SAFETY ASSESSMENT OF KOREAN NUCLEAR FACILITIES: CURRENT STATUS AND FUTURE

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This paper introduces the development of safety assessment technology in Korea, focusing on the activities of the Korea Atomic Energy Research Institute in the areas of system thermal hydraulics, severe accidents and probabilistic safety assessment. In the 1970s and 1980s, safety analysis codes and methodologies were introduced from the United States, France, Canada and other developed countries along with technology related to the construction and operation of nuclear power plants. The main focus was on understanding and utilizing computer codes that were sourced from abroad up to the early 1990s, when efforts to develop domestic safety analysis codes and methodologies became active. Remarkable achievements have been made over the last 15 years in the development and application of safety analysis technologies. In addition, significant experimental work has been performed to verify the safety characteristics of reactors and fuels as well as to support the development and validation of analysis methods.

KEYWORDS : Nuclear Power Plant, Safety Assessment, Safety Analysis, Thermal Hydraulics, Severe Accident, Probabilistic Safety Assessment, Risk-Informed Approach

1. INTRODUCTION

According to the International Atomic Energy Agency (IAEA) [1,2], safety assessment for nuclear power plants (NPP) refers to the following: (a) an assessment of all aspects of a practice that are relevant to protection and safety (siting, design and operation of the facility); (b) analysis to predict the performance of an overall system and its impact, where the performance measure is the radiological impact or some other global measure of the impact on safety; and (c) the systematic process that is carried out throughout the design process to ensure that all of the relevant safety requirements are met by the proposed (or actual) design. Safety analysis is a major part of the safety assessment; it is an analytical evaluation that is usually performed by means of computer codes, through which it is demonstrated how safety requirements are met for a broad range of operating conditions and various initiating events [2].

Safety assessments and/or analyses incorporate deterministic and probabilistic approaches, which complement each other. Deterministic safety analyses are usually performed for pre-selected sets of design basis accidents. They should be generally conservative; it this can be achieved by either a conventional conservative analysis or a new best-estimate analysis with uncertainty quantification. A probabilistic safety assessment (PSA) should handle all potentially important events as realistically as possible. Realistic deterministic analyses should be a part of the PSA procedure in order to determine the plant status for given scenarios.

Various computer codes are used in the process of deterministic safety analyses to analyze the reactor physics, fuel behavior, thermal hydraulics in systems and components, containment behavior, and structural integrity [3]. Different categories of computer codes are used in a PSA to evaluate the risk-related quantities.

Significant efforts have been made in Korea to establish safety assessment code systems, related technology, and experimental infrastructure. The relevant technical capability has reached a very high level so that technology transfer is frequently requested by developing countries.

This paper summarizes the major activities and achievements in the development and applications of safety assessment technology in Korea, focusing on the activities of the Korea Atomic Energy Research Institute (KAERI) in the areas of system thermal hydraulics, severe accidents and probabilistic safety assessments.

2. THERMAL-HYDRAULIC SAFETY ASSESSMENT

2.1 Historical Overview

Thermal-hydraulic safety assessments represent the most conventional and crucial step in the design and licensing of NPPs. In particular, the effectiveness of the emergency core-cooling systems (ECCS) against loss-of-coolant accidents (LOCA) has been one of the major topics of safety assessment for light water reactors. Thermal-hydraulic safety is assessed and verified through a combination of analytical and experimental methods, and these have resulted in various safety analysis codes and experimental facilities.

Several integral and separate effect test facilities have been constructed and operated since 1960s, providing very useful information and data on the accident behavior of light water reactors. They include Semiscale, LOFT (Loss-of-Fluid-Test), FLECHT (Full-Length Emergency Cooling Heat Transfer Tests), and TLTA (Two Loop Test Apparatus) in the United States; LSTF (Large-Scale Test Facility), CCTF (Cylindrical Core Test Facility) and SCTF (Slab Core Test Facility) in Japan; UPTF (Upper Plenum Test Facility) and PKL in Germany; and BETHSY in France [4, 5]. Several categories of thermal-hydraulic analysis codes have also been developed by industry, research organizations and regulatory organizations. Typical examples are FLASH, RELAP, TRAC, COBRA-TRAC, RETRAN, ATHLET, CATHARE, CATHENA and APROS [3].

In Korea, systematic system safety analyses started in the early 1980s, when safety analyses of Kori Unit 1 and Wolsung Unit 1 were among the major safety issues. The Nuclear Safety Center at KAERI established a safety analysis code system based on the Water Reactor Evaluation Model (WREM) system of the U.S. Nuclear Regulatory Commission (USNRC). The WREM system consisted of the RELAP4, CONTEMPT-LT and TOODEE2 codes. Since then, several safety analysis codes have been introduced from overseas vendors, research organizations and regulatory bodies. The RELAP5 code, which was introduced from the USNRC in the late 1980s, is the most widely used system analysis code in Korea. RETRAN is also widely used in industry applications.

Code development activities became active in the 1990s; MARS [6] and TASS [7] were typical domestic codes that were developed for system safety analyses of nuclear reactors. A realistic safety analysis methodology for a LOCA has also been investigated actively by KINS, KAERI, academia and industry. In addition, extensive experimental programs have been developed to support the development of nuclear reactors, advanced fuels and safety analysis codes.

2.2 Development of Analysis Technology

2.2.1 MARS

The MARS (Multi-dimensional Analysis of Reactor Safety) code was developed starting in 1997 for realistic multi-dimensional thermal-hydraulic system analyses of light water reactor transients [6]. MARS originated from consolidation of the RELAP5/MOD3.2.1.2 (one-dimensional two-fluid modeling of two-phase flows) and COBRA-TF (three-dimensional, two-fluid, three-field modeling) codes through integration of the hydrodynamic solution schemes and the unifying various thermal-hydraulic models, equations of state and input/output features. The sources of the original codes were fully restructured using the modular data structure and a new dynamic memory allocation scheme of FORTRAN 90. In addition, a new multidimensional fluid model was developed and implemented onto the RELAP5 system analysis module in an effort to overcome several limitations of the COBRA-TF 3-dimensional (3D) vessel module.

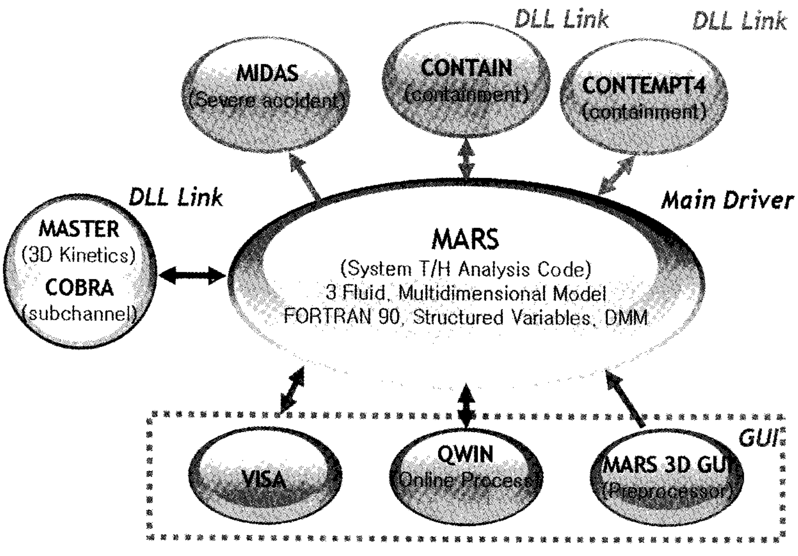

Fig.1. MARS Code and Link with Other Systems

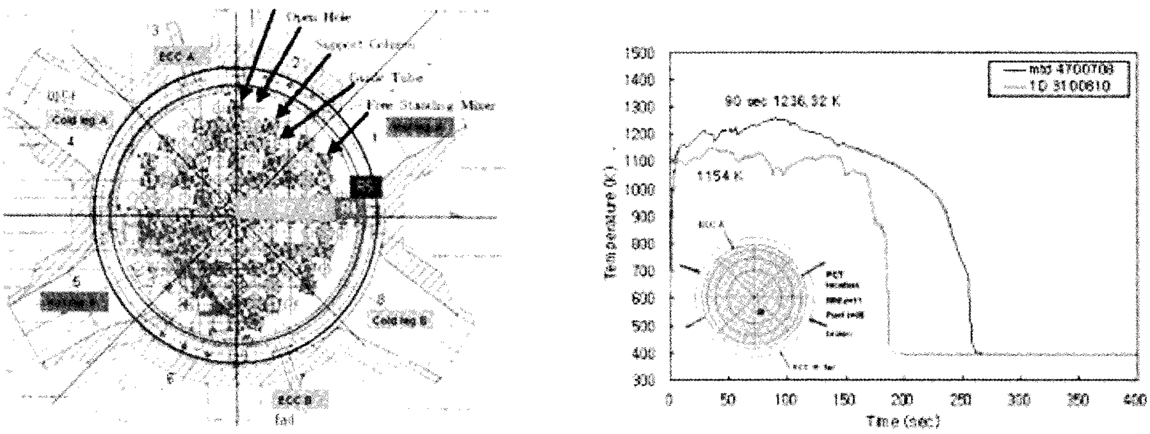

Fig. 2. MARS 3-D Analysis for LBLOCA of Kori-Unit 1 (UPI Plant) Compared to the 1-D Calculation [8]

MARS runs on a Windows platform. It is currently a popular multi-dimensional thermal-hydraulic tool used in analyses of reactor transients and experiment facility simulations and for various safety research purposes. As shown in Fig. 1, MARS can also be connected, by means of a dynamic linkage using dynamic-link libraries (DLL), to other codes such as a 3D kinetics code and containment analysis codes [8]. A simple graphical user interface (GUI) is provided for program execution and to display the calculation results online. To improve and extend the MARS modeling capability, various thermal-hydraulic properties and correlations have been added and incorporated into the code. With these improvements, the capability of MARS has been extended for an application to not only light and heavy reactors but also to research and advanced reactors such as HANARO, Sodium-Cooled, Pb-Bi-Cooled, and Gas-Cooled reactors [9].

The MARS code is now being used successfully in many key areas of safety research. The code is now transferred to the Korea Institute of Nuclear Safety (KINS) for use as an auditing analysis tool. Further systematic independent code qualification processes are on-going, and CANDU thermal hydraulic models are being implemented in an effort to create a unified auditing tool for all operational power plants in Korea [10].

2.2.2 TASS

TASS (Transients and Setpoints Simulation) was first developed in the early 1990s as an interactive and faster-than-real-time code for the Nuclear Steam Supply System transient simulation of both ABB-CE and Westinghouse plants [9]. It was a flexible tool for PWR analysis which gave the user complete control over the entire process of simulation.

The TASS/SMR (Transient and Setpoints Simulation Small and Medium Reactor) code was developed to perform safety analyses and performance evaluations of the integral-type pressurized water reactor known as SMART [11]. The code has been verified through the use of simplified and analytical problems, existing code calculations, and experimental data dedicated to SMART [12]. Assessments against separate and integral effect test data indicate that the code predicts the overall variation of important thermal hydraulic parameters well [12]. TASS/SMR is used as a basic safety analysis tool in the design of SMART.

2.2.3 ViSA

ViSA is a user-friendly computing environment for system thermal-hydraulics codes. It was developed by KAERI [13,14]. Currently, ViSA can be used with system thermal-hydraulics codes such as RETRAN, RELAP5 and MARS. The integration of ViSA with a system thermal-hydraulics code becomes a plant-independent nuclear plant transient analyzer. The key features of ViSA can be summarized as follows:

- plant-independent, as any plant can be modeled via code inputs
- the same level of accuracy as the original codes
- various graphical representations of the output
- on-line interactive control for a simulation of various operator actions

General code users can benefit from the input handling capability, reliable control editor, and more legible output presentation of the transient analyzer. ViSA has been used successfully in the training of plant safety personnel to allow for immediate on-site system analyses for a wide range of plant transients. ViSA integrated with RELAP5 has been exported to research institutes, regulatory agencies, and universities in several countries. Recently, IAEA has started to use this tool for operator training in member countries.

2.2.4 CUPID

A component-scale thermal-hydraulic code, CUPID (Component Unstructured Program for Interfacing

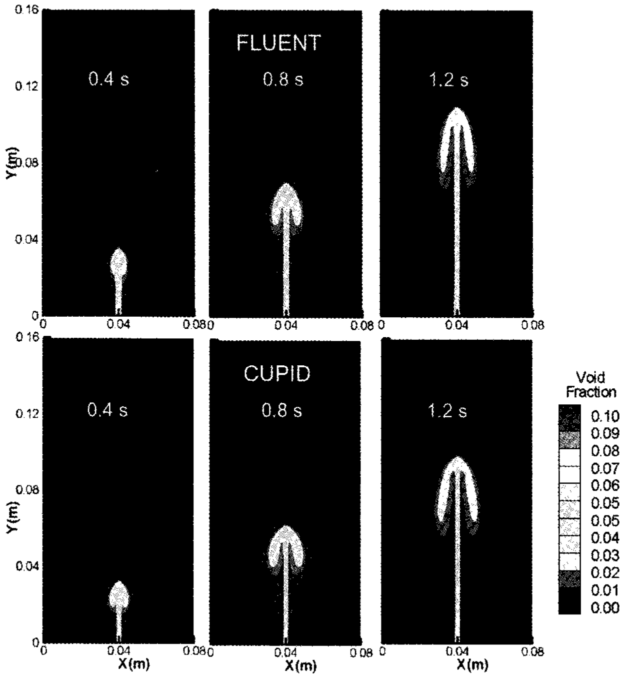

Fig. 3. Simulation of Air Injection by CUPID and FLUENT

Dynamics), was developed at KAERI starting in 2007. It can be used as either a three-dimensional computation module within a system code or a stand alone component-scale code [15]. In the CUPID code, a two-fluid three-field model is adopted for two-phase flows, and the governing equations are solved on unstructured grids, which is very useful for a flow in complicated geometries. A semi-implicit method is applied as a basic numerical scheme with some modifications for application to three-dimensional, unstructured grids. The numerical scheme has been proven to be very robust for complicated two-phase flow simulations [15, 16]. A modified SMAC and SIMLPE-based schemes have also been developed in parallel for the CUPID code.

Figure 3 illustrates a CUPID simulation of the air injection phenomena, compared to the prediction by the commercial CFD code FLUENT.

2.2.5 SPACE

The Korean nuclear industry has been using vendor codes from overseas sources for safety analyses of NPPs. The SPACE (Safety and Performance Analysis Code) code development program was launched in September of 2006 under the auspices of the Ministry of Knowledge Economy (MKE) to overcome the technical limitations of overseas codes and to secure its own safety analysis code system for pressurized water reactors [17].

While SPACE will mainly be used in safety analyses, the code with its best-estimate capabilities will be able to cover performance analyses as well. The code solves two-fluid, three-field governing equations in one- or three-dimensional geometries. It will incorporate the component models required for modeling a PWR, such as reactor coolant pump or a safety injection tank. The programming language used in the new code is C++, which is more familiar to the new generation of engineers.

According to the revised program, the code will be developed and analyzed in three phases:

- Phase 1 (Oct. 2006 ~ Mar. 2010): Development of the integrated safety analysis code, SPACE
- Phase 2 (Apr. 2010 ~ Dec. 2012): Code verification, validation and pilot applications to an actual plant
- Phase 3 (Jan. 2012 ~ Dec. 2015): Further validation and licensing of the code, application to actual projects

To accomplish these rigorous goals, several research and industrial organizations including KAERI, KHNP (Korea Hydro and Nuclear Power Co., Ltd.), KOPEC (Korea Power Engineering Company, Inc.), KNF (Korea Nuclear Fuel), and KEPRI (Korea Electric Power Research Institute) are participating in the SPACE code development project. The pilot code release and validation steps are expected in 2009 and 2012, respectively.

2.3 Experimental Support

An extensive experimental program has been operated to support the development of advanced reactor systems and fuels and to develop and validate advanced analysis codes. Major experimental works performed thus far include the following:

- Integral effect tests with the ATLAS facility to investigate the overall thermal-hydraulic behavior in reactor systems and to validate the safety analysis codes
- Separate effect tests to investigate and verify the performance of new design features of APR1400
- Separate effect tests for multi-dimensional effects in a reactor vessel, fuel assemblies, water pool and piping systems
- Heat transfer tests including the critical heat flux (CHF), post-CHF heat transfer, and reflood heat transfer
- Experiments for interfacial area transport modeling covering non-heating and heating conditions

Detailed information can be found in the literature including Refs. [18-25].

2.4 Applications

Domestic and foreign safety analysis codes have been widely used in safety analyses of new and operating reactors and in the development of advanced reactor systems. Among application-related activities in Korea, two activities are introduced briefly in this section: (a) the realistic evaluation methodology (REM) for LOCAs and (b) NPP simulators.

The realistic evaluation of LOCAs, which refers to the best-estimate calculation plus uncertainty quantification, has been an important subject in deterministic safety analyses for the past twenty years, since USNRC first allowed the use of this methodology in addition to the conventional conservative approach [26-28]. In the early 1990s, the KINS performed a research project to test the feasibility of REM on large- and small-break LOCAs in collaboration with the Korea Advanced Institute of Science and Technology and Seoul National University, as illustrated in reference [29]. The KINS REM evolved until recently by incorporating new findings and experiences; however, no published articles exist in archival literature. KAERI also developed a simpler methodology and applied this to an operating PWR [30]. Modified RELAP5/MOD3 codes were used in the REM of KINS and KAERI.

The Korean industry also developed its own REM for large-break LOCA analyses. This is known as the KEPRI Realistic Evaluation Methodology (KREM) [31]. It was demonstrated on Kori 3/4, which are three-loop PWRs, using a combined code of CONTEMPT4/MOD5 and RELAP5/MOD3.1. KREM follows the philosophy of CSAU [28] with a small number of improvements and adopts nonparametric statistical tolerance limits according to the Wilks formula. KREM has been applied to safety analyses of several operating plants for periodic safety reviews and power up-ratings. It is also being applied in the safety analysis of the new plant APR1400, with some improvements.

The development of NPP simulators has been lead by KEPRI, with the participation of KAERI. Advanced thermal-hydraulic models of best-estimate codes have been used, as discussed in Ref. [32]. The visual environment for best-estimate codes can also be used as a software simulator for operator training. An effort was made to develop a web-based nuclear simulator based on ViSA [33].

2.5 Future Prospects

The research and development related to thermal-hydraulic analysis codes and applications will continue to be important in the future. Major activities will include the following:

- Establishment of a regulatory auditing code system termed KINS-RETAS (Reactor Thermal-hydraulic Analysis System) using MARS as a pivotal code
- Development, verification, validation, application and licensing of the industrial safety analysis code SPACE
- Development of advanced analysis technology for more detailed analysis of the thermal hydraulic behavior in reactor systems
- Further enhancement and application of computational fluid or multi-fluid dynamics technology for reactor design and safety assessments

Extensive experimental work will also be required to support code development and validation.

It is expected that an advanced safety assessment system based on a multi-scale and multi-physics approach can be established in the late 2010s. The code should cover all of the relevant scales, including micro-scale (turbulence), meso-scale (component) and macro-scale (system).

3. SEVERE ACCIDENT ANALYSIS

3.1 Historical Overview

The occurrence of severe accidents became a pertinent issue in nuclear safety after the Three Mile Island (TMI) accident in 1979. Considering the wide range of phenomena and the need to handle molten nuclear materials, severe accident research has been performed mainly within the framework of international cooperation programs. Korea joined in the CSARP (Cooperative Severe Accident Research Program) of USNRC [34] in 1982, the PHEBUS Project of IPSN [35] in 1992, and the OECD/NEA RASPLAV Project [36] in 1996. Korea also participates in many other international projects.

A number of severe accident analysis programs were introduced through those programs, e.g., STCP (Source Term Code Package), SCDAP/RELAP5, CONTAIN, MELCORE, and ICARE. The industrial severe accident codes of MAAP and GOTHIC were also introduced; they have been used widely, mainly by the industry. Severe accident analysis codes have been assessed through international programs including MCAP (MELCORE Code Assessment Program) and the OECD International Standard Problem (ISP) or by benchmark problem exercises. The MELCOR and MAAP codes are widely used for regulatory and industrial purposes, respectively.

Severe accident research within the framework of government's mid- and long-term nuclear R&D programs seeks to understand the important relevant phenomena, the development of analysis technology, and the development of accident management strategies for the prevention and mitigation of potential risks. The research program includes experimental efforts along with the development of phenomena specific models. The main focus has been on the molten corium coolability inside and outside the reactor vessel and hydrogen combustion in the containment area. Significant experimental achievement and useful analytical tools have been developed during the last 15 years. Recently, several advanced codes for specific applications (e.g., GASFLOW and MC3D) were also introduced through international cooperation.

3.2 Development of Analysis Technology

The development of severe accident analysis codes has not been very active compared to the development of thermal-hydraulic analysis codes. There are many reasons for this, including the complexity of the phenomena involved, limitations in human and financial resources, and the less commercial characteristics of foreign codes. Three domestic severe accident analysis codes are discussed here:

- The ISAAC code for severe accident analyses of CANDU reactors
- MIDAS for integrated severe accident analyses
- TRACER for fuel-coolant interaction

3.2.1 ISAAC [37-39]

The ISAAC (Integrated Severe Accident Analysis code for CANDU plants) code was initially developed at KAERI for Wolsong units 1/2/3/4 PSA. It is being improved for the severe accident management of CANDU-6 reactors. A thermal-hydraulic module of the ISAAC code was imported from the MAAP/PWR computer code, which was developed by EPRI for pressurized water reactors. As the CANDU-6 design characteristics differ from typical PWRS, CANDU-6-specific features were modeled and added to the ISAAC code based on the current understandings of the unique accident progressions for the CANDU configuration. Currently, version 4 of the code is under development in KAERI.

3.2.2 MIDAS [9,40]

MIDAS (Multi-purpose IntegrateD Assessment code for Severe accidents) is an integrated severe accident code that analyzes and efficiently simulates severe accidents in NPPs. The code is based on the open source of MELCOR 1.8.5, which is distributed by USNRC. The data structure was restructured and new models such as a gap-cooling model were incorporated. In MELCOR 1.8.5, which uses pointer variables, it was difficult to create an addition of new models or a modification of existing models. The code modernization in the MIDAS code not only enables easy improvement or addition of models from a developer side, but also provides the friendliness from a user side. The MIDAS code can be adjusted to most NPPs in operation and under development including, SMART and Very High Temperature Gas-Cooled Reactors.

3.2.3 TRACER [41,42]

TRACER is a mechanistic code that predicts vapor explosion propagation during fuel-coolant interactions. It enables a multi-dimensional analysis of the behavior of four fluid phases: melt drop, fragmented debris, liquid coolant, and vapor coolant. It is being used in research and commercial applications in Korea. It has also been used in international benchmark problems related to steam explosions [43].

3.3 Experimental Support

There have been significant experimental works related to severe accident phenomena, as summarized in Ref. [44]. They include the simulation of naturally arrested thermal attacks in the vessel (SONATA) [45], tests for real molten corium interactions with water (TROI) [46], tests related to the in-vessel retention of molten corium by external reactor vessel cooling (IVR-ERVC) [47], and tests related to the hydrogen behavior inside containment areas [48].

3.4 Applications

Applications of severe accident analysis are done in several ways, including:

- analysis of the hydrogen distribution using lumped parameter codes and CFD codes [49]

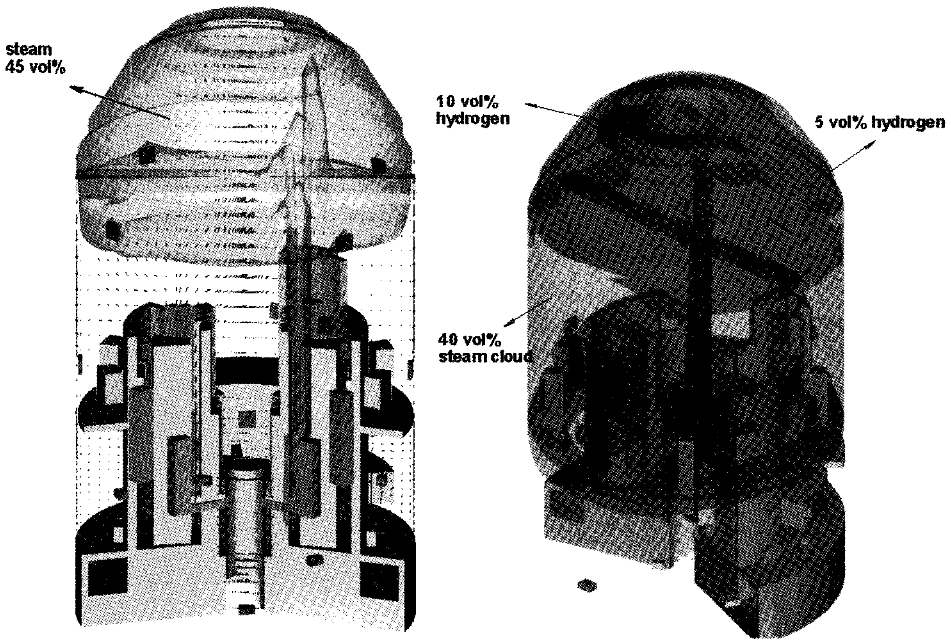

Fig. 4. Contour of the Steam and Hydrogen Concentration in the LOCA Sequence
(left: T=4000 sec., right: T=6200 sec.)

- analysis of the molten corium behavior inside a reactor vessel and the cooling performance on the reactor vessel, which is related to IVR-ERVC [50]
- analysis of steam explosions inside and outside the reactor vessel
- development of severe accident management procedures

Several categories of codes are used for these applications. These include integrated or specific phenomena analysis codes, lumped parameter or CFD codes, as well as domestic or foreign codes. Figure 4 illustrates the typical hydrogen distributions inside the containment obtained from a GASFLOW analysis of the LOCA sequence in APR1400 [49].

3.5 Future Prospect

A strategy for the development of severe accident analysis technology is presently under active discussion. As important severe accident scenarios are considered in designing new reactors, a comprehensive severe accident analysis code package covering both integrated and specific phenomena-related codes should be secured for the industry. The regulatory organization should also establish an appropriate severe accident code system for audit calculations.

However, code development efforts will be focused on several selected areas. Fuel-coolant interaction appears to be the most promising area for domestic code development. Incorporation of domestic plant-models into existing codes is also important from the viewpoint of practical applications. The experimental work should be closely connected with the development of analysis technology.

4. PROBABILISTIC SAFETY ASSESSMENT

4.1 Historical Overview

The first probabilistic safety assessment (PSA) for NPPs, termed the Reactor Safety Study or WASH-1400, was issued in 1975 [51]. After it became known that the TMI accident scenario was predicted by WASH-1400, the USNRC required that all utilities shall examine the vulnerability of their NPPs using PSA as a TMI action requirement. Through several years' effort, all PSA modeling and insight pertaining to U.S. NPPs were gathered, and the USNRC sought to use the valuable PSA insights and models not only to identify design vulnerabilities but also to improve the operation and maintenance.

The PSA policy statement [52] issued by the USNRC in 1995 states that "the use of PSA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PSA methods and data..." The PSA policy statement was the starting point of Risk-informed Regulation (RIR) in the U.S.A. The move toward RIR was a significant transition for the USNRC, driven by the premise that reductions in unneeded expenditures of resources on matters that are not significant in terms of safety are required so that nuclear power can be made safer [53].

Although the first Korean PSA was started before 1994, the two policy statements discussed below played a major role in introducing and establishing the PSA in Korea.

In 1994, the policy statement on the nuclear safety was promulgated, requiring that the utility should perform integrated safety assessments for NPPs using the PSA and that the regulator should implement a risk-informed and reasonable regulation considering the cost benefit. In 2001, the NSC (Nuclear Safety Commission) put into effect the Policy on Severe Accidents, in which the utility was requested to complete PSAs for all operating NPPs by 2006, to secure an ability to mitigate a severe accident, to establish a severe accident management plan, and to evaluate and monitor the risk level. Based on these two policy statements, in 2003 the utility announced an implementation plan for the PSA and the severe accident management. After confirming the PSA substructure enhanced by the utility, in 2006 the regulator announced a Korean RIR implementation plan [54]. The utility's implementation plan for risk-informed application was established according to the regulator's RIR implementation plan in 2006 [55].

Thus, the two policy statements deeply affected the direction of Korean regulations. Korean RIR & application started from these policy statements.

4.2 Development of Analysis Technology

Computer codes are required to develop the PSA models, and a reliability data base of the components is required to evaluate the PSA models. Thus far, a considerable amount of PSA software has been developed around the world. However, recently, only a small number of PSA codes are used widely; these were developed in the U.S.A., Sweden and Korea. Korea has shown success in the PSA software area. Korea has used PSA software developed domestically and has exported PSA software abroad. A brief history of Korean PSA software industry is shown in Fig. 5.

In Korea, the KAERI instigated the development of the PSA software in an effort to perform PSAs as well as to test and implement the new methodologies of the PSA. The first Korean PSA software, KIRAP, was put into development in 1987 by KAERI [56]. The level-2 PSA package CONPAS, including an event tree editor, was developed in 1996 [57]. The KIRAP/CONPAS package has been successfully used in Korea with the Yonggwang 3&4 PSA, the Ulchin 3&4 PSA, the Wolsong 2-4 PSA, and others. The KOPEC developed a cut-set generation engine termed FORTE [58] and PSA software termed SAREX. SAREX and FORTE have been used with the Kori 3&4 PSA, the Kori 2 PSA and the Ulchin 1&2 PSA.

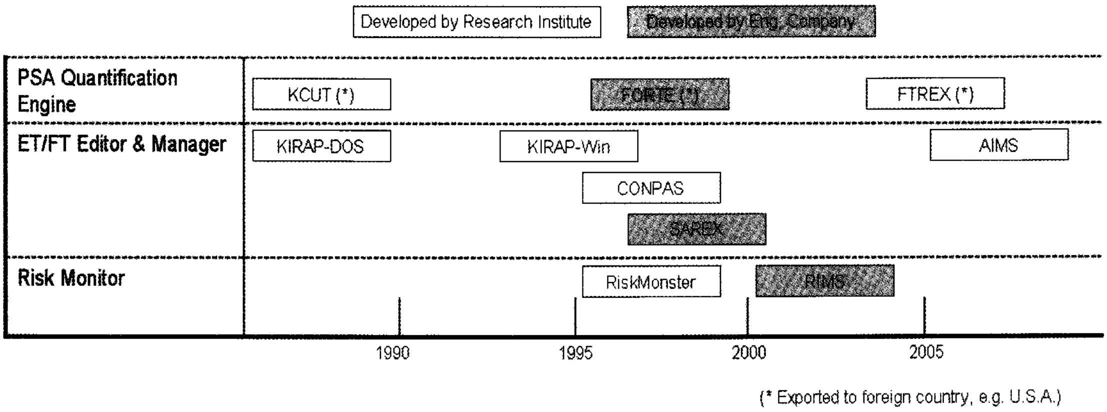

Fig. 5. History of Korean PSA Software

Recently, KAERI developed a cut-set generation engine known as FTREX [59]. As of 2009, FTREX is the fastest cut-set generation engine in the world. FTREX is used in more than 40% of sites in the U.S.A. thus far.

To evaluate plant-specific risk, the reliability data of a specific NPP is needed. After the Yonggwang 3&4 PSA was submitted to KINS, KINS requested the formulation of a plant-specific reliability database. To cope with such a request, KAERI developed a plant-specific reliability database that it termed KIND (Korea Integrated Nuclear reliability Database) [60]. Recently, KHNP developed the Plant Reliability data Information System (PRINS) [61], which is based on the architecture of KIND. KHNP is continuing in its formulation of the reliability database using PRINS for all NPPs.

4.3 Applications

4.3.1 PSA for a Construction Permit and Operating License

The first PSA application in Korea was the reliability analysis for the auxiliary feedwater system (AFWS) of Kori 1&2. According to this study, the enhancing test procedure and operator training were recommended to improve the reliability of AFWS.

As one of the TMI action requirements, the level 1 PSA was performed to evaluate the overall safety of Kori 3&4 and Yonggwang 1&2. As Korea launched a program for the design and construction of Yonggwang 3&4 as the forerunner of the OPR-1000 (Optimized Power Reactor 1000), the PSA faced a turning point; that is, the PSA was to be performed as a part of the design process of NPPs. The PSA of Yonggwang 3&4 was originally initiated to meet the supplementary conditions for the construction permit [62]. Based on the results of the PSA, the original design was changed and the procedures were updated. For all of the NPPs of the OPR-1000 series that were constructed after Yonggwang 3&4, the level 1 and 2 PSAs were performed as a part of the design process for the licensing pertaining to the construction and operation of these plants.

Recently, there are needs for the improvement of the quality of the PSA. KAERI performed the first research project to improve the PSA quality for the Ulchin 3&4 PSA model [63], and considerable efforts, such as the updating of existing PSA models, are on-going to improve the PSA quality for the implementation of RIR. The present status of the Korean PSA is summarized in Table 1.

4.3.2 Risk-Informed Regulation and Applications

Recently, many activities are on-going related to risk-informed regulation and applications in Korea.

The surveillance test intervals (STIs) of RPS/ESFAS (Reactor Protection System/Engineering Safety Features Actuation System) described in the Technical Specification were extended for the following NPPs: Kori 3&4, Yonggwang 1&2, Ulchin 3&4, and Kori 2. The Allowed Outage Time (AOT) extension for the instrument bus inverter of the Ulchin 3&4 was approved in 2007 [64]. The containment integrated leak rate test intervals (ILRT) [65] were extended from 5 years to 10 years for Yonggwang 1&2 in 2005, for Yonggwang 3&4 & Kori 3&4 in 2006, for Ulchin 1-4 in 2007, and for Kori 2 in 2008. The RI-ISI (Risk-informed In-service Inspection) for Ulchin 3&4 has been submitted to KINS, and KINS is presently reviewing the documents [66].

A risk monitor termed RIMS (RIsk Monitoring System) was developed and installed at Kori 3&4 [67]. However, RIMS has not shown impressive results thus far, as an on-line maintenance is not utilized as yet in Korea. A risk monitor known as ORION has also been developed to assist with a safe and fast maintenance and overhaul during a refueling period [68]. In 2001, the NSC recommended the introduction of the maintenance rule in Korea. A pilot

Table 1. Present Status of Korean PSA

Unit	Scope & Purpose of PSA	Completed Date
KR-1	PSA for TMI action Plan	'02.11
	PSA update for Risk monitor	'07.05
KR-2	PSA for TMI action Plan	'03.12
	PSA update for Risk monitor	'07.06
KR-3,4	PSA for TMI action Plan	'92.08
	PSA update for Risk monitor	'03.06
YG-1,2	PSA (LEVEL I) for TMI action Plan	'92.08
	PSA update (LEVEL I, II) for Severe Accident Policy statement	'03.12
	PSA update for Risk monitor	'07.12
YG-3,4	PSA (LEVEL I) for the requirement for construction & operation permits	'94.02
	PSA update (LEVEL I, II) for Severe Accident Policy statement	'04.12
	PSA update for Risk monitor	'05.06
YG-5,6	PSA (LPSD PSA) for the requirement for construction & operation permits	'00.12
	PSA update for Risk monitor	'05.12
UC-1,2	PSA for Severe accident policy	'05.12
	PSA update for Risk monitor	'06.12
UC-3,4	PSA (LEVEL I) for the requirement for construction & operation permits	'97.1
	PSA update (LEVEL I, II) for Severe Accident Policy Statement	'04.12
	PSA update for Risk monitor	'05.06
UC-5,6	PSA (LOSD PSA) for the requirement for construction & operation permits	'02.06
	PSA update for Risk monitor	'06.06
WS-1	PSA for Severe Accident Policy statement	'03.12
	PSA update for Risk monitor	'07.02
WS-2,3,4	PSA for the requirement for construction & operation permits	'97.1
	PSA update for Risk monitor	'07.02

study for the maintenance rule was performed for Ulchin 3&4 and Kori 3&4 [69].

KINS is now testing a new regulation method known as RIPI (Risk-informed Periodic Inspection), which uses risk insight in selecting the items for a periodic inspection [70]. It is tested and updated continuously.

4.4 Future Prospects

There are many studies related to the PSA and its application in Korea. An integrated research program for the PSA has been performed by KAERI starting in 2007. The main objectives of this research project is to develop technologies (1) to solve Korea-specific issues, (2) to contribute to the solution of common issues around the world, and (3) to develop a basis for a risk/performance assessment framework for the future. Additionally, research related to the PSA of new reactors and/or non power facilities has been conducted.

Research for Korean Specific Issues

- Seismic risk is highly plant- and site-specific. Research on the ground response spectrum, a realistic seismic fragility model, and a seismic risk evaluation model are being conducted in the research program.
- Human error is one of the main contributors to the risk of NPPs. Several studies are being performed for the development of a risk/performance analysis model related to human-induced/unplanned reactor trips.

Research for World Wide Issues

- Assessing the risk impact of digital instrument and control (DI&C) systems in NPPs is a worldwide issue. It is an urgent issue in Korea, as Korea is building new

NPPs using DI&C technology for the entire system. Thus far, there is no well-established or well-accepted PSA method for DI&C systems. KAERI is performing research in this area to cope with this issue.

- Recent safety assessments emphasize best-estimates rather than conservatism. In the best-estimate approach, the handling of uncertainty is a critical issue. There are studies to reduce two types of uncertainty: (1) research to reduce the uncertainty of level 2 PSA related to severe accident phenomena [71], and (2) research to analyze the parametric uncertainty of computer code [72].

Research for the Future

- The U.S. NRC and industries pointed out that the lack of an integrated/objective framework for risk-informed decision making is the main obstacle for the extension of RIR. Research is continuing in the development of a framework for integrated assessment of risk and performance. This framework will allow better integrated/objective risk-informed decision making.
- Generation IV NPP emphasizes the use of risk at the design stage. A framework is necessary so that risk insight can be used to achieve a "built-in" concept rather than an "add-on" concept [73]. In addition, the PSA of Generation III+ and/or IV requires new PSA technologies such as assessments of passive system reliability and knowledge on the subject of new severe accident phenomena. Research for the PSA of Gen-IV and SMART facilities is on-going in Korea.
- The areas of the risk assessment are extended continuously. Research on the risk assessment of radioactive waste disposal and decommissioning will commence soon in Korea.

After the events of September 11, 2001, the physical protection of NPPs became an important issue in the nuclear industry. There is research related to the use of PSA technology in identifying the vital areas of NPPs [74].

5. CONCLUSION

In this paper, the development safety analysis technology and related applications are reviewed with a focus on the activities of KAERI. It was found that significant achievements have been made in thermal hydraulics analyses, severe accident analyses, and PSA technologies. Extensive experimental programs over the past 15 years have also strengthened the establishment of analysis technologies. Additional improvements are necessary of the safety analysis technology in each area, and the feasibility of an integrated analysis of thermal hydraulics, severe accidents, PSA, and material behavior is required.

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