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Comparative Assessment of Radioactive Contamination in APR 1400 Reactor During SBO and TMI Accidents

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ABSTRACT

In this research, the radiological contamination caused by the Station Blackout (SBO) accident in the Advanced Power Reactor 1400 (APR 1400) reactor was investigated. The results of the investigation of the Fukushima accident in Japan showed that such an event can lead to severe radioactive pollution in the environment surrounding a nuclear power plant. This research assumes that the SBO accident happened at one of the reactors of Unit 1 at the Baraka Nuclear Power Plant. This research mainly focuses on the type and quantity of radioactive materials released into the environment after containment failure occurs. MELCOR 1.8.6 code calculations show that it takes approximately 77.56 hours from the onset of the accident until the failure of the reactor containment building and the subsequent release of radioactive materials. According to these calculations, the largest mass of released radioactive materials is related to noble gases, with about 364.98 kg entering the environment and 5.426 kg of cesium iodide aerosol are released as well. These results demonstrate that the extent of radioactive contamination depends strongly on the type of radioactive species released.

INTRODUCTION

In this research, the level of radioactive contamination resulting from two types of severe nuclear accidents involving the APR 1400 reactor is investigated. The APR 1400 reactor is assumed to be located in Unit 1 of the Barakah Nuclear Power Plant (BNPP) in the United Arab Emirates. The accidents modeled for this reactor are the SBO (Station Blackout) accident and the Three Mile Island (TMI) accident [1]. We compared and analyzed the effects of the resulting radiation. Table 1 compares TMI Unit 2 and BNPP Unit 1 specifications.

Table 1. Comparison of TMI Unit 2 and BNPP reactor [1].

Item	TMI Unit 2 reactor	BNPP reactor
Reactor type	PWR	PWR
Location	Pennsylvania, USA	UAE
Capacity	906 MWe	1400 MWe
Date of accident	March 28, 1979	February 3, 2024

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APR 1400

The APR 1400 reactor is a pressurized light water reactor with an electric power of 1400 megawatts, which was developed in South Korea in 2002. The name of the APR 1400 reactor stands for Advanced Power Reactor, and the number 1400 indicates the electrical power of this reactor. The design and development of the APR 1400 reactor were carried out using the experience in the construction of the first pressurized light water reactor in South Korea, the Optimized Power Reactor (OPR1000), with an electric power of 1000 megawatts. The OPR 1000 reactor was South Korea's first experiment in constructing light water reactors under pressure, which led to the development of the APR 1400 reactor with the improvement of safety systems and further optimizations. The Optimum Power Reactor 1000 was Korea's main power plant model for many years. However, the plant's cooling system broke down, leading to its unprecedented shutdown. The Korea Hydro and Nuclear Power Company improved the plant by developing the APR 1400 in response to the breakdown of the cooling system. During 2 years, the initial and conceptual design of the APR 1400 reactor was completed, and the first part of this research was completed in 1994. At this point, all the advanced plans regarding the design and construction of pressurized light water reactors were studied, and a general plan about the operation of safety systems and the selection operator for light water reactors was evaluated. By studying other development plans for advanced light water reactors, the safety and economic objectives of the APR 1400 reactor were determined. In the second part of the design, a basic plan that includes all the needs raised in the first part of the research was designed. Also, at this stage, practical tests were conducted on advanced features introduced in advanced light water reactors, including the nuclear steam transfer system, in parallel during production and design in the factory. And finally, according to the complete design of the reactor systems in the factory, a Standard Safety Analysis Report (SSAR) was prepared. And in 1999, this part of the design was finished. During the third part of the design, which continued until 2001, more optimizations were made in order to improve the safety and performance of the reactor, as well as to make the reactor more economical, and in 2002, it received the design approval of the Standard from the regulatory organization of South Korea [2,3].

Barakah Nuclear Power Plant (BNPP)

The United Arab Emirates signed a contract with South Korea on December 27, 2009, for the construction of 4 APR 1400 reactor units. The construction site of this nuclear power plant is located in an area called Barakah, which is west of Abu Dhabi. According to the contract between the United Arab Emirates and South Korea, the first APR 1400 reactor will be put into operation by 2017, and one reactor will be put into operation every year until 2020. The initial safety analysis report related to the APR 1400 reactor for the Barakah Nuclear Power Plant (BNPP) was submitted to the United Arab Emirates Nuclear Regulation Authority in 2010, and a construction permit was issued in 2012. The initial concreting for the construction of the power plant started in October 2012. The reactor core was placed in the power plant in July 2014, and a fuel loading permit was issued in October 2016, and an economic operation permit was issued on May 1, 2017. The design reference of the BNPP is based on the model of Shin Cori 3 and 4 power plants. But due to different climatic conditions, several changes were made in the design of the power plant. Some of these changes are in the part of the seismic response spectrum, the heat sink temperature of the power plant and the ambient air temperature, and the 50 Hz power transmission network. Also, in the issues related to the nuclear safety of the BNPP, terrorism issues such as plane crashes, terrorist operations, and cybersecurity have been seriously considered.

Station Blackout (SBO) accident

The shutdown of the station, if it is accompanied by the failure of the auxiliary power source, will cause the occurrence of serious nuclear accidents and meltdowns and the release of radioactive materials into the environment. Since many of the main and safety systems of nuclear reactors work with AC electricity, if the main and emergency power sources of the power plant are cut off, their operation will stop, and this issue causes very serious concerns in the design and operation of nuclear reactors. In the design safety plans of nuclear reactors, an emergency alternating current source is always considered so that the cooling system of the heart of the reactor takes the heat caused by nuclear decay, and if this emergency power source fails, the situation can quickly become critical and lead to severe nuclear accidents. In the article by F. Ghaderinia, the detailed modeling and stages of the SBO accident process in the APR 1400 reactor up to the failure of the concrete structure of the reactor containment are described in detail [1]. This article only mentions the level of contamination and release of radioactive materials.

Three Mile Island (TMI) accident

On March 28, 1979, an accident occurred at the Three Mile Island reactor in Pennsylvania. This accident has been one of the most severe nuclear accidents to date. As a result of this accident, a small amount of radioactive particles leaked into the outside environment. In this accident, a fault was found in the operation of the steam generator feeding system. Due to this failure, the turbine generator was automatically stopped, and the control rods were directed into the reactor to reduce its power. Three auxiliary feeding pumps were supposed to provide the necessary water, but they could not because one of the valves in the pipe that goes to the steam generator was closed by mistake. It took 8 minutes to discover this issue, and the valve was opened, and the steam generator dried up, thus the pressure of the cooling circuit increased to 160 atmospheres and caused the pressurizer relief valve to open. The pressurizer relief valve remained open for 2 hours, and the amount. A lot of cooling water left the reactor cooling circuit. By reducing the pressure of the cooling circuit to 109 atmospheres, the emergency cooling system of the core should have been used, but this did not happen due to the fullness of the pressurizer tank, and the reactor operator even turned off the main pumps of the cooling circuit. This leakage of water in the circuit and the simultaneous increase.

In water temperature, the effect of nuclear fission led to the melting of fuel rods and the release of radioactive fission fragments in the reactor building and leakage into the environment. In 2022, Mr. Almteiri modeled a TMI accident for the APR 1400 reactor using PC Transient Analyzer software (PC Tran) [2]. PC Tran is software that runs under Windows and can model a variety of transient and severe nuclear accidents for the APR 1400 reactor. Figure 1 shows the APR 1400 reactor along with all safety and cooling systems. The amount of radioactive material leakage calculated by PC Tran software is given in Table 2.

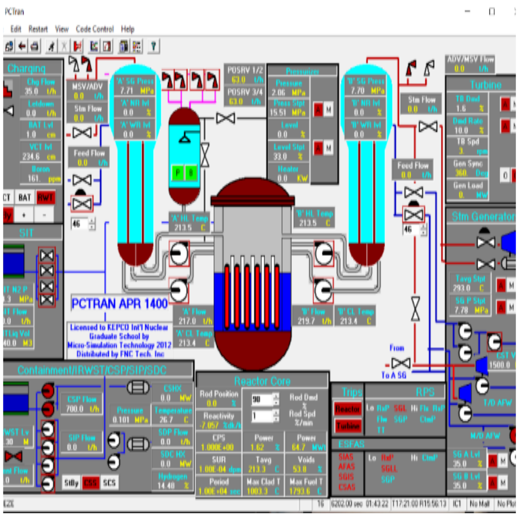

Fig. 1. PC Tran APR 1400 model [4].

Table 2. The amount of radioactive materials calculated by the PC Tran software [4].

Nuclides	Activity (Bq)	Nuclides	Activity (Bq)
I^{131}	$3.77*10^{10}$	I^{132}	$4.53*10^{10}$
I^{133}	$6.05*10^{10}$	I^{134}	$9.60*10^{10}$
I^{135}	$3.42*10^{10}$	Kr^{83m}	$1.14*10^{10}$
Kr^{85m}	$4.92*10^{11}$	Kr^{85}	$4.19*10^{11}$
Kr^{87}	$1.17*10^{10}$	Kr^{88}	$6.40*10^{12}$
Xe^{131m}	$1.79*10^{10}$	Xe^{133m}	$4.20*10^{11}$
Xe^{133}	$6.42*10^{12}$	Xe^{135m}	$1.17*10^{10}$
Xe^{135}	$1.84*10^{10}$	Xe^{138}	$1.64*10^{10}$
Co^{58}	$2.39*10^{3}$	Co^{60}	$2.39*10^{3}$
Rb^{86}	$2.39*10^{3}$	Sr^{89}	$2.39*10^{3}$
Mo^{99}	$2.39*10^{3}$	Cs^{134}	$2.39*10^{3}$
Cs^{136}	$2.39*10^{3}$	Cs^{137}	$2.39*10^{3}$

METHODOLOGY

APR 1400 MELCOR model

MELCOR code is a software used for accurate modeling of the process of severe nuclear accidents in light water reactors [5-10]. This code was prepared and developed at the request of the US Nuclear Regulatory Commission and Sandia National Laboratory, and this code was used as an integrated tool for risk assessment in second-generation reactors and as an alternative to other codes. By this code, a wide range of interactions and events that occur in severe nuclear accidents are modeled. This phenomenon includes the thermohydraulic response of the reactor cooling circuit, the destruction and movement of materials inside the reactor core, the interaction of molten materials inside the reactor core and concrete, the spreading and dispersion of dust and radioactive materials inside the reactor and in the environment, and hydrogen gas combustion. And other gases. The current application of the MELCOR code is to simulate severe nuclear accidents, perform sensitivity tests, and check the uncertainty of other parameters.

APR 1400 cooling circulation model in MELCOR

APR 1400 reactor is a PWR-type reactor and has two cooling circuits. Figure 2 shows a schematic diagram of the reactor cooling circuit modeled for simulation in the MELCOR code [1]. Each reactor cooling circuit loop has 11 control volumes. A hot leg, a steam generator inlet plenum, three control volumes for the SG U-tube hot side, one control volume for the Steam Generator (SG) U-tube cold side, an SG outlet plenum, two intermediate legs, and two cold legs. The pressurizer is connected to the hot leg of loop A. One Pilot-Operated Safety Relief Valve (POSRV) is on top of the pressurizer, and the POSRV is designed for controlling RCS pressure. POSRV fully opens at a high pressure of 17.51 MPa and closes when the reactor cooling circuit pressure is reduced to a blowdown set point of 17.50 MPa. APR 1400 containment is shown in Fig. 3 [1]. Containment is subdivided into 12 control volumes. Containment control volumes consist of cavity (CV801), chamber room (CV802), RPV annulus (CV803), refueling room (CV 804), two steam generator components (CV 805& CV 806), pressurizer component (CV 807), upper component (CV808), containment dome (CV 809), annular component (CV 810), hold up volume tank (CV 811), and IRWST tank (CV 812). The environment is modeled by CV 813.

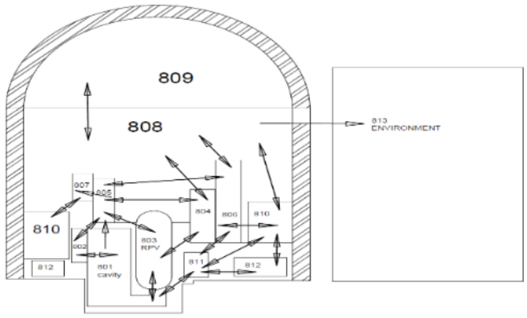

Fig. 2. APR 1400 RCS model in MELCOR.

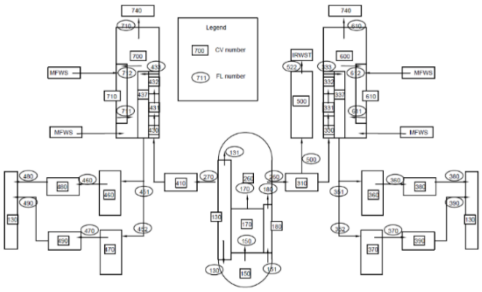

Fig. 3. APR 1400 containment model for MELCOR [1].

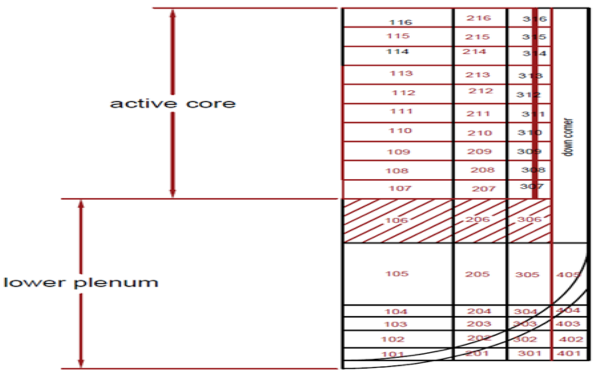

Fig. 4. Core nodalization schematic [1].

Valve FL 848 is used to model containment failure. FL848 is defined as a conditional valve in the reactor modeling in the MELCOR code, and when the pressure inside the containment building exceeds 1.4 MPa, this valve opens permanently. In Table 3, the volume control numbers used in the MELCOR code model for simulating the containment building circuit of the APR 1400 reactor are summarized. Core and lower plenum nodalization are shown in Fig. 4 [1]. The whole core is divided into 4 control volumes. The definition of these volume controls is as follows: CV1 is the core channel. CV2 is the downcomer channel, CV3 is the bypass channel, and CV4 is the lower head plenum. The core is radially divided into 5 rings and axially divided into 16 levels. 3 rings are in the active core region. Ring 4 is located in the bypass control volume, and ring 5 is located in the downcomer control volume. For axial levels nodalization, 6 levels are located in the lower head plenum, and 10 levels are inactive core and bypass, and downcomer regions [9,10].

RESULT AND DISCUSSION

Investigating and comparing the amount of radioactive contamination in the SBO and the Three Mile Island accidents in the APR 1400 reactor

In this section, the level of radioactive contamination in the APR 1400 reactor has been investigated. According to MELCOR code calculations, the source term and emission rate are much higher than the Three Mile Island accident in the APR 1400 reactor.

Table 3. APR 1400 containment nodalization in MELCOR 1.8.6 [1].

CV NUMBER	CV NAME	CV NUMBER	CV NAME
310	HOT-LEG-AB	600	SGAB-2ND
330	SGAB-HOT1	610	SGAB-2ND-DC
331	SGAB-HOT2	700	SGCD-2ND
332	SGAB-HOT3	710	SGCD-2ND-DC
337	SGAB-COLDTUBE	740	TURBINE
360	INTER-A1	801	Cavity
380	COLD LEG-A	802	Chamber room
370	INTER-B1	803	RPV annulus
390	COLD LEG-B	804	Refueling pool
410	HOT-LEG-CD	805	S/G loop A
430	SGCD-HOT1	806	S/G loop B
431	SGCD-HOT2	807	Pressurizer
432	SGCD-HOT3	808	Containment
437	SGCD-COLDTUBE	809	Containment dome
460	INTER-C1	810	Annular comp
480	COLD LEG-C	811	Holdup volume tank
470	INTER-D1	812	IRWST
490	COLD LEG-D	813	Environment
500	PRESSURIZER		

Table 4. Source term of nuclides.

Nuclide	Categories	Activity (Bq)
I^{131}	Halogens	5.26e+23
I^{132}	Halogens	9.29e+19
I^{133}	Halogens	1.30e+21
I^{134}	Halogens	6.28e+19
I^{135}	Halogens	3.80e+20
Xe^{131m}	noble gases	3.37e+20
Xe^{133m}	noble gases	6.86e+13
Xe^{133}	noble gases	1.07e+20
Xe^{35m}	noble gases	4.5e+22
Cs^{134}	Alkali metal	5.47e+13
Cs^{136}	Alkali metal	5.47e+13
Cs^{137}	Alkali metal	5.47e+13

In the scenario of the Three Mile Island (TMI) accident for the APR 1400 reactor, which was carried out by M. Almteiri and J. Kim, it was determined that the amount of 14.233 PBq of noble gas was released [4]. The emission of radioiodine was 187.21 GBq. These figures were less than 93 PBq noble gases and 560 GBq radioiodine generated during the TMI accident. However, the calculated amount of radioactive materials leaked from the APR 1400 reactor due to the SBO accident is much higher than the Three Mile Island (TMI) accident. The amounts of radioactive materials that have leaked out from the APR 1400 reactor in the SBO accident are summarized in Table 4. The values of this table are obtained by the MELCOR 1.8.6 code and from the SBO accident simulation for the APR 1400 reactor. The emission rate of noble gases leaked from the reactor due to the SBO accident is 188 ZBq. Because in this article, the worst-case scenario of the SBO accident modeling was considered, and it was assumed that all reactor safety systems were disabled due to a power outage, that there were no electrical batteries, and that the entire core had melted due to the accident. The noble gas release in the APR 1400 reactor during the SBO accident is approximately 1.5 million times higher than that in the Three Mile Island accident in this reactor. The amount of radioactive iodine released from the SBO accident in the APR 1400 reactor is also much higher than the TMI accident. The amount of iodine released in the SBO accident was 528 ZBq and 187.21 GBq in the TMI accident in the APR 1400 reactor. The contamination level of the SBO accident is significantly higher than that of the Three Mile Island (TMI) accident in the United States of America. In the TMI accident, 93 PBq of noble gas and 560 GBq of radioactive iodine were released.

Table 5. Comparison of the source term of nuclides in the APR 1400 reactor and the TMI reactor.

	APR1400 SBO	APR1400 TMI accident
Iodine Activity (Bq)	528e+21	187.21e+9
Nobel gases activity (Bq)	188e+21	14.23e+15

CONCLUSION

This research examines the amount of radioactive pollution caused by two accidents involving the APR 1400 reactor. The Station Blackout (SBO) accident was simulated by the MELCOR 1.8.6 code, and the amount of pollution was calculated in this simulation. The modeling of the Three Mile Island (TMI) accident for the APR 1400 reactor was done by Mr. Almteiri, and the source term was obtained using PC Tran. The actual emitted Source Term and amount of emission were comparable to the TMI accident emissions. The emission of noble gases was 14.233 PBq, while the emission of radioiodine was 187.21 GBq. These figures were less than 93 PBq noble gases and 560 GBq radioiodine generated during the TMI accident. The results of the calculations with the MELCOR code show that in the event of an SBO accident for the APR 1400 reactor, the amount of radioactive material leakage is much higher than that of the Three Mile Island (TMI) accident. The amounts of radioactive contamination in all 3 accidents are summarized in Table 5. The amount of iodine released from the SBO incident is 5.667 million times that of the Three Mile Island (TMI) accident. Also, the amount of noble gases released from the SBO incident is 3.357e+11 times that of the Three Mile Island (TMI) accident. In general, MELCOR's calculations show that in the event of an SBO accident in the APR 1400 reactor, the amount of radioactive material leakage is very high, and the dimensions of the accident are very large and serious.

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AUTHOR CONTRIBUTION

F. Ghaderinia, M. Rahgoshay, and J. Jafari contributed equally as the main contributors of this paper. All authors read and approved the final version of the paper.

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