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Analysis of Primary Containment Capture System for the propose Advanced Modern 600 nuclear power plant

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ABSTRACT

Analysis of global electric markets shows that the electrical grid size of many developing countries is too small or too distributed to accommodate nuclear power plants (NPP) with large unit sizes. Thus a modern NPP design with a smaller output (~600 MWe) as well as with proper confidence of public safety is of interest. But as the Fukushima Daiichi NPP accident demonstrates, natural disasters can severely challenge the existing safety philosophy of defense in depth at NPPs. To improve and reinforce defense in depth, a new barrier to fission product release, a long-term and passive system against environmental release - a Primary Containment Capture System (PCCS) for Advanced Modern 600 MW (AM600) is proposed. This system is designed to confine ex-containment release and to prevent containment failure from overpressure. In this study the thermal-hydraulics behavior of the PCCS of different length is studied. Analysis results show that a 1-km length with 8 m diameter gravel filled tunnel (PCCS) can maintain safe conditions for containment for 10 days following extended Station Blackout (SBO), and consequently the higher length for longer period without any release to the environment. The concept of the tunnel that is connected with containment will be suitable for multiunit site.

1. Introduction

Analysis of global electric market shows that smaller NPP in the range of 600 MWe is more suitable for developing countries. However, after the Fukushima Daiichi accident many are averted from NPP technology. Therefore, a robust and secure NPP system with smaller output (~600 MW) is of interest.

Though recent NPP designs include more passive safety features, still these safety features have limitations under extreme beyond design basis accident scenarios. The Fukushima Daiichi accident is one such example. In the prototypical NPP design, the containment building is the last defense barrier for radiation release. However, in the case of severe accident, the production of huge amounts steam and other gases can threaten the containment building with overpressure and failure. Ultimately huge radiation fallout can result and contaminate a large area. Therefore, a stronger barrier with the concept of zero radiation release up to a certain period is proposed. Such a design can enhance defense in depth of such NPPs. The idea of the PCCS is to open a controlled flow path to an external system or volume to relieve the steam pressure which is generated inside the containment building. Though Europe and Canada have already installed various types of Containment Filtered Venting System (CFVS), the PCCS for AM600 is a new, simple and robust design to enhance the safety of the plant for a coping period of more than 10 days without any active power.

The primary goal in the nuclear industry is to protect the public from the potential consequences of severe nuclear accident. Historically, the Rasmuseen Report [WASH 1400] indicated that a severe nuclear accident which includes core melting can cause a large release of radiation by containment failure due to overpressure. After the Three Mile Island Unit 2 accident in 1979 and the Chernobyl accident in 1986, the nuclear industry was spurred to design and installed CFVS type systems in NPP (e.g. Korea). The Fukushima Daiichi accident in March 2011 further accelerated this idea. Most European countries installed CFVSs for their NPP's in the 1980's and 1990's. After 2011 many other countries planned to install such system. All of these CFVSs use some type of filter before release of the gases to the environment. The concern in design is the proper filtration capability, even if the general public will not accept any release to the environment.

Containment is the last barrier to protect the environment from radiation hazards. However, containment can be threatened by overpressure after a postulated severe nuclear accident. The integrity of the containment can be maintained by controlled release of steam and radiation per the current severe nuclear accident management guidelines (Andreeva et al., 2008; Yuan et al., 2014, 2013).

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Most CFVS designs employ a wet scrubbing processes to improve the filtration. The main challenge of the wet filtering system is as the water approaches the saturation condition, the filtering efficiency rapidly decreases. In addition, for long term filtration, maintaining the sufficient supplies of filtering water can be another issue. Power sources may not be available for an extended period of time. During a severe nuclear accident only passive systems can be credited for long periods. Note that some of the European CFVS provide only 24 h functioning without operator action (Rust et al., 1995). There are other drawbacks of CFVS including containment sub-atmospheric pressure, hydrogen buildup due to condensation in the subcooled water pool of the CFVS, and unnecessary release of radioactive material due to inadvertent opening. These issues studied in the US (Schlueter and Schmitz, 1990; Jack Dallman et al., 1990; Kelly, 1991)

The decontamination rate is not 100% for any CFVS. This necessarily infers the release of radioactivity to the environment (Schlueter and Schmitz, 1990). Removal of organic iodine by the water scrubber has also been a technological challenge (Iodine Chemistry and Mitigation Mechanich (ICHEMM), 2003). In the long term operation of these filters may face aerosol clogging under adverse conditions (Thomasa and et al., 2001; Dillmann and Wilhelm, 1990; Frising et al., 2005). After a large amount of active aerosol filtration, the accumulated Fission Product (FP) decay heat might cause the temperature of aerosol to be high enough to re-vaporize or it may undergo partial melting (Auvinen, 2000).

A PCCS is proposed to prevent containment failure on overpressure and to confine ex-containment release for the proposed AM600 NPP. There is no release path for radiation to the environment and therefore issues for public concern can be greatly reduced. This study analyzed the thermal-hydraulics behavior of the proposed PCCS following a hypothetical severe nuclear accident for the proposed AM600 NPP. This study is done by considering the containment of typical 1000 MWe NPP.

The proposed PCCS is connected to the containment through valves (Fig. 1) which will open at a preset value i.e. 0.5 Mpa, 0.625 Mpa or 0.8 Mpa depending upon the design. The volume of the tunnel (PCCS) is sufficient to accommodate steam and gasses from the containment for several days without intervention after severe accident.

2. Description of AM600 PCCS

A simplified conceptual diagram of the PCCS system is shown in Fig. 1. The PCCS tunnel is connected to the containment by a pipe

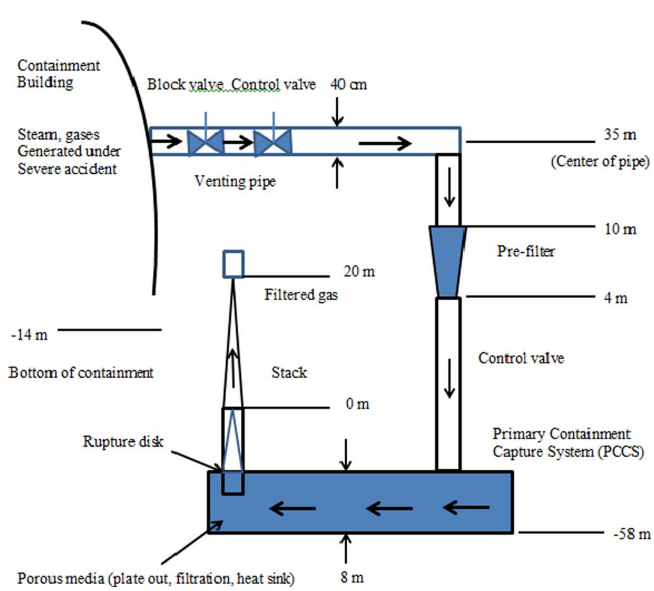

Fig. 1. Conceptual diagram of AM600 PCCS.

through a set of block valves and control valves which can be opened either manually or remotely. This connecting line is large enough to depressurize the AM600 containment system. Woon et al. (2014) reported that a vent flow of 17 kg/s of steam flow is sufficient to depressurize the containment in most of the late containment failure scenarios for the larger Optimized Power Reactor-1000 MW (OPR-1000) containment. A 60 cm line is more than adequate to pass the required steam flow to reverse containment pressurization. A stack is located at the end of the tunnel. To prevent failure of the PCCS due to overpressure, a rupture disk between the tunnel and vent stack is provided.

The length of the PCCS tunnel is initially assumed as 1-km or 1.5-km long. The diameter is taken as 8-m. The tunnel is designed as an ASME pressure vessel constructed of carbon steel and is located within an outer concrete tunnel. The primary tunnel will be filled with suitable gravel.

2.1. Gravel characteristics

Due to the availability of data and simplicity of calculation, concrete gravel is considered for sizing the AM600 PCCS. The gravel data shown in the Table 1 are used to model the 1 km and 1.5 km tunnels. The diameter of the gravel is assumed as 40 mm.

3. Methodology

The AM600 PCCS is analyzed using the severe accident analysis code MELCOR. MELCOR is a fully integrated, engineering-level computer code whose main purpose is to model the advancement of accidents in light water reactor nuclear power plants (Sandia National Laboratories, 2005). MELCOR is executed in two parts. The first part is called MELGEN, in which the majority of input is specified, processed, and checked. When the input checks are satisfied, a Restart File is written for the initial conditions of the calculation. The second part of MELCOR is the MELCOR program itself, which advances the problem through time based on the input to MELGEN and any MELCOR input. MELCOR input processing is illustrated in Fig. 2 (Sandia National Laboratories, 2005). MELCOR is divided into many packages. Only the Control Volume Hydrodynamics (CVH), Flow Path (FL), Radionuclide (RN), Control Function (CF), Tabular Function (TF), and Heat Structure (HS) packages are used here. The CVH and FL packages are responsible for modeling the thermal-hydraulic behavior of liquid water, water vapor, and gases in MELCOR. Connections between control volumes, through which the control volume contents may flow, are defined by input to the Flow Path package. Nodalization of AM600 PCCS is shown in Fig. 3.

Severe nuclear accident may progress by the rapid pressurization due to the production of significant quantities of steam air, carbon-monoxide, hydrogen, carbon dioxide, etc. The steam production rate from decay heat after hypothetical severe accident for proposed AM600 NPP is scaled using Advanced Power Reactor 1400 MW (APR-1400)

Table 1
PCCS tunnel properties.

Length	Parameters	Unit	Value
1000 m	No of gravels spheres in each volume	(−)	1.96×10^8
1500 m			2.94×10^8
Free area for venting flow		m^2	10.85
1000 m	Gravel volume for each section	m^3	6568.65
	Free volume in each section		5997.72
	Total volume for each section		12566.37
1500 m	Gravel volume for each section		9852.98
	Free volume in each section		8996.58
	Total volume for each section		18849.56
Porosity of the tunnel		(−)	0.477
Hydraulic diameter of flow area		m	0.006