MDEP
Technical Report
TR-APR1400-02

APR1400 Working Group activities

Technical Report on Background Information relevant to addressing Severe Accidents in the APR1400 design

Participation

Regulators involved in the MDEP working group discussions:	KINS (South Korea), FANR (United Arab Emirates) and US NRC (United States)
Regulators which support the present report:	KINS (South Korea), FANR (United Arab Emirates) and US NRC (United States)
Regulators with no objection:	-
Regulators which disagree:	-
Compatible with existing IAEA related documents:	Yes

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(1) Introduction

This report is intended to provide background information on factors that are relevant to the review of Severe Accident provisions that are either in place, for the existing APR1400 nuclear power plants in operation, or are proposed for those APR1400 nuclear power plants that are either under construction or undergoing design review in the MDEP member countries. This report has been compiled based on the inputs provided by the APR-1400 Severe Accident Technical Expert Subgroup members from the Republic of Korea, the United States of America, and the United Arab Emirates.

(2) Rationale for Document Development

The design and implementation of measures provided to prevent and/or mitigate the effects of Severe Accidents in Nuclear Power Plants are largely influenced by the governing legislative requirements of the country in which the Nuclear Power Plant is to be constructed. During initial discussion among the members of the APR-1400 Severe Accident Technical Expert Subgroup it was determined that, to perform activities effectively, some understanding of similarities and differences between Severe Accident requirements and provisions within the member countries was needed. Familiarity with these various factors which influence the outcomes of the Severe Accident analyses was expected to assist future discussion in the specific areas of interest related to Severe Accidents. To meet this objective it was decided to compile three documents which would cover:

- Severe Accident Regulatory Requirements applicable to the APR1400 design constructed, proposed or undergoing design review in the member countries.

- Severe Accident prevention and mitigation features of the APR1400 design constructed, proposed or undergoing design review in the member countries.

- Summary of Codes, Methodologies and Counter Measures for Severe Accident analysis at APR-1400 units constructed, proposed or undergoing design review in the member countries.

This document is intended to record the outcomes of these tasks and provides the following Tables:

- Table 1 - Regulatory Requirements related to Severe Accidents.
  The table is formatted to address regulatory requirements in the following areas:
  - General Requirements
  - In vessel melt retention
  - In-vessel and ex-vessel steam explosions
  - Combustible gas control
  - Molten core concrete interaction

  - High pressure melt ejection and direct containment heating
  - Containment performance
  - Accident management
  - Probabilistic requirements

- Table 2 - Severe Accident prevention and mitigation features of the APR1400 design

  The table is formatted to address prevention and mitigation features in the following areas:
  - Preventing and mitigating high pressure melt ejection,
  - Containment hydrogen control
  - Mitigating molten core concrete interaction
  - Containment depressurisation

- Table 3 - Summary of Codes, Methodologies and Counter Measures for Severe Accidents at APR-1400 units

  The table is formatted to address codes, methodologies and counter measures for severe accidents in the following areas:
  - Molten core concrete interaction,
  - Hydrogen control
  - Steam explosion
  - Direct Containment Heating/High Pressure Melt Ejection
  - Containment Performance
  - Equipment Survivability
  - Evaluation of External Injection capability to primary & secondary sides

Table 1. Regulatory Requirements related to Severe Accidents

	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
General Requirements	**Nuclear Safety Act Section 6 Accident Management of Reactor Facilities** **Article 85-18 (Scope of Application)** The rule and regulation of accident management for nuclear reactor facilities (hereinafter referred as "accident management") prescribed in Article 21 (1) 6 of the Act are subject to the provisions of Article 85-19 through 85-23. **Article 85-19 (Scope of Accident Management)** The scope of accidents subject to accident management shall be as follows. Design Basis Accident Accidents caused by multiple failure Beyond-Design Basis external events including natural and man-made hazards prescribed at Article 13. Beyond design basis significant core damage accident The specific provisions on the identification of accident sequences listed under Paragraph (1) 2 through 4 are provided by Nuclear Safety and Security Commission. **Article 85-20 (Facilities for Accident Management)** The systems and equipment necessary to implement the accident management prescribed from Article 85-19 (1) 2 to 4 shall be capable of performing the functions required under the severe accident conditions associated with severe core damage. The systems and equipment in the aforementioned Article 85-20 (1) shall be capable of test, surveillance, inspection and maintenance in accordance with the requirements. **Article 85-21 (Accident Management Strategy and Implementation System)** The accident management strategies shall comply with the following requirements: The essential safety functions to be maintained and restored for accident management shall be defined, in addition, operator actions taking into account human factors shall be included in the strategies. Include the technical basis of the accident management strategy and the procedures and instructions for creating the guidelines and the maintenance plan. The accident management program shall include technical bases for each strategy, a writer's guideline for procedure and guidance, and its maintenance program complying with following requirement.	<u>**FANR-REG-03, Regulation for the Design of Nuclear Power Plants, Version 0**</u> **Article (8): Principal Technical Requirements** 1. To ensure Safety, the following fundamental Safety Functions shall be performed in Operational States, in and following a DBA and, to the extent practicable, on the occurrence of those selected Accident Conditions that are beyond the DBAs: a. Control of reactivity; b. Removal of heat from the core; c. Confinement of Radioactive Materials and control of operational Discharges, as well as limitation of accidental releases. 2. A systematic approach shall be followed to identify the SSCs that are necessary to fulfil the Safety Functions at the various times following a PIE. **Article (10): Principal Technical requirements** **Item 3.** The Design shall have as an objective the prevention or, if this fails, the mitigation of radiation exposures resulting from DBAs and selected Severe Accidents. Design provisions shall be made to ensure that potential radiation Doses to the public and the site personnel do not exceed the criteria approved by the Authority **Article (12): General Design Basis** **Item 4.** The performance of the Nuclear Facility in Accidents beyond the design basis, including selected Severe Accidents shall also be addressed in the Design. Best-estimate methods and data; e.g., best estimate vs. Design allowable may be used for the purpose. **Item 5.** Consideration shall be given to the plant's full design capabilities, including the possible use of some systems (i.e. safety and non-safety systems) beyond their originally intended function and anticipated operational states, and the use of additional temporary systems, to return the plant to a controlled state and/or to mitigate the consequences of a severe accident, provided that it can be shown that the systems are able to function in the environmental conditions to be expected. **Article (24): Severe Accidents** Certain very low probability plant states that are beyond	<u>**Title 10 of the Code of Federal Regulations (10 CFR) Part 50—Domestic Licensing of Production and Utilization Facilities**</u> **Applications for Licenses, Certifications, and Regulatory Approvals; Form; Contents; Ineligibility of Certain Applicants** **§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.** (c) *Requirements*. (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system. **§ 50.63 Loss of all alternating current power.** (a) *Requirements*. (1) Each light-water-cooled nuclear power plant licensed to operate under this part, each light-water-cooled nuclear power plant licensed under subpart C of 10 CFR part 52 after the Commission makes the finding under § 52.103(g) of this chapter, and each design for a light-water-cooled nuclear power plant approved under a standard design approval, standard design certification, and manufacturing license under part 52 of this chapter must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors: (i) The redundancy of the onsite emergency ac power sources; (ii) The reliability of the onsite emergency ac power sources; (iii) The expected frequency of loss of offsite power; and

	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
	To secure the organizational and human resource and to identify the organization responsibility and empowerment necessary for the implementation of accident management strategies; To establish the necessary command and control system for the implementation of the accident management strategy; To monitor the operability and functionality of systems and equipment necessary for the implementation of designated accident management strategies and provide appropriate remediation actions in the case where abnormal conditions are identified; To Include, if necessary, the terms of functional integration of accident management strategies; To take into consideration the characteristics of multi-unit site, if the identified facility belongs to a multi-unit site; **Article 85-22 (Evaluation of Accident Management Capability)** The Accident Management Plan shall provide the assessment of the capability of accident management including systems and equipment, strategies and the implementation framework to be deployed, and demonstrate the compliance with the following safety goal: To prevent the substantial release of radioactive materials leading to public health risk, in particular to residents residing nearby the site, or to incur long-term environmental impact. To minimize the increase of risk to the health and environment of residents near the site due to the operation of a nuclear reactor and associated facilities. The evaluation of the compliance with the prescribed safety goal aforementioned provisions shall be based on deterministic and probabilistic approaches and specific provisions on the assessment are provided by NSSC (Nuclear Safety and Security Commission) **Article 85-23 (Accident Management Education and Training)** (1) The education and training plan established to maintain the effectiveness of the accident management plan shall comply with following requirements: Periodic training compatible with positions and his/her responsibilities and authorities prescribed the accident management plan. Training should be conducted within two years to validate the effectiveness of the accident management strategy and implementation framework.	DBA conditions and which may arise owing to multiple failures of Safety Systems leading to significant core degradation may jeopardise the integrity of many or all of the barriers to the release of radioactive material. These event sequences are called Severe Accidents. Consideration shall be given to severe accidents by providing in the design reasonably practicable preventive and/or mitigative measures. These measures need not involve the application of conservative engineering practices used in setting and evaluating DBAs, but rather should be based upon realistic or best estimate assumptions, methods and analytical criteria. On the basis of operational experience, relevant safety analysis and results from safety research, design activities shall take into account the following : a. Provisions to promote in-vessel core melt retention b. Provisions to prevent and/or withstand in-vessel and ex-vessel steam explosion c. Provisions for combustible gas control d. Provisions for mitigation of molten core debris concrete interaction e. Provisions to prevent and mitigate high pressure core melt ejection from the Reactor Pressure Vessel f. Provisions to prevent early containment failure under severe accident conditions g. Accident Management procedures shall be established, taking into account representative and dominant Severe Accident scenarios h. The effectiveness of the severe accident measures shall be confirmed by demonstrating that the Authority's safety target is met. i. Articles 51, 76 and 80 identify measures that intended to reduce the likelihood of some scenarios that were leading contributor to severe accidents, namely: Pressurised Thermal Shock (PTS) Anticipated Transient without Scram (ATWS) and Station Blackout (SBO). **Article (44): Safety Analysis** **Item 2.** The computer programmes, analytical methods and plant models used in the Safety analysis shall be verified and validated, and consideration shall be given to uncertainties. **Article (56): Emergency Core Cooling** **Item 3.**	(iv) The probable time needed to restore offsite power. (2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Licensees are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review. **§51.55 Environmental report-standard design certification** (a) Each applicant for a standard design certification under subpart B of part 52 of this chapter shall submit with its application a separate document entitled, "Applicant's Environmental Report—Standard Design Certification." The environmental report must address the costs and benefits of severe accident mitigation design alternatives, and the bases for not incorporating severe accident mitigation design alternatives in the design to be certified. **<u>§ 52—Licenses, Certifications, and Approvals for Nuclear Power Plants</u>** **Subpart B—Standard Design Certifications** **§ 52.47 Contents of applications; technical information** (a)(2) A description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. It is expected that the standard plant will reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations. Such items as the reactor core, reactor coolant system, instrumentation

	Country Requirement		
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	<u>**Regulation on detailed criteria for accident management scope and accident management capability evaluation**</u> <u>**[Enforcement 2016.7.3] [Nuclear Safety and Security Commission Notice 2016-2, 2016.7.3, enacted]**</u> **Chapter 1: General Rules** **Article 1 (Purpose)** The purpose of this regulation is to prescribe specific rules and regulations with regard to the evaluation of accident management capabilities delegated by Enforcement Regulation for the Nuclear Act 85-19 dealing with selection of accident sequences to be considered in establishing accident management and 85-22(2) associated evaluation of accident management capabilities. Regulation shall be applied for drafting accident management plan in accordance with Article 4 (6) (3) of the Enforcement Regulations of the Nuclear Safety Act (hereinafter referred to as the "Enforcement Regulations") or Article 9 (3) nuclear reactor facility and its associated facilities (Hereinafter referred to as "the nuclear reactor facility ") and for nuclear reactor facilities that submit an accident management plan in accordance with Article 16 (4) of the Enforcement Regulations. The extent of accident management described in Article 3 to Article 5 may be restricted, in cases where provisions cannot be practically implemented due to unique design principles and/or design differences, provided that nuclear reactor facilities can be shown not to pose an unsafe condition. **Chapter 2 Scope of Accident Management** **Article 3 (Scope of Accidents Due to Multiple Failures)** The accidents initiated from multiple failures to be applied with the reactor facilities in accordance with Article 85-19, 1, (2) are listed in the annexed Table 1. **Article 4 (Scope of Beyond- Design Basis external events including natural hazards and man-made hazards)** The beyond-design basis external and man-made hazard applicable to the nuclear reactor facilities according to Article 85, Paragraph 19, Paragraph 1, Item 3 of the enforcement regulation for Nuclear safety Acts are as follows: Natural hazards caused by geological, earthquake, meteorological, hydrological and marine	Adequate consideration shall be given to extending the capability to remove heat from the core so that, following a Severe Accident acceptable temperatures can be maintained in SSCs important to the safety function of confinement of Radioactive Materials. **Article (58): Emergency Core Cooling** **Item 4.** Adequate consideration shall be given to extending the capability to transfer residual heat from the core to an Ultimate Heat Sink so as to ensure that, in the event of a Severe Accident, acceptable temperatures can be maintained in SSCs important to the Safety Function of confinement of Radioactive Materials. **Article (70): Instrumentation and Control** **Item 2.** Instrumentation and recording equipment shall be provided to ensure that essential information is available for monitoring the course of DBAs and the status of essential equipment; and for predicting, as far as is necessary for Safety, the locations and quantities of Radioactive Materials that could escape from the locations intended in the Design. The instrumentation and recording equipment shall be adequate to provide information as far as practicable for determining the status of the Nuclear Facility in a Severe Accident and for taking decisions in Accident Management. **Article (71): Instrumentation and Control** 1. A control room shall be provided from which the Nuclear Facility can be safely operated in all its Operational States, and from which measures can be taken to maintain the Nuclear Facility in a stable, safe state or to bring it back into such a state after the onset of Anticipated Operational Occurrences, DBAs and Severe Accidents. Appropriate measures shall be taken and adequate information provided to safeguard the occupants of the control room against consequent hazards, such as undue radiation levels resulting from an Accident Condition or the release of Radioactive Material or explosive or toxic gases, which could hinder necessary actions by the operating personnel. 2. Special attention shall be given to identifying those events, both internal and external to the control room, which may pose a direct threat to its continued Operation, and the Design shall provide for reasonably practicable measures to minimise the effects of such events.	and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent. The following power reactor design characteristics will be taken into consideration by the Commission: (i) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials; (ii) The extent to which generally accepted engineering standards are applied to the design of the reactor; (iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials; and (iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable postulated site parameters, including site meteorology, to evaluate the offsite radiological consequences. The evaluation must determine that: A) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE); (B) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage)

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	phenomena. Unpredictable malicious aircraft impact associated with terrorism. Composite hazard by the aforementioned two hazards, necessary for the implementation of accident management strategies. **Article 5 (Threats to containment performance caused by significant core damage)** The accident management shall have enough capability to cope with the threats to containment performance due to the significant core damage initiated from an accident in accordance with Article 85, Paragraph 19, and Subparagraph 1. Table 2 provides the list of the threats to be considered in the accident management. **Chapter 3 Evaluation of Accident Management Capability** **Article 6 (evaluation of Severe Accident Prevention Capability)** The nuclear reactor facility for power generation shall have enough capability to prevent significant core damage at reactor system or spent fuel storage facilities due to multiple faults in accordance with Article 3. Regarding the events identified in Article 4 at nuclear facilities, the critical safety functions including cooling capability for reactor core and spent fuel pool and containment integrity shall be restored and maintained. **Article 7 (Evaluation of Severe Accident Mitigation Capacity)** The nuclear reactor facility shall be designed so as to prevent the loss of the containment performance to limit significant large release of radiological source term generated from significant core damage. **Article 8 (Evaluation of radiological consequences)** Radiation exposure dose of resident for each accidents listed in Article 85-19, paragraph 1 shall be evaluated by the deterministic approach and evaluated dose shall be limited within the criteria prescribed in the Enforcement Regulation Article 5-2 **Article 9 (Risk Assessment)** (1) The technical adequacy, quality and scope of the probabilistic safety assessment shall be suitable for the integrated risk evaluation of nuclear facilities. (2) The risk goal applicable to the probabilistic safety evaluation in Paragraph (1) are as follows. The early fatality and the cancer fatality (stochastic	3. The layout of the instrumentation and the mode of presentation of information shall provide the operating personnel with an adequate overall picture of the status and performance of the Nuclear Facility. Ergonomic factors shall be taken into account in the Design of the control room. 4. Devices shall be provided to give visual and, if appropriate, also audible indications of Operational States and processes that have deviated from normal and could affect Safety. **Article (90): Radiation Protection** Equipment shall be provided to ensure that there is adequate radiation monitoring in Operational States, DBAs and, as practicable, Severe Accidents: 1. Stationary Dose rate meters shall be provided for monitoring the local radiation Dose rate at places routinely occupied by operating personnel and where the changes in radiation levels in Normal Operation or Anticipated Operational Occurrences may be such that access shall be limited for certain periods of time. Furthermore, stationary Dose rate meters shall be installed to indicate the general radiation level at appropriate locations in the event of DBAs and, as practicable, Severe Accidents. These instruments shall give sufficient information in the control room or at the appropriate control position that Nuclear Facility personnel can initiate corrective action if necessary. 2. Monitors shall be provided for measuring the Activity of radioactive substances in the atmosphere in those areas routinely occupied by personnel and where the levels of Activity of airborne Radioactive Materials may on occasion be expected to be such as to necessitate protective measures. These systems shall give an indication in the control room, or other appropriate locations, when a high concentration of radionuclides is detected. 3. Stationary equipment and laboratory facilities shall be provided for determining in a timely manner the concentration of selected radionuclides in fluid process systems as appropriate, and in gas and liquid samples taken from plant systems or the environment, in Operational States and in Accident Conditions. 4. Stationary equipment shall be provided for monitoring the effluents prior to or during Discharge to the environment. 5. Instruments shall be provided for measuring radioactive surface contamination.	would not receive a radiation dose in excess of 25 rem TEDE; (a)(4) An analysis and evaluation of the design and performance of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of emergency core cooling system (ECCS) cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter; (a)(8). The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v); (a)(23). For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass; (a)(27) A description of the design-specific probabilistic risk assessment (PRA) and its results.

	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
	effect) risk of resident nearby the facilities due to the consequences shall be less than or equal to 0.1% of the overall risk level or meet the corresponding performance goal. The sum of the frequencies of the event which emit radionuclide Cs-137 in excess of 100TBq shall be less than 1.0 * 10-6 / year. (3) The results of the probabilistic safety assessment under Paragraph (1) shall be utilized to improve the capability to prevent and mitigate severe accidents at nuclear facilities. **Addenda <2016-2, 2016.7.3>** **Article 1 (Effective Date) This Regulation shall be effective from the date of notification.** **<u>Policy Statement on Severe Accident of Nuclear Power Plant</u>** 1. Background (see the policy statement) 2. Definitions (See the policy statement) 3. Severe Accident Policy 1) Safety Goal The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from all other accidents. The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes. In achievement of the above safety goals, the performance goals which are aimed at preventing the core damage and mitigating the fission product releases from the containment. are to be established. 2) Probabilistic Safety Assessment An owner of nuclear power reactor should assess safety of the nuclear power plant via probabilistic approach in order to find measures which can reduce the risk as low as possible. In particular, the design and operational procedures of nuclear power plant should be assessed to improve the accident prevention and mitigation capabilities for accident scenarios which have relatively high probability of core damage. The plant vulnerability should be fixed by cost-benefit consideration. 3) Severe Accident Prevention and Mitigation Capability Nuclear power plant should have a prevention capability of core damage to keep off severe accidents. Reactor containment should preserve its structural integrity and function as a barrier against fission product release in	6. Facilities shall be provided for monitoring of individual Doses to and contamination of personnel. **<u>FANR-RG-004, Evaluation Criteria for Probabilistic Safety Targets and Design Requirements, Version 0</u>** **Article (12)** 1. Because it takes time for an Accident to progress and the transport of radionuclides into the Containment is gradual and does not include the entire inventory due to deposition on colder surfaces in primary and secondary systems, and because of a better estimate of Containment performance under Severe Accident loads, releases to the environment and subsequent consequences are significantly reduced. (1) Accordingly, the Severe Accident considerations discussed in this article capitalise on the results of Severe Accident research that was conducted over the last 30 years. There is now a greater understanding of what happens during a Severe Accident, which allows a better estimate of Containment performance under Severe Accident loads, and more reliable Severe Accident management and Emergency Preparedness programmes. 3. The Severe Accident considerations discussed in this article are intended to ensure that the likelihood of an Accident having harmful consequences remains extremely low, i.e., reduce to low likelihood the probability of occurrence of core melt Accident and/or acute radiation exposures resulting in fatalities. The incorporation of such features provides Defence-in-Depth and helps compensate for phenomenological and other uncertainties (e.g., human error) that affect the risk from Severe Accidents. Designs meeting the evaluation criteria discussed below can be considered to have effective Severe Accident prevention and mitigation capabilities and provide adequate assurance of protecting public health and Safety. a) In-vessel Core Melt Retention During a core melt Accident, if the reactor vessel remains intact, molten core debris will be retained in the lower head and phenomena such as ex-vessel steam explosion, direct Containment heating, and core concrete interactions, which occur as a result of core debris relocation to the reactor cavity, can be prevented.	

	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
	order to mitigate the consequence (impact) of accident even in the case of core damage. 4) Severe Accident Management Program An owner of a nuclear power reactor should establish and implement severe accident management programs. The programs should include accident management strategies, accident management organization, guidelines, training and education program, instrumentation, and assessment of essential information, etc.	Measures to promote in-vessel retention, such as by flooding the reactor cavity with water to cool the core debris inside the reactor pressure vessel may be included for use as an Accident management strategy. However, with the low frequency of core melt Accidents specified in Article (6), and the Severe Accident mitigation features listed in items (e) through (j) below, additional in-vessel retention measures would not be warranted unless the PRA shows this to be a key feature for the protection of public health and safety. b) Steam Explosions • In-Vessel Steam Explosion During the initial stages of progression of Severe Accidents, molten debris from the damaged core would relocate to the lower plenum of the reactor pressure vessel. If a sufficient amount of water remained in the lower plenum, the molten core material falling into the water could generate steam and if severe enough, an explosion. This explosion could challenge the reactor vessel and Containment integrity. However, a recent assessment of this issue by a United States Nuclear Regulatory Commission sponsored steam explosion review group (Reference (8)[1]) concluded that this mode of Containment failure has a very low likelihood of occurring. It should be confirmed that the underlying assumptions in Reference (8) are applicable to the proposed Design. • Ex-Vessel Steam Explosion Reactor vessel failure at high or low pressure coincident with water present within the reactor cavity may lead to interactions between fuel and coolant with a potential for steam generation or steam explosions. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporization and acceleration of surrounding water creating substantial pressure and impact loads. It should be confirmed that the Design has been analysed for ex-vessel steam explosion and that the structural integrity of the Containment would be maintained in the event of an ex-vessel steam explosion. c) Combustible Gas Generation and Control	

[1] Reference 8: NUREG-1524, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of Broader Fuel-Coolant Interaction Issues," August 1996

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		The issue regarding combustible gas generation centres on the rate and quantity of hydrogen production and the associated hydrogen steam mass and energy release rates into the Containment during both in-vessel and ex-vessel phases of Severe Accidents. These parameters strongly influence the flammability of the Containment atmosphere and the magnitude, timing, and location of potential hydrogen combustion. Hydrogen combustion in the Containment could produce pressure and thermal loads that might threaten the integrity of the Containment boundary. There are uncertainties in the phenomenological knowledge of hydrogen generation and combustion. In order to ensure Containment integrity will be maintained, the Design should provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction. In addition, the Design should be capable of precluding uniform concentrations of hydrogen from exceeding 10 percent (by volume), or should provide an inerted atmosphere within the Containment. d) Core Debris-Concrete Interaction In the event of a Severe Accident in which the core has melted through the reactor vessel, it is possible that Containment integrity could be breached if the molten core is not sufficiently cooled. In addition, interactions between the core debris and concrete could generate large quantities of additional hydrogen and other non-condensable gases, which could contribute to the eventual overpressure failure of the Containment. Downward erosion of the basemat concrete could also lead to basemat penetration with the potential for ground water contamination and subsequent discharge of radionuclides to the surface environment. Thermal attack by molten corium on retaining sidewalls could produce structural failure within the Containment causing damage to vital systems and perhaps to failure of Containment boundary. Therefore, the applicant/licensee should assess a) reactor cavity floor space to ensure adequate area for debris spreading; b)means to flood the reactor cavity to assist in the cooling process; and c) impact of core concrete interaction with cavity walls on the Containment integrity. e) High Pressure Core Melt Ejection • A high pressure core melt ejection is the ejection of	

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		core debris and hydrogen from the reactor vessel at high pressure. High pressure core melt ejection could cause fragmentation and dispersal of core debris and hydrogen within the Containment atmosphere, termed direct Containment heating (DCH) that has the potential to cause early Containment failure. Containment failure could occur due to the heat-up and pressurisation of the Containment as a result of hydrogen combustion and core debris heat generation. Another potential consequence of high pressure melt conditions could be a thermally induced failure of steam generator tubes while the RCS is at high pressure, leading to Containment bypass. The likelihood of failure of the steam generator tubes depends on several factors including the thermal hydraulic conditions at various locations in the primary system which determines the temperature and pressure to which the steam generator tubes are subjected as the Accident progresses. The presence of defects in the steam generator tubes will increase the likelihood of failure. • The Design should include an AC-independent RCS depressurisation system for reducing the probability of high pressure melt conditions and the reactor cavity design should include features to enhance core debris retention in the reactor cavity (e.g., no direct pathway to the Containment atmosphere). f) Containment Performance under Severe Accident Conditions The Containment should be designed to have a high probability of withstanding the loads associated with Severe Accident phenomena. This should be done by demonstrating that the Containment will maintain its role as a reliable, low leakage barrier for approximately 24 hours following the onset of core melt accident. After 24 hours, releases from the containment should be controlled or ensure that a containment failure probability of 0.1 is achieved. The Containment should be assumed to have failed if any of the following conditions occur: • Containment structural failure • The Containment is bypassed • The Containment fails to isolate • The Containment seal materials fail as a result of over-temperature or pressure • The molten core debris melts through the	

	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
		concrete basemat into the subsoil g) Severe Accident Management The Design should include provisions to facilitate the management of Severe Accidents. This should include provisions such as instrumentation that can provide the operating staff with information on the Accident progression (e.g., parameter trends), provisions to supply water and electrical power from outside sources (e.g., fire protection system water, portable generators) and provisions to protect the operating staff from radiation and toxic gases such that they can safely perform the actions called for in the Accident Management programme. The Design provisions should be consistent with and support the NPP's Accident management programme. h) Release of Radioactive Material The annual risk to members of the public from the release of Radioactive Material from a Severe Accident should not exceed the risk equivalent to a Dose of 1 mSv/yr. Appendix A provides guidance on the methodology to be used in calculating the annual Effective Dose to members of the public.	
In-vessel melt retention		**FANR-REG-03** **Article (24) item a. (See general requirements.)** **FANR-RG-004** **Article (12) item 3.a. (See general requirements.)**	
In-vessel and ex-vessel steam explosion		**FANR-REG-03** **Article (24) item b. (See general requirements.)** **Article (66) Internal Structures of the Containment Item 2.** Consideration shall be given to the capability of internal structures to withstand the effects of a Severe Accident. **FANR-RG-004** **Article (12) item 3.b. (See general requirements.)**	**10 CFR 52.47(a)(23) (See general requirements.)**
Combustible gas control		**FANR-REG-03** **Article (24) item c. (See general requirements.)** **FANR-RG-004**	**10 CFR 50.44 Combustible gas control for nuclear power reactors.** **(c) *Requirements for future water-cooled reactor***

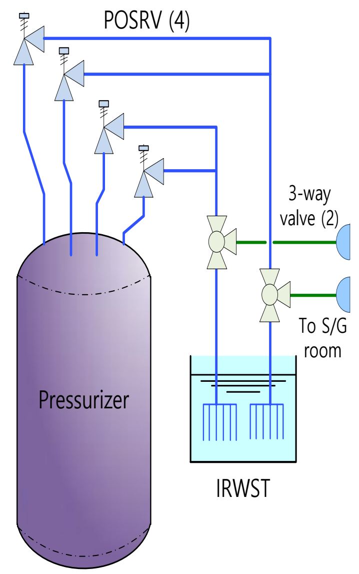
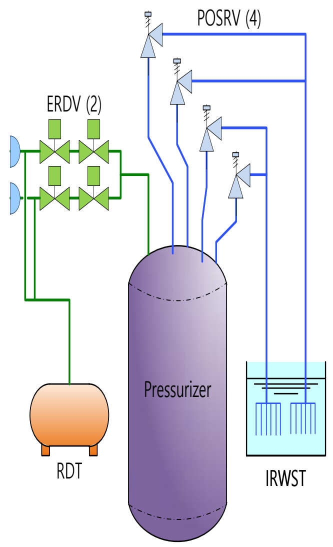
	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
		Article (12) item 3.c. (See general requirements.)	***applicants and licensees.*** (1) *Mixed atmosphere.* All containments must have a capability for ensuring a mixed atmosphere during design-basis and significant beyond design-basis accidents. (2) *Combustible gas control.* All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features. (3) *Equipment Survivability.* Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. Environmental conditions caused by local detonations of hydrogen must also be included, unless such detonations can be shown unlikely to occur. The amount of hydrogen to be considered must be equivalent to that generated from a fuel clad-coolant reaction involving 100 percent of the fuel cladding surrounding the active fuel region. (4) *Monitoring.* (i) Equipment must be provided for monitoring oxygen in containments that use an inerted atmosphere for combustible gas control. Equipment for monitoring oxygen must be functional, reliable, and capable of continuously measuring the concentration of oxygen in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning. (ii) Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning. **10 CFR 52.47(a)(23) (See general requirements.)**

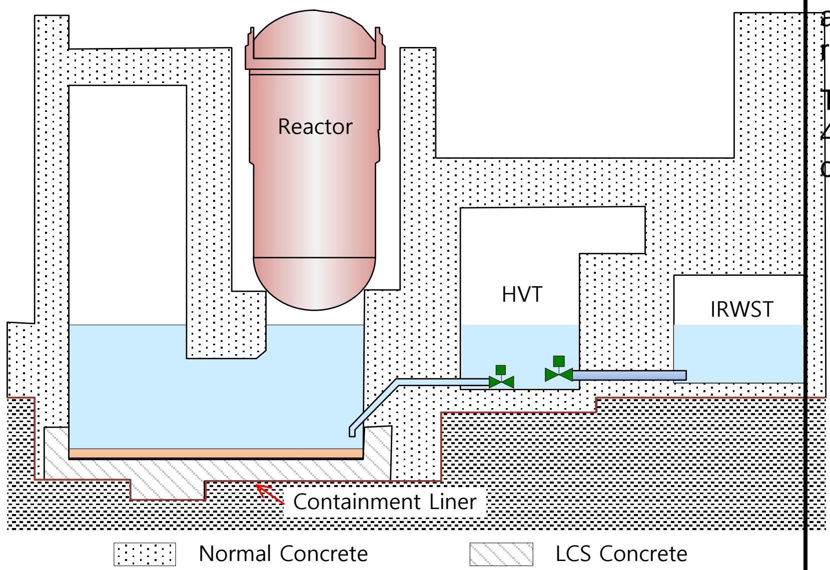
	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
Molten Core Concrete Interaction		**FANR-REG-03** **Article (24) item d. (See general requirements.)** **FANR-RG-004** **Article (12) item 3.d. (See general requirements.)**	**10 CFR 52.47(a)(23) (See general requirements.)**
High Pressure Melt Ejection and Direct Containment Heating		**FANR-REG-03** **Article (24) item e. (See general requirements.)** **Article (66) Item 2. (See in-vessel and ex-vessel steam explosion.)** **FANR-RG-004** **Article (12) item 3.e. (See general requirements.)**	**10 CFR 52.47(a)(23) (See general requirements.)**
Containment Performance		**FANR-REG-03** **Article (24) item f. (See general requirements.)** **Article (59) Containment system** **Item 2.** All identified DBAs shall be taken into account in the Design of the containment system. In addition, consideration shall be given to the provision of features for the mitigation of the consequences of selected Severe Accidents in order to limit the release of Radioactive Material to the environment. **Article (60) Containment system** **Item 2.** Provision for maintaining the integrity of the containment in the event of a severe accident shall be considered. In particular, the effects of any predicted combustion of flammable gases shall be taken into account. **Article (63) Containment penetrations** **Item 4.** Consideration shall be given to the capability of penetrations to remain functional in the event of a Severe Accident. **Article (64) Containment isolation** **Item 4.** Consideration shall be given to the capability of isolation devices to maintain their function in the event of a Severe Accident. **Article (65) Containment isolation**	**10 CFR 50 Appendix A** ***Criterion 16—****Containment design*. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
		Item 2. Consideration shall be given to the capability of containment air locks to maintain their function in the event of a Severe Accident. **Article (66) Item 2. (See in-vessel and ex-vessel steam explosion.)** **Article (67) Removal of Heat from the Containment** **Item 2.** Consideration shall be given to the capability to remove heat from the reactor containment in the event of a Severe Accident. **Article (68) Control and Cleanup of the Containment Atmosphere** **Item 2.** Consideration shall be given to the control of fission products, hydrogen and other substances that may be generated or released in the event of a Severe Accident. **<u>FANR-RG-004</u>** **Article (12) item 3.f. (See general requirements.)**	
Accident Management	**See general requirements.**	**<u>FANR-REG-03</u>** **Article (24) item g. (See general requirements.)** **Article (36) Other Design Considerations** Sharing of items Important to Safety between nuclear power plant units for the purpose of accident management shall be permitted only provided that it has been demonstrated that such sharing does not prevent the other units from performing all Safety functions on the assumption of a single failure. Systems that are not safety systems may be shared between several units provided that such sharing would not increase either the likelihood or the consequences of a severe accident. **Article (70) Item 2. (See general requirements.)** **Article 79: Emergency Control Centre** An on-site (within the site area) technical support centre, separated from the plant control room and an operation support centre and an off-site emergency operation centre shall be provided to serve as emergency facilities in the event of an Emergency. Information about important plant parameters and radiological conditions in the plant and its immediate surroundings shall be available there. Emergency power supply system should	

	Country Requirement		
Severe Accident Consideration	**Korea**	**UAE**	**USA**
		be equipped to cope with a loss of off-site power. The facilities shall provide means of communication with the control room, the supplementary control room and other important points in the plant, and with the on-site and off-site Emergency Response organizations. Appropriate measures shall be taken to protect the occupants for a protracted time against hazards resulting from a Severe Accident, where applicable. **<u>FANR-RG-004</u>** **Article (12) item 3.g. (See general requirements.)**	
Probabilistic Requirements	Currently, the legal provision for PSA is under the rule making process. However, all nuclear power plants in operation or under construction have submitted Level 2 PSA including external events and these have been reviewed by KINS based on the aforementioned “Policy Statement declared by MOST in 2001	**<u>FANR-RG-004, Evaluation Criteria for Probabilistic Safety Targets and Design Requirements</u>** **Article (6) probabilistic Safety Targets – Evaluation Criteria** 1. The NPP should be designed, operated and maintained so as to limit its overall core damage frequency (CDF) to < 10^{-5}/yr (mean value from the PRA considering internal and external events and all modes of Operation). 2. The NPP should be designed, operated and maintained so as to limit its overall large release frequency (LRF) to < 10^{-6}/yr (mean value from the PRA considering internal and external events and all modes of Operation). 3. The NPP should be designed, operated and maintained so as to avoid a disproportionate concentration of risk resulting from any single SSC failure or human action. 4. Sensitivity studies, using the PRA, should be performed to determine whether small variations in SSC and human performance (e.g., reliability, availability) would cause any of the above evaluation criteria to be exceeded. If the results of the sensitivity studies show any of the above evaluation criteria are exceeded, a review should be conducted and documented to see if corrective action is warranted. **Article (12) item 3.h. (See general requirements.)**	**<u>10 CFR 50.34</u> Contents of applications; technical information.** **(f) Additional TMI-related requirements.** (1)(i). Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. **<u>10 CFR 50.47(a)(27)</u> (See general requirements.)**

Table 2. Severe Accident prevention and mitigation features of the APR1400 design

SA Prevention and Mitigation Features	Korea	UAE	USA
Preventing and Mitigating High Pressure Melt Ejection (HPME)	**Emergency RCS Depressurization System** to cater for the unavailability of safety injection (Shin-Kori 5&6). The original design of the APR-1400 took into consideration the POSRV for the depressurization function. A concern related to the possibility of hydrogen, produced by the in-vessel metal water reaction, accumulation in the IRWST, and released via the POSRV, resulted in the introduction of operator controlled three-way valves to divert the effluent containing hydrogen to the Steam generator compartment. The Shin-Kori 5&6 design has installed additional emergency RCS depressurization valves to avoid operator action related to the 3-way valves. Shin-Kori 3&4  Shin-Kori 5&6  	**Emergency RCS Depressurization System**, which is similar to Shin-Kori 3 & 4, is designed to prevent HPME by depressurizing the RPV before its failure. The passage from the reactor cavity to the upper containment in the UAE design is a torturous path, which significantly limits the fraction of corium that participates in a direct containment heating accident, if a HPME were to occur.	**Emergency RCS Depressurization System**, which is similar to Shin-Kori 3 & 4, is designed to prevent HPME by depressurizing the RPV before its failure. The passage from the reactor cavity to the upper containment in the U.S. design is a torturous path, which significantly limits the fraction of corium that participates in a direct containment heating accident, if a HPME were to occur.
Containment Hydrogen Control	To prevent hydrogen accumulation leading to hydrogen detonation, The Korean APR-1400 design has installed 30 Passive Autocatalytic Recombiners (PARs) complemented by 10 glow plug igniters	To prevent hydrogen accumulation leading to hydrogen combustion, the UAE design consists of 30 PARs and 10 hydrogen igniters (HIs). 12 of the 30 PARs are also installed for DBA purposes. The igniters are classified as Non-1E and are powered from Class 1E buses which receive power from preferred power supply (PPS) I or PPS II (i.e. two distinct and separate sources of off-site power). In the event of a loss of off-site power, the igniters are powered from one of the two EDGs; a selector switch provides this functional capability. On loss of off-site power and failure of both of the EDGs to start or run (i.e., SBO), the igniters are powered from the alternate AC (AAC) diesel generator.	To prevent hydrogen accumulation leading to hydrogen combustion, the U.S. design consists of 30 PARs and 8 hydrogen igniters (HIs). Although HIs are classified as Non-class 1E, the electrical power for HIs is supplied from the Class 1E bus (Train A or B) with the electrical isolation device in order to enhance the reliability of HIs. At loss of offsite power and failure of the emergency diesel generators to start or run (station blackout), the HIs have the alternative power supply from the alternate alternating current (AAC) generator. During a complete loss of ac power including from the AAC generator, the HIs are powered from the DC battery.

SA Prevention and Mitigation Features	Korea	UAE	USA
Mitigating Molten Core Concrete Interaction (MCCI)	Mitigation measures against MCCI are the reactor cavity design and the cavity flooding system (CFS) used to cool ejected corium and slow down concrete erosion due to MCCI. The CFS provides a means of flooding the reactor cavity during a severe accident to cool the core debris in the reactor cavity and to scrub fission products. The water delivery from the IRWST to the reactor cavity is accomplished by means of operator actuated motor operated valves. The CFS uses water from the IRWST and directs it using gravitational head to the reactor cavity via the HVT by way of the two HVT spillways and two reactor cavity spillways. Flooding of the HVT progresses until the water levels in IRWST, HVT, and reactor cavity equalize at 6.4 m (21 ft.) above the reactor cavity floor.  In the original APR1400, the concrete composition of reactor vessel cavity wall and floor is the same as that of the containment building i.e. basaltic concrete. In SKN 3 and 4, one foot of Limestone Common Sand (LCS) concrete was added to the floor and wall of the cavity. SKN 3 and 4 also has a design modification to the cavity structure to prevent the intrusion of molten corium into the cavity sump by the installation of a blocking wall. Due to the design modification, the cavity floor area is reduced from 82 m^2 to 72 m^2, SKN 5 and 6 uses only LCS concrete in the floor and wall of the reactor cavity.	A large cavity floor area (80 m^2) is provided, designed to maximize the unobstructed floor area available for the spreading of corium which mitigates MCCI. The CFS provides a means of flooding the reactor cavity during a severe accident for the purpose of cooling the core debris in the cavity and scrubbing fission product releases. The water delivery from the IRWST to the cavity is accomplished by means of motor operated valves and gravity drainage of the IRWST. The CFS is designed to provide a supply of water to quench the core debris. After the operator manually operates CFS by opening reactor cavity flooding motor operated valves, water floors from IRWST to HVT and then to the cavity by gravity. Flooding of the HVT continues until the water levels in IRWST, HVT, and reactor cavity equalize at 6.4 m (21 ft.) above the reactor cavity floor. The reactor cavity floor and walls of the Barakah units 1 to 4 are constructed of Limestone Common Sand (LCS) concrete.	A large cavity floor area (80 m^2) provides spreading area to reactor power of 0.02 m^2/MWth, which mitigates MCCI. CFS is designed to flood the reactor cavity during a severe accident to minimize or eliminate corium-concrete attack; minimize the generation of combustible gases (hydrogen and carbon monoxide) and other non-condensable gases; scrub fission products; and remove heat from the core debris. After the operator manually operates CFS by opening reactor cavity flooding motor operated valves, water floors from IRWST to HVT and then to the cavity by gravity. Flooding of the HVT continues until the water levels in IRWST, HVT, and reactor cavity equalize at 6.4 m (21 ft.) above the reactor cavity. The reactor cavity floor and walls of the U.S. design are to be constructed of Limestone Common Sand (LCS) concrete.

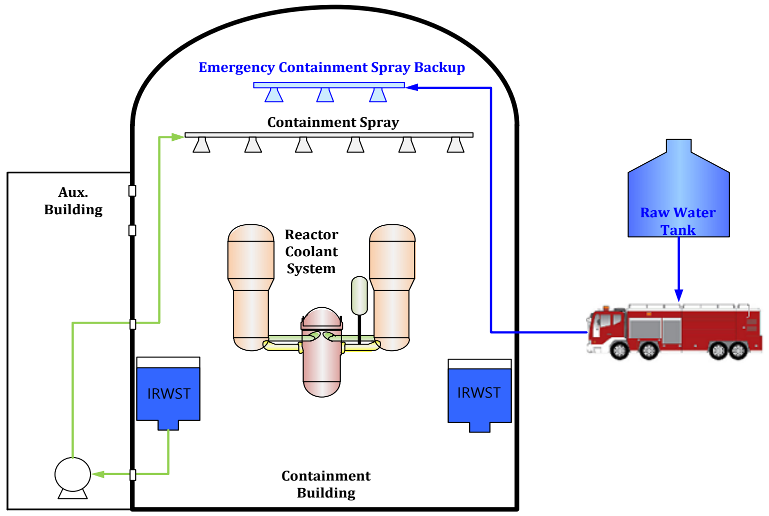
SA Prevention and Mitigation Features	Korea	UAE	USA
Containment Depressurization	The ECSBS is designed to protect the containment integrity against overpressure and prevent the uncontrollable release of radioactive materials into the environment. The emergency containment spray flow path is from external water sources (the reactor makeup water tank, demineralized water storage tank, fresh water tank, or the raw water tank), through the fire protection system line via the diesel-driven fire pump, to the ECSBS line emergency connection located at ground level near the auxiliary building. ECSBS operation begins 24 hours after the onset of core damage and is capable of controlling containment pressure and reducing containment atmospheric temperature for a period of 48 hours. The maximum pressure and temperature following the initial 24-hour period are enveloped by the maximum pressure and temperature during the initial 24-hour period. This prevents the uncontrolled release of fission products into the environment.  	The ECSBS in the UAE design is identical to that in the Korean design. Even though the maximum pressure and temperature following the initial 24-hour period are enveloped by the maximum pressure and temperature during the initial 24-hour period, uncontrolled release of fission products into the environment following the first 24 hour period is still under discussion between the applicant and regulatory body.	The ECSBS in the U.S. design is identical to that in the Korean design.

Table 3. Summary of Codes, Methodologies and Counter Measures for Severe Accidents at APR-1400 units

SA Phenomenology	Items	SKN 3&4	BNPP 1 - 4	US DCD	Remarks
Molten Core Concrete Interaction (MCCI)	Computational Code	MAAP 5.02	MAAP 5.03	MAAP 4.0.8	
	Counter Measures	• Cavity Flood System (CFS) • Design for preventing the intrusion of Molten corium into Reactor cavity sump • Additional embedment of 1ft-thick limestone concrete over the reactor cavity floor	• CFS • Standard cavity design • 90 cm limestone concrete layer protects the containment liner	• CFS • Standard cavity design • 90 cm limestone common sand concrete layer protects the containment liner	
Hydrogen Control	Computational Code	MAAP 4.06+	MAAP 5.03	MAAP 4.0.8	
	Counter Measures	• Passive Autocatalytic Recombiners (PAR) - 30 Units • Igniters - 10 Units • Safety Depressurisation and Vent System (SDVS) 3-way Valves	• PARs - 30 Units • Igniters - 10 Units • SDVS 3-way Valves	• PARs - 30 Units • Igniters - 8 Units • Power Operated Safety Relief Valves (POSRV)and 3-way Valves	
	FLC of containment for SA	105 psig	109 psig	109 psig	
Steam Explosion	Computational Code	• In vessel - TRACER-II • Ex vessel - TEXAS-V • Steam Spike : CONTAIN2.0	• In vessel - TRACER-II • Ex vessel - TEXAS-V • Steam Spike : CONTAIN2.0	TEXAS-V	
	Counter Measures		Reactor cavity design	Reactor cavity design	
Direct Containment Heating/High Pressure Melt Ejection (DCH/HPME)	Computational Code	MAAP 4.06+	MAAP 5.03	MAAP 4.0.8 (for rapid depressurization analysis)	
	Counter Measures	POSRVs	• POSRVs • Convoluted vent path	• POSRVs • Convoluted vent path	
Containment Performance (CP)	Computational Code	MAAP 4.06+	MAAP 5.03	MAAP 4.0.8	
	Counter Measures	Emergency Containment Spray Backup System (ECSBS)	ECSBS	ECSBS	
Equipment Survivability (ES)	Computational Code	MAAP 4.06+	MAAP 5.03	MAAP 4.0.8	
	Counter Measures	ES report	ES report		
Evaluation of External Injection capability to primary & secondary sides	Computational Code	MAAP 5.03	No analysis provided	RELAP5/Mod 3.3	
	Counter Measures	External cooling water injection to primary and secondary side of Reactor Coolant System (RCS)	External cooling water injection to primary and secondary side of RCS	External cooling water injection to primary and secondary side of RCS	

(3) Conclusion

The tables provided in this document provide background information to the members of the APR-1400 Severe Accident Technical Expert Subgroup on factors that are relevant to the review of Severe Accident provisions that are either in place, for the existing APR1400 nuclear power plants in operation, or are proposed for those APR1400 nuclear power plants that are either under construction or undergoing design review in the MDEP member countries. As part of preparatory efforts to promote common understanding among members, this document is intended to provide some familiarity with the various factors which may influence the outcomes of the Severe Accident analyses and is intended to assist future discussions in the specific areas of interest related to Severe Accidents.

(4) Revision Summary

Revision No.	Date	Summary of Changes
0	Septmember 2017	New document