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Assessment of passive safety system performance under gravity driven cooling system drain line break accident

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ABSTRACT

A generation III+ Boiling Water Reactor (BWR) which relies on natural circulation has evolved from earlier BWR designs by incorporating passive safety features to improve safety and performance. Natural circulation allows the elimination of emergency injection pump and no operator action or alternating current (AC) power supply. The generation III+ BWR's passive safety systems include the Automatic Depressurization System (ADS), the Suppression Pool (SP), the Standby Liquid Control System (SLCS), the Gravity Driven Cooling System (GDCS), the Isolation Condenser System (ICS) and the Passive Containment Cooling System (PCCS). The ADS is actuated to rapidly depressurize the reactor leading to the GDCS injection. The large amount of water in the SP condenses steam from the reactor. The SLCS provides makeup water to the reactor. The GDCS injects water into the reactor by gravity head and provides cooling to the core. The ICS and the PCCS are used to remove the decay heat from the reactor. The objective of this paper is to analyze the response of passive safety systems under the Loss of Coolant Accident (LOCA). A GDCS Drain Line Break (GDLB) test has been conducted in the Purdue University Multi-Dimensional Integral Test Assembly (PUMA) which is scaled to represent the generation III+ BWR. The main results of PUMA GDLB test were that the reactor coolant level was well above the Top of Active Fuel (TAF) and the reactor containment pressure has remained below the design pressure. In particular, the containment maximum pressure (266 kPa) was 36% lower than the safety limit (414 kPa). The minimum collapsed water level (1.496 m) before the GDCS injection was 8% lower than the TAF (1.623 m) but it was ensured that two-phase water level was higher than the TAF with no core uncovery.

1. Introduction

In these days nuclear energy is back on the policy agendas of many countries, with projections for new buildings similar to or exceeding those of the early years of nuclear power. However, as seen in the accidents at Three Mile Island, Chernobyl, and Fukushima, national and international anxiety about nuclear power stems directly from a fear of the release of radioactive material and its consequences on people and the environment. Hence, it is highly demanded that the new design combines improvements in safety with design simplification and component standardization to produce a safer, more productive, and more reliable nuclear power plant. Passively safe designs can achieve these goals with no active controls or human operational intervention to manage anticipated transients and Loss of Coolant Accident (LOCA). However, sensor and actuator functions which require Direct Current (DC) power are still needed to open valves at the right moment.

The passive safety system of the generation III+ BWR is introduced in Chapter Two and it was discussed in greater detail in a relevant paper (Lim et al., 2014).

A comprehensive test and analysis program should be carried out to study the thermal-hydraulic performance of the unique passive safety systems and interactions of their components in the event of LOCA. Since it is not practicable to build and test a full power prototypical system, a scaled integral system is required (Ishii et al., 1998).

Purdue University designed and constructed an integral test facility, i.e. PUMA (Purdue University Multi-dimensional integral test Assembly), sponsored by the United States Nuclear Regulatory Commission (USNRC). The facility contains all of the important safety systems of the generation III+ BWR that are pertinent to the postulated LOCA transient. A feasibility study of passive safety features has been experimentally performed in a postulated GDCS Drain Line Break (GDLB) accident (Ishii et al., 2008).

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Nomenclature

AC	Alternating Current
ADS	Automatic Depressurization System
BWR	Boiling Water Reactor
DC	Direct Current
DPV	Depressurization Valve
DW	Drywell
ECCS	Emergency Core Cooling System
EL	Elevation
GDCS	Gravity Driven Cooling System
GDLB	GDCS Drain Line Break
ICS	Isolation Condenser System
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MSIV	Main Steam Isolation Valve
PCCS	Passive Containment Cooling System
PUMA	Purdue University Multi-dimensional integral test Assembly
RELAP	Reactor Excursion and Leak Analysis Program
RPV	Reactor Pressure Vessel
SLCS	Standby Liquid Control System
SP	Suppression Pool
SRV	Safety Relief Valve
TAF	Top of Active Fuel
USNRC	United States Nuclear Regulatory Commission
VB	Vacuum Breaker
WW	Wetwell

The development of a well-balanced and justifiable scaling approach is essential. The PUMA scaling was based on the combination of the integral system scaling and the scaling of key local phenomena. The local phenomena should be scaled as accurately as possible and a well scaled integral test facility will produce valuable integral experimental data that simulates all the major phenomena of interest.

The description of PUMA facility and the integral test is presented in Chapters Three and Four. Subsequently, the conclusions obtained from the PUMA GDLB LOCA test are summarized in Chapter Five.

2. Passive safety system

In the 1990s, the generation III+ BWR design was developed relying to a large extent on passive features such as Isolation Condenser System (ICS), Automatic Depressurization System (ADS), Suppression Pool (SP), Standby Liquid Control System (SLCS), Gravity Driven Cooling System (GDCS), and Passive Containment Cooling System (PCCS). The GDCS and PCCS are unique to the generation III+ BWR serving as the Emergency Core Cooling System (ECCS) and containment cooling systems of currently operating BWRs. The ADS is actuated at a prescribed Reactor Pressure Vessel (RPV) downcomer collapsed liquid level condition (Level 1) which is 1 m above Top of Active Fuel (TAF) and depressurizes the RPV so that the GDCS can be actuated to inject highly subcooled water into the RPV. The ICS is functionally similar to that in operating BWR and acts as a decay heat removal system. The goals of the passive safety systems are to adequately cool the core by maintaining a water level above the active core and to provide a sufficient heat sink to keep the containment pressure and temperature below the design criteria.

The generation III+ BWR relies on natural circulation to provide flow to the reactor core different from conventional BWRs (Duncan, 1988). Natural circulation allows the elimination of several systems including recirculation pumps, safety system pumps and safety diesel generators that could possibly fail. The emergency core cooling and containment cooling systems do not have an active pump injecting flows and the cooling flows are driven by gravitational head.

Fig. 1 presents a schematic of the generation III+ BWR including passive safety system that requires only DC power from batteries but no external Alternating Current (AC) electrical power source or operator intervention. In particular, the PCCS, which is open to the Drywell (DW), receives a mixture of steam and non-condensable gas directly from the DW. Therefore, the PCCS operation requires no sensing, control, logic or actuated devices for operation. The passive safety system of the generation III+ BWR was discussed in greater detail in a relevant paper (Lim et al., 2014).

2.1. Integrated passive safety system performance during the LOCA

The most effective means of describing the function of each of passive safety systems is to relate their operations in response to a LOCA. As shown in Fig. 2, the LOCA transient is divided into the blow-down phase, GDCS injection phase and long-term cooling phase (Gamble, 2002). It is noted that GDCS initiates when the RPV pressure equalizes with that of the DW as indicated with a dotted line. The LOCA transient phases were discussed in greater detail in a relevant paper (Lim et al., 2014).

3. Experiment

The PUMA facility was designed based on the scaling and scientific design study for the generation III+ BWR (Ishii et al., 1996, 1998 and 2006). The PUMA facility is intended to operate at and below 1.034 MPa (150 psia) following scram with scaling ratios: pressure (1/1), temperature (1/1), level (1/4.5), volume (1/580), power (1/273.3) and time (1/2.12). A schematic of the PUMA facility is shown in Fig. 3.

The PUMA GDLB test initial conditions at 1.034 MPa are obtained from the RELAP5/MOD3.3 (patch03) simulation for the generation III+ BWR with appropriate scaling considerations when the generation III+ BWR RPV depressurizes from 7.171 MPa (1040 psia) to 1.034 MPa (150 psia). The RELAP5 code was developed for the USNRC and suitable for the analysis of all transients and postulated accidents in Light Water Reactor (LWR) systems, including both large and small break LOCAs. At the early stage of the blow-down phase, a critical flow is a dominant process and it can be predicted reasonably by the RELAP5 code.

It is noted that one ICS unit was assumed to be out of service and not available in the PUMA GDLB test as well as the RELAP5 application to the generation III+ BWR (GE, 2006).

The decay power curve for the PUMA GDLB test was scaled down from the generation III+ BWR. The generation III+ BWR decay power curve was obtained from the ANS Decay Heat Standard (Jo et al., 1996).

In the generation III+ BWR, the stored energy from the fuel rods in addition to the core decay heat will be released into the reactor, adding to the fluid enthalpy. The stored energy in the prototype fuel rods consisting of Uranium and Zircaloy cladding cannot be physically scaled in the PUMA electrical heaters. This additional source of heat needs to be properly scaled for balancing the total energy of the system. This amount of additional energy can be compensated