New AESJ Thermal-Hydraulics Roadmap for LWR Safety Improvement and Development after Fukushima Accident

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ABSTRACT

The Atomic Energy Society of Japan (AESJ) developed a New Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap (TH-RM) for LWR Safety Improvement and Development after Fukushima Accident by thoroughly revising the 1st version (TH-RM-1) prepared in 2009 under good collaboration of utilities, vendors, universities, research institutes and technical support organizations (TSO) for regulatory body. Thereafter, the revision has been made by three sub working groups (SWGs) of “safety assessment”, “fundamental technology” and “severe accident”, by considering the lessons learned from the devastating Fukushima-Daiichi Accident. The “safety assessment” SWG pursued development of computer codes mostly for safety assessment. The “fundamental technology” SWG pursued safety improvement and risk reduction via accident management (AM) measures by referring the technical map for severe accident established by “severe accident” SWG. Phenomena and components for counter-measures and/or proper prediction are identified by going through severe accident progression in both reactor and spent-fuel pool of PWR and BWR. Twelve important technology development subjects have been identified, which include melt coolability enhancement to maintain integrity of containment vessel. Work Description Sheet was developed for each of identified and selected R&D subjects. External hazards are also considered how to cope with from thermal-hydraulic safety point of view. This paper summarizes the revised TH-RM with several examples and future perspectives.

KEYWORDS

AESJ, Thermal-Hydraulics, Safety, Technological Strategy, Roadmap, Fukushima-Daiichi Accident

1. INTRODUCTION

In the utilization of light water reactors (LWRs), thermal-hydraulics is taking key roles such as stable reactivity control in the core, transport of heat generated in the core to turbine to generate electricity and assurance of core integrity by sufficient cooling in case of accident. Because of such important roles, thermal-hydraulics should provide fundamental technical information for risk assessment and effectiveness of safety features, and boundary conditions for fuel and structure materials.

In Japan, the safety research was started by the former Japan Atomic Industrial Forum, Inc. (JAIF) as "SAFE Project" in 1963 on such thermal-hydraulics as coolant behavior and core cooling during loss-of-coolant accident (LOCA), being funded from Japanese government and industry. Practically, Hitachi, Mitsubishi Heavy Industry (MHI) and former Nippon Atomic Industry Group (NAIG, currently Toshiba) performed fundamental experimental research on the prevention of core melting and radioactive material release in relation to the effectiveness of emergency core cooling system (ECCS). Since then, thermal-hydraulics safety research has been done extensively in Japan, partly leading to ROSA (Rig-of-Safety Assessment) Program that started in 1970 concerning effectiveness of ECCS during LOCA and 2D-3D Program concerning refill and reflood of PWR core during large break LOCA in the former JAERI (Japan Atomic Energy Research Institute).

Especially after TMI-2 (Three-Mile Island Unit-2) reactor accident in 1979, significant amount of technical information relevant to reactor safety and improvement had been gained through large-scale experiments, development of safety analysis tools, development of researchers and engineers for experiments, and international research collaboration. However, there still exist much of difficulties in exactly re-producing prototypical thermal-hydraulic phenomena by using computer codes because such phenomena inherently appear within large-scale systems of LWR with reciprocal interactions among components through 3-D two-phase flows. Therefore, large-scale demonstration proof tests are still required to confirm new design, although computer codes are used to design various types of components and reactor systems. Since the LWR technology itself has been recently considered matured because of rather long-term reactor operation since 1970's, however, research and development (R&D) became not so prominent recently. The development of New High-Performance LWRs (ABWR & APWR) [1] was started in 2008 based on Nuclear Energy National Plan [2] that was established in 2006 by METI based on Framework for Nuclear Energy Policy in 2005 [3].

All these activities came to reset because of significant lessons-learned from the occurrence of severe accident (SA) at Fukushima-Daiichi Nuclear Power Station (1F accident, hereafter) on March 11, 2011. New regulation law/standards [4] were established in 2013 by the Nuclear Regulation Authority (NRA) in Japan, which is currently pursued by utilities onto most of existing LWRs to restart. In particular, influences from such low-frequency yet high consequence event as external events that include earthquake, tsunami, tornadoes, typhoons and volcanoes as well as man-made hazards are taken into account by new counter-actions based on the lessons-learned from1F accident. Defence-in-depth (DiD) was also re-considered to follow up to level 5 from level 3, by referring IAEA INSAG-10 etc., and to incorporate measures for prevention and mitigation of SA into the new regulation law/standards. The AM measures provided by 2002 are thus greatly and continuously being updated.

Meanwhile, the 1st Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap (TH-RM-1) of the AESJ was established in 2009 [5], before the 1F accident. Collaboration was made among major relevant entities; utilities such as TEPCO, KEPCO and JAPC, vendors such as Hitachi, Toshiba and MHI, research institutes such as JAEA, CRIEPI and IAE, regulatory body TSO such as JNES, and 10 universities including Tokyo, Kyoto, Nagoya, Tsukuba etc. Main subjects were R&Ds necessary for the development of New High-Performance LWRs including passive safety features and for improvements of operating LWRs that include subjects related to high burn-up fuel and aging of components such as pressurized thermal shock (PTS) for PWR. As important technical subjects common to both new LWR development and operating LWRs, SA research and development/improvement of safety analysis tools such as best-estimate (BE) computer codes were considered to pursue. All the technical subjects were classified in a form of technology map that describes achievements, further R&D needs, R&D priority and entities that may carry out the required R&D to attain objectives, along with a chronological RM. The development plan of the New High-Performance LWRs was commonly referred in each of the chronological RM. All the chronological RMs are summed into one to visualize mutual relationship among the relevant subjects as shown in **Appendix**.

The TH-RM-1 was uploaded to the home page of the Thermal-Hydraulics (T/H) Division of AESJ, thus opened public, and referred by METI and NISA, for example, when they consider R&D related to the development of New High-Performance LWRs and safety improvement for the operating LWRs. Outline of the TH-RM-1 was presented further at the keynote lecture for NTHAS-7 [6].

The occurrence of the devastating 1F accident in 2011, however, and the lessons-learned from the 1F accident as well as the great changes in the government policy and regulation for nuclear energy utilization in Japan made us utterly re-consider the TH-RM contents as well as their role and significances. The main strategy to revise the TH-RM was re-defined to attain the world highest-level safety in the LWRs in Japan by promoting R&D to continuously improve safety in the operating LWRs, and to contribute for the establishment of standards as basis of relevant technologies. It appeared rather difficult situation for a continuous discussion because of frequent changes in the policy and counter-measures related to the LWR utilization, even after the restart of TH-RM revision work in 2012. Through the continuous effort taken by the members who have involved since the development of the TH-RM-1, however, the TH-RM has been successfully revised by March 2015 [7]. This paper describes the newly developed TH-RM, especially on the strategy, major outlines and contents.

2. REVISION OF THERMAL-HYDRAULICS TECHNOLOGICAL STRATEGY ROADMAP

Just after the completion of the 1st AESJ Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap TH-RM-1, the AESJ T/H Division started updating by newly establishing a working group (WG) in 2009. In the discussion for revision, focus was given onto human resource development mainly through research in universities by breaking down the major technical subjects given in the TH-RM-1 into fundamental phenomena and subjects. This method was called as “matching of needs and seeds”, being designated as main strategy for revision. The revision activity had then been pursued by three sub-working groups (SWGs) of “severe accident (SA)”, “scaling” and “plant improvement technology”, under the WG. The “scaling” SWG discussed needs and subjects related to the development of new safety analysis computer code in Japan. Each of SWG includes experts from academia, research organizations and industry.

In the course of the TH-RM-1 development, before the establishment, it appeared that most of major test facilities had been closed except few such as LSTF and THYNC in JAEA. The motivation to conduct experiments was found decreased, partly because of difficulties in obtaining sufficient budget. The situation of human resource was clarified further, especially from questionnaire for universities, as one of the most important subjects that thermal-hydraulics was facing to pursue with highest emphasis by confirming a consensus among all the members involved in the TH-RM-1. Related to this significant concern, the composition of technical subject of the TH-RM-1 was considered somehow inappropriate as long as fundamental research is considered, because detailed phenomena that should be investigated related to each technical subject were not well described. The TH-RM-1 was then recognized more suitable for vendors to undergo technical development than for fundamental research.

The 1F accident being followed by the great changes in the government policy and regulation for nuclear energy utilization made us to re-define the main strategy to revise the TH-RM to attain the world highest-level safety in the LWRs in Japan by promoting safety-related R&D to continuously improve safety for current LWRs. The revision strategy of “matching of needs and seeds” defined before the 1F accident, however, was considered to maintain along with the new strategy, because the importance of human resource development became more prominent after the 1F accident. The TH-RM revision work was re-started in 2012. The previous SWG names of "scaling" and "plant improvement technology" were changed respectively to "safety assessment" and "fundamental technology" to indicate the objectives so that they pursue the safety improvement more clearly, while "SA" SWG was kept unchanged. The key issues and examples in the revised TH-RM are shown in **Fig. 2-1**. The progress of revision work was presented as a Keynote Lecture at NTHAS-9 in 2014 [8].

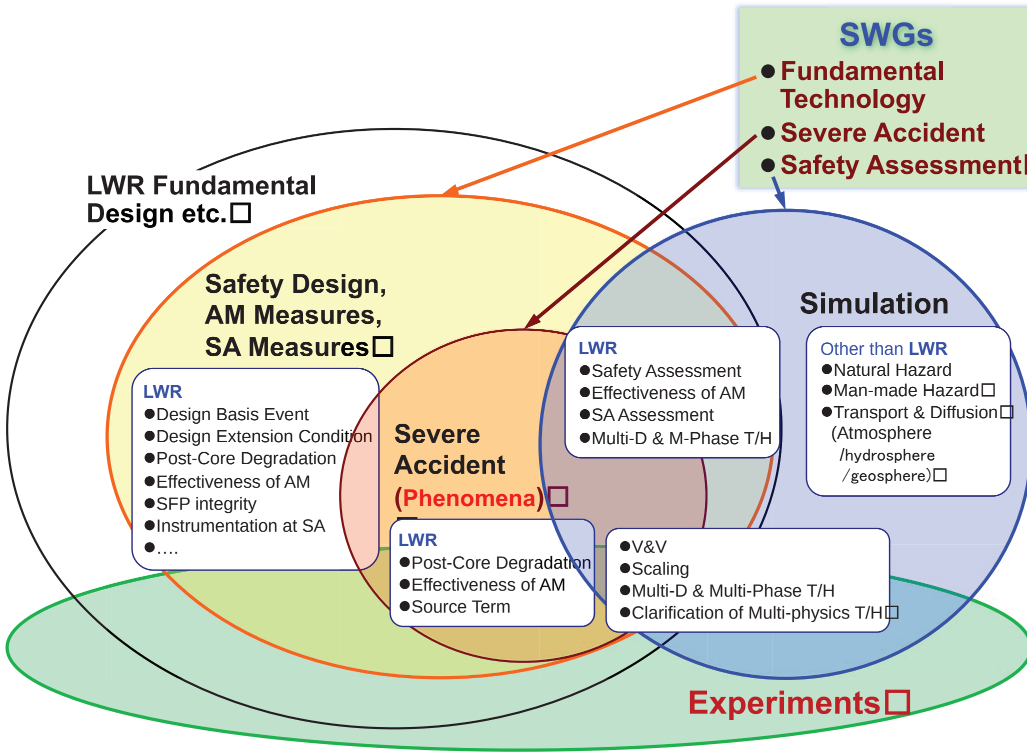

Figure 2-1 Key Issues and Examples related to 3 SWGs for Revision of TH-RM-1

Figure 2-2 shows summary of chronological RMs identified in the revised TH-RM. Practical time length is indicated up to mid-2020's to which selected technical subjects are compared in one diagram.

Each technical subject should make its own progress and have its own milestone. In the revised TH-RM, however, milestones common to all the subjects are considered. Expected progress in nuclear technology and development and important requirements for such milestones may include "immediate (within few years)" to respond to new regulation law/standards for re-starting of existing LWRs, new technologies that may improve safety, continuous safety improvement after re-starting, the 1F decommissioning RM including debris removal, and technology feedback from new LWR development. Most of them are noted in **Fig. 2-2** as well as safety research done by the NRA.

The selected technical subjects include technologies related to SA countermeasures and thus AM improvements including those for clarification of the 1F accident and development and improvements of simulation techniques that may address clarification of influences from various external events, especially from the points of view that may continuously contribute safety improvements based on lessons-learned from the 1F accident. Development of human resource as well as knowledge base, test facilities and code & standards is considered too as the important and necessary subjects to accomplish the LWR safety improvements.

External events with extremely low frequency yet significantly high consequences are considered in the revised TH-RM as the most important lessons-learned from the 1F accident. Practical counter-measures are newly required by new regulation law/standards by the NRA. As for thermal-hydraulics, however, research has been rather scarce except hydraulic (and neutronic) response during earthquake. The detailed investigation has thus been done to prepare Work Description Sheets (WDS); one for each event of tsunami etc., which includes the current status of knowledge or so. This is explained in more detail in Section 2.3.2. Based on such information investigation, cooperation and collaboration with other academic societies specific to each of external events is planned to enhance capability of AESJ to effectively decrease risks from the external events. Internal cooperation is also considered as well.

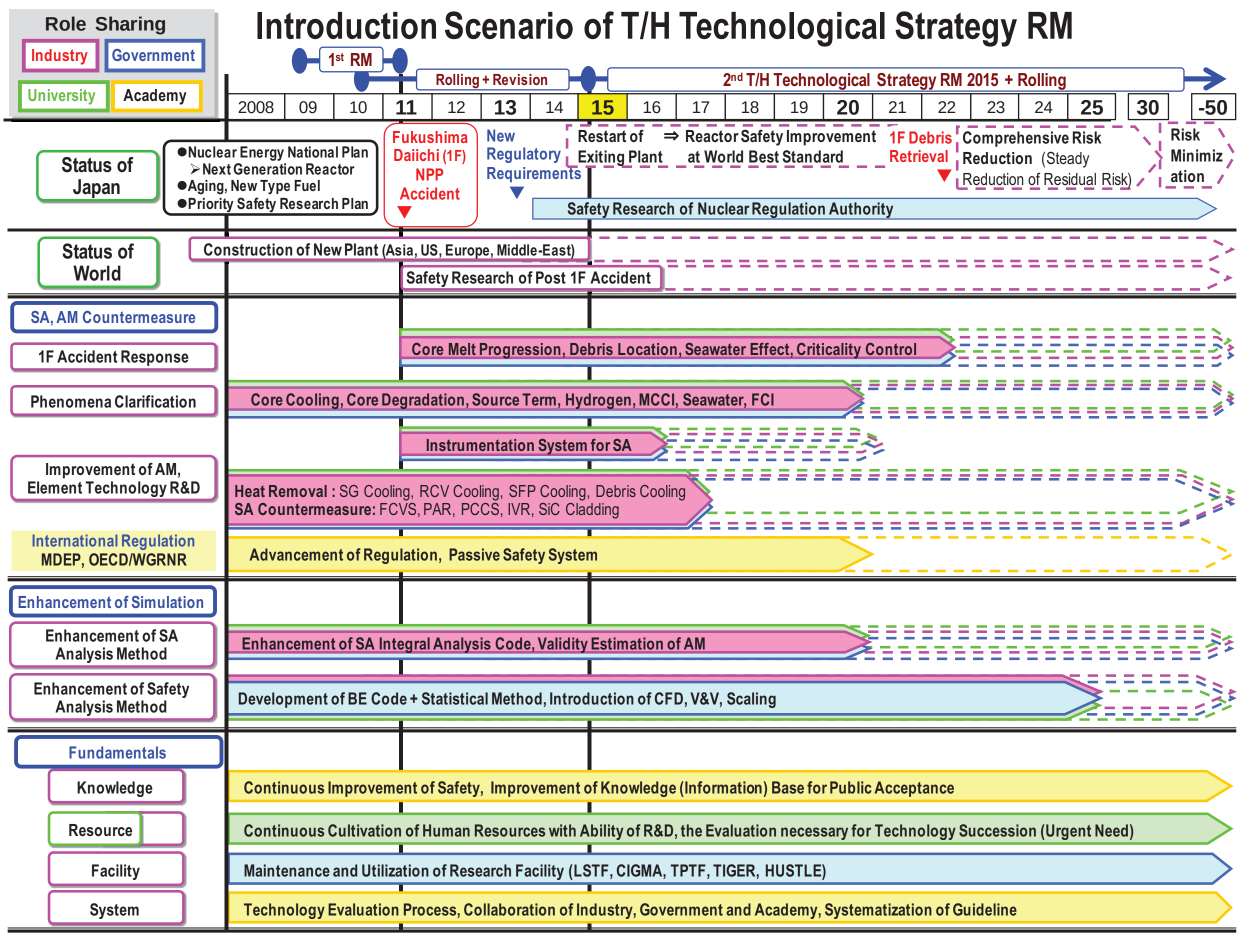

Figure 2-2 Sum of Chronological Roadmaps

2.1. Technology Map for Severe Accident

The Severe Accident SWG completed to prepare their technology map in 2013, by breaking-down various SA phenomena, AM measures and their responses into fundamental (elementary) process to provide Phenomena Identification (PI) Table, following the strategy of "matching of needs and seeds" [9]. Such breaking-down, however, was performed after the re-evaluation and consideration to complement SA phenomena and counter-measures to (prevention of) SA, in light of 1F accident.

Major SA phenomena and technologies dealt with in the technology map are following 10 subjects as;

- • Core degradation, • Re-criticality of melt/debris, • In-vessel Retention (IVR),
- • High pressure melt ejection (HPME) & Direct containment heating (DCH),
- • Molten core concrete interaction (MCCI), • Fuel-coolant interaction (FCI, steam explosion),

• Air-tightness of containment vessel (CV), • Hydrogen behavior, • Source term and • Instrumentation

Based on the SA phenomena observed in the 1F accident, following 3 subjects were re-evaluated;

• Core melting progression, • Influences of Sea-water Injection, and

• Re-criticality & Source-term (influences of pool-scrubbing during depressurization boiling etc.)

Following 3 subjects were complemented further;

• Countermeasures to suppress leakage by maintaining CV airtightness,

• Methods to minimize Environmental Impact and • Instrumentation during SA

When the technology map was prepared, following items were reviewed well to judge priority of each fundamental subject; status and degree of sufficiency of state-of-the-art knowledge including database as well as risk significance and severity of consequence as probabilistic risk assessment (PRA) viewpoints. An example of such technology map is shown in **Table 2-1** on hydrogen. After the preparation of the technology map, however, chronological roadmap was not prepared. The technology map was referred by fundamental technology SWG when technical subjects were considered and reviewed.

Table 2-1 An Example of Technology Map for SA (ex. Hydrogen, a part, as of 2013)

Technical Subject	State-of-Art of Subject	Database and/or Knowledge Base	Extent of Influences	Evaluation Method		Uncertainty		R&D Priority
				Integral Code	Detailed Code	Phenomena	Code Maturity	
Mixing	• Many experiments • CFD prediction of stratification, mixing, wall condensation is premature (ISP-47 etc.) • In sufficient counter-measure for H2 accumulation in RB	• Refs. on experiments and CFD analysis on H2 distribution in CV • Code analysis by former JNES • OECD/THAI Projects	Small influence onto CV failure frequency, however, Risk of large release in case of DDT in CV	Lumped-parameter code such as MELCOR and MAAP may be possible	CFD prediction may perform detailed multi-D analysis	**Small:** Prototypal-size exps. done	**High:** Many codes and experiences	**Low:** Many experiments and code analyses done
Combustion	• combustion limit, combustion form, frame propagation, DDT have experiments respectively • Model developments done • Small knowledge on combustion during spray, OECD-NEA/THAI2 may provide data	• Experiments on deflagration such as NUPEC, NTS, BMC and LSVCTF • Experiments on detonation such as SNL, BNL and RUT • Analysis on detonation by JNES		Lumped-parameter code such as MELCOR and MAAP may be possible	Several codes such as SODIV	**Medium:** Prototypal-size exps. done Uncertainty in DDT analysis	**Medium:** Code developed for deflagration and detonation	**Low:** Experiments on combustion and model developments done
•••								

2.2. Roadmap for Fundamental Technology

As the first step of the RM development for fundamental technology, the Fundamental Technology SWG prepared a list of candidates of the AM measure from the viewpoints of the safety enhancements against SAs, similar to the SA-SWG Technology Map. These candidates cover the AM technologies that function in each phase of SA progression, which are pre-core damage phase, post-core damage phase without vessel failure, post-vessel failure phase and pre-containment failure phase. The measures for the cooling of spent fuel pool (SFP) are considered as well.

These AM technology candidates are prioritized for further R&D by evaluating three factors; extent of potential risk reduction, extent of uncertainties with AM and extent of maturity of performance evaluation for the technology. The evaluated results are reviewed by the SWG members. The prioritization is to be re-evaluated periodically and continuously following the progress of the RM utilization.

Twelve AM technology candidates are identified and selected to have high priority for further R&D,

which are illustrated in **Fig. 2-3**. Each technology is judged to have potentially large impact on safety enhancement by enhancing the capabilities for preventing core damage and by maintaining the accident progression within in-vessel, maintaining the integrity of CV and SFP. In order to facilitate the R&D for these technologies, a Chronological RM and a WDS have been generated for each of identified and selected R&D subjects, to be described in the following sub-sections. Expected outcome and application are provided in the WDS, which are reviewed and revised periodically by monitoring their achievements against these expectations, the industry and/or regulatory needs and budget circumstances.

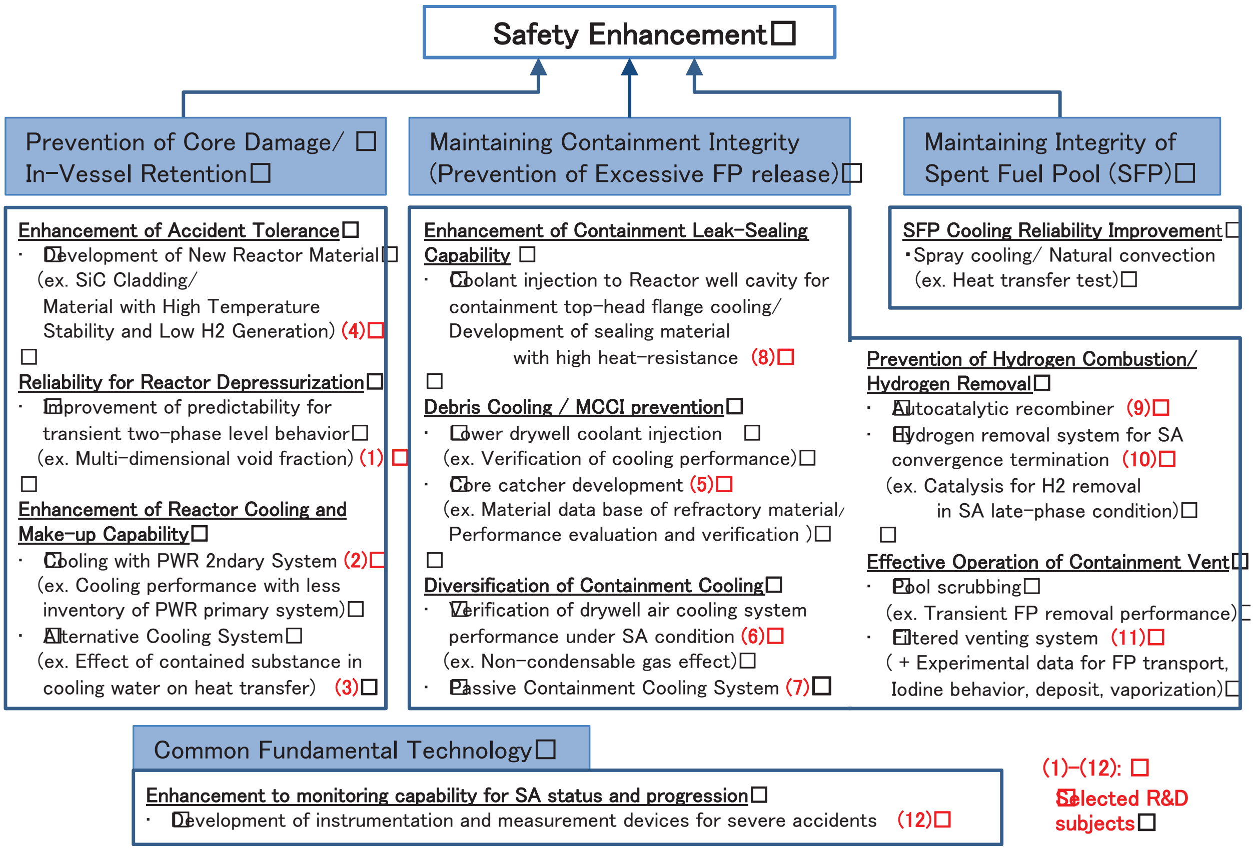

Fig.2-3 Selected R&D Areas and Subjects

The RMs and WDSs can support the planning (preparation and maintaining) of the experimental facilities to be used for future R&D and the planning of the resource development in both vendors and academia. It is expected as well that the achievement of each R&D subject can enhance the capability to prevent or mitigate SA consequences and can be utilized to establish standards for these selected technologies.

2.2.1. Chronological Roadmap:

The followings are brief description of chronological RMs for some of selected areas, which may contain more than one of the selected subjects. (Diagrams are not shown.)

Reactor vessel (RV) integrity

This area relates to accident progression behavior in RV, and contains R&D subjects such as investigation of RV failure mechanism or coolability effectiveness of seawater. The RV failure condition such as failure location and area still has large uncertainty; especially knowledge about the lower head failure condition for BWR is limited. Furthermore, although seawater injection was conducted in the 1F accident, knowledge about how seawater and impurities influence fuel cooling characteristics is also limited.

Therefore, SA progression within RV, such as core melting, failure mechanism of RV and influences of seawater injection is regarded as very important issue that should be well understood.
Based on the knowledge by the R&D, it would be effective to enhance SA analysis methods from the viewpoint of molten debris retrieval in the Fukushima Daiichi NPP.

Containment integrity (molten core cooling)
This area relates to coolant injection into containment as a means of AM following reactor vessel failure, and contains R&D subjects such as MCCI or core catcher design. Melt coolability still has a large uncertainty, especially under the unfavorable situation, for instance, delayed coolant delivery. Hence, coolability enhancement and uncertainty reduction are desired to secure CV integrity during SA. As for the CV of existing LWRs, confirmation of effectiveness of early coolant injection (or advance flooding) to suppress MCCI may contribute higher confidence level in safety evaluation for beyond design basis event (BDBE). For new reactor design, several concepts for effective melt cooling have been proposed. Material property is the key information for core catcher configuration. Because of geometrical restriction, further simplification is required to accommodate these concepts into operating reactors. Analytical model enhancements are necessary too for proper evaluation of material behavior, heat transfer and hydraulics regarding melt cooling, as well as their integration as a system model.

Hydrogen risk reduction
This area relates to reduction of risk associated with hydrogen combustion, which requires estimation of such related phenomena as hydrogen generation, transfer, mixing and burning, including performance of hydrogen processing equipment such as PAR (Passive Autocatalytic Recombiner). Various related studies such as OECD/ NEA THAI project have been carried out. There still remain, however, many challenges, for instance reduction in uncertainty of the concentration/ distribution of hydrogen by mixing, stratification etc. with atmosphere containing vapor and performance degradation of PAR by aerosol etc. Also, even in the inerted CV of BWR, generated hydrogen is likely to remain at a high concentration for a long period of time, managing hydrogen safely in the accident atmosphere has become an issue. On the other hand, reduction of the hydrogen generation itself by replacing Zircaloy with SiC for the fuel cladding material has been proposed as a fundamental measure. When overcoming challenges relating to processing hydrogen, analytical approach using 3-D CFD analysis has come to be taken in addition to experimental approaches.

2.2.2. R&D Subjects:

The followings are brief descriptions of the selected R&D subjects.

Melt coolability enhancement and reduction of uncertainty
Important phenomena related to this R&D subject are, melt gravitational transport, jet impingement, spreading, ablation and gas release, heat transfer to coolant, material properties, and structural integrity. Based on the review of current states of knowledge regarding these phenomena and their importance for safety enhancement, possible research topics for this subject are;

- Effectiveness of early coolant injection (or advance flooding)
- Melt-coolant heat transfer and hydraulics (containment / core catcher configuration specific)
- Refractory material properties (thermal and chemical)

Examples of expected outcomes or their applications from these topics are;

- SAMG optimization and its technical bases consolidation
- Design data for core catcher
- Model / correlation improvement and database for code verification

As for implementation status, tests have been started to investigate thermal hydraulics in the cooling channel of core catcher, and to obtain refractory material data as projects supported by government.

Passive autocatalytic recombiner (PAR)

PAR is developed to prevent the hydrogen combustion in CV at accident and is adopted mainly in European plants. Based on lessons-learned from the 1F accident, PAR has been adopted in the domestic existing PWR plants with large dry CV. As for the domestic existing BWR plants, PAR has been installed in the reactor building to restrain hydrogen combustion. However, it is desirable to enhance resistance against the catalyst poison and/or to obtain in depth understanding on the new catalyst by promoting advances in the PAR system performances with further improvements in the performance evaluation method. Japanese new PAR system, based on the automotive exhaust gas purification catalyst, is under development by government fund.

Cooling with PWR Secondary System

Cooling with steam generator (SG) secondary system is one of the most important AM measures for PWR because that makes it possible to remove the core decay heat without losing the inventory of the primary system and to reduce pressure of the primary system promoting coolant injection by low-pressure injection equipment. Effectiveness of the cooling using SG secondary system has been indicated by various integrated system tests [10], but the data that can cover a wide range of SA scenarios is insufficient, for example when the primary system inventory is small or core decay heat is small. Therefore, the integrated system tests covering wide range of SA conditions using the ROSA-LSTF of JAEA have been planned and performed, which demonstrate the effectiveness of improved AM measures and also expand the database for the validation of BE analytical methods.

2.3. Roadmap for Safety Assessment

Roadmap for safety assessment is revised by the Safety Assessment SWG to develop computer codes for the evaluation of nuclear reactor safety including simulation methods for natural hazards such as flooding, tsunami, high wind, typhoon & tornadoes, and fires on- & off-site.

Firstly, the SWG members discussed freely and listed up the following 3 technical subjects on thermal-hydraulics simulation identified in light of the 1F accident, especially for the subjects 2 and 3.

1. Phenomena in RV, CV and containment building
   phenomena during DBA and SA, transfer and/or diffusion of radioactive materials,
   hydrogen behavior, structure integrity against core melt and debris,
   TH behaviors in spent fuel storage pool and fluid-structure interaction
2. Decommissioning technology for 1F NPP
3. External events
   Tsunami, volcano, tornado/high wind, fire, (internal) flooding, cold/severe heat wave, etc.

Secondly, we consolidated above identified technical subjects from view points of time constraint and maturity of simulation methods into an arrangement table similar to the SA-SWG Technology Map, and prepared four Organized Information Tables (OITs) of phenomena in RV before and after core degradation/ melting, phenomena in CV and SFP, similar to the WDSs for Fundamental Technology SWG. Five OITs were prepared further for external events of tsunami, volcano, tornado/high wind, internal flooding and fire. Each information table described current status of relevant simulation methods, remaining gaps from intended needs and issues for developing simulation methods. Simulation methods concerning phenomena in RV before and after core degradation/melting, CV and SFP have been studied actively so far compared with a study concerning external phenomena. Therefore, on the arrangement table, the simulation methods/computer codes under utilization/ development/ improvements were put in order every industrial sector, government and academic sector.

Finally, desirable features that future computer codes should possess such as 3D-multifield model including coupling between microscopic and macroscopic models were pointed out.

The result of the thermal-hydraulic study for nuclear reactor safety is finally gathered via model development and V&V into the thermo-hydraulic system code. In other words, the system code shows the ”integrated capability” of thermal-hydraulics for nuclear reactor safety of the country, and therefore each country has developed their own system code (e.g.: TRACE, CATHARE, ATHLET, MARS, COSINE). It is necessary to develop the system code made in Japan.

2.3.1. Technical Problems/Challenges in Safety Assessment:

Phenomena in RV after core degradation

The new NRA regulation law/standards require that a nuclear power plant and its auxiliaries shall equip measures to prevent and to mitigate SAs, and effectiveness of the measures for maintaining CV integrity shall be evaluated by using appropriate BE methods, whose validity and applicability to simulate phenomena in RV after core degradation have been confirmed. It is also required that consideration of sensitivity of models in the methods on evaluated results appropriately, when the models have large uncertainty and/or models are applied to conditions beyond applicable parameter ranges. Key survey and discussion results summarized in an OIT for simulating phenomena in RV after core degradation are following.

US-developed SA codes of MAAP (FAI), MELCOR (NRC/SNL) and RELAP/SCDAP (ISS) are widely used. The MAAP uses a lamped parameter model for modeling plant system. The MELCOR basically uses a 1-D node-junction model for plant system modeling, however, it can use a 2-D node-junction model packages for solving core melting and relocation behavior. The RELAP/SCDAP can solve thermal interaction between a dropped melted core and PV lower head in 3-D by coupling with a CFD code. THALES2 (JAEA) and SAMPSON (IAE) are Japan-developed SA codes. Latter especially is a mechanistic model code, being furnished with detailed models such as 2-D, 3-phase, multi-component model to simulate core melting and relocation behavior, hydrogen generation and PV lower head failure by high-temperature melt, while it utilizes the RELAP5 for system thermal-hydraulic response analysis. In order to estimate PV failure location with accuracy, however, higher level of detail is required in geometric and thermal-hydraulic modeling than those of above codes. Molten core relocation behavior is quite complex because it is affected by geometries and material physical properties. Detailed simulation of thermal-hydraulics and neutronics coupling is also important to estimate 3-D power profile when core degradation takes place. New experiments that simulate the progress of core degradation is then desirable to obtain model validation data that can be used without scaling considerations by using actual fuel with dimensions as close to prototype as possible. It is also desirable to conduct additional material experiments to clarify multi-component interactions when melting and oxidation coexist etc.

Development of new Japan’s own integral system analysis code applicable to SA is underway. Code development, maintenance and application to various activities should contribute to continuous safety improvement in LWR design and AM measures as well as systematic human resource development not only for R&D area but also for training and supporting decision-making.

Phenomena in CV

Time scale of SA widely ranges from several hours to several days. Integrated analysis code, such as MAAP or MELCOR, is applied for the evaluation of plant status by following the long-term accident progress in CV and the evaluation of source term (FP behavior) and hydrogen. The integrated code is designed to calculate SA phenomena in relatively small number of nodes for rather short time, because empirical correlations and/or physical models are prepared based on rather short-term experiments. Most of such physical models are macroscopic and are not sufficient to understand the details of phenomena that may occur in CV. Although there is no integrated code composed only of detailed mechanistic models, current SA analysis approach as above would be reasonable if available computational power is considered. Along with the improvement of computer performance, in the near future, the integrated SA analysis code with the detailed mechanistic models is expected/desirable.

Specific problems/challenges of SA analysis in CV identified in the TH-RM are as follows;

a. On such SA phenomena as hydrogen combustion, MCCI, FCI, DCH, FP behavior, improvement of corresponding models is required. Relevant experiments are also required to obtain validation data.

b. To analyze molten core behavior in detail, more detailed modeling of CV internal structure is required. The conventional lumped-parameter model and the node-junction model should have difficulties in the prediction of the SA event in the true sense.

Future activities from above problems/ challenges are as follows;

a. Validation of SA code including model improvements is necessary in response to the progress in understanding the situation of the Fukushima Daiichi nuclear power station.
b. Corresponding various model improvement requirement that may arose from a. above, relevant experiments and model improvements must be done successively yet in timely manner.
c. Development of detailed mechanistic code must be pursued to complement lack in the experiments for the validation and improvement of current macroscopic SA codes.

2.3.2. Compilation and Evaluation of Major External Events:

The latest information of simulation technology about the external events (tsunami, volcano, tornado, flooding, fire) which was not developed to a technology map, but was arranged to the OIT as an "information search” results. The database for field and phenomena to simulate, the state-of-art of simulation technology, a model V&V and extension of experiment database, priority and division of rules, etc. were investigated like the internal events. As an example, the contents of the OIT about an internal flooding are summarized.

(1) Field and phenomena to simulate

Contents to be evaluated as an internal flooding due to the pipe rupture in the reactor building are checked to a new Japanese regulation standard. "The internal flooding impact evaluation guide of a nuclear power plant" is shown from the NRA as an internal flooding guide. This guide suggests that utility should evaluate the influence of submergence, water and steam to equipment and confirm whether the function is secured. The guide suggests following explanation about the influence of steam with acceptable evaluation methods using computer code.

- An appropriate estimation area should be set up when diffusion area is assumed to evaluate influences of steam by using a suitable calculation method. When the diffusion area is computed by using a 3-D flow software etc., utility should indicate it’s applicability and evaluation conditions to the vapor diffusion calculation.

The guide further requires that SFP should be taken into account as a source of flooding water when it leaks including water sloshing due to seismic motion.

(2) The State-of-art Simulation Technology

The SWG examined possible calculation methods for the vapor diffusion evaluation being used by PWR utilities, and confirmed that a 1-D code of node-junction method is used. Simulation of actual vapor flow conditions in a complicated reactor building using 3-D CFD code should be more effective to quantitatively estimate local influences of vapor than by 1-D code.

Sloshing of SFP is evaluated with 3-D CFD code to estimate wave amplitude and the amount of leaking water. Since the best evaluation method is not specified yet, continuous review of evaluation method; those used in regulation examination for example, is good to find the one.

(3) A model V&V and extension of experiment database

Since evaluation method for internal flooding has not been well established/specialized in nuclear power plant technology, it needs further investigation including experiments to provide data for model validation (ex. CFD grade) along with the model development as nuclear power plant-related R&D.

(4) Priority and roles (sharing), etc

Priority of improvement/development of simulation technology for internal flooding is high because safety evaluation is required by regulation law/standard. Both of utility (industry) and regulatory body (government) should bear appropriate simulation technology for their own use. New model/code and detailed data for validation could then be shared among industry and academia.

2.4. Rolling of Roadmap (Utilization and Revision)

Figure 2-4 is a diagram for rolling (utilization and revision) of the established new TH-RM which can be downloaded from the website of AESJ T/H Division [7] so that everyone can share. The TH-RM is also explained to relevant authorities and academic societies as well as being reported to several conferences and journals. Stakeholders use it for the planning and proposal of R&Ds for continuous safety improvements. Authorities (government) such as METI, MEXT and NRA and/or industries may provide budget to carry out the proposed R&D. Each R&D subject should have it's own objectives to contribute to the safety improvements, even when it is decided to perform by referring the criterions defined and expected and/or available resources suggested in the TH-RM. A few important subjects such as (1), (3), (5) and (12) in Fig. 2-3 have started R&D activity in a form of national project being funded from both of industry and government. Following feedbacks from the results and progress in the performed R&D subjects as well as the change in social needs, the TH-RM is reviewed and revised periodically yet continuously under the AESJ T/H Division, concerning the industry and/or regulatory needs, the achievements of the R&D and R&D budget status.

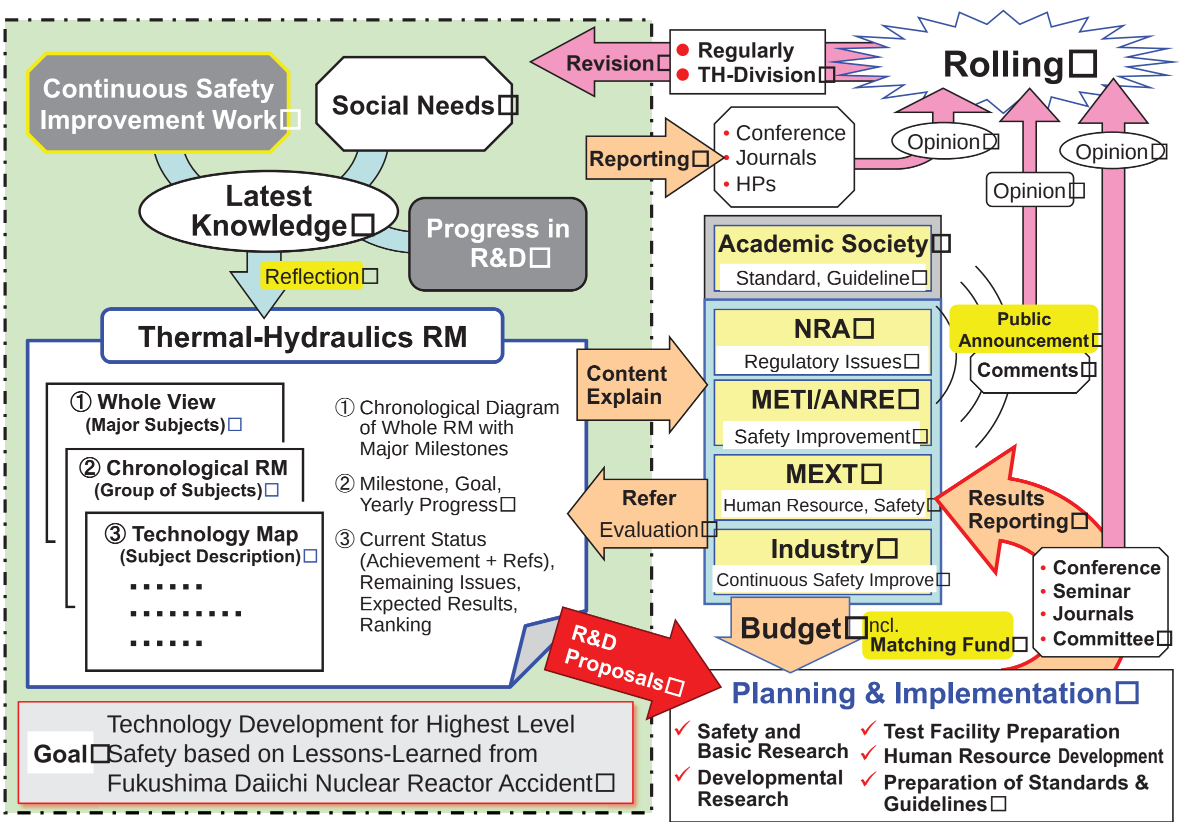

Fig. 2-4 Diagram for Rolling (Utilization and Revision) of TH-RM

3. CONCLUSIONS

The T/H Division of AESJ developed the new Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap (TH-RM) for LWR safety improvement and development after the 1F accident by thoroughly revising the 1st version (TH-RM-1) prepared in 2009. Key issues to pursue have been re-organized under collaboration among industry, government and academia as a basis for safe and stable operation of LWR in light of lessons-learned from the significant 1F accident.

The revised TH-RM compiles near and mid-term challenges to attain the world highest-level safety in the LWRs in Japan, by promoting newly identified R&D subjects to continuously improve safety in the operating LWRs and by keeping "matching of needs and seeds" in mind for continuous and effective human resource development from academia. The three SWGs of Severe Accident, Safety Assessment and Fundamental Technology identified the technical subjects and related phenomena for R&D to implement rigorous, fundamental yet profound counter-measures onto the operating LWRs to never cause disastrous accident again.

The T/H Division of AESJ continues effort to promote and summarize the results from identified R&D subjects and continuously/periodically revise the TH-RM following the progress in the technology development relevant to the identified R&D subjects as well as the social needs. International cooperation and collaboration is pursued to find better counter-actions for further improvement in LWR safety.

ACKNOWLEDGMENTS

Authors are thankful to the T/H Division of AESJ and the members of "Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy" WG for their continuous support with valuable comments and discussions to develop the revised TH-RM considering the lessons-learned from the accident at Fukushima-Daiichi nuclear power station.

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APPENDIX The 1st T/H Technological Strategy Roadmap TH-RM-1

Roadmap s relevant to LWR technology were developed first in 2006 for advanced utilization of current

LWR, in the major 3 technical fields of high-burnup fuel, aging management and power uprate that is for thermal-hydraulics (T/H). In 2007, government & utilities announced "Development of next-generation light-water reactor capable of winning a world standard” to start operation on 2030. The T/H field was then expected to play a major role in the development & verification of fundamental technology to support stable, economic & safe high-performance LWRs (PWR & BWR). A special committee was established at AESJ in 2007 and formulated 1st AESJ Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap (TH-RM-1) in March 2009 under collaboration among industries, government and academia, which may cover LWR thermal-hydraulics R&D in Japan. The AESJ T/H division took over the role of revision of TH-RM afterwards.

Whole technical subjects dealt with by the TH-RM-1 are compiled in a chronological manner in **Fig. A-1**. Major milestones followed that for the development of next-generation light-water reactors. Technical subjects of the TH-RM-1 were composed of the following three major parts;

I. Subjects specific to Reactor Design
   (1) New LWR to develop next generation LWRs, natural circulation reactors, small reactors etc.
   (2) Existing LWRs to deal with power uprate, high burn-up fuel, aging management etc.

II. Subjects common to LWRs
   (3) New and Existing LWRs to address for the development of advanced safety analysis method, post-accident long-term cooling, severe accident management (SAM) etc.

The developed TH-RM-1 indicated entities (utilities, vendors, research institutes, universities, TSOs that support regulatory bodies) who should take a responsibility to conduct the indicated tasks.

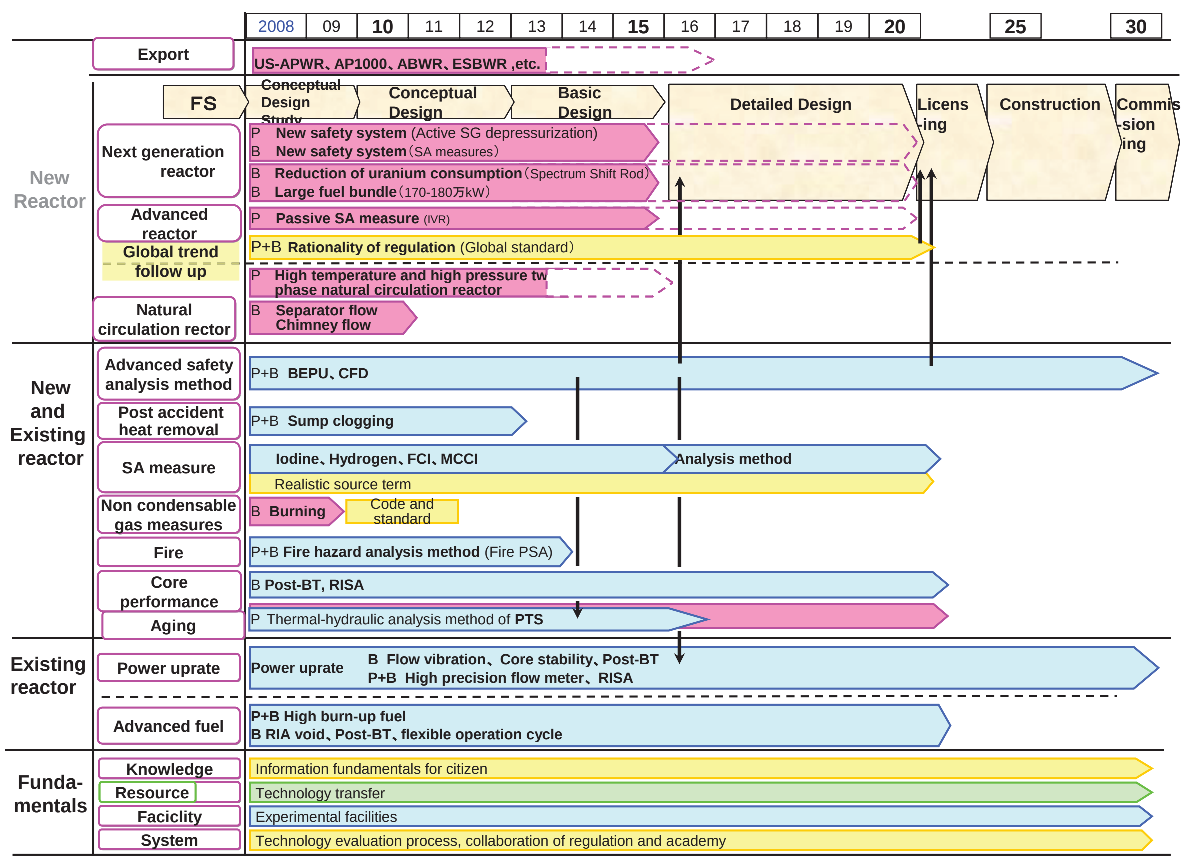

Figure A-1 Full Contents of 1st T/H Technological Strategy Roadmap (TH-RM-1)