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Assessment of PASS Effectiveness under Severe Accidents in Nuclear Power Plants

Yu Jung CHOI [a*], Sung Bok LEE[a], Hyeong Taek KIM [a], Jin Yong LEE [b]
[a]*Korea Hydro & Nuclear Power Co., LTD.*
25-1 Jang-dong, Yusung-gu, Daejon, 305-343 Korea
[b]*FNC Technology Co., LTD.*
Seoul National University, Bldg. #135 San 56-1 Shilim-dong, Gwanak-gu, Seoul, 151-742 Korea
[*]*Corresponding author: yujungchoi@khnp.co.kr*

1. Introduction

Following the accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979, the USNRC formed a lessons-learned Task Force to identify and evaluate safety concerns originating with the TMI-2 accident. NUREG-0578[1] documented the results of the task force effort. One of the recommendations of the task force was for licensees to upgrade the capability to obtain samples from the reactor coolant system and containment atmosphere under high radioactivity conditions and to provide the capability for chemical and spectral analyses of high-level samples on site. NUREG-0737[2] contained the details of the TMI recommendations that were to be implemented by the licensees. Additional criteria for post accident sampling system(PASS) were issued by Regulatory Guide 1.97[3]. As the results, PASS has been installed on nuclear power plants(NPPs) in Korea as well as United States.

However, significant improvements have been achieved since the TMI-2 accident in the areas of understanding risks associated with nuclear plant operations and developing better strategies for managing the response to potential severe accidents at NPPs. Thus, the requirements for PASS have been re-evaluated in some reports. According to the reports, the samples and measurements from PASS do not contribute significantly to emergency management response to severe accidents due to the long analyzing time, 3 hours. Hence, this paper focused on the development of the quantitative analysis methodology to analyze the sequence of the severe accident in Yonggwang nuclear power plants (YGN) and presented the results of the analysis according to the developed methodology.

2. The Development of the Quantitative Analysis Methodology

2.1 Development of Accident Scenarios

First of all, the accident scenarios to be analyzed were developed based on a review of probabilistic safety assessment (PSA) results. According to the Level 1 PSA results of YGN, there are various accident scenarios to bring the core damage. Among these scenarios, 11 accident scenarios were selected as initiating events except the scenarios, which were impossible to be quantitated with computation code. The selected scenarios were listed in Table 1. The list included all severe accident scenarios according to the loss of the heat removal of the primary and secondary systems.

Table 1. The Selected Severe Accident Scenario from the PSA Results

Initiation Events
1. Large Loss of Coolant Accident
2. Medium Loss of Coolant Accident
3. Small Loss of Coolant Accident
4. Steam Generator Tube Rupture
5. Large Secondary Side Break
6. Loss of Feedwater
7. Loss of Instrument Air
8. Loss of Component Cooling Water
9. Loss of Offsite Power
10. Station Blackout
11. Interfacing Systems Loss of Coolant Accident

After the core damage, the status of the reactor cooling system and the containment was affected by three main factors; the time of the core damage, the pressure of the reactor cooling system, and the heat removal from the primary and secondary system. Thus, from the scenarios in Table 1, 16 main accident scenarios for the analysis were developed to cover the various associations of three factors, as listed in Table 2.

2.2 Development of the Analysis Models

For evaluating the effectiveness of PASS, the modular accident analysis program (MAAP) 4.0.4 was utilized. The MAAP code is a computational code to simulate the reactor cooling system and containment, and this code includes important models to simulate the phenomena of the severe accidents.

For simulating the scenarios with MAAP code, the analysis models of the reactor cooling system and the containment for YGN were developed. Especially, the existing containment analysis models, which were composed 5 nodes were modified to the multi-compartment containment models which were composed 21 nodes, considering the sampling points of PASS in the containment.

Table 2. The Severe Accident Scenario for the Analysis

Case No	Initiating Events
L-1 [a, c]	Large Break Loss of Coolant Accident Size of 6" in cold-leg
L-2 [a, d]	Large Break Loss of Coolant Accident Size of 6" in cold-leg
L-3 [b, d]	Large Break Loss of Coolant Accident Size of 6" in cold-leg
L-4 [a, c]	Medium Break Loss of Coolant Accident Size of 2" in cold-leg
L-5 [a, d]	Medium Break Loss of Coolant Accident Size of 2" in cold-leg
L-6 [b, d]	Medium Break Loss of Coolant Accident Size of 2" in cold-leg
L-7 [a, c]	Small Break Loss of Coolant Accident Size of 3/8" in cold-leg
L-8 [a, d]	Small Break Loss of Coolant Accident Size of 3/8" in cold-leg
L-9 [b, d]	Small Break Loss of Coolant Accident Size of 3/8" in cold-leg
N-1 [a, c]	Station Blackout
N-2 [b, c]	Station Blackout
N-3 [a, e]	Loss of Feedwater Failure of both MD- and TD-AFW
N-4 [b, e]	Loss of Feedwater Failure of both MD- and TD-AFW
N-5 [a, c]	Steam Generator Tube Rupture One tube Failure of 1.33" in hot side of SG
N-6 [a, d]	Steam Generator Tube Rupture One Tube Failure of 1.33" in hot side of SG
N-7 [b, d]	Steam Generator Tube Rupture One Tube Failure of 1.33" in hot side of SG

a : No operator action
b : Operator action
c : Failure of all safety systems initially
d : Failure of all safety systems in recirculation mode
e : No failure of all safety systems initially

3. Analyses Results of the Accident scenarios

The analyses about the 16 accident scenarios were carried out with MAAP code. The analysis results of YGN units 1 and 2 were listed in Table 3. Because the results of YGN units 3 and 4 showed quite similar trends compared to YGN units 1 and 2, the analysis results of YGN units 3 and 4 were not presented in this paper.

Table 3. The Period of the Accident Progress

Case No	Period of Accident Progress [hr]						
	A	B	C	D	E	F	G
L-1	0.04	-	0.96	0.19	0.67	1.08	29.16
L-2	0.06	-	2.67	0.38	0.89	59.32	-
L-3	0.06	-	2.67	0.38	-	-	-
L-4	0.4	-	0.40	0.17	0.51	1.79	23.72
L-5	3.16	-	0.74	0.52	0.89	59.42	12.15
L-6	3.16	-	0.74	28.3	-	-	-
L-7	1.04	0.39	0.5	0.22	0.78	0.61	18.97
L-8	28.35	-	10.75	5.51	4.05	-	15.31
L-9	28.35	-	10.75	5.51	-	-	-
N-1	0	-	2.46	0.45	1.13	0.71	20.05
N-2	0.06	-	2.40	0.45	1.19	10.14	-
N-3	9.57	4.13	0.16	0.45	1.36	5.63	-
N-4	9.57	4.13	0.16	0.45	-	-	-
N-5	0.89	1.24	1.19	0.28	0.79	1.07	25.5
N-6	4.25	-	1.13	0.66	0.91	62.43	11.38
N-7	4.25	-	1.13	0.66	-	-	-

A : Period from the beginning time of accident to the time of reactor coolant pump off
B : Period from the time of reactor cooling pump off to the drying time of the steam generator
C : Period from the drying time of the steam generator to the time of core uncovered
D : Period from the time of core uncovered to the time of core exit temperature above 922 K
E : Period from the time of core exit temperature above 922 K to the time of core relocated
F : Period from the time of core relocated to the time of reactor vessel failed
G : Period from the time of reactor vessel failed to the time of containment failed

According to the specific criteria for PASS capability delineated in NUREG-0737 and Regulatory Guide 1.97, all samples and measurements were to be taken and analyzed within 3 hours of the decision to do so except for chlorides which were to be taken and analyzed within 24 hours. However, as shown in Table 3, almost all of the periods of each accident progress were less than 3 hours except the L-8 and L-9 Case. Therefore, the PASS information under severe accidents is able to very mislead the operator action. Based on the analyses results, it could be concluded that PASS was unavailable for the most severe accident scenarios.

4. Conclusion

To re-evaluate the requirements for PASS in YKN, the quantitative analysis methodology to analyze the sequence of the severe accident was developed. The severe accident scenarios for the analysis were determined from the level 1 PSA results of YKN. And, the multi-compartment analysis models of MAAP code were developed.

According to the results, almost all of the severe accident scenarios were progressed rapidly, so that the periods of each accident progress were less than 3 hours. Thus, the samples and measurements from PASS within 3 hours could not contribute significantly to emergency management to severe accidents.

REFERENCES

[1] "TMI Lessons Learned Task Force Report and Short-Term Recommendations," NUREG-0578, July 1979.
[2] "Post Accident Sampling Guide for Preparation of a Procedure to Estimate Core Damage," supplement to NUREG-0737, March 1982.
[3] "Regulatory Guide 1.97: Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S Nuclear Regulatory Commission, May 1983.