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Survey on Research and Development Status of Japanese Small Modular Reactors in OECD/NEA Activities (2022-2023)

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An important theme of Japan's 6th strategic energy plan is to indicate the energy policy path towards carbon neutrality by 2050. Policy responses for Japan's nuclear energy research and development (R&D) towards 2030 contain the demonstrations of technologies for small modular reactors (SMRs) through international cooperation by 2030. In light of this energy plan, basic policy initiatives over the next 10 years have been compiled to realize Green Transformation (GX), which simultaneously achieves decarbonization and economic growth. Looking overseas, activities of SMR R&D are active internationally, mainly in the US, Canada, Europe, China, and Russia. These activities are not only by heavy industry manufactures and R&D institutes, but also by venture companies.

Under these circumstances, the NEA CSNI has gathered an Expert Group on SMRs (EGSMR) to help estimate the safety effects of SMRs. The EGSMR efforts required the submission of responses to several questionnaires whose main purpose was to collect the latest information on the efforts of SMR deployment and research. The first author of this report responded to this based on information from Hitachi-GE Nuclear Energy, Ltd. and Mitsubishi Heavy Industries, Ltd. as well as JAEA. Most of the responses from Japan to the questionnaires are the information that serves as the basis of CSNI Technical Opinion Paper No. 21 (TOP-21). In this report, the Japan's publicly available responses to the questionnaires arranged and additional information are explained, which complements some of the content of the TOP-21. In this manner, the investigation results of R&D related to SMR in Japan, focusing on the EGSMR activities (2022-2023), are summarized. The target of this report is to provide useful information for future discussions on international cooperation concerning SMR as well as nuclear power field human resources development internationally and domestically.

In this study, as key topics regarding R&D on Japanese advanced reactors, major gaps have been identified between the technologies needed for practical application and their present status for high temperature gas-cooled reactor (HTGR) and sodium-cooled fast reactor (SFR). Moreover, in relation to the interconnections between HTGR and the hydrogen production facility, the safety implications of leakage of combustible gas from the hydrogen production facility, its abnormal occurrence, etc. are organized.

Keywords: Carbon Neutrality, Green Transformation, Small Modular Reactor, NEA, CSNI, EGSMR, HTGR, SFR, Hydrogen Production, Combustible Gas

+ International Affairs Department

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OECD/NEA 活動における日本の小型モジュール炉に関する研究開発状況の調査 (2022-2023 年)

日本原子力研究開発機構 安全研究・防災支援部門
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（2024 年 7 月 18 日受理）

日本の第 6 次エネルギー基本計画では、2050 年までのカーボンニュートラルを目指したエネルギー政策の道筋を示すことが重要なテーマとなっている。2030 年に向けた日本の原子力研究開発(R&D)への政策対応には、国際連携による 2030 年までの小型モジュール炉(SMR)技術の実証が盛り込まれている。これを踏まえ、脱炭素化と経済成長を同時に達成する Green Transformation(GX)の実現に向けて、今後 10 年を見据えた取組の基本方針が取りまとめられた。海外に目を向けると、米国、カナダ、欧州、中国、ロシアを中心に、重工メーカーや R&D 機関のみならずベンチャー企業も含めて、国際的に SMR の R&D 活動が活発である。

このような状況下で、原子力機関(NEA)の原子力施設安全委員会(CSNI)は、SMR の安全性への影響評価を支援するために、SMR に関する専門家グループ(EGSMR)を招集した。EGSMR の取組として、SMR の導入や研究活動に関する最新情報の収集を主目的とした数回にわたるアンケートへの回答の提出が求められた。これに対して、筆頭著者から、JAEA に加えて日立 GE ニュークリア・エナジー株式会社、三菱重工業株式会社からの情報に基づき回答した。アンケートに対する日本の回答の多くは、CSNI Technical Opinion Paper No. 21 (TOP-21)のベースとなる情報である。本報告書では、整理した公開可能な日本のアンケート回答と付加情報を示し、TOP-21 の記載内容の一部を補完した。これにより、EGSMR の活動(2022-2023 年)を中心とした日本における SMR に関する R&D の調査結果をまとめた。本報告書は、SMR に関する今後の国際協力の議論や国内外の原子力分野の人材育成に役立てることを目的としている。

この中で、日本の革新炉の R&D の主なトピックスとして、高温ガス炉(HTGR)とナトリウム冷却高速炉(SFR)に関して、実用化に必要な技術と現状のギャップを同定している。また、HTGR と水素製造施設の相互接続に関連して、水素製造施設からの可燃性ガスの漏洩と異常発生が安全性に与える影響等について整理している。

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Acronyms and Abbreviations

ABWR	Advanced Boiling Water Reactor
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
BWR	Boiling Water Reactor
CCVM	CSNI Code Validation Matrix
CRP	Coordinated Research Project
CSNI	Committee on the Safety of Nuclear Installations
DEC	Design Extension Condition
EGSMR	Expert Group on Small Modular Reactors
EMDAP	Evaluation Model Development and Assessment Process
EPZ	Emergency Planning Zone
ESBWR	Economic Simplified Boiling Water Reactor
FBR	Fast Breeder Reactor
GIF	GEN IV International Forum
GX	Green Transformation
HTGR	High Temperature Gas-cooled Reactor
HTTR	High Temperature Engineering Test Reactor
IAEA	International Atomic Energy Agency
ICS	Isolation Condenser System
JAEA	Japan Atomic Energy Agency
LFR	Lead-cooled Fast Reactor
LOCA	Loss of Coolant Accident

LOFC	Loss of Forced Cooling
LSTF	Large Scale Test Facility
LWR	Light Water Reactor
METI	Ministry of Economy, Trade and Industry
MEXT	Ministry of Education, Culture, Sports, Science and Technology
MOU	Memorandum of Understanding
MRX	Marine Reactor X
NEA	Nuclear Energy Agency
NEXIP	Nuclear Energy x Innovation Promotion
OECD	Organisation for Economic Co-operation and Development
PAZ	Precautionary Action Zone
PIRT	Phenomena Identification and Ranking Table
PRISM	Power Reactor Innovative Small Module
PWR	Pressurized Water Reactor
R&D	Research and Development
RBWR	Resource-renewable Boiling Water Reactor
RCP	Reactor Cooling Pump
RPV	Reactor Pressure Vessel
RVACS	Reactor Vessel Auxiliary Cooling System
SDC	Safety Design Criteria
SDG	Safety Design Guideline
SDHS	Stand-alone Direct Heat Removal System
SFR	Sodium-cooled Fast Reactor
SG	Steam Generator
SMR	Small Modular Reactor
TOP-21	CSNI Technical Opinion Paper No. 21
TRL	Technological Readiness Level
USNRC	United States Nuclear Regulatory Commission
V&V	Verification and Validation
VHTR	Very High Temperature Reactor

1. Introduction

As the importance of nuclear power is reaffirmed towards carbon neutrality, small modular reactors (SMRs) are attracting attention. SMRs are defined by IAEA as small nuclear reactors that produce electricity of up to 300 MWe per module [1], and are characterized by their high safety, high factory productivity, and flexibility in energy system. SMR research and development (R&D) activities are active internationally, chiefly in the US, Canada, Europe, China, and Russia, and are not only conducted by heavy industry manufactures and R&D institutes, but also by venture companies. Additionally, a lot of countries are taking part in the activities of international bodies (e.g., OECD/NEA and IAEA), and considerations concerning the SMR deployment are underway.

An Expert Group on SMRs (EGSMR) was convened by the NEA Committee on the Safety of Nuclear Installations (CSNI) in June 2021 to help evaluate the safety impacts of SMRs [2]. SMR designs have been furnished with a variety of coolants and reactor sizes. The EGSMR addresses a wide range of technologies, from water-cooled reactors to innovative reactors under development. In the EGSMR initiatives, target SMRs are broadly divided into water-cooled SMRs and advanced technology SMRs including GEN IV reactors. The objectives of EGSMR are to provide methods for coordinating activities of CSNI and NEA efforts relevant to SMR safety, and to coordinate information exchange among participating organizations and countries. The submission of responses to several EGSMR's questionnaires in 2022-2023 was requested in the EGSMR efforts for this reason. The main purpose of the submission of the responses is to gather the most recent information on the SMR deployment and research efforts among the participating organizations and countries.

The first author of this report is the contact person for the EGSMR in Japan. The questionnaire responses were made in collaboration with Hitachi-GE Nuclear Energy, Ltd. and Mitsubishi Heavy Industries, Ltd., as well as Fast Reactor Cycle System Research and Development Center, and HTGR (High Temperature Gas-cooled Reactor) Research and Development Center of JAEA. After aggregating the questionnaire responses from individual contributors, the overall contents were discussed in the meetings within JAEA. The questionnaire responses were finally submitted to the NEA Secretariat, following confirmation by the Nuclear Regulation Authority, Japan.

The CSNI Technical Opinion Paper No. 21 (TOP-21) titled ‘Research Recommendations to Support the Safe Deployment of Small Modular Reactors’ was published in October 2023 as a product of the EGSMR, on the basis of discussions on the questionnaire responses [3]. The TOP-21 provides a concise summary of the findings on activities that should be taken to support SMR deployment and the recommendations to CSNI, putting an emphasis on SMR safety. Hence, not all questionnaire responses from each country presented during the review process are described in the TOP-21. Meanwhile, many of Japan's responses to the EGSMR's questionnaires summarized Japan's efforts in SMR R&D that have been published, and so forth, which serve as the foundation for compiling the TOP-21.

In this report, the Japan's publicly available responses to the questionnaires organized and further information are explicated, centering around Japan's endeavors in SMR R&D and knowledge gaps for SMR. This supplements some of the description of the TOP-21. In this way, Japanese SMR R&D was surveyed, with the focus on the EGSMR activities (2022-2023), and the outcomes were summed up. The objective of this report is to provide a summary of the status of Japanese SMR R&D, and to disseminate information for the following purposes. (I) Exploit discussions on international cooperation in the future by indicating the Japan's status in SMR R&D in countries that are fostering the R&D and examining the deployment of SMRs. (II) Use to generate programs related to the development of nuclear power human resources internationally and domestically.

Chapter 2 of this report denotes an overview of the TOP-21. Annex A of the TOP-21 mentions the EGSMR's first questionnaire on the SMR deployment and research status. **Chapter 3** of this report indicates Japan's responses to the following items of the EGSMR's first questionnaire. (i) SMR deployment landscape. (ii) International research activities relevant to SMR safety. (iii) Major gaps in knowledge for SMR technologies. (iv) Impacts of multiple module configurations on SMR safety. (v) Applicability of knowledge base to support safety of water-cooled SMR designs. (vi) Extent of knowledge base to support confidence in safety case of advanced technology SMRs. (vii) SMR issues/topics requiring specific attention compared to safety cases of current designs.

The TOP-21 identified the following four major areas of interest by analyzing the responses from the participating organizations to the EGSMR's first questionnaire. (1) Regulatory harmonization. (2) Cross-cutting safety issues. (3) Experimental campaigns. (4) Benchmarking for computer code verification and validation (V&V). The targets of the EGSMR's second questionnaire are to review the areas of interest and develop suggested actions taken by CSNI to address them. **Chapter 4** of this report highlights Japan's responses to the following items of the EGSMR's second questionnaire. (a) Correspondence with IAEA's broad guidelines for advanced reactors, linked to (1). (b) Safety implications on interconnections between SMRs and associated heat application facilities, linked to (2). (c) Effective utilization of experimental data for SMR designs, linked to (3). (d) Reliability requirements of SMR passive safety systems, linked to (4).

2. Overview of CSNI Technical Opinion Paper No. 21 (TOP-21) [3]

This chapter provides an outline of the TOP-21. The EGSMR is comprised of CSNI participants (23 organizations from 15 countries) with an interest in SMRs, as well as international bodies (IAEA and the European Commission). The EGSMR mandate is to evaluate the needs of NEA member countries for SMR safety research. The EGSMR is tasked with recommending CSNI actions that can help countries address identified knowledge gaps and safety challenges. The EGSMR executed its assigned mandate through a process of collecting information, in a form of questionnaire, and generating recommendations through the information evaluation in 2022-2023. The consequences were compiled and the report was open as the TOP-21 in October 2023.

Through discussions on the basis of the responses to the questionnaires, the areas of interest and proposed actions identified in the EGSMR focus on the following four categories.

1. Regulatory harmonization: Foster international regulatory harmonization regarding SMR licensing.
2. Cross-cutting safety issues: Support understanding of broad cross-cutting SMR safety issues that impact many different designs.
3. Experimental campaigns: Organize reviews of international experimental campaigns and define new campaigns to improve phenomenological understanding and experimental validation of SMR designs.
4. Benchmarking for computer code V&V: Support international benchmark activities for computer modelling V&V for SMR designs.

Regarding the cross-cutting safety issues mentioned above, the EGSMR identified the following eight issues and prioritized them in accordance with their safety importance, timeline for resolution, and relevance to the CSNI mandate and capabilities. As a result, the highest priority was given to (1), that is, the application of defense in depth levels when passive safety is applied at multiple levels.

(1) Application of defense in depth levels when passive safety is applied at multiple levels.
(2) Probabilistic safety analysis for innovative/first-of-a-kind designs.
(3) Emergency planning zone (EPZ) and emergency response requirements for SMRs.
(4) Fuel safety of SMRs.
(5) Human factors (including remote operations and multi-unit/multi-module plants).
(6) Multi-unit/multi-module design aspects and impacts on safety.
(7) Transport of pre-fueled/spent nuclear modules and transportable/floating SMRs.
(8) Associated process applications: hydrogen production and process heat (e.g., chemical, mining, and district heating).

For all the above four areas of interest, the EGSMR recommends efforts to focus on two parallel tracks: water-cooled SMRs and advanced technology SMRs, with further priority given to supporting technologies and designs that are close to deployment.

The detailed areas of interest and suggested actions were reviewed, and initial key actions were prioritized based on safety significance and relevance to the CSNI mandate. The EGSMR recommends the following key suggested actions for the CSNI.

(a) Gather phenomena identification and ranking tables (PIRTs) to identify unique phenomena related to SMR safety and prioritize future efforts. Annex E of the TOP-21 represents an initial effort in collection of the PIRTs for the EGSMR.
(b) Start working on the high-priority cross-cutting safety issues.
(c) Review and update the CSNI code validation matrix (CCVM) for SMR safety.
(d) Collect and generate experimental data through joint research projects to support SMR safety. Annex C of the TOP-21 stands for an initial effort in collection of experimental facilities relevant to SMR safety for the EGSMR.

3. Japan’s Main Responses to EGSMR’s First Questionnaire Related to TOP-21

The items asked for responses in the EGSMR’s first questionnaire, which was indicated in January 2022, contained the following.

(i) SMR deployment landscape such as designs planned/constructed/operating, technological readiness level (TRL) for proposed technologies, and major knowledge gaps.
(ii) Important safety impacts of SMRs based on: application of SMR, operating location, or potential for multi-module configurations.
(iii) Planned research activities related to SMR safety (including international efforts) and availability of experimental facilities/data to support code validation to benefit SMR safety.
(iv) Applicability of current knowledge base and required extensions to support safety of water-cooled SMR designs.
(v) Extent of knowledge base needed to support confidence in safety case of advanced technology SMRs.
(vi) SMR issues/topics requiring specific attention (or which are potentially less important) compared to safety cases of current designs.

Not all responses from Japan against the EGSMR’s first questionnaire are reported in the TOP-21. This chapter presents the contents organized by Japan based on their technical knowledge according to the EGSMR’s first questionnaire.

3.1 SMR Deployment Landscape

Japan has not made any plans for SMR deployment over the next 10 years at this time. Meanwhile, the trends of the Japanese government with respect to SMRs are indicated below.

Japan’s 6th strategic energy plan was approved by the Cabinet in October 2021 [4]. The key theme is to show the path of the energy policy to realize carbon neutrality by 2050 and to reduce greenhouse gas emissions by 46% by 2030, while continuing to make strenuous efforts to meet the lofty goal of cutting emissions by 50%. Policy responses for Japan’s nuclear energy R&D towards 2030 looking ahead to 2050 are in the following. (I) Steady promotion of fast reactor development by utilizing international cooperation. (II) Demonstration of technologies for SMRs by 2030 through international cooperation. (III) Establishment of component technologies related to hydrogen production at HTGRs by 2030.

In accordance with Japan’s 6th strategic energy plan, the Japanese cabinet approved the ‘Basic Policy for the Realization of Green Transformation (GX)’ in February 2023 [5]. The GX basic policy includes initiatives for the next 10 years to concurrently attain decarbonization and economic growth. Restarting nuclear reactors with safety in mind will be prioritized in nuclear power, with the goal of reaching 20-22% of the country’s power mix by 2030. Concerning the development and construction of next-generation reactors, a concrete plan will be taken to rebuild the reactors on the premises of nuclear power plants that have been determined to be decommissioned.

The priority for the SMR type in Japan has not been officially decided. In Japan's 'Green Growth Strategy through Achieving Carbon Neutrality in 2050', the focus is placed on R&D of small modular light water reactor (LWR), HTGR, and fast reactor [6]. The Japanese government will actively support the efforts of Japanese companies in collaboration with overseas demonstration projects in countries such as the US and Canada, which aim to implement the small modular LWR by around 2030. In the 2030s Japanese nuclear industry will demonstrate the connective technologies of a carbon-free hydrogen production plant and the HTGR, and conduct the required verification for implementation. Hence, from the viewpoint of its high TRL, it is presumed that the small modular LWR will be deployed in the near future first, followed by the HTGR.

As for the HTGR, the OECD/NEA/CSNI loss of forced cooling (LOFC) project [7] is being fulfilled using JAEA's high temperature engineering test reactor (HTTR). Based on knowledge and experience gained through the LOFC project, CSNI is able to advance further technical reviews on HTGRs.

Japan's initiative 'NEXIP' (Nuclear Energy x Innovation Promotion) was launched in fiscal year 2019 to help accelerate the development of innovative nuclear technology. METI (Ministry of Economy, Trade and Industry) and MEXT (Ministry of Education, Culture, Sports, Science and Technology) are jointly moving forward with the NEXIP initiative.

The thermal-hydraulic study of an offshore floating nuclear power plant was among the researches covered in the NEXIP [8]. In the 1980s-2000s, a design study of the integral-type pressurized water reactor (PWR) 'MRX' (Marine Reactor X) was carried out based on experience of the nuclear ship 'MUTSU' [9]. The knowledge of MRX and MUTSU could be beneficial in examining the feasibility of an offshore floating nuclear power plant.

In the meantime, JAEA is providing support for the NEXIP. JAEA is conducting R&D on advanced reactors, such as HTGR and sodium-cooled fast reactor (SFR), to contribute to the construction of a technological basis for SMRs.

JAEA has executed safety demonstration tests for HTGR [10], experiments for taking helium gas at 950°C outside of the reactor pressure vessel (RPV) [11], etc. utilizing the HTTR with 30 MW thermal power. For this reason, JAEA has evaluated that the TRL for HTGR implementation is high. The major remaining issue is to perform experiments to demonstrate the utilization of heat (e.g., hydrogen production) coupled with the HTTR. The HTTR heat application test project was officially started in 2022, with the goal of establishing a safety design for the coupling of hydrogen production plant and HTGR by 2030. In case of a small modular HTGR deployment in Japan, the HTTR's experience will be instrumental in designing and constructing a demonstration reactor for the small modular HTGR. **Appendix A** describes the major features of the HTTR.

The TRL evaluation was executed in 2018 through the framework of the 'Fast Reactor Strategic Working Group' under Japan's METI for SFR technology and its fuel cycle technology, which were developed in Japan [12]. Most of SFR technologies were roughly evaluated to be at the level of TRL 6 (i.e., the technology demonstrated in a relevant environment) to TRL 7 (i.e., a system prototype demonstration in an operational environment).

JAEA has recently been examining a small reactor concept that combines SFR and molten salt thermal storage [13]. The possibility of sodium-water reactions in a steam generator (SG) has been eliminated. Seismic resistance has been improved through the use of floating seismic isolation buildings.

3.2 International Research Activities Relevant to SMR Safety

This section gives a summary of the main subjects that are driving international cooperation in the area of SMR safety.

(1) Small Modular LWR

a. In December 2021, GE Hitachi Nuclear Energy was selected by Ontario Power Generation as the technology partner for 'BWRX-300' at the Darlington site. BWRX-300 is a small BWR with an electric power of 300 MW class. Integral RPV isolation valves to mitigate the influences of a loss of coolant accident (LOCA) and an isolation condenser system (ICS) as a passive safety system are adopted to achieve high level safety and economy. Hitachi-GE Nuclear Energy, Ltd. has been working with GE Hitachi Nuclear Energy to develop BWRX-300.

b. Since 2016, Mitsubishi Electric Corporation [14] has participated in the development of PWR-based SMR 'SMR-160' with a rated electrical output of 160 MWe, which has been promoted by Holtec International in the US. Mitsubishi Electric Corporation is responsible for the design engineering of the instrumentation and control system for Holtec's SMR-160.

c. Since 2021, JGC Holdings Corporation and IHI Corporation have participated in the development of PWR-based SMR 'VOYGR' [15], which has been advanced by NuScale Power, LLC in the US. The VOYGR components (vessel, SGs, pressurizer, etc.) are housed in an integrated containment. The containment is about 4.5 m in diameter and about 23 m in height. Each module has an electrical power of 77 MWe. The design can accommodate up to 12 modules. In view of passive safety, cooling water circulates through the nuclear core by means of natural convection, eliminating the need for pumps.

d. Since 2022, the Japanese and US industries (JGC Holdings Corporation, IHI Corporation, Regnum Technology Group, and NuScale Power, LLC) have supported the SMR feasibility study in collaboration with the Ghanaian government through its agencies [16].

(2) HTGR

a. OECD/NEA/CSNI LOFC project [7] is being implemented using JAEA's HTTR. The purpose of the LOFC project is to demonstrate the safety inherent to HTGR and contribute to the international standardization of safety standards that reflect the HTGR safety characteristics necessary for its implementation.

b. JAEA has put forward a safety requirement for the design of HTGR, which was drafted under the Atomic Energy Society of Japan, and has been leading the discussions on SMR safety standards under the IAEA's Department of Nuclear Safety and Security. The results are summarized in the IAEA-TECDOC-1936 [17].

c. JAEA took part in the IAEA's Coordinated Research Project (CRP) on 'Development of Approaches, Methodologies and Criteria for Determining the Technological Basis' for the EPZ for SMR deployment, conducted in 2018-2021. The main objective of the participation is to organize the idea of eliminating the need for emergency evacuations by leveraging the HTGR's high safety. Deterministic selection methods for EPZ evaluation target events for the HTGR have gained international support. Thus, JAEA considers that sufficient consequences were attained through participation in the IAEA's CRP when reviewing the EPZ for the HTGR.

(3) Fast Reactor

a. Hitachi-GE Nuclear Energy, Ltd. has been considering the introduction of small modular SFR 'PRISM' (Power Reactor Innovative Small Module) [18] to Japan in collaboration with GE Hitachi Nuclear Energy.

b. Hitachi-GE Nuclear Energy, Ltd. is collaborating with the US's academia (i.e., UCB and UM) and the UK's academia (i.e., University of Cambridge) to benchmark nuclear analysis codes for light water-cooled fast reactor 'RBWR' (Resource-renewable BWR) [18].

c. On January 26, 2022, JAEA, Mitsubishi Heavy Industries, Ltd., and Mitsubishi FBR Systems, Inc. signed a memorandum of understanding (MOU) with TerraPower in the US, aimed at cooperating in the development of SFRs, including refueling machines and damaged fuel detection systems. The MOU has been expanded on October 31, 2023, as it became clear that Japan will begin conceptual design for fast reactor demonstration program in 2024. The revised MOU includes metal fuel safety, as well as an increase in size of the fast reactor conceptual design for enhanced cost competitiveness.

d. JAEA, Mitsubishi Heavy Industries, Ltd., and Mitsubishi FBR Systems, Inc. have collaborated with France on the development of design and related technologies for the advanced sodium technological reactor for industrial demonstration (ASTRID) project [19][20]. JAEA, Mitsubishi Heavy Industries, Ltd., and Mitsubishi FBR Systems, Inc. have continued to collaborate with France on R&D for SFR technology even after the ASTRID project was frozen in June 2018. In July 2023, a pool-type SFR proposed by Mitsubishi FBR Systems, Inc. was selected as the subject of demonstration SFR conceptual design. The pool-type SFR has been developed based on the experience of domestic SFR development and the outcome of the Japan-France collaboration.

(4) GIF

a. GEN IV International Forum (GIF) is being operated based on the technical reactor type (cooling method, neutron spectrum, etc.) and the technical classification (economy, non-electric application, etc.) [21]. GIF activities that closely relate to SMR include the ‘Economics Modelling Working Group’ and the ‘Non-Electric Application of Nuclear Heat Task Force’. GIF has already issued ‘Advanced Nuclear Technology Cost Reduction Strategies and Systematic Economic Review’ [22] and ‘Position Paper on Flexibility of GEN IV Systems’ [23]. In cooperation with other international organizations such as IAEA, GIF provides opportunities to explain the situation regarding SMRs from IAEA and understand the needs of the industry in the monthly GIF webinars (GIF webinar Series 43: Overview of Small Modular Reactor Technology Development [24]; GIF Forum with Industry 2022 [25]).

b. JAEA has been involved in the development of international safety design criteria for GEN IV reactors, such as SFR, lead-cooled fast reactor (LFR), and very high temperature reactor (VHTR), under GIF. NEA’s ‘Working Group on the Safety of Advanced Reactors’ has reviewed safety design criteria (SDC) and safety design guideline (SDG) for SFR and LFR, which were drafted under GIF.

3.3 Major Gaps in Knowledge for SMR Technologies

The evaluation of SMR safety research requires the identification and understanding of technical knowledge gaps. This section presents major gaps in previous knowledge concerning SMR technologies as far as possible.

(1) Small Modular LWR

There is no public information about major gaps between the technologies necessary for practical use and their current status for small modular LWRs being funded by Japan's NEXIP. The development of the small modular LWRs is ongoing to make sure their deployment.

(2) HTGR

JAEA has drafted a safety requirement for the design of HTGR under the Atomic Energy Society of Japan, taking account of the highly heat-resistant coated fuel particles and the inherent safety established by the HTTR [26]. According to the safety requirement of the HTGR design, there are major gaps between the technologies needed for practical use and their present status as follows.

a. For LWR, design extension conditions (DECs) are divided into DEC-A (i.e., no significant damage to fuel) and DEC-B (i.e., core melt). In contrast, HTGR has a single DEC category (i.e., no significant damage to fuel and core) because the design is required to prevent considerable damage to the fuel and core.

b. For LWR, it is postulated that the core melts while making efforts to maintain its integrity even in accidents. In contrast, HTGR must retain the confinement capability of radioactive substances and fuels even in accidents.

c. With regard to isolation of both water system and containment system in accidents, loss of coolant should be prevented for LWR. In contrast, for HTGR, requirements are made to inhibit water and air ingress into the primary cooling system, from the perspective of suppressing oxidation of core graphite in accidents specific to the HTGR.

(3) SFR

Major gaps between SFR technologies required for practical use and their current status, which include low TRL items through the TRL evaluation, are listed in the following.

I. Development of mass production technology for oxide dispersion strengthened cladding to achieve high burnup for economic improvement, and acquisition of related irradiation data.
II. Construction of data on structural materials with high temperature resistance under various conditions, including accidents.
III. Development of evaluation technology for thermal-hydraulic behavior of liquid metal sodium that enables compact reactor vessel. The subjects for the evaluation include thermal stratification in the reactor vessel, gas entrainment from the free liquid surface, sloshing in an earthquake, and natural circulation flow characteristics.

IV. Development of technology for seismic isolator for improving seismic resistance.
V. Acquisition of data and development of evaluation tools for damaged core behavior during severe accidents.
VI. Development of technology for prevention of core damage by virtue of its inherent reactivity characteristics or passive mechanisms (e.g., natural circulation cooling technology and passive safety technology).
VII. Development of measurement and maintenance technology used in sodium environments (e.g., sodium leak detector and sensor for under-sodium visualization).

3.4 Impacts of Multiple Module Configurations on SMR Safety

The output scale can be modified by modularization, which is a feature of SMR. This section represents typical safety-related matters that should be examined when modularizing.

a. In the case of a single module, precautionary action zone (PAZ) is small. In contrast, simultaneous disasters should be considered in the case of multiple module configurations according to the current regulatory assessment practice in Japan. Resultantly, the PAZ may be enlarged in response to the sum of the mass inventory of multiple reactors.

b. In general, the application of SMR should have no specific impact on safety cases, except for accidents or external hazards related to its application. When multiple reactors are densely located, the following matters should be considered in light of the lessons learned from the Fukushima Daiichi Nuclear Power Station accident. (i) Countermeasure against the simultaneous loss of cooling function of multiple reactors due to earthquake and tsunami. (ii) Issue with sharing equipment between reactor units. (iii) Difficulty in responding to severe accidents of multiple reactors simultaneously. Hence, it is needed to collect and analyze design information, including licensing experience for the design of multiple reactors [27].

3.5 Applicability of Knowledge Base to Support Safety of Water-cooled SMR Designs

The knowledge base is an important element in supporting the demonstration of reactor safety. This section outlines the main application examples and issues of the knowledge base that are being considered for water-cooled SMR designs.

a. Most of the systems and components of ‘BWRX-300’ [18] are built on proven technologies used in the advanced BWR (ABWR)/BWR fleet, and economic simplified BWR (ESBWR) which received design certification from USNRC. Therefore, the safety cases of BWRX-300 are an extension of those of current LWRs, and knowledge from LWRs can be applied to the BWRX-300. Verifications are required for technologies and components that are newly deployed with the BWRX-300. Integrated RPV isolation valves are a specific example. These valves can mitigate the influences of LOCA. The reactor concept with the integrated RPV isolation valves was approved on November 18, 2020 through a review of the Licensing Technical Report by USNRC. In addition, the BWRX-300 uses natural circulation for core cooling, and an ICS as a passive safety system. Although these technologies are adopted in the ESBWR and certificated by USNRC, there is a possibility of required verifications due to the lack of usage experience in Japan.

b. Mitsubishi Heavy Industries, Ltd. has been working on an integrated modular water reactor called 'Mitsubishi Small-PWR' [28]. The reactor integrated the PWR primary coolant pipe and the main components, such as a SG, into the reactor vessel. Accordingly, LOCA has been essentially eliminated. Additionally, the reactor coolant pump (RCP) is removed by employing two-phase natural circulation for primary coolant circulation in the reactor. As a result, the elimination of RCP locked rotor accidents and RCP seal LOCA has been substantially attained. In case of accidents (e.g., a main feedwater line break), passive heat removal is possible by utilizing the SG inside the reactor vessel. The feasibility of long-term heat removal with the passive heat removal system has been confirmed by virtue of experiments simulating a stand-alone direct heat removal system (SDHS).

3.6 Extent of Knowledge Base to Support Confidence in Safety Case of Advanced Technology SMRs

It may be difficult to develop the same level of knowledge base for the design of water-cooled SMRs regarding the design of advanced technology SMRs. This section states the main application examples and issues of the knowledge base that are being examined for the designs of HTGR and fast reactor.

(1) Common

It is crucial to practice systematic analysis tool development methodologies like the evaluation model development and assessment process (EMDAP). It is also essential to accumulate experience in verifying methodologies through international benchmarks, etc. as part of the practice.

(2) HTGR

The core of HTGR employs graphite that can withstand high temperatures. Even in case of accidents, the core temperature changes slowly owing to its large heat capacity, leading to no fuel failure (i.e., no core melt). Thus, the HTGR provides superior safety characteristics that are inherent. The inherent safety of the HTGR has been quantitatively demonstrated through safety demonstration tests [10] utilizing JAEA's HTTR.

(3) Fast Reactor[1]

a. For the small modular SFR ‘PRISM’ [18], the reactor vessel auxiliary cooling system (RVACS) is an important passive safety system for decay heat removal, which requires the provision of long-term core cooling in case of accidents. Along with the use of metallic fuel, which has greater safety features, it enables small modular SFRs that minimize initial investment and have a high level of inherent safety and reliability without requiring a power supply or human operation. The RVACS uses natural circulation of air outside the containment vessel to remove decay heat from the core. Verification is needed for the RVACS, which is a technology peculiar to the PRISM, in addition to technologies that are relevant to conventional SFRs.

b. The particle type metallic fuel sodium-cooled reactor [29] utilizes the decay heat removal using the secondary sodium system employed for conventional SFRs, on the basis of the established technology. The particle type metallic fuel has fission gas plenum below the active core region and sodium plenum is located just above the active core region so that neutrons easily leak in abnormal events, giving rise to higher inherent safety.

c. The core of the light water-cooled fast reactor ‘RBWR’ [18] has been rearranged to optimize the burning of trans-uranium elements. Meanwhile, commercially proven BWR technologies are being employed for components such as turbines and safety systems other than the core.

3.7 SMR Issues/Topics Requiring Specific Attention Compared to Safety Cases of Current Designs

The safety cases of designs for SMRs differ significantly from those for existing LWRs. This section shows the draft safety requirements being considered for the designs of HTGR and fast reactor.

(1) Common

In the case of a multi-module configuration, it is key to assume that an accident occurs owing to a common cause. This is probably due to external hazards such as earthquake and tsunami. If there is insufficient resistance to the common cause, the establishment of PAZ may be needed assuming simultaneous disasters of multiple modules.

(2) HTGR

JAEA has drafted a safety requirement for the design of HTGR under the Atomic Energy

[1] *In July 2023, a pool-type SFR proposed by Mitsubishi FBR Systems was selected as the subject of demonstration SFR conceptual design.*

Society of Japan [26]. The major characteristics of the HTGR safety design requirement are as follows. DEC is defined as a single condition category (i.e., no significant damage to fuel and core). A requirement is made to maintain the confinement capability of radioactive substances and fuels even in accidents. Accidents specific to HTGR (e.g., air-ingress accident and water-ingress accident) should be taken into consideration when evaluating the oxidation of the core graphite in the accidents. Furthermore, leakage of combustible gas or toxic gas from the hydrogen production facility is one of the events that should be considered for safety when connecting the hydrogen production facility, etc. to the HTGR. Therefore, as the safety requirement, it should be taken into account to ensure the safety of the HTGR against the leakage of combustible gas or toxic gas.

(3) Fast Reactor

JAEA has drafted a safety requirement for the design of SFR under the Atomic Energy Society of Japan, which was reflected in SDC and SDG for GIF SFR [30]. The safety evaluation events for SFR are quite different from those for existing LWRs. Analytical technologies for the SFRs have been developed over many years of R&D, resulting in the accumulation of usage experience. The analytical methodologies for design basis accidents have almost been completed. Hence, it is considered that international benchmarks and similar methods are effective in developing evaluation methodologies for DECs, sodium combustion, and sodium-water reaction. The DECs include reactor shutdown due to inherent reactivity characteristics or passive mechanisms, and the progress of core damage accidents. IAEA is implementing several international benchmarks for fast reactors as CRPs [31]. It is thus necessary to set issues that do not overlap with the IAEA's CRP activities.

4. Japan’s Main Responses to EGSMR’s Second Questionnaire Related to TOP-21

The TOP-21 determined the following four major areas of interest by discussing the responses from the participating organizations to the EGSMR’s first questionnaire. The EGSMR’s second questionnaire, which was denoted in July 2023, is obligated to review and refine the recommendations to CSNI.

(i) Forster international regulatory harmonization and standardization regarding SMR licensing, which includes collaboration with IAEA to collect information on phenomena.
(ii) Support understanding of broad cross-cutting SMR issues, which include understanding of safety implications on interconnections between SMRs and associated heat application facilities.
(iii) Organize international experimental campaigns to improve phenomenological understanding and experimental validation of SMR designs, which include collection of new data and aggregation of past data for water-cooled SMRs and advanced technology SMRs.
(iv) Support international benchmark activities for computer modelling V&V for SMR designs, which include collecting data to qualify passive systems and providing recommendations on reliability requirements.

Not all responses from Japan against the EGSMR’s second questionnaire are explained in the TOP-21. This chapter mentions the contents arranged on the basis of the technical information by Japan in accordance with the EGSMR’s second questionnaire.

4.1 Correspondence with IAEA's Broad Guidelines for Advanced Reactors

While IAEA has established safety standards for existing large LWRs, they have not yet been prepared for advanced reactors and SMRs. This section explains the review status with regard to the development of safety standards.

a. JAEA has been involved in the development of international safety design criteria for GEN IV reactors, such as SFR, LFR, and VHTR, under GIF. Both NEA’s ‘Working Group on the Safety of Advanced Reactors’ and IAEA have been reviewing the SDC and SDG for SFR and LFR, which were drafted under GIF.

b. IAEA analyzes the applicability of existing safety standards to advanced reactors and establishes safety reports. In the future, it would be needed to recognize a gap regarding the applicability of IAEA safety standards to advanced reactors such as HTGR and SFR. Additionally, IAEA is planning to collect information on actual examples of advanced reactor design to utilize as reference when applying the IAEA safety standards. In case of cooperating with such an IAEA plan, it is preferable to collect design information for advanced reactors and analyze their compliance with IAEA safety standards. GIF, research institutes in participating countries involved in the development of advanced reactors, and vendors for advanced reactors are the sources of information.

4.2 Safety Implications on Interconnections between SMRs and Associated Heat Application Facilities

SMR applications are being examined not only for power generation, but also for heat supply to industrial processes, hydrogen production, district heating, and other related uses. This section discusses safety considerations for the connection of hydrogen production facility.

JAEA plans to establish a safety design for HTGR through the licensing the HTTR related to the connection of the hydrogen production facility [32][33]. The safety design enables the application of general industrial regulations, that is, the High Pressure Gas Safety Act to hydrogen production facilities in response to requests from Japanese industry. In the HTTR, the high-temperature primary helium gas extracted from the reactor transfers heat to the secondary helium gas via an intermediate heat exchanger (see **Fig. A-1**). The secondary helium gas passes through newly installed high-temperature helium piping and supplies reaction heat to the hydrogen production facility to produce hydrogen. When designing, manufacturing, and managing hydrogen production facilities based on the High Pressure Gas Safety Act and its technical standards, the following events should be considered.

A. Variation in heat removal amount and increase in load of the secondary helium cooling system due to abnormality in the hydrogen production facility.
B. Ingress of combustible gas into the reactor building through the secondary helium piping.
C. Fire or explosion triggered by combustible gas leaking into the environment.
D. Poisoning of operators at control room owing to toxic gas leaking into the environment.
E. Impact of hydrogen production facility abnormalities caused by external events on nuclear reactor facilities.

In addition, the influence of abnormalities at hydrogen production facilities on nuclear reactor facilities is planned to be assessed as an external event under the Nuclear Regulation Act.

The verified technology for connecting to the hydrogen production facility utilizing the HTTR can be applied not only to the HTGR but also to other types of reactors.

4.3 Effective Utilization of Experimental Data for SMR Designs

Experiments form the foundation of evidence to demonstrate the safety of SMRs, and thus the effective use of experimental data in SMR designs is emphasized. This section points out important notices when exploiting test data for SMR designs.

a. In conducting safety analyses for SMRs, it is essential to validate the predictive capability of thermal-hydraulic codes leveraging experimental data under a wide range of conditions. It is also necessary to elucidate thermal-hydraulic phenomena involved through experiments that simulate systems like passive safety systems relevant to water-cooled SMRs and advanced technology SMRs, utilizing integral thermal-hydraulic test facilities. There are elements such as comparisons between calculations and data acquired from experiments simulating the whole system and component(s) of the SMR which are required to validate the thermal-hydraulic code for the SMR safety analysis. **Appendix B** shows the preparation procedure for the thermal-hydraulic code for the safety analysis.

b. To support SMR safety research, experimental data have been obtained utilizing the Japan's test facilities (e.g., experimental facilities for small modular LWRs, experimental HTGR, experimental fast reactor, and experimental facilities for SFR). Annex C of the TOP-21 [3] provides details of these test facilities. Each country's experimental facilities may have different scaling ratios and scaling methods to target reactors. It is desirable to verify the methodology for scalability of phenomena of interest by sharing, aggregating, and comparing experimental databases gained from these test facilities, which are open to publics. The open data contain data that has been acquired from associated research articles and reports, frameworks of IAEA's CRPs, and exercises of OECD/NEA/CSNI International Standard Problems.

c. As for GEN IV reactors (e.g., SFR and VHTR/HTGR), the participating countries are promoting the use of the relevant experimental data in manners such as the framework of GIF, bilateral cooperation between Japan and France, and cooperation with IAEA.

4.4 Reliability Requirements of SMR Passive Safety Systems

Passive safety is an important element in improving the safety of nuclear reactors, and is also taken into account for water-cooled SMRs and advanced technology SMRs. Such passive safety is highly dependent on the SMR designs. The main examples of passive safety being considered are represented.

a. Inherent reactivity characteristics and passive mechanisms are subject to the uncertainty of physical phenomena. This is because they do not require detection systems or utility power, and they naturally function as physical phenomena in response to system state changes. Therefore, it is necessary to evaluate the reliability by analyzing the propagation of uncertainty in physical phenomena. Benchmarking, information collection and analysis of the inherent reactivity characteristics and passive mechanisms could be among the major themes. Accident simulation experiments conducted by installing adopted passive safety system in integral thermal-hydraulic test facilities will be effective in demonstrating the passive safety system. **Appendix C** gives an outline of experiments performed with the large scale test facility (LSTF) on the passive safety systems in the 1990s.

b. ‘BWRX-300’ [18] employs natural circulation to cool the core, and an ICS as a passive safety system. Although ESBWR has adopted these technologies and received design certification from USNRC, design verification may be required owing to the lack of operational experience in Japan. To evaluate the coolant flow rate and heat removal performance under accident conditions, it is necessary to use experimental data or analysis methods verified by experiments. Nuclear regulatory authorities must approve the evaluation method.

c. ‘Mitsubishi Small-PWR’ [28] can remove heat passively by utilizing SG as a large-scale heat exchanger inside the reactor vessel in accidents such as a main feedwater line break. The passive SG heat removal system has a closed loop with the SG, and the heat exchanger is located in the pool outside the reactor vessel. The core decay heat is released to the atmosphere by means of natural circulation in the closed loop through the heat exchanger in the pool. In the early stage with a high decay heat level, heat is removed by cooling the water, causing a decrease in the pool water level. In the long term, when the decay heat level lowers, heat is removed by air cooling to cool the core. In addition, the reliability requirements of the passive SG heat removal system include adequate heat removal capacity during water and air cooling, the duration of heat removal, and the stability of the system (i.e., the stability of the coolant’s natural circulation). The reliability requirements have been confirmed through element experiments, which indicate that heat removal is feasible during air cooling as the most severe condition, and the evaluation method has been verified. The method for evaluating heat removal performance is subject to regulatory approval.

d. In the SMR designs, it is essential to unify the methodology for evaluating features such as heat removal performance based on the experimental data or analysis methods, which will lead to licensing. To achieve this, it is crucial for Technical Support Organizations, research institutes, and vendors to coordinate and advance discussions.

5. Summary

The investigation consequences for Japanese SMR R&D were summed up, centering on the EGSMR activities (2022-2023). Japan's responses to the following items in the EGSMR's first questionnaire are organized.

(i) SMR deployment landscape.
(ii) International research activities relevant to SMR safety.
(iii) Major gaps in knowledge for SMR technologies.
(iv) Impacts of multiple module configurations on SMR safety.
(v) Applicability of knowledge base to support safety of water-cooled SMR designs.
(vi) Extent of knowledge base to support confidence in safety case of advanced technology SMRs.
(vii) SMR issues/topics requiring specific attention compared to safety cases of current designs.

Regarding item (iii), particularly for technologies of advanced reactors, the draft safety requirement for the HTGR design was created by the Atomic Energy Society of Japan. HTGR has a single DEC category (i.e., no significant damage to fuel and core) because the design is necessary to avoid markedly damage to the fuel and core. HTGR is also required to maintain the confinement capability of radioactive substances and fuels even in accidents. Furthermore, from the perspective of suppressing the oxidation of core graphite in accidents peculiar to HTGR, there are requirements for inhibiting water and air ingress into the primary cooling system. Major gaps include the development of the following SFR technologies involving low TRL items. (I) Mass production technology for oxide dispersion strengthened cladding. (II) Evaluation technology for thermal-hydraulic behavior of liquid metal sodium. (III) Technology for seismic isolator. (IV) Technology for prevention of core damage by virtue of its inherent reactivity characteristics or passive mechanisms. (V) Measurement and maintenance technology used in sodium environments.

Additionally, Japan's responses to the following items of the EGSMR's second questionnaire involving major areas of interest identified by the EGSMR are arranged.

(a) Correspondence with IAEA's broad guidelines for advanced reactors, which is relevant to regulatory harmonization.
(b) Safety implications on interconnections between SMRs and associated heat application facilities, which are relevant to cross-cutting safety issues.
(c) Effective utilization of experimental data for SMR designs, which is relevant to experimental campaigns.
(d) Reliability requirements of SMR passive safety systems, which are relevant to benchmarking for computer code V&V.

Especially, concerning item (b), a safety design for HTGR is planning to establish through the licensing the HTTR relevant to the connection of the hydrogen production facility. The safety design enables the application of general industrial regulations (i.e., the High Pressure Gas Safety Act) to hydrogen production facilities. The following events should be examined for hydrogen production facility.

(A) Variation in heat removal amount and increase in load of the secondary helium cooling system due to abnormality in the hydrogen production facility.
(B) Ingress of combustible gas into the reactor building through the secondary helium piping.
(C) Fire or explosion triggered by combustible gas leaking into the environment.
(D) Poisoning of operators at control room owing to toxic gas leaking into the environment.
(E) Impact of hydrogen production facility abnormalities caused by external events on nuclear reactor facilities.

Additionally, the effect of abnormalities at hydrogen production facilities on nuclear reactor facilities is planned be evaluated as an external event under the Nuclear Regulation Act.

In relation to item (i) of the EGSMR's first questionnaire, there are currently no concrete plans for SMR deployment in Japan over the next 10 years. Meanwhile, through a variety of initiatives according to Japan's 6th strategic energy plan and the basic policy for the realization of GX, the following are thought to be key benefits that SMRs offer to Japan in the future deployment of nuclear energy.

(1) The SMR is expected to be a promising nuclear technology for revitalizing the nuclear power industry.
(2) The accumulation of technological basis, including advanced reactors (HTGR, SFR, etc.), is being advocated by nuclear companies and research institutes.
(3) Generally, the maintenance and development of the world's leading human resources are expected, while inheriting nuclear technologies that have been cultivated over many years.

Consequently, it should be considered that it is important to promote R&D on SMR while leveraging international collaboration, with the aim of contributing to carbon neutrality in the future.

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Appendix A Major Features of HTTR

Table A-1 compares the major features of the high temperature engineering test reactor (HTTR) and pressurized water reactor (PWR). Coolant is helium gas for the HTTR, which is chemically inert and has no phase change during operation, and water for the PWR. The coolant temperature at the reactor outlet is 950°C (maximum) in the HTTR, and 325°C in the PWR. The coolant pressure is 4 MPa in the HTTR, and 15.5 MPa in the PWR. For the HTTR, the fuel is low-enriched UO_2 with tri-isotropic coatings of SiC and PyC because of its good retaining performance for fission products at high temperature. In contrast, for the PWR, the fuel is low-enriched UO_2 with clad of Zircaloy-4. The material of the moderator and reflector is graphite for the HTTR which has a large heat capacity and excellent thermal resistance, and water for the PWR. The control rod (CR) clad is made of ferritic superalloy with excellent thermal resistance for the HTTR, and stainless steel for the PWR.

As a unique CR insertion method for the HTTR, two-step CR insertion is adopted to avoid CR replacement at the reactor scrams [A-1]. The outer nine pairs of CRs in the replaceable reflector region are inserted into the core first. The inner seven pairs of CRs in the fuel region are inserted into the core 40 min after the outer nine pairs of CRs are inserted or when the reactor outlet coolant temperature becomes less than 750°C.

Figure A-1 illustrates the schematic diagram of the reactor cooling system of the HTTR. The main cooling system consists of the primary cooling system, secondary helium cooling system, and pressurized water cooling system [A-2]. Two independent vessel cooling systems indirectly cool the reactor core during non-forced cooling accidents in which the auxiliary cooling system is no longer able to cool the core [A-3]. Each vessel cooling system is composed of a water-cooling loop and cooling panels located around the reactor pressure vessel (RPV). The cooling panels cool the RPV by means of radiation and natural convection and remove the decay heat from the reactor core during non-forced cooling accidents.

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Table A-1 Major features of HTTR and PWR

Items	HTTR	PWR
Reactor outlet coolant temperature	950°C (maximum)	325°C
Coolant pressure	4 MPa	15.5 MPa
Fuel	Low-enriched UO_2 with tri-isotropic coatings of SiC and PyC	Low-enriched UO_2 with clad of Zircaloy-4
Moderator and reflector	Graphite	Water
Coolant	Helium gas	Water
Control rod clad	Ferric superalloy	Stainless steel

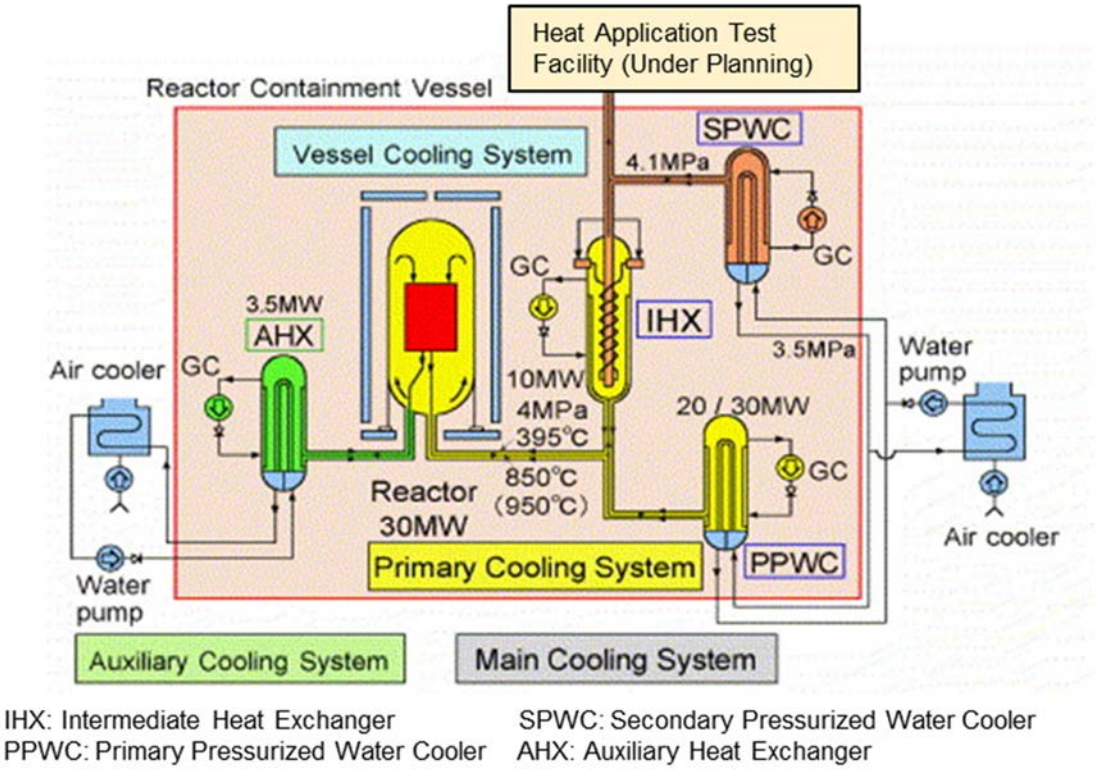

Fig. A-1 Schematic diagram of reactor cooling system of HTTR

Appendix B Preparation Procedure for Thermal-hydraulic Code for Safety Analysis

Safety evaluations are being pursued making use of thermal-hydraulic codes and other codes to obtain licenses of nuclear power reactors. Safety margin is the distance between the regulatory acceptance criterion and the safety variable value at which the system or barrier stops functioning [B-1]. The evaluation model (EM) code and best-estimate (BE) code are included in the thermal-hydraulic codes. The conventional EM code safety analysis employs conservative assumptions for the initial and boundary conditions to provide conservative results. Accordingly, the EM code is not suitable for quantitatively estimating the safety margin. In contrast, the BE code on the basis of models such as two-fluid model is utilized under realistic assumptions for the initial and boundary conditions. The purpose of best-estimate plus uncertainty (BEPU) is to quantitatively evaluate the safety margin using statistical treatment of uncertainties. The BEPU approach, which is considered a mature technology [B-2], is expected to be used in SMR designs. To assess the predictive capability of the BE code, it is important to utilize data obtained from experiments conducted under a wide range of conditions.

Figure B-1 presents the elements necessary for validation of the thermal-hydraulic code for safety analysis. Experiments employing scale-downed thermal-hydraulic test facilities are roughly classified into an integral effect test (IET) and a separate effect test (SET) [B-3]. The IET and SET simulate the entire system and component(s) of the targeted reactor, respectively. The posttest analysis of the IET is carried out using the thermal-hydraulic code with two-phase models and equations under the identical condition as the IET. If the calculated results of the thermal-hydraulic code are not in agreement with the IET results, experimental data through the SET should be acquired to improve the analytical models and correlations, paying attention to differences in scaling. If the thermal-hydraulic code is able to reproduce the consequences of IET, it is considered 'completion of thermal-hydraulic code'. The reactor safety assessment requires appropriate measures for input data, nodalization, selection of physical models, etc. for the thermal-hydraulic code.

Reference

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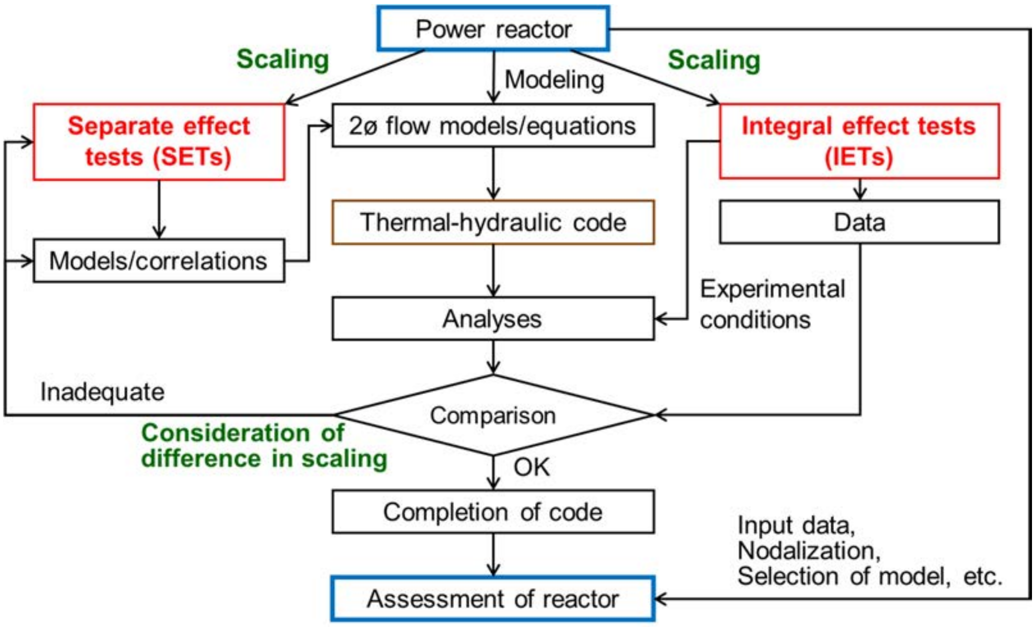

Fig. B-1 Elements needed for validation of thermal-hydraulic code for safety analysis

Appendix C Outline of Experiments with LSTF on Passive Safety Systems in 1990s

In the 1990s, the large scale test facility (LSTF) [C-1] was employed to conduct many experiments on small-break loss-of-coolant accidents (SBLOCAs) utilizing the rig-of-safety assessment (ROSA)/advanced passive 600 MWe reactor (AP600) testing program. The LSTF, built originally to simulate 4-loop PWR, was modified by adding components that were specific to the Westinghouse AP600 design [C-2], as viewed in **Fig. C-1**. The modified LSTF provides a full pressure, full height, and 1/30.5 volumetrically-scaled simulation of AP600. A passive residual heat removal (PRHR) system in the Westinghouse AP600 advanced passive reactor design is a natural circulation-driven heat exchanger cooled by water in an in-containment refueling water storage tank (IRWST). The LSTF test results confirmed the AP600 passive safety components generally operated as designed, producing the core cooling being maintained. The final design approval (FDA) for the AP600 standard design was granted to Westinghouse Electric Company by USNRC in 1999 [C-3].

In the 1990s, furthermore, thermal-hydraulic research was executed utilizing the ROSA/LSTF on PWR designs that adopted passive safety systems [C-4]. The passive safety systems include the secondary-side automatic depressurization system (SADS) of the steam generator (SG) and the gravity-driven safety injection system (GDIS), as seen in **Fig. C-2**. The SADS depressurizes the primary system via natural circulation cooling, while employing the SG secondary-side as a heat removal source. The GDIS has a tank that holds water at ambient temperature and pressure. The top of the tank is open to the atmosphere. An injection line connects the bottom of the tank to the vessel downcomer or cold leg. Various experiments simulating SBLOCAs were performed with the ROSA/LSTF. The thermal-hydraulic behaviors of passive safety systems were revealed in the LSTF test results. The primary pressure was successfully decreased to the GDIS injection pressure of 0.2 MPa by only the SADS. Long-term passive core cooling was ensured by the GDIS injection. Consequently, the effectiveness of the combined use of SADS and GDIS was verified experimentally.

References

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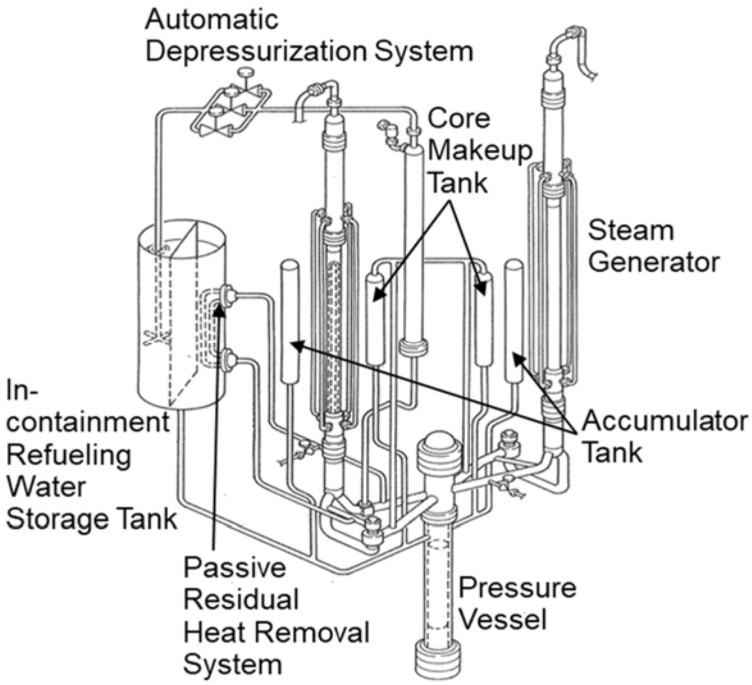

Fig. C-1 Schematic diagram of LSTF with passive safety systems for AP600

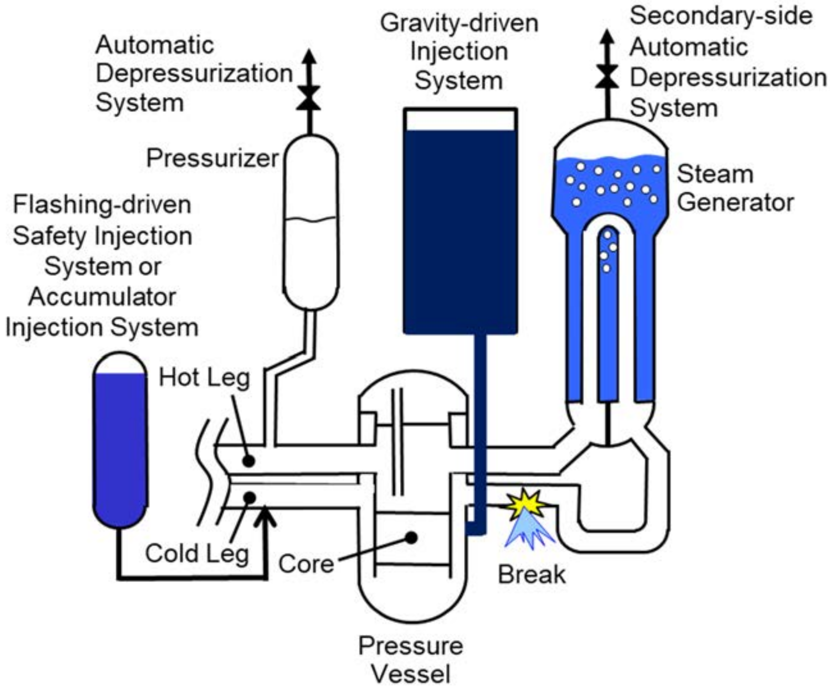

Fig. C-2 Schematic diagram of LSTF with passive safety systems for PWR