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A Study on the Mitigating Capability of an Auxiliary Feedwater System During SBO for APR1400

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ABSTRACT

The objective of this paper is to establish an auxiliary feedwater (AFW) operational technical bases for the Korean Next Generation Reactor (APR1400) by modeling the plant, and by analyzing station blackout (SBO) using the MELCOR code. For the integrity of the reactor vessel and containment safety against severe accidents, it is essential to understand the severe accident sequences and to assess accident progression accurately using computer codes. Furthermore, it is important to attain the capability to analyze the advanced nuclear reactor design for the severe accident prevention and mitigation. Accident analyses are also undertaken to find out how effective AFW is mitigating in severe accident progresses. A nominal base case for SBO without AFW, time interval between feedwater stop and reactor vessel failure is 12,740 seconds. When AFW operates to mitigate the SBO accident progression 2, 4 and 8 hours after SBO starts, the reactor vessel failure is delayed for 20,415 seconds, 22,633 seconds and 26,508 seconds, respectively thus the operator has more time available for AC recovery and accident mitigation to prevent reactor vessel failure.

KEY WORDS: auxiliary feedwater, station blackout, Next Generation Reactor, integrity of the reactor vessel, accident prevention, accident mitigation

1. INTRODUCTION

Since the Three Mile Island (TMI) accident, the importance of accident management in nuclear power plants has increased. Many countries, including the United States (US), have focused on understanding severe accidents, in order to identify ways to further improve the safety of the plants [1]. It has been recognized that plant-specific safety assessments are beneficial in understanding plant-specific vulnerabilities to severe accidents.

The severe accident phenomena of a nuclear power plant have large uncertainties. For the retention of containment integrity and the improvement of nuclear reactor safety against severe accidents, it is essential to understand the severe accident phenomena and to be able to access the accident progression accurately using computer code. Furthermore, it is important to attain a capability for developing techniques and assessment tools for an advanced nuclear reactor design as well as for severe accident prevention and mitigation. Also, an Auxiliary feedwater system (AFWS) is one of the important systems related to the severe accident prevention and mitigation during SBO. The objective of this paper is to establish a technical bases for an application of the MELCOR[1] code to the Korean Next Generation Reactor (APR1400) by modeling the plant and by analyzing the plant steady state. Thus, this paper is undertaken to describe the safety analysis of the nuclear power plants, to identify plant response and vulnerabilities via analyzing the anticipated results, and to set up a framework for an accident management program based on those analysis results. In this paper, we have estimated the optimization of the auxiliary feedwater system of APR1400 using the safety analysis approach. The operating duration of an auxiliary feedwater system after SBO has been classified as 2 hrs, 4 hrs and 8 hrs to identify accident mitigation of the APR-1400. APR1400 is a typical pressurized water reactor in Korea. Finally, conclusions are presented in Section 4.

2. METHODOLOGY

2.1 Methodology of Analysis

Basically the scope and methodology used to perform this analysis are similar to those of the conventional safety analysis report. Therefore, a state-of-the-art safety analysis technique has been used in this study to demonstrate its application feasibility for accident management [2] using the MELCOR code.

2.2 Initial conditions and assumptions of plant features

The APR1400 is a two-loop, 4000MWt pressurized water reactor, which is an extension of the Korean Standard Nuclear Plant (KSNP) design. An evolutionary approach that relies on the actual experience from a KSNP is utilized for the development, and therefore, configuration of the reactor coolant system (RCS) is basically the same as that of the KSNP such as a reactor vessel, a pressurizer, two steam generators with four reactor coolant pumps, two hot legs and four cold legs, etc.. The changes are made in the component sizes and design parameters primarily to increase the heat transfer for additional power, i.e. increase of the steam generator heat transfer area; higher flow reactor coolant pump; large pressurizer; and reduction of the hot-leg temperature. In addition, new and advance design features are introduced including the adoption of pilot operated safety relief valves (POSRVs), four-train safety injection system (SIS) with direct vessel injection (DVI), fluidic device in safety injection tank (SIT), in-containment refueling water storage system (IRWST), and an integrated head assembly (IHA). Also, design improvements are made to the chemical and volume control system (CVCS), shutdown cooling system (SCS), containment spray system (CSS), and containment hydrogen control system(CHCS).

Assumptions and initial conditions for the SBO are as follows :

- no RCS failure assumed due to high temperature and overpressure
- no AFW (Both of motor and Turbine-driven Pump)
- POSRV available, no hydrogen detonation considered
- no steam explosion
- considered transient initiates at 100 sec after steady state achieves stable condition**. (**no main feedwater , RCP Stop , turbine trip, MSIV Close at 100 sec)

The station blackout (SBO) sequences consists of a total loss of all AC power, including those from emergency diesel generators with a failure of all engineering safety features. This event is defined as TMLB' by WASH-1400 [3]. It is assumed that the operator fails to actuate the Safety Injection Pump, the Containment Fan Cooler (CFS) and Containment Spray Pump.

3. RESULTS

This section describes the analysis results of the SBO. These event sequences are summarized as table 1. Major plant behaviors are as follows.

3.1 Primary System Behavior

The SBO results in the unavailability of all engineered safety systems. As a result of the unavailability of the feed water to the steam generators, approximately one hour (3,682 seconds) after the SBO the steam generators dry out and heat removal from the RCS is lost. Loss of heat removal results in a repressurization of the RCS up to the Pilot Operated Safety Relief Valve (POSRV) setpoint pressure. The cycling of the POSRVs at 4,750 seconds after the SBO brings on a continuous loss of RCS inventory and core uncovery (5750 seconds). Without operation of any engineered safety feature systems, the fuel rapidly heats up, melts, relocates to the lower plenum and fails in the RV lower head at approximately 12,840 seconds after the accident. A summary of the MELCOR predicted sequence of the key events and their timings are provided in Table 1. A summary of the key parameters are provided in Table 2. The RV failure mechanism is initially assumed to be a failure of a single lower head penetration.

Table 1. Major sequence-SBO with AFW / without AFW

Events Sequence	AFW Availability			
	No-AFW	2hr-AFW	4hr-AFW	8hr-AFW
TMLB' Transient Initiated				
Steam Generator Dryout	3,682	15,726	24,129	40,207
Pressurizer POSRV Open	4,750	17,347	26,047	42,516
Core Uncovery Start	5,797	18,816	27,633	44,327
Complete Core Uncovery	7,837	21,472	30,616	47,792
UO2 Melting Start	8,543	22,324	31,549	48,785
Core Support Plate Fail	12,832	27,615	37,025	55,300
Lower Head Penetration Failure	12,840	27,622	37,033	55,308

Table 2. Major Comparisons Elements SBO with / without AFW

Comparison Elements	AFW Availability			
	No-AFW	2hr-AFW	4hr-AFW	8hr-AFW
Time available between AFW Stop and RV rupture (sec)	12,740	20,415	22,633	26,508
Containment Pressure at RV Rupture (Pa)	2.9277E5	3.0378E5	3.4295E5	2.2798E5
Corium mass in low plenum at RV rupture(kg)	100,348	100,694	101,571	100391
H2 mass generated in core before RV Rupture (kg)	596	605	619	607

3.2 Steam Generator Behavior

Meanwhile a steady-state condition keeps the secondary water level constant the in steam generator, without such a feedwater inventory decrease rapidly to a dryout state of SG at 3,700 seconds after the SBO event occurs. As pressure of the SG secondary side would rise up to 8.27 Mpa from the onset of the SBO.

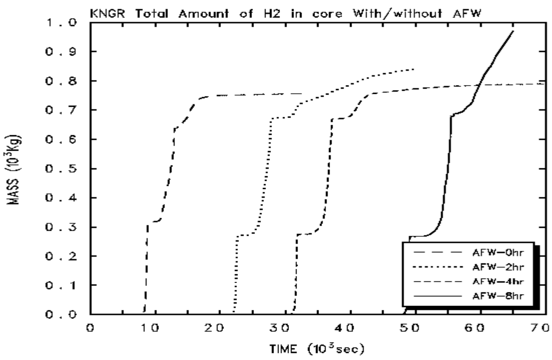
3.3 Cavity and Containment Response

The containment performance during this SBO sequence demonstrates the passive plant capabilities of plant. The upper compartment pressure for this event is presented in Fig. 2. Pressures and temperatures in other containment locations are similar. For the SBO scenario, discharge from the primary system are ducted via the pressurizer pressure relief piping into the In-containment Refueling Water Storage Tank (IRWST). During the early part of the transient, the IRWST water remains subcooled since there is no significant discharge from the RCS. During the early stages of this discharge, steam discharged into the IRWST is condensed due to the subcooling in the IRWST water. Therefore, the containment pressures remain near initial conditions until the IRWST reaches the saturation conditions. At that time the IRWST water begins to boil, adding steam mass into the containment atmosphere. Without containment heat removal (containment sprays unavailable), this steam addition is seen to directly result in a small containment pressure increase. At reactor vessel breach (12,840 seconds), a rapid containment pressurization is observed at 3.0 MPa . Other pressure peaks come from hydrogen burn., and the corium debris deposited into the flooded cavity.

3.4 Assessment of AFW mitigating effect during SBO

As a result of core heat removal via the stem generators, the RCS pressure is maintained below 2250 psia (158 Kg/cm2) during the time period of auxiliary feedwater availability. Following reactor vessel rupture, corium release to the cavity occurs in a short time and Particulate Debris relocation to the lower Plenum increases rapidly. Around 8,000 sec after the SBO initiates, H_2 Production Begins.

Calculation results show that the steam generator dry-out time is very important. When the temperature becomes saturated in the RCS, core uncovery begins. RPV wall temperature increases rapidly, Vessel integrity fails. Table 1, 2 and Fig. 1 ~ Fig. 6. show how much AFW mitigate the accident progress of the SBO event. As shown in table 1, the accident sequence time is very delayed with AFW. Time available to RV rupture increases with the AFW running. Nominal base case of SBO without AFW, time interval between feedwater stop and reactor vessel failure is 12,740 seconds. . When AFW operates to mitigate the SBO accident progression 2, 4 and 8 hours after the SBO starts, the reactor vessel failure is delayed for 20,415 seconds, 22,633 seconds and 26,508 seconds, thus respectively the operator has more available for AC recovery and accident mitigation to prevent reactor vessel failure. Fig. 1 shows the trend of total hydrogen mass in core during SBO which are similar regardless of the operation timing of the AFW. In the case of SBO with 4 hrs AFW operation time, the containment pressure (fig. 2) is relatively higher than that of other cases. But corium mass in low plenum, H_2 mass in the core, and the temperature and pressure are similar (fig. 3 and 4). Total of particulated debris (fig. 5) in the lower plenum shows slightly different trends (110-130 e3 Kg) in both of the SBO with and without AFW. Fig 6 shows that the range of integrated mass release to the cavity is 150-160 e3 Kg.



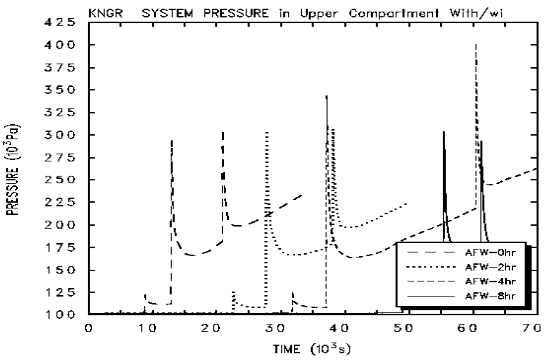


Fig. 1. Total amount of H_2 in Core Fig. 2. Containment Pressure

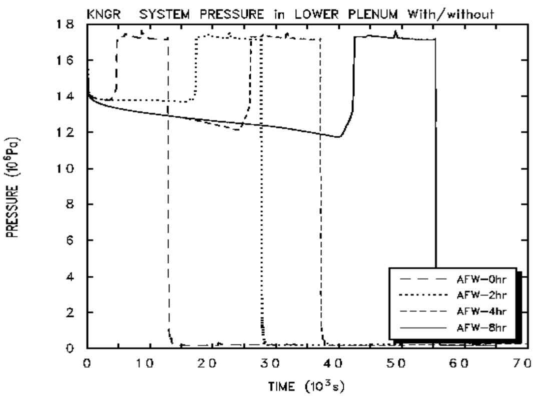
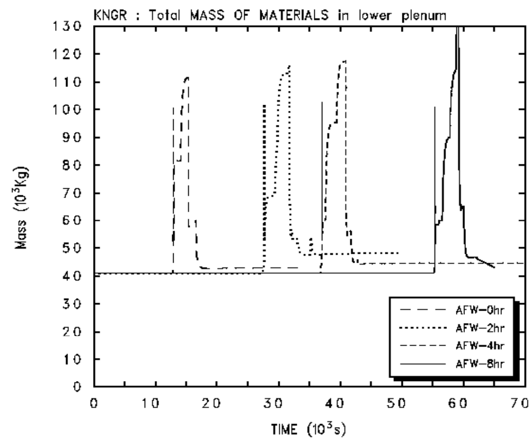


Fig. 3. Pressure in Core / lower Plenum

Fig. 4. Particulated Debris Temp



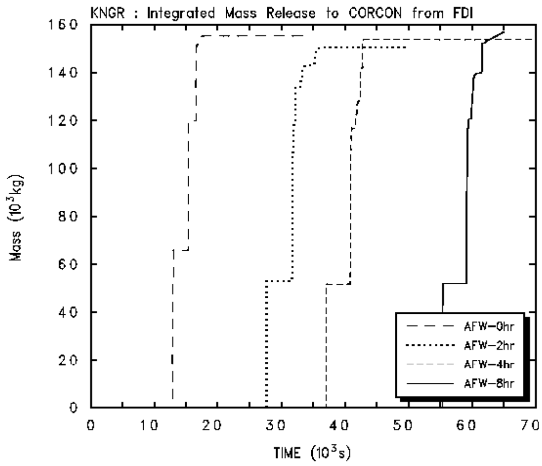


Fig. 5. Total mass of Particulated Debris in Lower Plenum   Fig. 6. Integrated mass Release to Cavity

4. CONCLUSIONS

Accident analyses are undertaken to find out how effective AFW is in mitigating the severe accident progresses. A nominal base case of SBO without AFW, time interval between feedwater stop and reactor vessel failure is 12,740 seconds. When AFW operates to mitigate the SBO accident progression at 2, 4 and 8 hours after SBO starts, the reactor vessel failure is delayed for 20,415 seconds, 22,633 seconds and 26,508 seconds respectively, thus the operator has more available for AC recovery and accident mitigation to prevent reactor vessel failure. As the result, the probability of not recovering AC power becomes less with the operation of the AFW than without.

ACKNOWLEDGMENTS

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