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Severe Accident Source Term Analysis for a 4000 MWt Evolutionary Pressurized Light Water Reactor

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Abstract

In order to get an insight on the feasibility of NUREG-1465 accident source term, fission product releases into containment of a 4000 MWt evolutionary pressurized light water reactor design under severe accidents are analyzed by using MAAP4 computer code. The fission product release timing and fraction thus estimated are compared with NUREG-1465 especially for iodine case. For the analysis, accident sequences are selected based on the core damage frequency of the plant design. Results show that severe accident source term of a large evolutionary pressurized light water reactor design is well represented and even bounded by NUREG-1465-defined values.

1. Introduction

After Three-Mile Island accident, even though the importance of the severe accident source term became of great interest, relatively small amount of iodine were released into the environment than the previous licensing calculation addressed in the Regulatory Guides which assumes instantaneous release of fission product. Based on the finding, a number of observers have claimed about the conservatism inherent in the previous regulation [1].

Results of researches driven by USNRC on the fission product release timing and magnitude (NUREG-1150 [2], NUREG/CR-5747 [3]) are concentrated into NUREG-1465 which represents the source term in the risk-significant accident sequences of the typical nuclear power plants operating in the United States..

The accident source term defined in the NUREG-1465 have been derived from examination of a set of severe accident sequences for light water reactors (LWR) of current design. However, because of general similarities in plant and core design parameters, these results are also considered applicable to evolutionary LWRs such as ABWR and System 80+. [4]

Thus it can be stated that the NUREG-1465 could be readily applicable to the Korean Next Generation Reactor (KNGR) design which is a 4000 MWt evolutionary pressurized LWR comparable to the western large advanced power reactors. However, in order to achieve an insight on the fission product magnitudes of the plant, severe accident source terms are estimated by using MAAP4 [5] computer code and the results are compared with NUREG-1465 values.

2. Source Term Analysis

In order to define fission product release timing and fraction representing operating reactors in the NUREG-1465, source term analysis results for the dominant accident sequences of several PWR (pressurized water reactor) and BWR (boiling water reactor) plants has been referred. Among them, Surry, Zion, Oconee3 and Sequoyah are considered as PWR plants, and the accident sequences can be largely classified into small LOCAs (Loss of Coolant Accident) and transients. For LOCA, hot leg large break LOCA is included only in the sequences of Surry and Sequoyah, and most of the other sequences are small break LOCAs.

For the KNGR, according to the NUREG-1465 approach, accident sequences dominating in core damage frequency are selected and the description is shown in Table 1 [6]. The fission product release fraction and timing are calculated using MAAP4 computer code, and the results are compared with NUREG-1465 values mainly focused on iodine release which is an important radioisotope with respect to 10CFR100 offsite dose calculation.

MAAP4 provides integrated plant model which has been extensively used over the ten years for source term analysis. Also, the code has been benchmarked against experimental studies related to severe accidents and industry experience such as TMI-2 accident, as well as RETRAN and RELAP5 computer codes. Previous comparisons between generic MAAP4-calculated source term and those defined in the NUREG-1465 indicated general agreement between the two methodologies which suggests that MAAP4 can be used to generate plant-specific source terms for regulatory applications [7-11].

One difference between NUREG-1465 and MAAP4 is that, as shown in Table 2, NUREG-1465 presents the fission products in their elemental forms and provides guidance on the expected chemical transformations, while MAAP4 combines the elements into compounds based on consideration of chemical properties and mass balances. NUREG-1465 source term is divided into eight fission product groups, while MAAP4 calculations use twelve groups. However, all the important fission products are equally represented by both NUREG-1465 and MAAP4. For the Caesium (Cs) and Iodine (I), MAAP4 considers them as a compound, CsI, although it can exist as aerosol or vapor according to the competition of CsI vapor pressure and containment pressure.

Another difference is that NUREG-1465 defines five distinct release phases for the fission products. In contrast, MAAP4 calculates the reactor fuel rod and coolant system heat-up, fission product release from the core, transport to the containment, leakage to the reactor or environment, including models for engineered safety systems and active and passive fission product retention mechanisms. MAAP4 allows each fission product group to exist in three possible states (aerosol, vapor and deposited to the heat sink) and calculates the mass transfer between these states [5].

2.1 Accident Scenarios Considered

For the sequences dominating in CDF shown in Table 1, not all the sequences need to be analyzed since a sequence can be bounded by another sequence due to following reason:

MLOCA-S03 and LLOCA-S03 are sequences where hot leg injection to prevent boron precipitation failed after safety injection into direct vessel injection (DVI) line is successfully delivered into reactor vessel downcomer. Based on the design basis large LOCA analysis, the boron precipitates 3.4 hr post-LOCA and thus gap fission product release into containment will occur after at least 3.4 hrs for these two sequences. Therefore, it can be known without analyses that gap and early in-vessel fission product release will be much slower than NUREG-1465-defined in-vessel phase of 2 hours.

SGTR-S28 and SLOCA-S22 are the sequences where RCS injection using safety injection system or shutdown cooling system failed even though aggressive secondary cooling is successful. These sequences are very slow and we core uncover is not expected during 12 hrs of calculation using MAAP4.

SGTR-29 is a sequence where steam generator tube rupture occurs, and after that aggressive secondary cooling and safety injection system failed. Thus SGTR-S29 is similar to SLOCA-S23 in core melt progression except that SGTR is a bypass sequence.

ATWS-S12 is a sequence where secondary cooling is successful but the boron injection and primary depressurization fails. Since secondary cooling is successful for this sequence, the accident progression will be slower than LOFW-S17 even though steam generator tube rupture might be induced. However, after that, the core melt progression of this sequence would be similar to SGTR-29.

2.2 Assumptions for MAAP4 calculation

Based on the description stated in the previous section, LOFW-S17, SLOCA-S23, SGTR-S28, LODC-S18 among the sequences listed in Table 1 are finally selected for this source term analysis and we used following assumptions for each sequence:

(1) LOW-S17: Primary safety depressurization using pressurizer relief valves is assumed to begin at the automatic opening set point of the pressurizer safety valves.
(2) SLOCA-S23: Break size of 0.05 ft^2 on cold leg is assumed.
(3) SGTR-S28: It is assumed that 2 tubes are ruptured at the highest point of the U-tubes.
(4) LODC-S18: The initiating event is loss of 125V DC and aggressive secondary cooling. Also primary safety depressurization subsequently failed. However, for simplicity we assumed the sequence as a total loss of feed water with a failure of primary depressurization.

Sequences and assumptions stated above are summarized in Table 2. For the MAAP4 calculation, the default fission product release model is used: CORSOR-M model for noble gases and CsI and CsOH, and CORSOR-O model for the rest of the fission product groups. These models have been benchmarked against ORNL fission product release tests VERCOS.

3. Results and Discussion

For the LOFW-S17, the starting point of gap release is 4601 sec (1.28 hr) and after gap release the containment release rate after core damage is slower than NUREG-1465 as shown in Table 3 and Fig.1. This is mainly due to safety injection from safety injection tanks (SIT) after successful primary depressurization at 4807 sec (1.34 hr). The fraction of CsI release into containment until two hours is 24.2% which is smaller than 40% for 1.8 hours defined as an in-vessel release phase in NUREG-1465.

For the SLOCA-S23, due to the break on the reactor coolant system, the core uncover occurs at 1961 sec (0.55 hr) faster than 3747 sec (1.04 hr) for LOFW-S17, and the fission product release into containment begins at 2612 sec (0.73 hr) which is earlier than LOFW-S17. However, the reactor coolant depressurization through the break enables earlier injection of SIT water (2797 vs. 4807 sec). Thus, fractional CsI release into containment is about 10% for 2 hours into the accident, which is far less than 40% for 1.8 hours as defined in the NUREG-1465.

For the SGTR-S28, core uncover is not predicted for 12 hours of calculation and thus this sequence is of minor importance in comparison with NUREG-1465 in-vessel phase.

For the LODC-S18, the primary pressure is retained at the pressurizer safety valve set point due to failure of safety depressurization. Thus the core uncover occurred at 4584 sec (1.27 hr) which is far later than LOFW-S17 and SLOCA-S23. The fission product begins to release into containment after 5787 sec (1.61 hr) and the CsI release fraction is less SLOCA-S23.

4. Conclusion

Severe accident fission product release into containment is analyzed by using MAAP4 computer code for the KNGR and the results are compared with NUREG-1465 values. For the analysis, accident sequences dominating in core damage frequency based on the KNGR probabilistic safety assessment are considered. The analysis results show that source term of the dominant severe accidents in the KNGR containment is well represented and even bounded by NUREG-1465 values.

Among the sequences considered, the largest in-vessel release fraction for 2 hours into the accident is 24.2% for the KNGR, and it is much smaller than 40% during 1.8 hours as specified in NUREG-1465. Thus it can be stated that the severe accident source term in the KNGR containment is well represented and even bounded by NUREG-1465 values.

References

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Table 1 KNGR dominant severe accident sequences considered

No.	Accident Sequence	Description
1	LOFW-S17	Loss of main feed water, failure of aggressive secondary cooling, and success of primary depressurization
2	SLOCA-S23	Small loss of coolant accident, failure of safety injection system and failure of aggressive secondary cooling
3	SGTR-S28	Steam generator tune rupture, failure of safety injection system, success of aggressive secondary cooling, and failure of shutdown cooling system injection
4	SLOCA-S22	Small loss of coolant accident, failure of safety injection system, success of aggressive secondary cooling and failure of shutdown cooling system injection
5	ATWS-S12	ATWS and failure of secondary cooling with successive failure of boron injection and safety injection system
6	MLOCA-S03	Medium break LOCA, success of one safety injection system train, failure of hot leg injection
7	RVR-S01	Spontaneous reactor vessel failure
8	SGTR-S29	Steam generator tune rupture and failure of aggressive secondary cooling and safety injection system
9	LODC-S18	Loss of 125V DC, failure of aggressive secondary cooling, failure of safety depressurization
10	LLOCA-S03	Large LOCA, success of 2 trains of safety injection tanks and systems, failure of hot leg injection

Table 2 Comparison of NUREG-1465 and MAAP4 fission product groups

NUREG-1465	MAAP4
Group 1: Noble Gases (Xe, Kr)	Group 1: Noble Gases (Xe, Kr) and Non-radioactive Aerosols
Group 2: Halogens (I, Br)	Group 2: CsI + RbI
Group 3: Alkali Metals (Cs, Rb)	Group 6: CsOH + RbOH
Group 4: Tellurium Group (Te, SB, Se)	Group 3: Tellurium Oxide (TeO2) Group 10: Antimony (Sb) Group 11: Elemental Te (Te2)
Group 5: Barium Group (Ba, Sr)	Group 4: Strontium Oxide (SrO) Group 7: Barium Oxide (BaO)
Group 6: Noble Metals (Ru, Rh, Pd, Mo, Tc)	Group 5: Molybdeum Oxide (MoO2)
Group 7: Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	Group 8: Lanthanides (La2O3, Nd2O23, Sm2O3, Y2O3)
Group 8: Cerium Group (Ce, Pu, Np)	Group 9: Cerium (CeO2) Group 12: Uranium and Transuranics (UO2, NpO2, PuO2)

Table 3 Accident progression for each sequence for source term analysis

	Accident Progression Time (sec)			
	LOFW-S17	SLOCA-S23	SGTR-S28	LODC-S18
Accident initiation	0	0	0	0
Reactor Trip	19	3	145	0
Auxiliary Feed Water System Start	-	-	145	-
Pressurizer Safety Valve Open	2878	-	-	-
RCP Trip	3389	941	19932	4023
Core Uncovery	3747	1961	> 6hr	4584
Core Exit Temperature 1200 F	4476	2497	> 6hr	5461
Start of Fission Product Release	4601	2612	> 6hr	5787
Safety Injection Tank Injection	4807	2797	> 6hr	-
Reactor Vessel Failure	20136	24842	> 12 hr	12790

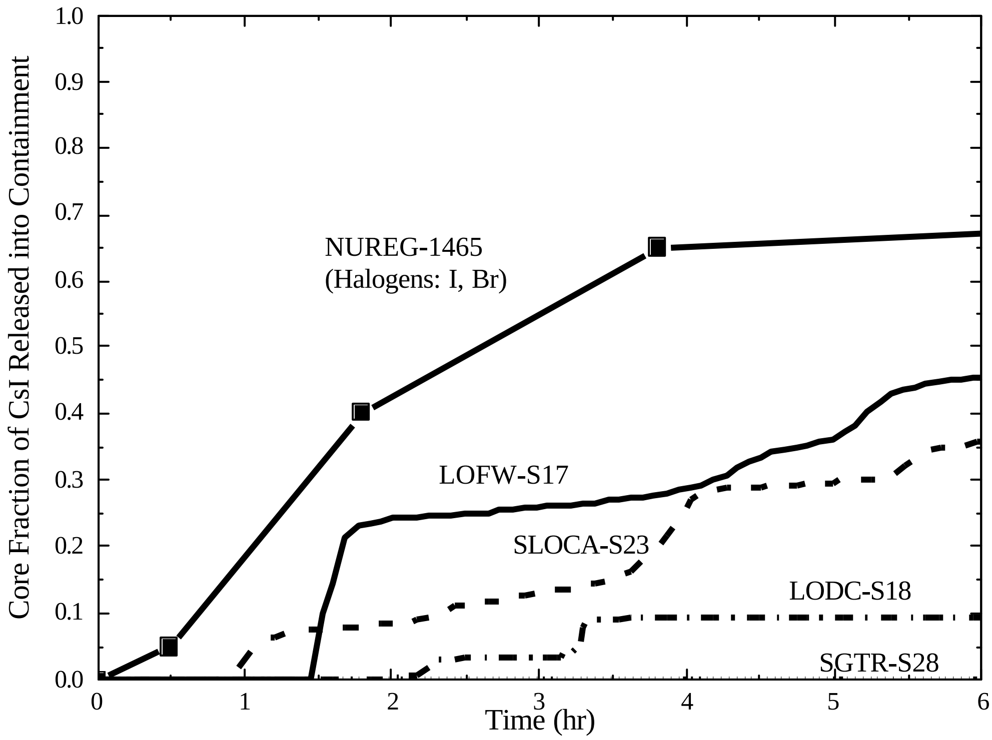

Fig. 1 Comparison of timing and fraction of CsI release into containment for the KNGR dominant accident sequences with Halogen release specified in the NUREG-1465.