Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 18-19, 2017

Status of Severe Accident Analysis for New Accident Management Plan

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1. Introduction

After the Fukushima accident, the Korean regulatory body requested that it should be necessary to revise the Safety Goal and develop the Accident Management Plan (AMP) including the inter-connection of Emergency Operation Procedure (EOP), Severe Accident Management Guideline (SAMG), and Extensive Damage Mitigation Guideline (EDMG) for domestic nuclear power plants. According to this, the AMPs for all domestic NPPs should be developed until June of 2019.

The current developing regulatory standard and guide require that AMP should identify the safety of nuclear power plants based on both the deterministic and probabilistic approaches.

The probabilistic method requires the model modification reflects a list of the facility improvements resulted from the post Fukushima actions first of all, and then development of realistic scenarios reflecting the main operator actions that include the SAMG. The intent of Probabilistic Safety Assessment (PSA) is to figure out the risk factors and to keep a low level as possible. Also, PSA results should meet the Safety Goal as below;

1) CDF < 1E-4 (1/10 in case of newly constructed plant)
2) LERF < 1E-5 (1/10 in case of newly constructed plant)
3) Frequency of Cs-137 release bigger than 100TBq < 1E-6

The deterministic method-based evaluation is to verify containment coping capabilities about the phenomena of the main current issues selected as the management objects that threaten the containment integrity. The Regulatory Standard 19.2.1 requires evaluating the threat factors needed to manage accidents where the frequency of occurrence or its influence is highly considerable after conducting the PSA.

The purpose of the Deterministic Safety Assessment (DSA) is to prove that the containment still remains intact against each of the threat factors when taking accident response actions into consideration.

This paper illustrates the analysis methodology to satisfy the deterministic Severe Accident analysis and impact evaluation required in the Accident Management Plan and discusses the recently raised issues. Also, the preliminary evaluation applying the methodology of the DBA dose evaluation is performed about the field of the deterministic offsite effect analysis of Severe Accident in which any concrete methodology has not been determined yet and its results are discussed.

2. Mitigation Capability Assessment

2.1 Selection of Accidents As Managed Objects

The Regulatory Standard 19.2.1 requires identifying containment integrity carrying out the evaluation for the containment challenges as follows.

1) Combustion and explosion of combustible gases
2) High temperature and pressure of containment buildings
3) Core melt and MCCI
4) High pressure melt ejection and HPME/DCH
5) Core melt and FCI
6) Creep fracture of SG heat pipe and isolation boundary bypass of containment building

Other than the factors mentioned above, the AMP requires conducting an evaluation about threat factors that need accident management due to its great frequency of occurrence and impact on the PSA for the purpose of considering a maximal range of the factors with both deterministic and probabilistic factors.

However, there is significant difference between DSA and PSA when it comes to the purpose and the process of the analyses. The DSA is generally to assess containment integrity with certain types of phenomena. Thus, it must verify the integrity in case where the certain phenomena occur by conservatively supposing the requirements and occurrence time of the threat factors.

On the other hand, the PSA focuses on evaluating probabilities that the reactor core or containment building might get damaged in the accident sequences by probabilistically modeling how the systems function from the point of occurrence of initiating events. That is to say, the PSA results itself suggest the possibility that the containment gets damaged as conclusion. Therefore, the method of the impact evaluation of Severe Accident and acceptance criteria need to be discerned from those of PSA although there is no problem with the method of deterministic analysis after selecting main accident sequences from the perspective of frequency of occurrence and core damage on the PSA.

2.2 Accident Evaluation Method

The Severe Accident coping design for domestic plants has been applied since APR1400 type reactors. APR1400 reflects the coping capability to the design carrying out the integrated analysis of Severe Accident at the design stage. The analysis of Severe Accident mitigating capability must be implemented in order to develop the AMP because the plants built prior to APR1400 do not reflect the requirements of Severe Accident.

The methodology used in the integrated analysis of Severe Accident for APR1400 is compared in table 1 while the evaluation methodology has not been certainly determined for the Severe Accident mitigating capability evaluation for the plants in operation to this point in time.

When it comes to methodology of the Severe Accident mitigating capability analysis, the first problem is whether or not it is reasonable to apply that of newly constructed plants.

The second problem is that as shown in table 1 there is great difficulty in evaluation at home and abroad even today due to its high uncertainty of the phenomena in which the momentum of HPME/DCH and FCI varies significantly and whose pressure fluctuates greatly. Due to these problems the Physical Code is usually used but it takes considerable time even only for developing a basic input model.

Thirdly, it is still not clear if the bypass accidents are the threat factors to the deterministic containment integrity. The bypass accidents are caused by the penetration or SGTR rather than by the problem with containment integrity. Therefore, it is difficult to determine if these accidents damage the integrity and to evaluate deterministic impacts without any experimental and computational codes models in the existing Severe Accident analysis. Surely, the accidents can be regarded important in reference to a release of radioactive materials but it is not reasonable to say that the bypass accidents could have a bad impact on core melt or containment integrity itself. Also, the DSA does not seem to be conducted since the accidents have already been known as important ones in PSA and have been specified in terms of mitigating strategies in the AMP.

3. Impact Analysis of Severe Accident

3.1 Considerable Requirements

The impact assessment of Severe Accident is stated in the Regulatory Standard 16.4. The result of it requires that the management criteria of radiation exposure dose at a boundary of limited areas be lower than 250mSv. The lists of basic assumptions are as follows.

1) Consideration of a potential release of radioactive materials to the external regions during the process until the safe environment is ensured
2) Reflection of accident sequence selections and accident management actions that include the accidents sorted into multiple failures, extreme natural hazard, Severe Accident in terms of radiation exposure level.
3) Possibility of taking accident management actions into account in order to reduce radiation exposure level (needed to be specified in the guide)
4) Possibility of using conservative assumptions and methodologies or those that have better reliabilities by means of uncertainty/sensitivity analysis applying realistic assumptions and methodologies.
5) Validation through the verification with experiment results and computational codes internationally used with the latest experiment and research results reflected
6) Assumption of conservative operation parameter for the inventory of fission products existed in the nuclear fuel.
7) Reflection of progression based on the accident sequences for locations and paths where radioactive materials are released (including leak rate at high pressure and so on)

The guidelines for the selection of accident sequences in the impact assessment of Severe Accident state that severe accidents and severe accident sequences must be selected based on the source terms and the movements of radioactive materials respectively among the accident sequences that largely contribute to the CDF.

However, the aforementioned provisions belong to the selection requirements of accident sequences of dose assessment based on the method of probabilistic assessment. In other words, the deterministic dose assessment should satisfy the criteria for radiation exposure levels with containment integrity identified after conducting the evaluation for mitigation capability based on the methodology of deterministic analysis. The issue of how the source terms of deterministic Severe Accident should be defined has still been pending.

Also, the evaluation cannot be implemented with the current level of technology if the definition of the perspective of radioactive materials movements includes phenomenological effects such as generation of fission products, collapse inside the containment building, condensation and re-evaporation. As a result, it is seen that these provisions should be re-discussed and redefined.

3.2 Preliminary Impact Assessment of Deterministic Severe Accident

The preliminary assessment has been conducted by using the following methodology to carry out the impact assessment of deterministic Severe Accident required in the AMP.

The source terms defined in TID-14844 have been used in the existing design basis accidents for the dose assessment when the accidents occur.

Source terms for Severe Accident have not been developed yet although the Alternate Source Term (AST) has been created. In Europe, the Reference Source Term have been developed and used for dose assessment in case of Severe Accident. Since it is predicted to take a long time for the RST development applicable in Korea the alternative way is to select the source terms amongst the PSA source terms based on the amount of emission of fission products during the accidents where the containments are intact.

The targeting plant is Westinghouse type NPP. The accident scenarios and evaluation methods are as follows.[3]

1) Initiating Event : SBO
2) Accident Sequence : reactor rupture at high pressure of RCS after core damage
3) Application of source terms : calculation of sixty nuclides with the ORIGEN code and sorting into nine nuclide groups, development of the nuclide library considering thermal power and the library of dose shift factors
4) Dose assessment districts : EAB, LPZ, MCR
5) Atmosphere dispersion factor : Use of the ARCON96/PAVAN codes, production of input data by making use of recent five-year meteorological factors
6) The motions of radionuclides in a timely manner : release fraction of each of the nuclides at 72 hours

3.3 Outcome of Preliminary Impact Evaluation of Deterministic Severe Accident

The dose assessment has been implemented applying the basic assumptions and methodology used when evaluating the DBA doses. The results are depicted in the following, Table 2 and Table 3

Table 2. Radiation Decay not considered

time	type	EAB	LPZ	MCR
Worst two hours	effective dose	3.8%		
	Thyroid gland	0.2%		
	Whole body	3.7%		
Total Release rate	effective dose	93.6%	7.4%	36.2%
	Thyroid gland	0.8%	0.1%	13.8%
	Whole body	92.8%	7.2%	29.0%
Acceptance Criteria	effective dose	25 rem	25 rem	5 rem
	Thyroid gland	300 rem	300 rem	50 rem
	Whole body	25 rem	25 rem	5 rem

Table 3. Radiation Decay considered

time	type	EAB	LPZ	MCR
Worst two hours	effective dose	3.0%		
	Thyroid gland	0.2%		
	Whole body	2.9%		
Total Release rate	effective dose	8.3%	0.7%	7.8%
	Thyroid gland	0.8%	0.1%	11.2%
	Whole body	7.8%	0.6%	1.4%
Acceptance Criteria	effective dose	25 rem	25 rem	5 rem
	Thyroid gland	300 rem	300 rem	50 rem
	Whole body	25 rem	25 rem	5 rem

As shown in Table 2 and Table 3, every evaluation results are shown to satisfy the acceptance criteria.

3. Conclusions

Even though the AMP for domestic nuclear power plants in operation should be developed by June 2019, the relevant regulation standards are still being revised and are not fixed yet. Therefore, in this paper, the issues for Deterministic approach required in the regulatory standard and guide were discussed. And the alternative methodologies for deterministic offsite dose calculation during the severe accident were proposed since it is expected to take the long time to determine the source term for severe accident.

The AMP related regulatory standards are likely seen to cause problems with consistency of assessment scenario selection and methodology because the probabilistic evaluation factors are mixed with the deterministic analysis. Particularly, those problems pertaining to the evaluation field of containment integrity and offsite effect evaluation seem to be urgently solved. The standards that can reflect the realistic assumptions on evaluation methods and accident mitigation actions must be more specifically defined.

It is seen that many discussions are needed for the evaluation methodology of each of the accidents that need Severe Accident management for the plants in operation. Above all, the proper measures should be made by admitting that it would take more time to carry out Severe Accident analysis in more detail.

Defining the source terms for Severe Accident impact analysis and developing the evaluation scenarios are also the pending issues.

The currently regulated dose limits are satisfied in the consequences of the preliminary assessment that used the existing DBA methodology and the source terms in case where the containment is intact. However, there is a need to discuss if it is possible to define the source terms as the representatives of the deterministic offsite impact analysis, which were used in the methodology.

Explicit technical standards must be established through in-depth talks between the utility, regulatory bodies, and experts in the field of Severe Accident in order to develop the AMP for the entire plants by June, 2019.

Also, there is a matter for discussion to develop the AMP step by step to reduce trial and errors. Just as used in developing the SAMG for domestic plants, it seems reasonable that the generic AMP should preferentially be developed and then modified and revised. Furthermore, developing a specific SAMG for individual plants is also regarded as an appropriate approach.

REFERENCES

[1] Accident Management Regulatory Standard, KINS/RS-N19.0. Rev.0, KINS, 2016.11

[2] Accident Consequence Assessment Regulatory Guide, KINS/RG-N16.04, Rev.0, KINS, 2016.11

[3] Technical Support for Dose Estimation in Condition of RER, 2017-50003339-전-0087TC, KHNP, 2017.01