Review of practicing Level-2 probabilistic safety analysis for Chinese nuclear power plants

PENG Changhong, ZHANG Ning, and YANG Yinghao

China Nuclear Power Technology Research Institute, Shenzhen 518026, China (pxm321@163.com)

Abstract: Existing studies about Level-2 PSA (Probabilistic Safety Analysis) in the world, covering NUREG-1150, IAEA-SSG-4, 50-P-8 and Level-2 plant PSA reports for AP1000 and EPR, serve in this paper as the basis of a literature study aimed at determining guidelines to practice Level-2 PSA in Chinese commercial nuclear power plants. We recapitulate the main technical elements composing Level-2 PSA: the familiarization with plant data and systems, interface with Level-1, containment performance analysis, accident progression and containment event tree analysis, source term and release category analysis, sensitivity, importance and uncertainty analysis, and the relationship between them. At last, outcomes of Level-2 PSA are presented.

Keywords: Level-2 PSA; containment performance; event tree analysis; source term; release category

1 Introduction

Probabilistic safety analysis (PSA) of a nuclear power plant provides a comprehensive and structured approach to identifying accident scenarios and deriving numerical estimates of the risk emerging from the operation of the plant and to which the public is exposed. The insights gained from PSA are used in conjunction with those from deterministic analysis in the decision-making process regarding safety issues. Nowadays, China is the most rapidly developing nuclear power country in the world. PSA technology is constantly being explored, developed and applied. In the Regulations on Design Safety for Nuclear Power Plants (HAF102) issued in April 2004, the National Nuclear Safety Administration (NNSA) explicitly required both deterministic and probabilistic analysis methods to be used in safety assessment. In addition, the “Review Principles of Nuclear Safety in the Generation II+ Nuclear Power Project” issued by the NNSA also explicitly required that the internal event Level-1 PSA should be carried out according to the requirement of HAF102-2004, and that the Level-1 PSA should be improved to reach the Level-2 and 3 PSA gradually.

Level-2 PSA models the phenomena following the onset of core damage that have the potential to challenge the containment integrity and lead to a release of radioactive material into the environment. In Level-2 PSA, the source term and the frequency are among the main elements on which the establishment of emergency planning in nuclear power plants is based. Emergency planning is an important part in the application process to get a construction license. Establishing emergency planning zones is the basis of emergency planning and should rely on the output of source term in the nuclear power plant. However, the detailed study of Level-2 and 3 PSA, which includes the analysis of source term release and environmental consequences, has not been conducted for a special nuclear power plant in China until now. The plants in China basically borrow the source term results originating from the same type of plants in other countries or from some classical PSA studies. For example, the Daya Bay and Ling’ao plants use the S3 source term employed for the same type of plants in France. However, because of the differences in location, design and operation, the risk for each plant is different. A reasonable and effective emergency planning must be based on plant-specific risk assessment results. The conclusions of Level-2 PSA can support the prevention and mitigation of severe accidents. They can also give input data for the Level-3 PSA. For all these reasons, the study of Level-2 PSA is an imperative work in China.

Reviewing and learning is the first step to carry out Level-2 PSA. In this paper, some studies about Level-2 PSA are reviewed, and then the technical procedure is underlined. Moreover, applications of

Received date: December 14, 2010
(Revised date: January 19, 2011)

Level-2 PSA are given. All this will give a basis for the development of Level-2 PSA in China.

2 Review of each technical element

Studies relevant to Level-2 PSA were reviewed, including NUREG-1150[1] "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants", "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants"[2], "Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level-2): Accident Progression, Containment Analysis and Estimation of Accident Source Terms (50-P-8)"[3] and others [4-7], such as AP1000 and EPR Level-2 PSA reports. The research framework is shown in Fig. 1. The main technical elements and the relationship between them are introduced as follows.

2.1 Familiarization with plant data and systems

The first task of Level-2 PSA is to familiarize with the design and operation of the plant as well as the plant's response to severe accidents, and to study the plant design characteristics, which are of great importance in the development of severe accidents and the subsequent containment response. These design characteristics include:

(1) The design and operation of systems that may be used during a severe accident to mitigate its consequences, such as containment structure and parameters, design data of safety systems and containment systems, system capacity, operating limits and actuation criteria, the possibility of containment bypass, hydrogen control systems, *Et al*;

(2) Important plant and containment characteristics, which may provide insights on the progression of the accident and potential vulnerabilities, such as the core structure, the materials and quality of each components, RCS pressure/temperature/ volume; concrete composition, *Et al*;

(3) The study of Severe Accident Management Guidance (SAMG);

(4) The selection and analysis of reference plants.

2.2 Interface with Level-1

The interface between Level-1 and Level-2 PSA is often accomplished through the definition of Plant Damage States (PDS), which should represent groups of accident sequences that result in a similar event progression and radiological source terms, and provide the initial and boundary conditions for undertaking severe accident analysis. The objective of the PDS analysis is to combine event sequences from the Level-1 that result in similar severe accident progressions, containment responses, and fission product releases into the environment. The Level-1 results are sorted according to the physical state of the plant systems that were in activity prior to the onset of the core damage, and the availability of systems that could be actuated subsequently to mitigate the consequences of the core damage.

Typical grouping criteria used for LWRs include:

- The type of initiating event (intact RCS or LOCA);
- The RCS pressure (high or low) at the time of the core damage;
- The status of safety systems (such as the safety injection system) and support systems (such as power and components cooling systems) at the time of the core damage and during the

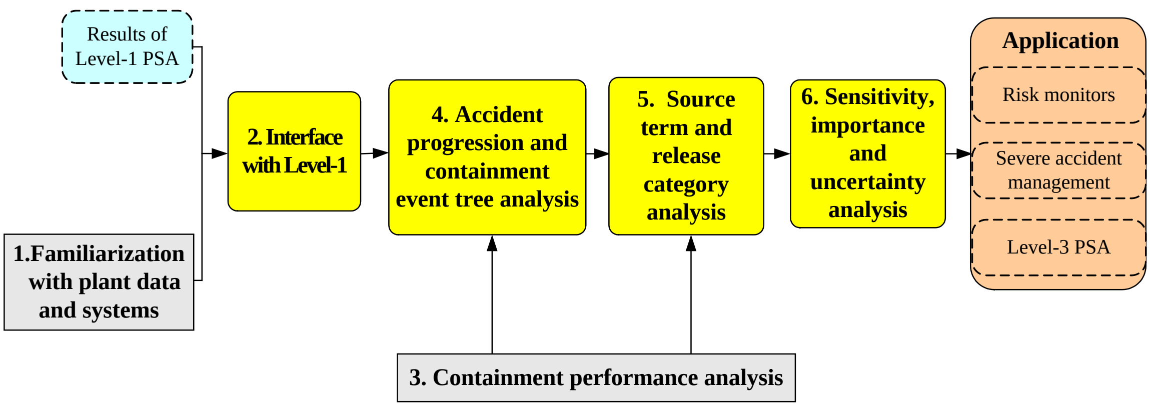

Fig.1 Research framework of Level-2 PSA.

progression of the accident;
- The availability of the containment protection and mitigation systems (such as containment sprays hydrogen mixing/ recombiners /ignitors);
- The integrity of the containment.

The main content of this task includes:
(1) Perfecting the Level-1 PSA model, and identifying the PDS attributes;
(2) Analyzing the core damage sequences, and determining the PDSs for each sequence according to the PDS attributes;
(3) Developing a tool to classify the PDSs into groups, such as a bridge tree or a general containment event tree;
(4) Forming the PDS groups using the tool developed in step 3.

2.3 Containment performance analysis

The primary factor affecting the source term in case of severe accident is the containment failure mode. Containment is the final boundary defending against the potential release of the fission product into the environment. Containment failure could be divided into two types, namely early containment failure and late containment failure in chronological order. All early containment failures will induce large radioactive source terms. In contrast, in case of late containment failure, various mechanisms can significantly help retain the fission products inside of the containment, and thereby greatly attenuate the release into the environment.

There are several containment failure modes. Among these, containment bypass failure mode will be considered in Level-1 PSA and will be included in the definition of the PDS because this mode is related to the containment condition at the moment of the core damage. For those PDSs in which the integrity of the containment is still guaranteed, high pressure and high temperature will be generated in the containments during various accident sequences, so it is necessary to carry out a containment performance analysis to determine whether the containment pressure boundary will be able to withstand these (and other) loads, and to identify if the containment has failed or not. Because challenges to containment integrity can take many forms, the analysis of containment performance limits must address several topics. Typically, the challenges to the containment can be classified under the following points:
- Slow overpressure: steady build-up of heat and gases in the containment atmosphere;
- Rapid overpressure: steam explosions, hydrogen burns, DCH;
- Underpressure: condensing of steam in absence of incondensable gases;
- Containment bypass: the creep rupture of steam generates tube;
- Overheating: degradation of containment systems and structures through elevated temperatures;
- Basemat penetration: core debris melting through the basemat of the reactor building;
- Missile generation, especially from in-vessel steam explosion or catastrophic RPV failure.

The main research contents of this task include:
(1) Conducting containment performance analysis under severe accidents;
(2) Analyzing the containment isolation failure using a fault tree;
(3) Developing a model for the analysis of containment bypass failures.

2.4 Severe accident progression and containment event tree analysis

The deterministic analysis of reactor and containment behavior during given accident sequences represents the principal basis for CET (containment event tree) analysis and quantification in a Level 2 PSA. Such analyses provide a plant specific technical basis for distinguishing the individual event branch probabilities based on the phenomena involved, and can be used to determine the success criteria for the CET branches. A Level-2 PSA requires the analysis of complex physical and chemical processes, the interaction between them, and their impact on the containment boundary. The phenomena to be considered in the course of the accident after the onset of core degradation can be grouped into two categories:
(1) Phenomena associated with the thermal-hydraulics of the accident progression and the associated containment response. These phenomena range from hydrogen generation and core-material relocation during the in-vessel phase to

containment failure due to loads generated by the core destruction process.

(2) Phenomena associated with the chemical processes affecting a) the release and composition of the radionuclides during the accident; and b) the transport of the radioactive material held in the fuel through the containment and into the environment.

Computer codes that address the entire spectrum of processes include MAAP, MELCOR, ESCADRE and THALES-2. Consequently, these codes provide an integrated framework for evaluating the timing of key accident events, thermodynamic histories of the reactor coolant system, core and containment, and corresponding estimates of fission product release and transport.

The primary function of a probabilistic model for evaluating containment performance is to provide a structured framework for organizing and displaying the alternative accident progressions that may evolve from a given core damage sequence or a plant damage state. This framework generally takes the form of containment event trees. These logic structures are the backbone of the Level 2 PSA model. The CET structure and nodal questions must address all of the issues relevant to the progression, containment response/failure, and source terms of severe accidents. This includes the important time phases of the progression of severe accidents, the consistency in the treatment of severe accident events from one time frame to another, and the interdependencies of phenomena.

The nodes in the CET follow the chronology of the accident's progression from the core damage through failure of the reactor pressure vessel (RPV) to the failure of the containment in the short term or in the long term. The time frames, which are defined to mark the important stages of the progression of a severe accident and the times of major changes in the fission product release, typically include the following:

− From the occurrence of the initiating event up to the start of core damage;
− From the start of core damage but before the failure of the RPV;
− Immediately following the failure of the RPV;
− In the longer term when there is molten core material outside the RPV.

The containment event tree nodes are usually a set of questions that relate to whether particular phenomena occur in each of the time frames addressed in the analysis, whether any systems credited in the Level-1 PSA have been recovered, whether severe accident management actions have been carried out, and whether failure or bypass of the containment has occurred. Hence, an adequate number of time frames and nodes need to be defined to allow all the significant phenomena that are relevant for each time frame to be addressed.

The next stage of the Level-2 PSA is to quantify the analysis to determine the frequency of the various sequences identified in the containment event trees. The data required for this step are the frequencies of the PDSs, which are derived in the Level-1 PSA, and the conditional probabilities of the event tree branch points. The quantification of the event trees needs to be supported by information derived from several sources, including severe accident analysis, containment performance analysis, and fission product release and transport analysis. The quantification of the event trees also needs to take account of the interdependencies between the nodes in the event trees. These can arise due to dependencies between the support systems, the phenomena that could occur in successive time frames and between human actions when carrying out severe accident management actions.

The development and the quantification of the CETs require a large number of plant and containment states to be handled. For the development of the Level-2 PSA model, the same software model is used as for the Level-1 model (such as for the definition of the risk spectrum). One of the advantages of this state of fact is to make the development of Level-1/Level-2 PSA integration convenient. The end states of the containment event trees define the sequence of events and the final state of the integrity of the containment. It is necessary to group the end states of CET into release categories for a large quantity of sequences. The grouping of the end states is further discussed in the section on source term and release category analysis.

The contents of this task include:

(1) Analyzing the progression of severe accidents;
(2) Developing methods of Human Reliability Analysis (HRA) in the eventuality of severe accidents;
(3) Evaluating the Human Error Probabilities (HEPs) according to three time periods (core damage, RPV damage, containment damage);
(4) Analyzing the impact of environmental changes on human operation;
(5) Developing a containment event tree;
(6) Identifying the typical accident sequences depending on the result of level 1 PSA and the accident process;
(7) Quantifying the containment event tree.

2.5 Source term and release category analysis

Source term (ST) and release category (RC) analysis evaluate the fission products released into the environment. It serves as input to the Level-3 PSA for the purpose of identifying the risks regarding public health and economic consequences. Source terms determine the quantity of radioactive material that is released from the plant into the environment. According to the scope of the PSA, several additional characteristics of the release may be defined such as the timeframe of the severe accident in which the release begins, the reactor vessel pressure during the core damage, the modes/mechanisms of leakages from containment, the specialized equipment providing mechanisms to contain radioactive material. Since the CETs have a large number of end states, the latter shall be grouped for practical reasons into release categories. The source term analysis is then carried out for the release categories. In this manner, it is necessary to select the typical accident sequences first and perform source term analysis with serious accidents process. Then, one should develop a rapid source term analysis program to conduct source term analysis on the other accident sequences, and finally classify the CET end states in the Release Categories so as to provide the interface to Level-2 PSA and Level-3 PSA.

The CETs have a large number of end states, each of which represent a sequence of events that has occurred following core damage. However, many of the CET end states are identical or similar in terms of phenomena that have occurred and the resulting release of radioactive material to the environment. These similar end states should be grouped or ‘binned’ together to reduce the number of distinct accident progressions requiring deterministic source term analysis. These attributes are used to define the release categories.

- Containment bypass versus no bypass
- Time frame in which the containment failure occurs
- Containment failure category
- Melt retained in-vessel
- Molten Core Concrete
- Interaction occurs
- Melt flooded ex-vessel (covered by water)
- Source term mitigated by sprays or scrubbing (for bypass sequences)

Then, it is possible to classify the CET End States into Release Categories. The categorization scheme is usually comprised of two distinct steps. The first groups the CET end-points on the basis of similar source term phenomena to form source term categories (STCs), while the second groups STCs on the basis of similar environmental consequences to form release categories. The allocation of STCs to RCs is based on the potential of each source term to cause adverse effects. The CET end points are categorized according to a number of attributes related to fission product release, retention and transport mechanisms through each of the major barriers into the environment. The purpose of this categorization (also referred to as source term binning) is to allow practical source term analysis to be performed for each predefined RC. The key attributes include:

- The timing of the release;
- The status of the containment (that is, whether containment isolation has occurred, whether containment failure has occurred giving rise to enhanced leakage or a large leakage area, and whether molten core material released from the reactor pressure vessel is challenging the integrity of the basemat);
- The way the release is occurring (such as high pressure melt ejection, dry core concrete interaction, and core concrete interactions from submerged corium);

- The fission product removal mechanisms (such as containment sprays, or retention in the secondary containment or reactor building);
- The pressure suppression pool (for boiling water reactors).

If the level-2 PSA is to be taken into a level-3 PSA, additional attributes may need to be defined. They include the height of the release, the location of release, the energy of the release, and the release duration.

Each CET end states within a particular bin is expected to have similar radiological release characteristics and off-site consequences for the source term analysis carried out for the group to characterize the entire set of CET end states within the group and thereby reduce the amount of source term analysis to be performed. By summing the frequency of all the CET end states assigned to the RC, we can get the frequency of the RC.

Plant specific source term analysis should be carried out to determine the magnitude and attributes of the source term for each of the release categories. This should be done using an integral code (such as, MAAP) capable of modeling the integrated behavior of severe accident phenomena – that is, simultaneously calculating reactor thermal-hydraulic response, core heat-up, fuel damage and material relocation, containment response, radioactive material release from fuel, and radioactive aerosol and vapor transport through the RCS and containment. In the source term analysis, the fission products need to be grouped by chemical and physical characteristics in order to deal with them.

Source term analysis needs to model the whole process which affects the release and transportation of the fission products in the containment and associated buildings. It includes:

- The release of fission products in the vessel
- The retention in the reactor coolant system
- The release ex-vessel
- The retention in the containment and associated buildings

The calculation and plant model should determine the spatial distribution of radionuclide species within the reactor coolant circuit and containment as well as the quantity released into the environment; and represent radioactive source terms in the form of one or more initial share of core reserves.

After the Level-2 PSA Model has been developed, the Level-1/Level-2 PSA integration model, which could directly calculate the frequency and the release of the source term attributes, should be implemented.

Hence this part of the process involves:

(1) Defining the release categories;
(2) Grouping of the CET end states into the release categories;
(3) Carrying out the source term analysis for each of the release categories;
(4) Developing the Level-1/Level-2 PSA integration model.

2.6 Sensitivity, importance, and uncertainty analysis

The Level-2 PSA requires identifying the dominant sources of uncertainty in the analysis and quantitatively characterizing the effects of these uncertainties on the baseline (point estimate) results. This is typically accomplished by two methods: Sensitivity analysis and Uncertainty analysis. Sensitivity analysis is a useful tool to guide the selection of sources of major uncertainty for the containment event tree. Sensitivity analysis is used to measure the extent to which results would change if alternative models, hypotheses or values of input parameters are selected. Uncertainty analysis examines a range of alternative models or parameter values, assigns each model/value a probability and generates a distribution of results, within which the baseline results represent one possible outcome.

The overall objective of sensitivity analysis is to show the potential impact of important assumptions and uncertainties on the results. The objective of importance analysis is to determine the importance of components and systems. The overall objective of uncertainty analysis is to assess the uncertainty in the output of PSA (such as the PDS frequency, the RC frequency or the final risk).

In order to perform uncertainty analysis on the source term, it is useful to develop a process based on either the Monte Carlo random sampling or the Latin hypercube sampling methods, which are widely used nowadays.

This involves the following tasks:
(1) sensitivity analysis on the source term;
(2) importance analysis on the source term;
(3) development of source term uncertainty analysis code;
(4) uncertainty analysis on the source term.

2.7 Outcome of Level-2 PSA

The outcomes of Level-2 PSA can be listed as follows:
(1) Identify vulnerabilities in the plant protection and risk mitigation mechanisms, take targeted preventive measures, improve the design of plants, raise the level of safe operation and reduce the risk induced by the plant, protect the health and security of the plant's staff and of people living nearby;
(2) Provide solutions to prevent and mitigate severe accidents through management measures. In order to prevent severe accidents or mitigate their consequences, many measures are used, such as preventing failures of the vessel or the containment, controlling the spread and release of radioactive material. The Level-2 PSA can be used to determine the effectiveness of the severe accident management measures;
(3) After developing the Level-2 PSA model, nuclear power plant risk monitoring and management procedures can be implemented. They could be used to analyze the events taking place in the plant, then classify the event sequences into PDS, then into RC, and then analyze the rhythm of changes in RC frequency and containment failure probability under a specific event. The source term and frequencies are the basis of emergency planning. According to the result of Level-2 PSA, the reference scenario of the emergency planning can be determined and implemented, including the study of potential choices for the emergency planning zone;
(4) The Level-2 PSA provides input to the Level-3 PSA. The Level 3 PSA models the consequences of a release of radioactive material into the environment and provides an estimate of the public health and other social risks such as the contamination of land or food.

3 Conclusion

This study demonstrates the necessity of performing Level-2 PSA in China through the survey of related research on the Level-2 PSA. Then, we exposed the technical route and the research contents of each major technology element related to Level-2 PSA. This study just is an introductory work for the development of a detailed Level-2 PSA model. It can be used as reference by other researcher. Based on the study, we could conclude that the major elements of Level-2 PSA are the following:
(1) Familiarization with plant data and systems,
(2) Interface with Level-1,
(3) Containment performance analysis,
(4) Accident progression and containment event tree analysis,
(5) Source term and release category analysis, sensitivity,
(6) Sensitivity importance and uncertainty analysis,
(7) Application of Level-2 PSA.

References

[1] UNITED STATES NUCLEAR REGULATORY COMMISSION: Severe Accident Risks, An Assessment for Five US Nuclear Power Plants, Rep. NUREG-1150[R]. USNRC, Washington D.C. 1990.
[2] INTERNATIONAL ATOMIC ENERGY AGENCY: Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants[R], IAEA Specific Safety Guide No. SSG-4, 2010.
[3] INTERNATIONAL ATOMIC ENERGY AGENCY: Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2): Accident Progression, Containment Analysis and Estimation of Accident Source Terms; IAEA Safety Series No 50-P-8, 1995.
[4] LIU, T.: Suggestions for the Development of Level-2 PSA in China[C]. The 1st PSA Technology Forum, 2008. [in Chinese]
[5] QIAO, M., QIU, Z., MEI, Q., and LIU, H.: Nuclear Power Station Level-2 PSA-the Calculations and Analysis of the Source Terms and LERF. The 1st PSA Technology Forum, 2008.[in Chinese],
[6] OECD NUCLEAR ENERGY AGENCY: Level 2 PSA methodology and severe accident management [R]. Rep. NEA/CSNI/R(97)11, OECD/NEA, Paris, 1997.
[7] OECD NUCLEAR ENERGY AGENCY: Recent Developments in level 2 PSA and severe accident management[R]. OECD/NEA, 2007.