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Design Concepts and Safety Concerns of the Small and Medium Size Reactors (SMR)

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Abstract

The small and medium size reactors (SMR) and interface facilities such as desalination plant are expected to be located near the population area because of restrictions in transporting the plant products such as fresh water to long distance area. To protect the public around the plant facility from the possible release of radioactive materials, the design development of the SMR is focusing on an enhancement of the safety and reliability as well as the economics. In this study, the major safety concepts of the SMR designs significantly different from the current PWR designs are investigated and the safety concerns applicable to the integrated SMR design of Korea (called SMART), were identified. Those safety issues include the use of proven technology, application of strengthening defense in depth, event categorization and selection, simplification of emergency planning, determination of accident source terms and so on. The efforts to resolve the safety concerns in the design stage will provide an improvement of the safety of the SMART design.

I. Introduction

The small and medium size reactors (SMR) are developed worldwide for various application purposes such as district heating, seawater desalination, steam production for industrial use, nuclear ship propulsion, as well as electricity production. The power range of the SMR is generally considered as less than about 700 MW. In applying the SMR to the district heating or seawater desalination, there are some restrictions in transporting the plant products such as the hot steam or fresh water to long distance area because the construction cost of the transport system is too high. Thus, the SMR and interface facilities are expected to be located near the population area. It implies that the public around the plant facility should be protected in depth from the possible release of radioactive materials under any plant condition. In addition, the heat grid or plant products should be prevented from radioactivity contamination. To meet these requirements, the SMR designs are focusing on an enhancement of the safety and an improvement of the reliability as well as the economics. In particular, the SMR designs adopt extensively inherent safety characteristics and passive safety concepts using natural forces such as natural convection, gravity or stored energy. The SMR designs being developed worldwide are summarized in Table I, which represents the development status and the design characteristics of the PWR-typed SMR [1]. Russia and Italy are developing the various thermal power ranges of SMR designs to apply to the district heating and seawater desalination. Especially, NIKA of Russia with 70 to 300 MWt [2, 3] and NILUS of Italy with 50 to 1000 MWt [4, 5] are adopting extensively the inherent safety concepts. The marine reactor (MRX) of Japan [6] for the propulsion of nuclear ship, the NHR of China [7] and CAREM of Argentina [8] for the district heating or seawater desalination are in basic design stage. Korea is also developing a system integrated modular advanced reactor (SMART) with 330 MWt. In this study, based on the analysis of SMR designs adopting significantly different from the current light water reactors (LWR), the major safety concepts of the PWR-typed SMR designs are investigated, and the safety concerns applicable to the integrated SMR design of Korea, SMART, are identified. The efforts to resolve the safety concerns in the design stage will provide an improvement of the safety of the SMART design.

Table I. Development Status and Design Characteristics of PWR-typed SMR

Design Name	Thermal Power	Reactor Type	Core Cooling	SG Type	Refueling Period	Design Stage	Designer
AP 600	1,940 MWt	Loop	Forced	U-tube	24 months	Detailed	Westinghouse
QP 300	1,000 MWt	Loop	Forced	U-tube	12 months	Detailed	China, SNERDI
AST-500	500 MWt	Integral	Natural	OT tube	2 years	Detailed	Russia, OKBM
KLT-40	40 MWt	Loop	Forced	Coil OT tube	2-3 years	Detailed	Russia, OKBM
PIUS	2,000 MWt	Integral	Forced	OT tube	12 months	Basic	Sweden, ABB
NHR-200	200 MWt	Integral	Natural	U-tube	3 years	Basic	China, INET
CAREM 25	100 MWt	Integral	Natural	OT tube	13 months	Basic	Argentina, CNEA
MRX	100 MWt	Integral	Forced	Coil OT tube	44 months	Basic	Japan, JAERI
ABV	38 MWt	Integral	Natural	OT tube	4-5 years	Basic	Russia, OKBM
VPBER-600	1,800 MWt	Integral	Forced	OT tube	18 months	Conceptual	Russia, OKB ME
SPWR	1,800 MWt	Integral	Forced	Coil OT tube	2 years	Conceptual	Japan, JAERI
SIR	1,000 MWt	Integral	Forced	OT tube	2 years	Conceptual	UK and USA
ISIS	650 MWt	Integral	Forced	Coil OT tube	18 months	Conceptual	Italy, ANSALDO
ATS 150	536 MWt	Integral	Natural	OT tube	2 years	Conceptual	Russia, EMBDB
MARS	600 MWt	Loop	Forced	OT tube	17 months	Conceptual	Italy, ENEA
RUTA NHP	20 MWt	Integral	Natural	Coil OT tube	5 years	Conceptual	Russia, RDIPE
SAKHA-92	7 MWt	Integral	Natural	Coil OT tube	20-25 years	Conceptual	Russia, OKBM
UNITHERM	17 MWt	Integral	Natural	Coil OT tube	20 years	Conceptual	Russia, RDIPE
NILUS	50/200/1000	Integral	Natural	OT tube	5 years	Conceptual	Italy, PDM
SMART	330 MWt	Integral	Forced	Coil OT tube	4.5 years	Conceptual	Korea, KAERI
NIKA	70/300 MWt	Integral	Natural	Coil OT tube	4.5-5 years	Conceptual	Russia, RDIPE

ISIS: Inherently Safe Immersed System
SMART: System Integrated Modular Advanced Reactor
MARS: Multipurpose Advanced Reactor
AP : Advanced Passive
NILUS: Natural Circulation Integrated Layout Ultimate Safety Reactor
NHR: Nuclear Heating Reactor
MRX: Marine Reactor X
PIUS: Process Inherent Ultimate Safety
OT tube: Once Through Tube

II. Design and Safety Characteristics of the SMR

II.1. Integrated Arrangement of the Reactor Coolant System

Almost SMR are designed with integrated arrangement of main components such as pressurizer, main circulation pumps (MCP), and steam generators (SG) as shown in Table I. This integrated arrangement simplifies the reactor coolant system (RCS) and allows elimination of large pipes such as hot leg and cold leg. Especially, in case of adopting natural circulation system as core cooling mode, the MCP are also eliminated. Therefore, large break LOCA or MCP failures, which is considered as important design basis accidents in current LWR design, are excluded. In addition, the reactor pressure vessel (RPV) is relatively larger in size as the main components are installed inside the RPV. The reactor vessel penetrations are also minimized and located on the upper part of the RPV. These features provide the sufficient amount of the reactor coolant to cover the core in any LOCA condition. Also, in case that the control rod drive mechanisms (CRDM) are placed inside the RPV, the control rod rejection accident would be excluded. Besides, the core reactivity is controlled by control rods instead of boron solution during normal operation and the primary system pressure is self-controlled by a partial pressure of nitrogen and saturate steam corresponding to the core outlet temperature. Therefore, the boron control system is simplified and the pressurizer heater and spray to control the system pressure are eliminated. Also, in the SG, the coiled once-through typed tubes are adopted to enhance the heat transfer from the primary coolant to secondary side, instead of the current type of U-tube. As a result, these integrated RCS design significantly different from the current LWR design provide an improvement of safety and reliability of the SMR systems.

II.2. Core Cooling by Natural Circulation

There are two types of reactor coolant circulation system in cooling the core, one is natural circulation system using the density gradient and the other is forced circulation system using the MCP. In general, the natural circulation system is possible due to the very low head loss of the integrated primary system and it allows the simple reactor design and easy operation. However, the natural circulation system requires relatively higher reactor size to establish the sufficient natural circulation flow. Meanwhile, the forced circulation system allows more efficient and compact reactor design although the RCS becomes complicated. In recent, based on the core power of the integrated SMR, the economic performance was analyzed [8]. It indicates that the optimized parameters to minimize the cost of electric generation, system pressure and temperature and geometry dimension of SG tube and fuel rod are strongly dependent on the coolant circulation system. It also indicates that, for the smaller reactors than about 330 MWt, the advantage of forced circulation system is not enough to overcome the simplicity of natural circulation system and, for the larger reactors than about 500 MWt, the natural circulation system gives too low efficiency. As shown in Table II, the several reactors of less than about 300 MWt adopt the natural circulation system for the core cooling and then the reactor vessel size becomes higher and slimmer than those with forced circulation system. In addition, the SMR designs adopting the natural circulation system has smaller core flow rate than that of the forced circulation system because the temperature difference between the core inlet and outlet is high at the same thermal power. For example, the CAREM with the natural circulation system has about three times lower flow rate than that of the MRX. As a result, the natural circulation system with no RCP could reduce a potential for the rapid loss of flow in the core, but it has also disadvantage that the construction cost increases because of the larger size of reactor.

II.3. Adoption of Passive Safety Concepts

The passive system is to actuate a system using a natural driving force such as natural convection, gravity, or stored energy. In general, the passive systems could be classified as two groups, one is a system activated only by the process parameter variation without operators' intervention and the other is a system activated by opening a valve at certain set point. The core cooling or the residual heat removal system using the natural circulation is included in the first group. The pressure self-control using nitrogen solubility in the pressurizer is also included in this passive concept. The performance of these systems are generally confirmed by various tests or experiments. The second group of passive systems includes the safety systems to inject the coolant into the RCS by the compressed gas force and gravity. The CRDM to drop the control rods by the spring force and gravity into the core region is also included in this group. For the second group of passive systems, an inadvertent actuation of the passive component is generally analyzed to ensure the safety of SMR design.

Table II. Core Cooling Mode and Operating Conditions of Integrated SMR

Design Name	Thermal Power [MWt]	Core Cooling Mode	RPV Height/Dia [m]	Core Power Density [kW/l]	Prim/Secd Pressure [MPa]	Core ΔT [°C]	Core Temp Inlet/Outlet [°C]	Core Flow Rate [kg/s]
PIUS	2,000	Forced	58.0/12.2	72.0	9.0/4.0	30	260/290	13,000
VPBER-600	1,800	Forced	20.2/5.97	69.4	15.7/6.38	31	294/325	10,140
SPWR	1,800	Forced	29.0/6.6	65.1	13.8/5.6	26	288/314	12,300
SIR	1,000	Forced	23.8/5.8	54.6	15.5/-	24	294/318	7,500
ISIS	650	Forced	26.5/4.9	70.0	14.0/-	39	271/310	2,911
SMART	330	Forced	10.2/4.1	62.6	15.0/3.0	40	270/310	1,556
MRX	100	Forced	9.4/3.7	42.0	12.0/4.0	15	282/297	1,250
NILUS	1000	Natural	24.5/4.56	29.0	15.5/6.5	40	289/329	4,303
ATS 150	536	Natural	16.7/5.3	38.5	15.8/4.5	75	265/340	-
AST-500	500	Natural	16.4/4.82	27.0	1.96/1.2	77	131/208	1,548
NILUS	200	Natural	13.7/3.32	29.0	15.5/6.5	36	293/329	938
NHR-200	200	Natural	13.6/5.0	36.2	2.5/3.0	59	154/213	640
CAREM 25	100	Natural	11.0/2.84	55.0	12.25/4.7	42	284/326	410
NIKA	70	Natural	-	40.0	15.0/3.0	40	260/300	-
NILUS	50	Natural	12.8/2.4	29.0	2.5/4.8	34	182/216	329
ABV	38	Natural	4.8/2.6	43.0	15.4/3.14	82	245/327	85
RUTA NHP	20	Natural	15.0/4.8	16.8	0.1/0.4	35	60/95	136
UNITHERM	17	Natural	-	15.0	16.0/3.6	75	255/330	42
SAKHA-92	7	Natural	4.23/1.86	17.1	14.0/3.2	32	304/336	-

In general, the passive concept simplifies the plant design and operation and provides high reliability of the systems. Therefore, the SMR designs adopt extensively the passive safety features as much as possible, especially in removing the decay heat under accident condition. Two types of passive residual heat removal system (PRHRS), the RPV dry type adopting in SMART, NIKA, NHR designs and the RPV pool type adopting in DRX, PIUS, ISIS designs, are applied as shown in Fig. 1. In the case of RPV dry type, the decay heat is transferred from the core to containment through SG tube or independent heat exchanger by the natural circulation in the RVP. The residual heat transferred to the containment is removed ultimately to atmosphere by using steam condensation on inside containment wall and natural convection on outside containment, which is called passive containment cooling system (PCCS). In the case of RPV pool type, the decay heat is transferred from the core to the containment water through the SG tube or the RPV wall by the natural circulation. It is ultimately transferred to atmosphere by the natural circulation of the containment water, which is called containment water cooling system (CWCS). In general, the RPV pool type of PRHRS has advantages that the decay heat is removed without emergency water injection into the RCS because the primary coolant flow out is limited.

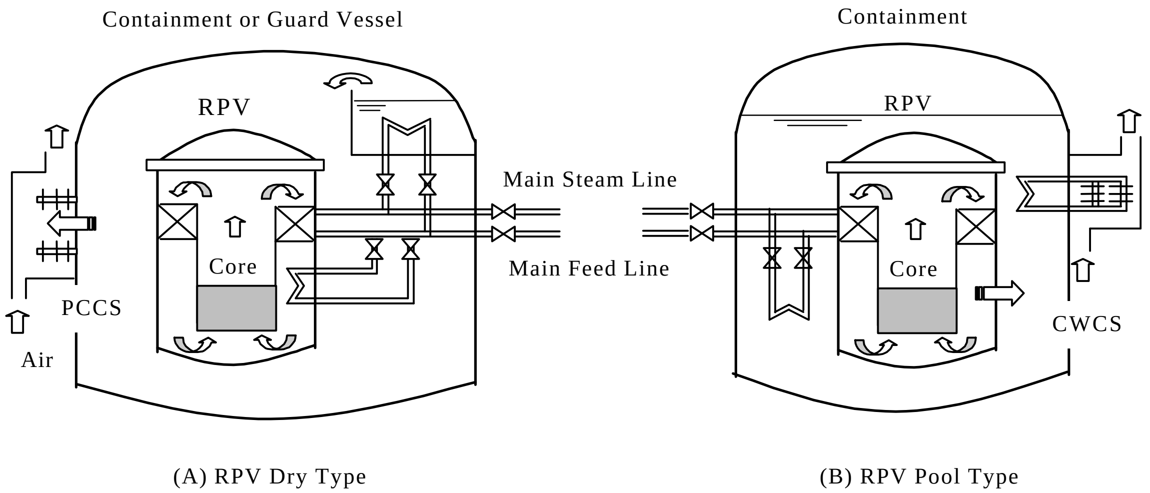

Fig.1 Passive Residual Heat Removal Systems

II.4. Enhancement of Containment Function

In order to prevent releases of radioactive materials in normal and abnormal conditions, the SMR designs adopt an enhanced level of defense in depth. In addition to the current multiple barriers, the guard vessel surrounding the RPV is added or the water-filled containment is adopted. These systems are protective passive feature, which contain the primary coolant and keep the core below the water level following the loss of coolant accidents. The current containment surrounds the guard vessel and plays a role of an external barrier for the retention of the radioactivity in case of beyond design basis accident and also mitigates any impact from external events such as airplane crash. Besides, in order to protect the interface facilities such as desalination plant from the radioactivity, the intermediate circuit is also adopted between the RPV and the interface facility. The immediate circuit would reduce the potential for the direct releases of the RCS coolant to the interface facility and eventually protect the heating grid or the water product from the pollution of radioactivity. In some plants such as NHR and RUTA, the pressure in the intermediate circuit is designed to be higher than that in the primary circuit. The coolant leakage would be directed toward the primary side.

III. Safety Concerns of the SMR

III.1 Use of Proven Technology

The SMR designs are expected to utilize inherent, passive, or other innovative means to accomplish their safety functions. In principle, the technologies incorporated in reactor design should be proven or qualified by experience, testing, or analysis [9]. If possible, the equipment shall be designed according to applicable approved standards and be of a design proven in previous equivalent applications. In particular, if significantly new design features are adopted, they shall be introduced after through research and prototype testing at the component, system, or plant level. For example, the passive safety features, which are expected to provide greater simplicity and higher reliability of the systems, should be evaluated to confirm the required performance and reliable operation under appropriate operation conditions by test and analysis. It is because the driving force in passive fluid systems and their flexibility in abnormal condition are lower than those of the active systems. In some cases, an active system may be required additionally to perform the safety functions.

III.2 Application of Strengthening Defense-in-Depth

In general, there are several levels of protection and multiple barriers to prevent releases of radioactive materials and to ensure that failures leading to significant radiological consequences are of very low probability. In current LWR plants, the physical barriers are in the form of the fuel matrix, the fuel cladding, the RCS boundary, and the containment. In the SMR designs, because the facilities are expected to locate near the population area and the plant products such as fresh water are required to prevent from the radioactivity contamination, the defense in depth concept is necessary to be strengthened to protect the public. For example, guard vessel surrounding the RPV and intermediate circuit between the RPV and the interface facilities are expected to provide more efficient barrier to confine the radioactive materials. In addition, the guard vessel is designed to perform passively the containment cooling under accident conditions, while the current reactor building performs the protection function from external hazards. In these designs, the containment volume is markedly smaller than that of current LWR and the active safety-grade containment coolers or spray systems are not provided. Thus, after an accident, the containment may be maintained at high pressure for a longer period than the current large containment and the fission products released from the RCS could be improperly spread inside containment. Therefore, new containment design deviated from the current practices needs to be reviewed to ensure a level of safety at least equivalent to that of the current LWR. Especially, if the non-safety grade equipment is used to mitigate the consequence of an accident, it would be needed to establish the safety requirements to treat the non-safety systems

III.3 Event Categorization and Selection

In principle, it is required that the beyond design basis accidents (beyond DBA) be considered in the SMR design as well as the DBA. In order to assess the safety of the SMR design using the passive or innovative means to accomplish the safety functions under accident conditions, all events applicable to the SMR design should be categorized according to the expected frequency of occurrence. For the each event category, acceptable criteria for the accident consequences should be established for core damage and dose limits. The events and sequences should be selected deterministically in supplementing with the insights from design specific PRA. Also, accident source term and analysis methodologies should be determined to calculate the radiological consequences or siting area. In several SMR designs, the accident sequences of a lower likelihood than traditional DBA are considered and an allowable evaluation methodology and acceptable criteria corresponding to the events are provided. In addition, events within a category equivalent to the current DBA category may be excluded or added because of the unique design features to cope with the accident. Therefore, the appropriate event categorization, the associated frequency ranges, and the acceptable criteria for the events must be established based on the current safety requirements.

III.4 Simplification of Emergency Planning

The Environmental Protection Agency of U.S. recommends that offsite protective measures be prepared when does consequences are estimated above a lower level emergency protective guidelines (EPG), 1 rem whole body and 5 rem thyroid at the site boundary after any accident. The SMR designs with enhanced safety features such as passive reactor shutdown and cooling systems and longer core heatup time than that of current LWR are expected not to exceed the lower level EPGs at the exclusive area boundary (EAB). The emergency planning zone (EPZ) is also expected to reduce to the EAB. It implies that some protective measures such as rapid notification, detailed evacuation planning and periodic exercises for the public need not be required by regulation. Therefore, it should be reviewed that the simplification of the emergency planning requirements, including the notification requirements, the size of EPZ, and frequency of exercises, is allowable and appropriate. This safety concerns are related to the accident evaluation and source terms.

III.5 Determination of Accident Source Term

The accident source terms are used to calculate not only the release of radioactive materials into the containment but also the potential radiological consequences after the postulated accidents. Because the performance of the fuel, reactor, or containment and the transport of fission products to environment are significantly different from the current LWRs, the new accident source terms would be required. For current operating plants, TID-14844 source terms determined from some conservative assumptions had been utilized to calculate the accident consequences. For evolutionary and passive reactors, NUREG-1465 source terms have recently developed based on realistic and mechanistic assumptions. Therefore, the new accident source term would be needed to evaluate the safety of the SMR design. Because the realistic source term is generally lower than the conservative source term, the accident consequences are expected to reduce for the integrated SMR designs with passive safety concepts.

III.6 Other Issues

Recently, the digital instrumentation and controls are adopting in the SMR designs as an advanced technology because of the ease of data processing. However, the reliability of software and the common mode failures of redundant equipment are raised as an important safety concerns. Also, integrated arrangement of main components or new components such as helical coiled tube may require different test methods and criteria from the current LWR plants. In particular, to demonstrate the system operability and to confirm the performance of the safety systems, a prototype or some other demonstration facility may be needed. Then, the demonstration facility and the safety test program will be required to review to ensure the safety of the facility. Also, the interface facilities such as desalination plant may affect on the operation of the nuclear steam supply system and then the interface effects would be assessed and the interface requirements should be established.

IV. Conclusions

The major safety concepts of the SMR designs with significantly different design concepts from the current LWR were investigated. Basically, the SMR reactor and interface facilities such as the desalination plant are expected to be located near the population area because of some restrictions in transporting the plant products such as the fresh water. To protect the public around the plant from the possible release of radioactive materials, the SMR designs adopt new design safety concepts, such as the integrated arrangement of main components, core cooling by natural circulation, passive safety systems, guard vessel and intermediate circuit to limit the release of radiological materials. Based on the new design concepts, the safety concerns applicable to the integrated SMR design of Korea, SMART, were identified. The safety issues include the use of proven technology, application of strengthening defense in depth, event categorization and selection, simplification of emergency planning, determination of accident source terms and so on. The efforts to resolve those safety concerns in the design stage will provide an improvement of the safety of the SMART design.

References

1. IAEA, “Design and Development Status of Small and Medium Reactor Systems 1995,” IAEA-TECDOC-881, IAEA, May 1996.
2. A.N. Achkasov, G.I. Grechko, and V.A. Shishkin, “Nuclear Steam Supply System NIKA-70 for Floating NPP with the Integral Reactor,” RDIPE, ICONE-6, ICONE-6392, May 10-14, 1998.
3. L.A. Adamovich, G.I. Grechko, V.L. Pavlov, and V.A. Shishkin, “NIKA-150 Design Concept of NSSS for Co-generation NPP,” RDIPE, ICONE-6, ICONE-6391, May 10-14, 1998.
4. C. Lombardi, and M.E. Ricotti, “The NILUS Project: Preliminary Study for Medium and Small Size Innovative PWRs-The NILUS-1000,” ICONE-6, ICONE-6459, May 10-14, 1998.
5. M.E. Ricotti, “The NILUS Project: Design Features of the Small Size, Natural Circulation, Integrated PWR for Heat & Electricity Production,” ICONE-6, ICONE-6458, May 10-14, 1998.
6. K.Sako, et al., “Advanced Marine Reactor MRX,” JAERI, Proceedings of International Conference on Design and Safety of Advanced NPPs, Tokyo, Japan, October 25-29, 1992.
7. D. Dong, D. Zhang, S. Wu and L. Wang, “Analysis of NHR-200 Nuclear Seawater Desalination and Its Interface Design,” IAEA-SM-347/46, Symposium on Desalination of Seawater with Nuclear Energy, Taejon, Korea, May 26-30, 1997.
8. P.R. Rubiolo, P.C. Florido, J.P. Ordonez, and G.G. Lavalle, “Evaluation of Conceptual Designs of Integrated PWRs and CAREM 25 Project,” IAEA-SM-347/60, Symposium on Desalination of Seawater with Nuclear Energy, Taejon, Korea, May 26-30, 1997.
9. IAEA, “Development of Safety Principles for the Design of Future Nuclear Power Plants,” IAEA-TECDOC-801, June 1995.