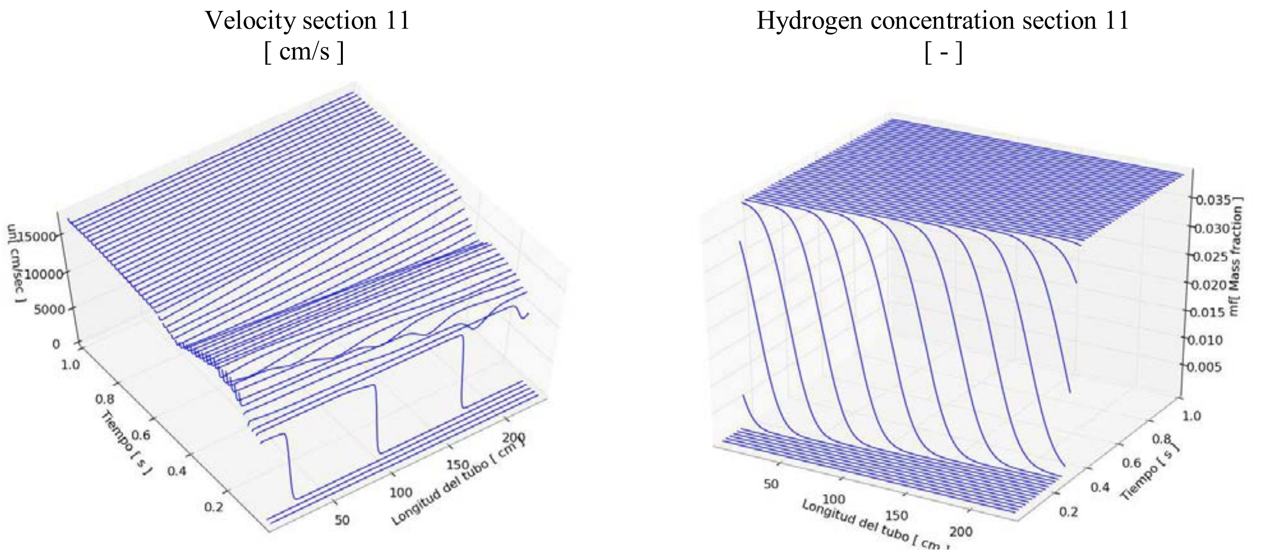


FIG. 7. Velocity and Hydrogen concentration profiles.

6. ASSESSMENT OF THE FILTER EQUIPMENT EFFECTS

To assess the effect of the filter equipment on the primary containment depressurization rates, an approach as described above can be followed. OPENFOAM models are being developed to calculate the pressure loss versus mass flow relationship on steady state conditions for each component of the filters with a focus on the multi-venturi and sand bed concepts [2]. The objective of the CFD models is twofold, in one hand to generate the parameters to be used in the systems simulation programs (RELAP/MELCOR) an on the other, to estimate the spatial distribution of the variables inside the filters to calculate the thermal, mass flow and gases concentrations loads on the filtering media. In this approach the mechanisms related to aerosols transport will be solved based on the steady flow patterns(uncoupled eulerian – lagrangian approach).

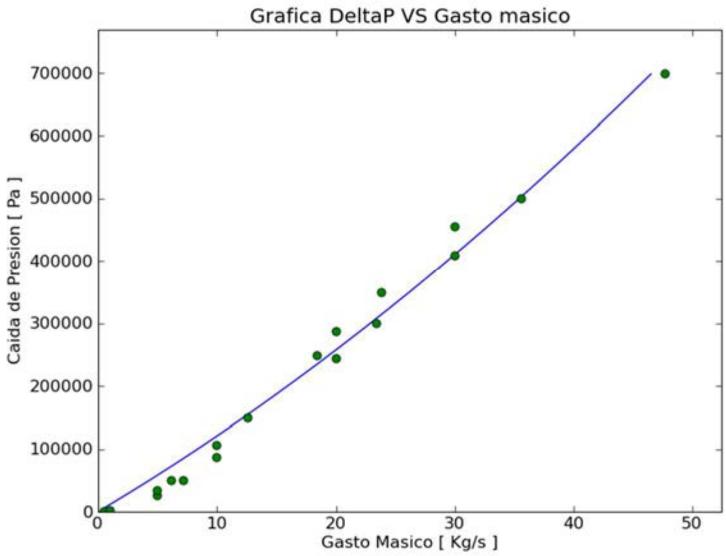


FIG. 8. Delta Pressure VS mass flow. The points belong to both venting pipes.

Table 2 reports the OPENFOAM results for a 7 sectors multiventuri system based on the steady state conditions in each single venturi pipe [3] using one of the compressible gas solvers("reactingFoam").

TABLE 2. MULTI VENTURI ACTIVATED SECTORS FLOW CAPACITY

Activated sector number	Mass flow (kg/s)	Used Capacity
1	0.1091	2.43%
2	0.8480	9.75%
3	2.3349	21.95%
4	3.2905	34.14%
5	5.9865	51.21%
6	9.5255	73.17%
7	13.933	100%

In the case of a Sand Bed filter, Figure 9 [4] shows the pressure loss versus mass flow rate using the OPENFOAM porous steady state solver and the typical dimensions reported in [2].

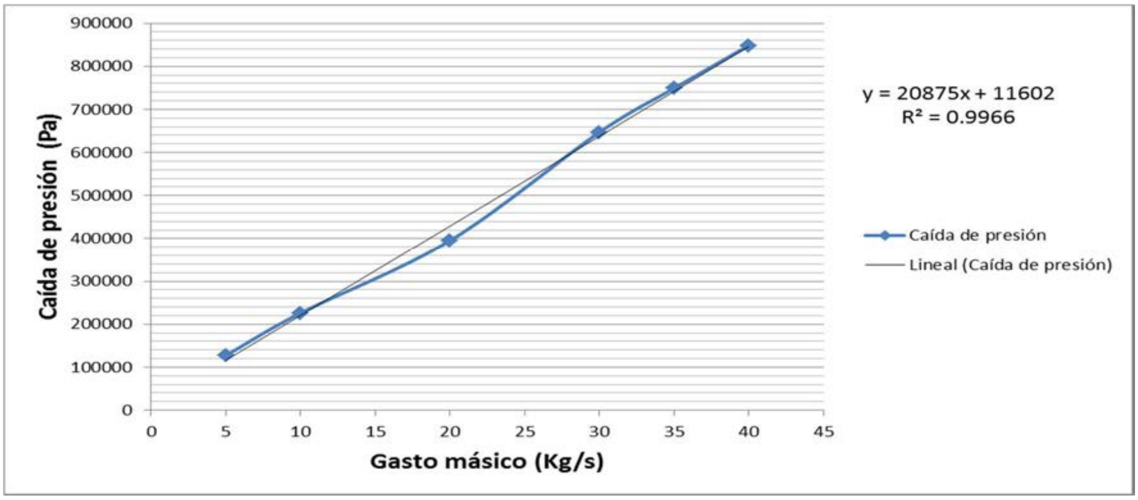


FIG. 9. Delta Pressure along filter deep VS mass flow.

7. CONCLUSIONS

The main conclusion is to justify the uncoupling method that can be applied in the analyses of the primary containment depressurization transients based on the use of independent models for the containment and for the venting pipes and it is expected to use the same approach with the filters. This simplification is necessary to perform in an efficient way the parametric studies over the severe accident scenarios and venting strategies spectrum postulated for a BWR Mark II.

ACKNOWLEDGEMENTS

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[1] GÓMEZ TORRES, A.M., SÁINZ MEJÍA E., Cálculo y Análisis de Concentraciones Volumétricas de Hidrógeno en el Tubo de Venteo Rígido de la CNLV en Condiciones de Accidente Severo, IT.CA/CNSNS-024-13 Vols. 1Y 2, Agosto 2013.

[2] OECD/NEA/CSNI Report, Status Report on Filtered Containment Venting, OECD/NEA/CSNI/R(2014) 7 July 2014 .

[3] REYES A.A., SAINZ E., Cálculo Hidrodinámico de Filtros Tipo Lavado en Líquidos Usados en los Sistemas de Venteo de la Contención, XXVI Congreso Anual de la Sociedad Nuclear Mexicana, Julio 2015.

[4] CUEVAS, D., SAINZ, E., Cálculo Hidrodinámico de Filtros Tipo Lecho de Arena Usados en los Sistemas de Venteo de la Contención, XXVI Congreso Anual de la Sociedad Nuclear Mexicana, Julio 2015.

Annex I

OVERVIEW OF RELATED IAEA PUBLICATIONS

The IAEA Safety Standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. The IAEA Safety Standards Series have three categories:

(1) Safety Fundamentals present the fundamental safety objective and ten principles of protection and safety, and provide the basis for the safety requirements;
(2) Safety Requirements establish the requirements that must be met to ensure the protection of people and the environment, both now and in the future;
(3) Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures).

Among the IAEA scientific and technical publications of interest to this publication are the following:

I-1. Safety of Nuclear Power Plants: Design (IAEA Safety Standards Series No. SSR-2/1, Rev. 1)

IAEA Safety Standards Series No. SSR-2/1 (Rev. 1) [3] establishes the design requirements for the structures, systems and components of a nuclear power plant, as well as for the procedures and organizational processes important to safety that are required for ensuring safe operation and preventing events that could compromise safety, or for mitigating the consequences of such events, if they occur.

SSR-2/1 (Rev. 1) introduces the design requirements in five sections following the introduction. Section 2 describes the safety objective, safety principles and concepts that form the basis for deriving the safety function requirements that must be met for the nuclear power plant, as well as the safety design criteria. Section 3 establishes the general requirements to be satisfied by the design organization in the management of safety in the design process. Section 4 establishes requirements for the principal technical design criteria for safety, including requirements for the fundamental safety functions and for the application of defence in depth. It also includes provision for construction and requirements for interfaces of safety with nuclear security and with the State system of accounting for and control of nuclear material, and for ensuring that radiation risks arising from the plant are maintained as low as reasonably achievable. Section 5 establishes requirements for general plant design that supplement the requirements for the principal technical design criteria to ensure that safety objectives are met and the safety principles are applied. The requirements for general plant design apply to all items (i.e. structures, systems and components) important to safety. Section 6 establishes requirements for the design of specific plant systems such as the reactor core, reactor coolant systems, the containment system, and instrumentation and control systems.

Containment venting, hardened venting or filtered venting is not explicitly defined in SSR-2/1 (Rev. 1). However, the need and specifications for hardened or filtered containment venting for the operating reactors may be inferred from Requirements 53–55 of SSR-2/1 (Rev. 1):

- Requirement 53. "The capability to transfer heat to an ultimate heat sink shall be ensured for all plant states";
- Requirement 54. "A containment system shall be provided to ensure, or to contribute to, the fulfilment of the following safety functions at the nuclear power plant: (i) confinement of radioactive substances in operational states and in accident conditions, (ii) protection of the reactor against natural external events and human induced events and (iii) radiation shielding in operational states and in accident conditions";
- Requirement 55. "The design of the containment shall be such as to ensure that any radioactive release from the nuclear power plant to the environment is as low as reasonably achievable, is below the authorized limits on discharges in operational states and is below acceptable limits in accident conditions".

I-2 Design of Reactor Containment Systems for Nuclear Power Plants (IAEA Safety Standards Series No. NS-G-1.10)

IAEA Safety Standards Series No. NS-G-1.10 [4] applies to the most common types of containment and includes some general recommendations for the design of containment systems for new nuclear power plants with the aim of dealing with severe accidents.

The recommendations in IAEA Safety Standards Series No. NS-G-1.10 focus on the design of containment systems and seek to ensure or contribute to the achievement of the three main safety functions:

(1) Confinement of radioactive substances in operational states and in accident conditions;
(2) Protection of the plant against external natural and human induced events;
(3) Radiation shielding in operational states and in accident conditions.

NS-G-1.10 [4] also underlines the importance of maintaining the structural integrity of the containment envelope and ensuring that the specified maximum leak rate not be exceeded under any condition pertaining to the design basis and severe accidents considered in the design. Features for the management of radionuclides should be such as to ensure that the release of radionuclides from the containment envelope is kept below authorized limits. In order to reach this goal, specific systems may be necessary, such as a ventilation system. Furthermore, the containment systems should permit the reduction of temperature and pressure within the containment when necessary. The containment design should not depend on venting as a means of maintaining structural integrity in any design basis accident condition. However, if venting is necessary during a severe accident the releases should be filtered to control the release of radionuclides to the environment. The Safety Guide addresses typical filter systems including sand, multi-venturi scrubber systems, high efficiency particulate air (HEPA) or charcoal filters, or a combination of these systems.

I-3 Severe Accident Management Programmes for Nuclear Power Plants (IAEA Safety Standards No. NS-G-2.15)

IAEA Safety Standards Series No. NS-G-2.15 [5] focuses on severe accident management programmes. It presents elements of accident management programmes, and gives examples of how to prepare, develop, implement and review accident management programmes. It has two main sections following the introduction. Section 2 presents the overall concept of an accident management programme as well as the high level considerations. Section 3 provides a detailed description of the process of development and implementation of an accident management programme, including verification and validation, roles and responsibilities of involved teams/organizations, education and training and the role of instrumentation and control.

NS-G-2.15 [5] foresees a structured top–down approach in the development of the accident management guidance. The approach should begin with the objectives and strategies, as well as procedures and guidelines, and should also cover both the preventive and the mitigatory domains. The objectives of accident management at the top level are defined to be: (i) Preventing significant core damage; (ii) terminating the progress of core damage once it has started; (iii) maintaining the integrity of the containment as long as possible; (iv) minimizing releases of radioactive material; and (v) achieving a long term stable state.

A severe accident as a result of multiple failures of safety systems leading to significant core degradation may result in jeopardizing the integrity of many or all of the barriers to the release of radioactive material. NS-G-2.15 calls for attention to potential design changes that could mitigate the consequences of a severe accident; such design changes would need to be evaluated and implemented if reasonably practicable. One example of a possible design change that can be implemented in existing plants is a hardened and/or filtered containment vent. NS-G-2.15 notes that a filtered containment vent can be used to prevent containment overpressurization, but also to release hydrogen (or oxygen) to reduce the hydrogen risk, to prevent unfiltered leakage from existing openings or from a containment that has a pre-existing (relatively) large leakage rate, or to prevent basemat failure — if anticipated to occur — at an elevated containment pressure. NS-G-2.15 highlights the conditions for initiation of containment venting with the aim of the protecting the structural integrity of the fission product barrier — accepted as the most important barrier in the defence in depth concept —at a time and at a containment pressure level that give confidence that the structural integrity of the containment will not be lost.

I-4 Off-Gas and Air Cleaning Systems for Accident Conditions in Nuclear Power Plants (IAEA Technical Reports Series No. 358)

This publication [8] summarizes many aspects of off- gas and air cleaning systems in terms of their potential use during a severe accident. It presents a spectrum of design basis and severe accidents and discusses source terms during minor accidents as well as design basis and severe accidents, reflecting the knowledge base available in 1993. The retention strategies in containment and confinement, containment venting systems for severe accidents and control room ventilation are discussed in detail. The design of ventilation systems is based on their functional requirements and requirements for process equipment as well as their testing. The design of engineered safety features for gas cleaning is discussed in terms of the control of airborne contamination in design basis accidents.

With regard to severe accidents, the report reviews processes for fission product release and behaviour in the containment. Addressing the controlled release of activity from the containment into the environment, the publication presents the filtered containment venting system (FCVS) technologies available in 1993 and their implementation status in different countries.

One important conclusion of this Safety Guide is that the installation of FVCSs in nuclear power plants is strongly recommended. The report cites the generally accepted values for filtration efficiencies (decontamination factor, DF) for containment venting filters (in the range of 100–1000) and for FCVSs. It provides the following three general criteria to be fulfilled in FCVS designs:

- The decision whether the venting should be initiated automatically or manually depends on the time available, the required reliability of the system and the access to the isolation valves.
- The components should be mechanically robust and resistant to pressure and heat.
- The system should provide a heat sink and should have the capacities for filtration of particulates (including aerosols) and vapour/gas (especially various forms of iodine).

I-5 Passive Safety Systems and Natural Circulation in Water Cooled Nuclear Power Plants (IAEA-TECDOC-1624)

As early as the mid-1980s, it was recognized that the application of passive safety systems (i.e. those whose operation takes advantage of natural forces such as convection and gravity), can contribute to simplification and potentially improve the economics of new nuclear power plant designs. Advanced designs currently available effectively utilize passive safety systems and natural circulation as a means to remove core power or decay heat during normal operation and accidents. IAEA-TECDOC-1624 [9] describes passive safety systems in a wide range of advanced water cooled nuclear power plant designs with the goal of gaining insights into system design, operation and reliability. The publication describes several passive systems. However, the report does not make the explicit recommendation that passive systems should be preferred to active systems. It reviews the types of advanced reactor passive safety systems for removing decay heat, and containment cooling and pressure suppression. It also features a brief summary of related thermal-hydraulic phenomena and establishes a link between the systems described and phenomena.

I-6 Mitigation of Hydrogen Hazards in Severe Accidents in Nuclear Power Plants (IAEA-TECDOC-1661)

The combustion of hydrogen, produced primarily as a result of heated zirconium metal reacting with steam, can create short term pressure or detonation forces that may exceed the strength of the containment structure and lead to early containment failure. For most NPPs, severe accidents lead to hydrogen release rates that exceed the capacity of hydrogen control measures for a conventional design basis accident (DBA). Local hydrogen concentrations can reach high levels in a short time, leading to combustible gas mixtures in the containment. Moreover, long term pressure buildup may occur as a result of steam generation through decay heat and/or through the generation of non-condensable gases, including carbon monoxide, carbon dioxide and hydrogen arising from the interaction of the molten core with the containment basemat concrete.

The main objective of IAEA-TECDOC-1661[10] is to contribute to the implementation of relevant IAEA Safety Standards, in particular two requirements:

(1) Performing computational analysis of severe accidents, and notably all problems related to hydrogen sources, hydrogen distribution, hydrogen combustion, hydrogen control and mitigation measures;

(2) Development and implementation of accident management programmes in NPPs, notably those measures aimed at mitigation of hydrogen in the reactor containments.

IAEA-TECDOC-1661 describes potential hydrogen sources during a severe accident at various stages covering: in-core degradation to core-concrete interaction; the distribution and combustion of hydrogen; calculation tools; and related experimental facilities. It presents in detail the effects of release mode, spraying on hydrogen distribution, containment layout effects and combustion modes such as deflagration, detonation and flame acceleration. Furthermore, it discusses the possible risks from hydrogen combustion and the evaluation of risk due to static and dynamic loads on the containment. A review of hydrogen measurement systems, systems qualification and probe position is provided. Finally, various techniques for the control of hydrogen and mitigation measures such as inertization, post-accident dilution, early venting and hydrogen removal are introduced. The report also provides a summary of hydrogen behaviour.

Annex II

AGENDA FOR THE TECHNICAL MEETING

The meeting agenda listing the titles of presentations and speakers is shown below.

Monday 31 August 2015		
09:30 – 10:30	Opening Remarks Introductions	Mr Jon Phillips, Acting Director, Division of Nuclear Power, Department of Nuclear Energy. Mr Gustavo Caruso, Special Coordinator of the Nuclear Safety Action Team (NSAT), Department of Nuclear Safety and Security. Mr Chad Painter, Scientific Secretary, Nuclear Power Technology Development Section.
Session I – Background and Regulations **Chair: Mr. Wiktor Frid**		
10:30 – 11:15	Summary of OECD/NEA Status Report on Filtered Containment Venting	Mr Thambiayah Nitheanandan, Chalk River Laboratories Canadian Nuclear Laboratories Canada
11:30–12:00	Filtered Containment Venting in Sweden; Historical Background, Regulatory Requirements and Design	Mr Wiktor Frid, Consultant, Formerly with Swedish Radiation Safety Authority Sweden
12:00 – 12:30	Bangladesh Perspective on the Requirement for Filtered Vent Systems on RNPP	Mr Md. Abdus Salam, Center for Research Reactor (CRR) Atomic Energy Research Establishment (AERE) Bangladesh
14:00 – 14:30	Filtered containment venting system for existing nuclear power plants: design parameters and associated safety criteria in Belgium	Ms Martina Adorni, Bel V, a subsidiary of the FANC (Federal Agency for Nuclear Control) Belgium
14:30 – 15:00	Regulatory Oversight of Filtered Venting and Containment Integrity at Canadian NPPs	Mr Samuel Gyepi-Garbrah, Reactor Behaviour Division Canadian Nuclear Safety Commission (CNSC) Canada
15:30 – 16:00	Regulations and Improvements in Hungary related to Severe Accidents and Filtered Venting	Mr Tamas Czerovszki, Hungarian Atomic Energy Authority Hungary
16:00 – 16:30	Regulatory Challenges of Filtered Containment Venting in Indonesia	Ms Helen RAFLIS, Nuclear Energy Regulatory Agency BAPETEN Indonesia

Tuesday 1 September 2015		
09.30 – 10:00	Regulatory Requirements and Actions Related to Improve Containment Venting for Laguna Verde NPP	Mr Yuri Raul Mamani Alegria, Comisión Nacional de Seguridad Nuclear y Salvaguardias Mexico
10.00 – 10:30	Status of Spanish Regulations and Industry actions related to Filtered containment Vent Systems	Ms Sara Gonzales Veci, Consejo de Seguridad Nuclear (CSN) Spain
11:00 – 11:30	Results of Regulatory Review of Activities for Implementation of Containment Filtered Venting for Ukrainian NPPs	Mr Dmytro Gumenyuk, State Scientific and Technical Center for Nuclear and Radiation Safety Ukraine
11:30 – 12:00	Provision of Containment Integrity at Russian WWER NPPs under BDBA Conditions	Mr Mikhail Maltsev, Atomenergoproyekt State Atomic Energy Russian Federation
Session II – Long Term Containment Response to Severe Accidents in Light of Fukushima Chair: Mr. Gustavo Rubio		
14:00 – 14:30	Behaviour of Containment System under Long Term Station Black Out Conditions	Mr Vishnu Verma, Bhabha Atomic Research Centre (BARC); Reactor Safety Division India
14:30 – 15:00	Comparative Analysis on Containment Over-Pressurization Scenarios using MAAP Code	Mr Sangwon Lee, Korea Hydro and Nuclear Power Central Research Institutes Republic of Korea
15:30 – 16:00	An Investigation of Pool Stratification and Vent Heat Transfer on BWR Wetwell External Venting using GOTHIC	Mr Thomas George, Zachry Nuclear Engineering United States of America

Wednesday 2 September 2015		
Session III – Strategies to Ensure Containment Integrity for Existing Plants during Severe Accidents Chair: Mr. Toshihiro Matsuo		
09:30 – 10:00	Experience with Containment Venting at Malaysian Research Reactor	Mr Mohammad Suhaimi bin Kassim, Malaysian Nuclear Agency Ministry of Science Malaysia
10:00 – 10:30	Strategies to Ensure Containment Integrity for ABWRs in Kashiwazaki-Kariwa Nuclear Power Station	Mr Toshihiro Matsuo, Deputy Manager Nuclear Reactor Safety Engineering Group Tokyo Electric Power Company Japan
Session IV – Strategies in Containment Cooling and Energy Management for Advanced Reactor Designs Chair: Jeff Taylor		
11:00 – 11:30	Strategies to Ensure Containment Integrity for EC6	Ms Liliana Comanescu, CANDU Energy Inc Canada
11:30 – 12:00	The Response of AP1000 under the Condition of Fukushima Accident	Mr Jiayun Wang, Shanghai Nuclear Engineering Research and Design Institute (SNERDI) China
12:00–12:30	ATMEA1 Strategy for Containment Cooling	Mr Eric Mathet, ATMEA France
14:00 – 14:30	Long Term Containment Protection Strategies for the AP1000® Plant Design	Mr Jeff Taylor, Westinghouse Electric Company United States of America
Session V – Filtered Venting Technology Chair: Mr. Salih Guentay		
14:30 – 15:00	KAERI Activities on the Filter Containment Venting System Development	Mr Kwang Soon Ha, Severe Accident & PHWR Safety Division, Korea Atomic Energy Research Institute (KAERI) Republic of Korea
15:00 – 15:30	The European PASSAM project: Experimental Studies for Atmospheric Source Term Mitigation with focus on Filtered Containment Venting Systems	Mr Thierry Albiol, Institut de Radioprotection et de Surete Nucleaire (IRSN) France
16:00 – 16:30	Advances on understanding of pool scrubbing for FCVS based on the PASSAM project	Ms Terttaliisa Lind, Paul Scherrer Institut Switzerland

Thursday 3 September 2015		
09:30 – 10:00	Design of Containment Filtered Venting System (CFVS) for TAPS-1&2 to limit the containment pressure below design pressure during design extension conditions	Mr Manoj Kansal, Nuclear Power Corporation of India Limited India
10:00 – 10:30	Filtered System for Purging Vapor-gas discharges from Containment of WCR at Severe Accidents	Mr Yuri Dulepov, Joint Stock Company Sverdlovsk Scientific Research Institute Russian Federation
10:30 – 11:00	Characterization of the Performance of Wet Scrubbers used in FCVS	Ms Terttaliisa Lind, Paul Scherrer Institut Switzerland
11:15 – 11:45	Lessons learned from analytical Re-evaluation of a Venturi Scrubber Venting System implemented in German NPPs through COCOSYS Analyses	Mr Siegfried Schwarz, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH Germany
Additionally	CFD Calculations in a MARK II Venting Systems	Sainz Eduardo, Instituto Nacional de Investigaciones Nucleares Mexico
11:45 – 12:15	Closing Remarks	Mr Monti Stefano, Section Head, Nuclear Power Technology Development Section (IAEA) Mr Gustavo Caruso, Special Coordinator of Nuclear Safety Action Team (NSAT), Department of Nuclear Safety and Security Mr Chad Painter, Scientific Secretary, Nuclear Power Technology Development Section (IAEA)

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