Bachelorarbeit

Design and Development of Safety Concepts for Pressurized Water Reactors from Generation II to Generation III+

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Abstract

The safety of nuclear power plants has been and still is an important topic in our society. It has also been subject to rapid advancements in the last decades. The aim of this thesis is to describe safety systems of three different pressurized water reactors from Generations II, III and III+, and moreover to demonstrate the differences and improvements made throughout these iterations. To this end, a wide range of technical literature was examined. The Generation II KWU pre-convoi reactors feature a balanced set of passive and active safety systems. Those are able to counteract design basis accidents, whereas the KWU plants lack systems that are designed for severe accidents, including core melt accidents. With the Generation III APR-1400 and the Generation III+ EPR, improvements were made by simplifying key safety systems, further reinforcing the containment and integrating arrangements preventing and mitigating severe accident scenarios, especially core melt accidents. The Generation III+ AP-1000 incorporates extensive passive design elements and safety systems. Their working principle is based on natural forces, offering advantages regarding their self-sufficiency.

Kurzfassung

Die Sicherheit von Kernkraftwerken war und ist ein bedeutendes Thema unserer Gesellschaft. Sie war auch Gegenstand rasanter technologischer Weiterentwicklung in den letzten Jahrzehnten. Das Ziel dieser Arbeit ist, die Sicherheitskonzepte von drei verschiedenen Druckwasserreaktoren der Generationen II, III und III+ zu beschreiben und darüber hinaus deren Unterschiede und Verbesserungen aufzuzeigen. Zu diesem Zweck wurde eine umfangreiche Auswahl an Fachliteratur betrachtet und analysiert. Die KWU Vor-Konvoi Reaktoren der Generation II weisen ein ausgewogenes Spektrum an passiven sowie aktiven Sicherheitseinrichtungen auf. Diese sind konzipiert, um Auslegungsstörfallen entgegenzuwirken. Es fehlen jedoch Einrichtungen, welche für schwere Störfälle bzw. Unfälle wie einen Kernschmelzunfall entworfen sind. Mit den Reaktoren APR-1400 der Generation III sowie EPR der Generation III+ wurden sicherheitstechnische Verbesserungen durch die Vereinfachung wichtiger Sicherheitssysteme sowie durch die Verstärkung des Sicherheitsbehälters erreicht. Um schweren Unfällen, insbesondere Kernschmelzunfällen entgegenzuwirken, werden nun spezielle Systeme eingesetzt. Der Generation III+ Reaktor AP-1000 verwendet tiefgreifende passive Sicherheitseinrichtungen. Deren Funktionsprinzip basiert auf Naturkräften, wodurch diese Systeme weitgehend autark arbeiten können.

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List of Abbreviations

ABWR	Advanced Boiling Water Reactor
AGR	Advanced Gas-Cooled Reactor
ALWR	Advanced Light Water Reactor
AP1000	Advanced Passive 1000
APR-1400	Advanced Power Reactor 1400
AVS	annulus ventilation system
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium
CF	capacity factor
CFS	cavity flooding system
CHRS	containment heat removal system
CSS	containment spray system
DBA	design basis accident
DCH	direct containment heating
EBR-I	Experimental Breeder Reactor I
ECCS	emergency core cooling system
EFWS	emergency feedwater system
EPR	European Pressurized Water Reactor
ERVCS	external reactor vessel cooling system
ESF	engineered safety feature
FBR	Fast Breeder Reactor
FD	fluidic device

GCR	Gas Cooled Reactor
GWe	gigawatt-electric
HLW	high-level waste
IAEA	International Atomic Energy Agency
IEA	International Energy Agency
IRWST	in-containment refuelling water storage tank
KEPCO	Korea Electric Power Corporation
Korea	Republic of Korea
KSNP	Korean Standard Nuclear Power Plant
KWU	Kraftwerk Union AG
LBLOCA	large-break loss-of-coolant accident
LEU	low-enriched uranium
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LWGR	Light Water-Cooled Graphite-Moderated Reactor
LWR	Light Water Reactor
MOX	mixed oxide
MPa	Megapascal
MSLB	main steam line break
Mtoe	megatonne of oil equivalent
MWe	megawatt-electric
MWth	megawatt-thermal
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
PCCS	passive containment cooling system
PHWR	Pressurized Heavy Water Reactor
POSRV	pilot operated safety relief valve
PRHR HX	passive residual heat removal heat exchanger
PRHRS	passive residual heat removal system

PWR	Pressurized Water Reactor
PXS	passive core cooling system
RBMK	Reaktor Bolshoy Moshchnosti Kanalnyy
RCC	rod cluster control
RCP	reactor coolant pump
RCS	reactor coolant system
RHRS	residual heat removal system
RPS	reactor protection system
RPV	reactor pressure vessel
RY	reactor-year
SBLOCA	small-break loss-of-coolant accident
SCS	shutdown cooling system
SDVS	safety depressurization and vent system
SGTR	steam generator tube rupture
SIP	safety injection pump
SIS	safety injection system
SIT	safety injection tank
SMR	Small Modular Reactor
TMI-2	Three Misle Island Unit 2
TWh	terrawatt-hour
UAE	United Arab Emirates
UK	United Kingdom of Great Britain and Northern Ireland
USA	United States of America
VVER	Vodo-Vodyanoi Energetichesky Reaktor

1. Introduction

1.1. History

Fermi, Amaldi, Agostino, *et al.* [1] found in 1934 that by targeting elements with recently discovered neutrons, artificial and radioactive isotopes could be created. Fermi observed that heavier elements were formed, but for uranium, both heavier and lighter isotopes could be detected [2]. Performing similar experiments, Hahn and Strassmann demonstrated in 1938 that by bombarding uranium (Z=92) with neutrons, lighter elements like barium (Z=56) and others were produced. Hahn subsequently wrote to his long-time friend Lise Meitner about this unexpected observation and asked for her opinion. [3] Frisch and Meitner explained this "fission" process in their groundbreaking publication "Disintegration of Uranium by Neutrons: a New Type of Nuclear Reaction" by suggesting that the uranium nucleus captures a neutron and splits into two nuclei, which are roughly of equal size. The kinetic energy released by this process was calculated in the paper from the nuclear radius and charge of the repelled parts, and was determined to be around 200 MeV [4]. Frisch would experimentally confirm this figure later in January 1939. In the same year, Hahn and Strassmann would also demonstrate with other experiments that additional neutrons were released during the fission process. [2] Bohr delivered the important theory that a nuclear fission was more likely to happen with slow neutrons and with the isotope uranium-235, compared to fast neutrons and uranium-238 [2]. Szillard and Fermi proposed using a moderator to slow down or "thermalize" the neutrons [2]. Consequently, Bohr and Wheeler developed the theory of nuclear fission, which lead to their landmark paper "The mechanism of nuclear fission," published in the year 1939 just a few days before the outbreak of the Second World War [6, p. 20]. As mentioned in [2], the British physicist community recognized the potential of a nuclear

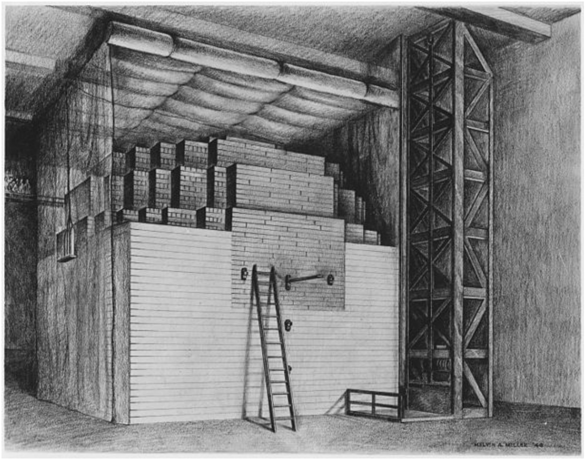

Figure 1.1.: Drawing of the first artifical reactor named Chicago Pile-1 [7]

fission device designed as a bomb, and thus a committee was formed in order to research the feasibility of such a nuclear bomb, which would be built and deployed during the war. This MAUD Committee concluded in two summary reports in July 1941 that [2]:

1. A nuclear bomb was feasible. A bomb containing 12 kg of active material would be as destructive as 1800 tons of TNT.
2. The controlled fission of uranium could be used for peaceful purposes like providing energy and further generating electricity.

The Americans heavily committed to develop and deploy a nuclear bomb with the "Manhattan Project" when they entered World War II. As part of their weapons development program, the world's first controlled nuclear reactor was designed and built under Enrico Fermi at the University of Chicago in December 1942: The Chicago Pile-1, pictured in Figure 1.1. [2] With the Second World War over, the attention of nuclear energy shifted to the peaceful usage, mainly to producing electricity by controlled nuclear reactions, although the development of nuclear and thermonuclear weapons would proceed in the era of the Cold War. The Experimental Breeder Reactor I (EBR-I) would become the first reactor generating electricity, when started up in December 1951. It was also the first reactor based on the "breeding" principle, producing more fissile material than consuming during operation.

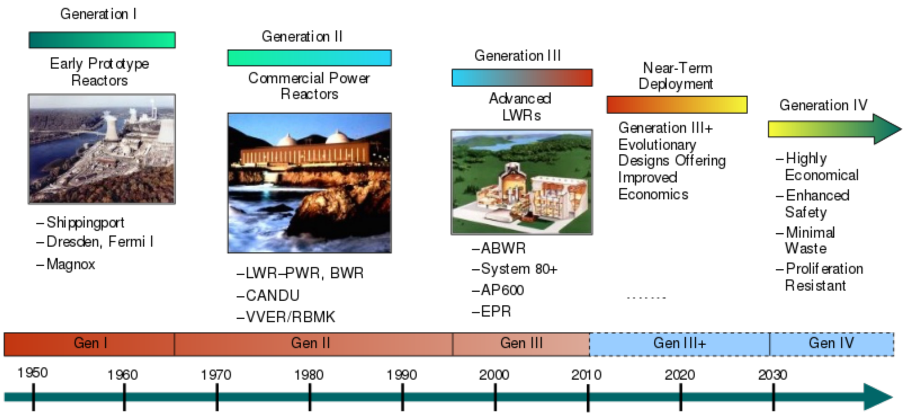

Figure 1.2.: The Evolution of Nuclear Power Reactor Generations [9]

Nuclear energy would also be valuable for naval propulsion, with the US developing the Pressurized Water Reactor (PWR) for boats and ships. In 1954, this design would power the world's first nuclear submarine, the USS Nautilus, and was later adapted for stationary power plants. [2] Being the first PWR and the third nuclear power plant (NPP) for commercial use in the United States of America (USA), the Yankee Rowe Nuclear Power Station went operational in 1960 [3]. Light Water Reactor (LWR)-designs with the utilization of low-enriched uranium (LEU) would become the prevalent reactor type around the world, with very few countries deviating from this approach, e.g. Canada with their Canada Deuterium Uranium (CANDU) design, which uses natural uranium as fuel and heavy water (D_2O) as the moderator. The two established LWR-designs are the PWR and the Boiling Water Reactor (BWR), with the PWR generating 69% of the current world's capacity and the BWR 20% respectively. [2]

1.2. Nuclear Reactor Generations

As described by Goldberg and Rosner [8], nuclear reactors are most commonly divided into five different generations (see Figure 1.2), based on their design and safety characteristics:

Generation I: During the 1950s and 1960s, first commercial prototype reactors were built. Those reactors would demonstrate the ability of nuclear energy to be used for a civil purpose ("proof of concept") and operated at rather low power levels. Examples include Shippingport, Dresden-1 and Fermi-1 in the USA, as well as Calder Hall-1 in the United Kingdom of Great Britain and Northern Ireland (UK). [8]

Generation II: This reactor generation was optimized for commercial use, mainly improving economics and reliability as well as implementing active safety features, which were controlled automatically or could be triggered by an operator. The operational lifetime was projected to be at least 40 years. The first units went into operation in the late 1960s, and typical reactor types for this generation include PWRs (see chapter 3), BWRs, CANDU reactors, Advanced Gas-Cooled Reactors (AGRs) as well as the Soviet version of a PWR, the Vodo-Vodyanoi Energetichesky Reaktor (VVER), and the Soviet Reaktor Bolshoy Moshchnosti Kanalnyy (RBMK). [8] As seen in Figure 1.3, the majority of today's nuclear power reactors supplying the biggest share of capacity are between 30 and 40 years old and can be therefore categorized as Gen II reactors. The biggest circle in Figure 1.3 represents 32 reactors supplying 31.96 gigawatt-electric (GWe) at 36 reactor-years (RYs), followed by 30 reactors providing 28.63 GWe at 37 RYs [10].

Generation III: As described in [8], significant improvements were made in comparison to the previous generation regarding fuel technology, thermal efficiency, standardization and safety, especially by utilizing passive safety systems. The expected operational lifetime was extended to 60 years. This generation was developed in the 1990s, and criticality of the first commercial unit was reached in 1996 in Japan. [8] This generation was dominated by LWRs, and examples of this generation include the Advanced Boiling Water Reactor (ABWR) and the Advanced Power Reactor 1400 (APR-1400) (see chapter 4).

Generation III+: This generation was a subsequent development of existing Gen III designs with the main goal to further improve plant efficiency and safety by integrating extensive passive safety systems. Examples are the EU-ABWR, the VVER-1200, the European Pressurized Water Reactor (EPR) (see section 5.1) and the Advanced Passive 1000 (AP1000) (see section 5.2). Small Modular Reactors (SMRs) were also further developed based on

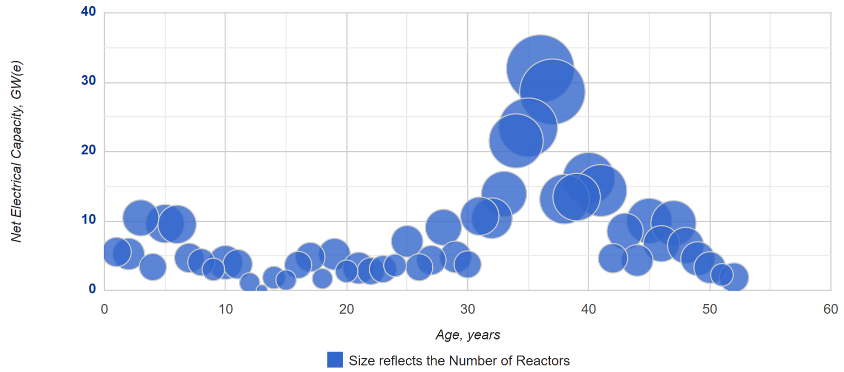

Figure 1.3.: Operational reactors by age and net electrical capacity (in GWe) [10]

down-scaled Gen III and III+ designs. SMRs are expected to be deployed in the near future, although some designs are based on Gen IV technology and will therefore not be deployable in the near term. [8]

Generation IV: As outlined in [8], the future reactor generation will implement all the advantages and safety characteristics of Gen III+ reactors but feature various different reactor concepts to ensure safer, broader and more flexible usage of nuclear power, e.g. the operation at high temperatures for hydrogen production and thermal energy off-taking. As described in [8], [11], proposed fast neutron reactor concepts could potentially reduce the quantity of high-level waste (HLW) in used fuel and burn actinides. Prototypes are currently being built and operated in different countries, inter alia China, India and Russia, although further extensive research and development is necessary [8].

1.3. Nuclear Power Today

In 2020, the total world's electricity production equalled 26 730 terrawatt-hour (TWh), with nuclear power supplying around 10% [12]. In 2018, NPPs generated 2710 TWh [13]. In 2019, this figure stayed nearly the same at 2657 TWh, until decreasing slightly in 2020, with an output of 2553 TWh [12]. The relative share of nuclear power compared to the total electricity production worldwide has decreased slowly from 15% in 1985 to today's figure of 10%. With 10.4%, nuclear power is the second-largest contributor to low-carbon electricity, exceeded by hydropower with a share of 15.8%. Around 37% of the total electricity is provided by low-carbon sources, while 63.3% is generated from fossil fuels, consisting primarily of oil, coal and gas. This ratio has been virtually unchanged since 1985, partially because of the decline in share of nuclear power while the output from other renewables increased. [14] When the nuclear and low-carbon share of the total primary energy supply is compared to the fossil fuels', this picture solidifies even more drastically: Only about 4.9% (706.8 megatonne of oil equivalent (Mtoe)) originated from nuclear in 2018 [13], and 4.3% in the following year [15]. In 2019, low-carbon sources contributed 15.7%, while more than 84% was provided by fossil fuels, with oil being the largest contributor (33.1%), followed by coal and gas [15]. The world's total energy supply in 2018 was 14 282 Mtoe [13].

As of September 2021, the nuclear share in the world's electricity production is provided by 444 power reactors with a combined total net capacity of 394 586 megawatt-electric (MWe). These units are operated in 32 different countries (plus Taiwan) worldwide, while the total RYs of experience amount to 19 084. [16] The majority of today's operational reactors are PWRs with 303 units providing a net electrical capacity of 288 GWe (see Figure 1.4). Furthermore, there are 62 BWRs operational with a net electrical capacity of 63 GWe. The Pressurized Heavy Water Reactor (PHWR)-design is mostly deployed in Canada and India, and worldwide a total number of 49 reactors of this type provide a comparably small capacity of 24 GWe. Light Water-Cooled Graphite-Moderated Reactors (LWGRs), Gas Cooled Reactors (GCRs) and Fast Breeder Reactors (FBRs) are currently of smaller significance regarding the commercial use of nuclear power. [17]

The USA is still the world's largest producer of nuclear power. It has the largest net electrical capacity of around 95.5 GWe and the highest number of operational reactors at 93 [18], while the relative nuclear share was 19.7% in 2020 [19]. The USA is followed by France (61.4 GWe and 56 operational units) and China (49.6 GWe and 51 units) [18], although France has the highest nuclear share in the world, equalling 70.6% in 2020, while China's relative output was at only 4.9% [19]. As a direct repercussion of the Fukushima Daiichi accident in March 2011, eight reactors were shut down in Germany.

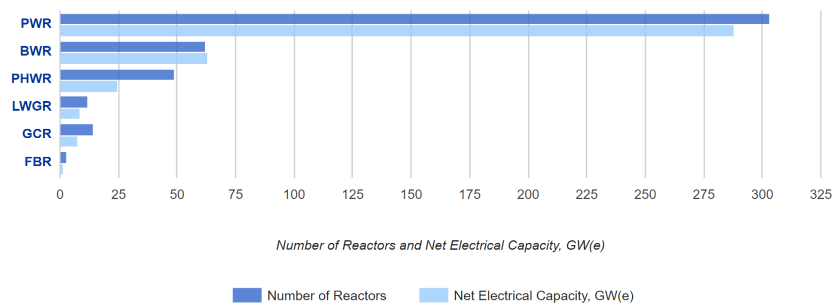

Figure 1.4.: Number of operational reactors and net electrical capacity in 2021 [17]

The European country decided to phase out its nuclear energy generation until 2022 as part of its new energy policy, and will therefore subsequently shut down all the remaining reactors. [20] As of Fall 2021, six operational reactors are providing 8.1 GWe in Germany, which amounts to a relative share of 11.4%, compared to the 30 shutdown reactors, which once provided a combined capacity of 18.26 GWe [18]. In Finland, four reactors have a net installed capacity of 2.8 GWe, and 33.9% of the country's electricity was generated by those units in 2020 [18]. The United Arab Emirates (UAE) has two operational APR-1400s with a capacity of 2.69 GWe, representing 1.1% of the electricity supplied [18]. Units one and two were connected to the grid in 2020 and 2021 respectively [21]. This reactor type and its inherent safety features will be examined and discussed in more detail in chapter 4.

1.4. Nuclear Power in the Future

The International Energy Agency (IEA) predicts in its report *World Energy Outlook 2020* [22, p. 344] that the global electrical capacity will reach 13 418 GWe by 2040, while the nuclear capacity will grow by 15% from 2019 to 2040. This would result in a generating capacity of about 479 GWe, therefore contributing a relative share of 8.5% to the global capacity. The increase in global production will be driven primarily by Asia, in particular by India and China [23]. With the goal of slowing down growth in global carbon dioxide emissions, there is a clear need for a timely transition from fossil fuel to low-carbon energy sources. The "Sustainable Development Scenario"

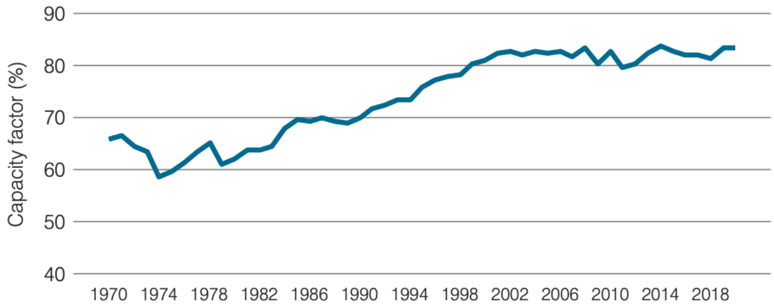

Figure 1.5.: Global average capacity factor [24, Fig. 4]

was introduced by the IEA and describes the measures needed for reaching certain climate goals while providing energy access [23]. It envisions nuclear electrical capacity to increase to 599 GWe by 2040 [22, p. 345].

In general, additional nuclear capacity from existing plants is created by improving the capacity factor (CF), by upgrading older plants and by extending the lifetime to up to 80 years. This is done particularly often in the USA, and is a cost-effective way of increasing capacity [23]. In the *World Nuclear Performance Report 2021* [24, pp. 6-7] it is illustrated that the global average capacity factor has been at a high level since the beginning of the 2000s, and the data indicates that there is no significant deviation depending on the reactor age (see Figure 1.5). Currently, there are around 50 reactors under construction in 19 different countries. About 100 more are on order or planned, and over 300 more are proposed, mainly in China, India and Russia. [23] Originally there have been four AP1000 reactors under construction in the USA, but following the bankruptcy of "Westinghouse Electric Company" two units were abandoned, while the units Vogtle 3 and Vogtle 4 were set to start commercial operation in 2022 [25]. This timeframe has been pushed back, and the start of commercial operation is expected for 2023 [26]. Two EPRs are being built in Europe: Olkiluoto 3 (1720 MWe) in Finland and Flamanville 3 (1650 MWe) in France (see Table 1.1). The UAE has ordered four APR-1400s, which are located at the NPP Barakah: Two units are operational since 2020 and 2021 [27], while Units 3 and 4 are under construction and will be connected to the grid in 2023. Currently, there are also four APR-1400 units under construction in the Republic of

Start	Country	Reactor	Model	Gross MWe
2022	Finland	Olkiluoto 3	EPR	1720
2022	Korea	Shin Hanul 1	APR-1400	1400
2023	Korea	Shin Hanul 2	APR-1400	1400
2023	Korea	Shin Kori 5	APR-1400	1400
2023	UAE	Barakah 3	APR-1400	1400
2023	UAE	Barakah 4	APR-1400	1400
2023	USA	Vogtle 3	AP1000	1250
2023	USA	Vogtle 4	AP1000	1250
2023	France	Flamanville 3	EPR	1650
2024	Korea	Shin Kori 6	APR-1400	1400
2026	UK	Hinkley Point C1	EPR	1720
2027	UK	Hinkley Point C2	EPR	1720

Table 1.1.: Selection of nuclear reactors currently under construction or awaiting grid connection [28]

Korea (Korea). [28] Table 1.1 consists of reactors under construction or of reactors which are expected to be connected to the grid in the near future. The mentioned reactor types will be of significance in subsequent chapters. Table 1.1 does not represent a complete list, and "Start" in it refers to the estimated year of grid connection.

1.5. Aim of this Thesis

The aim of this thesis is to give a structured overview of the design and development of safety concepts of PWRs, which characterize and distinguish different reactor generations. Furthermore, emphasis is placed on comparing similar safety designs, as well as on outlining improvements in safety design, philosophy and technical layout. Regarding Generation II safety systems, the KWU-PWR (NPPs Biblis and Grohnde) will be examined in more detail (see chapter 3). For Gen III safety systems, the APR-1400 design (see chapter 4) and for Gen III+, the EPR as well as the AP1000 (see chapter 5) will be discussed thoroughly.

2. Functionality and General Safety Concepts of PWRs

A PWR can be separated into two systems: The primary and the secondary system (see Figure 2.1) [29]. The most important elements of the primary system will be introduced in the following sections and the secondary system will be described briefly, although some arrangements and inherent safety features like the emergency core cooling system (ECCS) (see subsection 3.2.4) will be described in subsequent chapters.

2.1. Primary System

As outlined in [29], the reactor coolant system (RCS) (also called primary coolant system or primary loop) is a closed loop and has the main task of cooling the core and transporting the heat away from the fuel to the steam generators, where the heat is transferred to the secondary system. This closed environment also confines most of the radioactivity to this section of the plant [29]. The layout of the primary loop is shown in Figure A.1 and its main components include the reactor pressure vessel, steam generators, the reactor coolant pumps (RCPs), the pressurizer and the piping. As mentioned in [30], a PWR uses light water (H_2O) both as a neutron moderator and as its primary coolant. Furthermore, the water reaches temperatures of around 330 °C after passing the fuel [30], thus the RCS is kept under a pressure of around 150 bar via the pressurizer to prevent the coolant from boiling. The RCS operating pressure of the NPP Grohnde is 158 bar [31, p. 38], while it is 155 bar (15.5 Megapascal (MPa)) for the APR-1400 [32, p. 4], the EPR [33, p. 43] and the AP1000 [34, p. 30]. The power output of the reactor determines the number of primary loops [29]. In Figure A.2 a four-loop Westinghouse

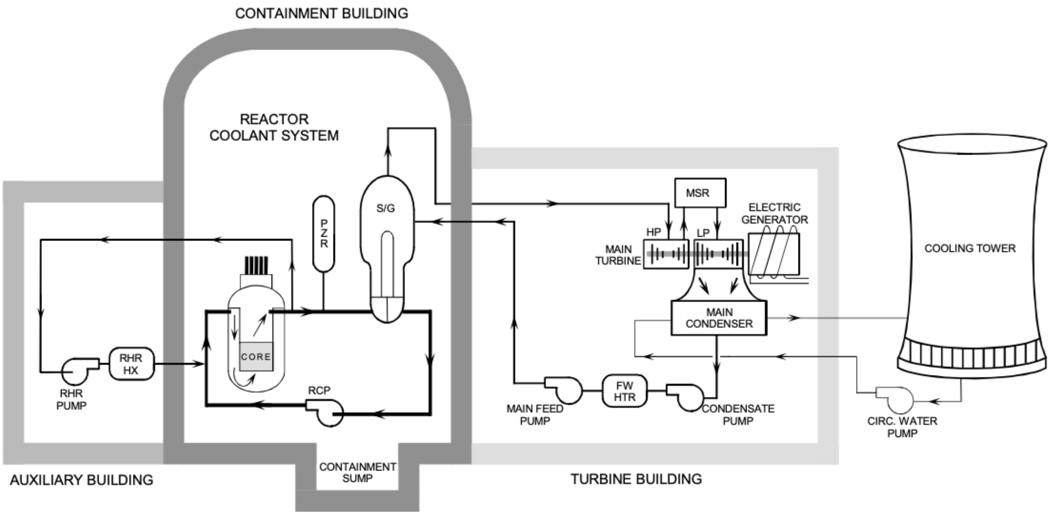

Figure 2.1.: Design-Layout of a PWR [29, p. 2]

system consisting of four steam generators, four RCPs and a pressurizer can be seen. This configuration is used for plants with a core that contains around 193 fuel assemblies (each consisting of 17x17 fuel elements) and with an electrical output rating ranging from 950 to 1250 MWe [29].

2.1.1. Reactor Pressure Vessel

As depicted in Figure A.3, the reactor pressure vessel (RPV) is a cylindrical structure made of manganese molybdenum steel with a removable top head for refueling purpose. It contains the reactor core, the core barrel and the upper internals package, which guides the control rods. The reactor coolant enters the vessel through the inlet nozzle (cold leg), passes through the reactor core and the fuel assemblies, where the heat is transferred and removed. The hot water exits the RPV through the outlet nozzle (hot leg) and transports the heat to the steam generator. [29] The configuration of a primary loop can vary depending on the technical layout of different reactor types: Kraftwerk Union AG (KWU)-PWRs [31, p. 137] and the EPR [33, Fig. 2.1-1] have a separate hot and cold leg, a steam generator and a RCP per loop (four loops in total), while the APR-1400 [32, pp. 15-16] and the AP1000

[34, pp. 3-4] feature a two-loop design, each consisting of a single hot leg, a steam generator and two cold legs with two RCPs (see Figure C.1).

2.1.2. Fuel and Control

The fuel is arranged in the nuclear core in the form of several fuel assemblies (around 193 for the four-loop Westinghouse system). A fuel assembly is a multi-component structure (see Figure A.4), housing e.g. 17x17 fuel rods. Each rod contains uranium-dioxide (UO_2) pellets and is enclosed by fuel cladding, which is typically made of a zirconium alloy, termed "Zircaloy". This material is selected because of its low absorption cross-section for neutrons and its corrosion resistance. [35] For LWRs, more specifically for PWRs, the enrichment grade of the deployed fuel is most commonly LEU. As described by Kok [35], the initial enrichment of the uranium of each fuel assembly ranges from around 2.10 % to 3.10 % U-235 for a modern-generation PWR. Other fuel materials, most notably mixed oxide (MOX) fuel, consisting of an additional oxide like PuO_2 to the UO_2, are also used in LWRs.

The reactivity and the power output of the core is controlled by the rod cluster control (RCC) assemblies. They are inserted into the core from the top by a drive shaft and the reactivity control is achieved by control rods, which are made of the material hafnium because of its high neutron absorption cross-section. In case of an emergency, also called a "trip", the RCC assemblies fall into the core by gravitational force and thus shutting it down. [35]

2.1.3. Steam Generators and Pressurizer

Coming from the core, the hot coolant reaches the steam generators, where it flows through tubes. These tubes are surrounded by the secondary coolant, which is also called feedwater. The heat is transferred to the secondary system and thus the feedwater starts to boil. The moisture from the steam is separated afterwards and is guided to the high-pressure turbine. This is done in order to prevent the blades of the turbines from being damaged. [29]

The pressurizer is connected to the primary loop via the surge line. It ensures that the pressure stays at a constant level, preventing the primary coolant from boiling. The system works with the use of electrical heaters and a pressurizer spray, which counteract pressure changes in the system. Furthermore, the pressurizer is connected to a pressurizer relief tank. [29]

2.2. Secondary System

The secondary system consists of the main steam system and the condensate/feedwater system, which are both located outside the containment building at the turbine building (see Figure 2.1). The dry steam from the steam generators passes through the high-pressure main turbine. It is then directed to another moisture separator/reheater system and afterwards to the low-pressure turbines. The steam has now transferred most of its energy to the turbines, which are connected to an electric generator, and is condensed into water in the main condenser. This is where the condensate/feedwater system takes over. The condensate is cleaned, heated up by low-pressure feedwater heaters and the pressure of the water is increased by the main feedwater pumps. The water is then pumped through the steam generators, where the loop starts again. [29]

2.3. General Safety Concepts

As described in [36], the on-site staff of a NPP as well as the public health in general have to be protected from the harmful effects of ionizing radiation. The safety philosophy of “Defense in Depth” is an integral part of the design-process, operation and maintenance of nuclear facilities. Furthermore, the objective of this approach includes protecting the environment while securing the functionality of the plant. “Defense in Depth” is achieved by the following design and safety principles [36], [37]:

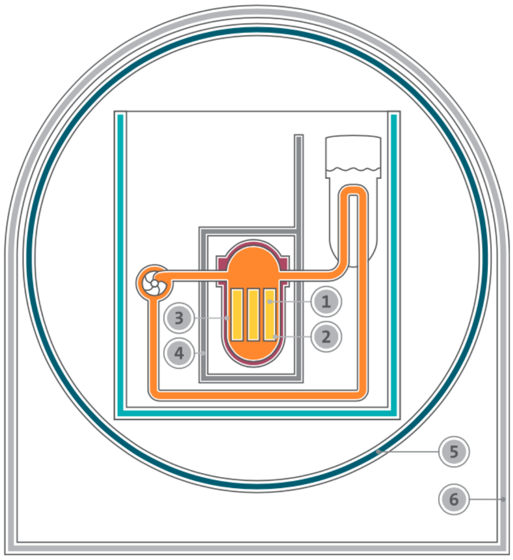

Figure 2.2.: Multiple safety barriers to prevent the release of radioactive material [38, Fig. 9]

Multiple Barriers: Multiple layers of safety barriers prevent the release of radioactive material. The most important barriers can be seen in Figure 2.2 and according to [37] consist of the:

1. **Structure of the Fuel:** Contains fission products and radiation.
2. **Fuel Cladding:** Fission products e.g. noble gases are kept within the fuel rods and do not come into contact with the primary coolant.
3. **Reactor Pressure Vessel:** All the integral parts of the core are housed within the RPV. It ensures the circulation of the coolant to remove heat and the steel wall absorbs radiation.
4. **Biological Shield:** Absorbs radiation from the core and is made of thick concrete.
5. **Containment:** This steel structure is airtight and prevents the release of radioactive material in case of a major rupture or failure within the primary circuit or other major components like the steam generators, or main steam lines.
6. **Reactor Building:** The reinforced concrete protects the reactor of external forces.

Redundancy and Diversity: For important safety-related mechanisms, there are multiple systems installed which serve the same purpose (*redundancy*). These systems do not only work independently, but also achieve their desired task by different ways to minimize the danger of common-mode failures (*diversity*). [36]

Physical Separation: Besides the existence of multiple backup mechanisms, those systems are also separated physically. This ensures that e.g. in the case of a fire or flooding of an area, not all systems that serve a specific purpose are destroyed. [37]

Fail-Safe-Principle: The design of relevant safety components should ensure that in case of an emergency, the reactor transitions by itself into a safe state. E.g. in case of a power outage, the control rods are inserted into the core by gravitational force and thus the nuclear reaction ceases. [37]

Automation: In case of a safety relevant incident or accident, the reactor protection system is designed to counteract and initiate safety measures automatically without the need of an active intervention by working personnel for at least 30 minutes. This mitigates the risk of human errors under significant pressure. [37]

Self-Sufficiency: The functionality of vital safety systems (e.g. the residual heat removal of the core after shutdown) has to be ensured for an extended period of time, even in case of a loss of electrical power. This is achieved by backup power supply systems like a diesel generator. [37]

3. Safety Concepts of Generation II Reactors

In 1969, KWU was founded as a subsidiary company of Siemens and AEG [39]. This company would, inter alia, develop Generation II PWRs, including the NPPs Biblis and Grohnde. The KWU-PWRs would later represent an integral part of Germany's nuclear fleet. The NPP Biblis is located in Hesse, Germany and comprises two nearly identical reactors, Biblis-A and Biblis-B [40]. NPP Grohnde consists of a single pre-convoi reactor[1] and is located in Lower Saxony, Germany [41]. More details can be seen in Table 3.1. Because of the high power density of the RCS in PWRs, extensive safety measures have to be undertaken. The safety concept of Generation II NPPs can be separated into three isolated systems: **Passive safety systems** (see section 3.1), **active safety systems** (see section 3.2), and the **reactor protection system** (see section 3.3). Safety systems differ from operating systems by their definition and their main design goals. The definition is set by rule 3501 of the German nuclear commission "Kerntechnischer Ausschuss (KTA)". [31] According to Möller [31, pp. 72-73], the safety goals include ensuring the:

Reactor	Commercial Operation	Net MWe	Status
Biblis A	26 February 1975	1167	Licensed decommission
Biblis B	31 January 1977	1240	Licensed decommission
Grohnde	01 February 1985	1360	Shutdown

Table 3.1.: Key figures for NPPs Biblis and Grohnde [42]

[1] In this thesis, the term "KWU plant/reactor" will be used synonymously with the more specific term "KWU pre-convoi reactor". The simplified name must not be mistaken for the more advanced "KWU convoi reactor", which is also called "KWU-Baulinie 80".

- power reduction, respectively, the emergency shutdown of the reactor (also called SCRAM). This can be achieved by inserting all control rods into the core and the injection of boric acid into the RCS for long-term reactivity control.
- required minimum level of primary coolant and its thermal transfer via the ECCSs, the additional boric acid system and the RCPs.
- heat extraction by the secondary circuit via the steam dump system.
- water flow to the steam generators via the emergency feedwater system (EFWS).
- pressure limitation of the primary system.
- safe enclosure of the radioactive inventory and fission products by the passive safety system.
- integrity of the containment building.
- functionality of the electrical systems.

For Generation II NPPs, the main design basis accidents (DBAs) are large-break loss-of-coolant accidents (LBLOCAs) [43, p. 198], although the possibility of a main steam line break (MSLB) is also taken into account. As described by Smidt [44, pp. 36, 60–61], a rupture of the RPV and the pressurizer can be eliminated by the means of strict quality control during production and installation and by selecting adequate materials. The components are additionally monitored and regularly checked. Although the RPV itself is not redundant, the different approaches regarding the quality of production and monitoring can be considered as diverse and redundant. As a result of this safety net, also termed as “basic safety”, failures can be theoretically excluded [44]. The manufacturing and installation of a RPV that suffices these standards poses a very challenging task. The precise work is of utter importance, as a failure of the RPV could have catastrophic consequences for the safety goal of containing radioactivity. Additionally, the operational lifespan of a NPP is in general economically defined by the lifetime of the RPV [45].

3.1. Passive Safety Systems

The design principle and goal of passive safety systems is that the inherent protective function is available at all times, without the need of additional energy sources or activation by plant personnel [31]. The principle of multiple barriers has been discussed in section 2.3 and represents the underlying design of the related systems.

3.1.1. Containment

According to Laufs [46], the containment design of all KWU plants following the prototype PWR Obrigheim features a nearly circular, gas tight double containment, which encloses the primary circuit. The diameter of the containment is thus defined by the power rating of the reactor and the contents of the RCS. The arrangement of the used double layer containment can be seen in Figure B.1. It consists of a highly gas-tight inner containment, a ventilated annulus with a filter system, which is connected to a chimney and serves as a suction in case of a leakage, and an outer containment. The outermost layer is a cylindrical, gas-tight structure made of reinforced concrete. [46] It is 1.8 m thick and designed to withstand external forces, including floods, earthquakes, tornados, airplane crashes and explosions [47, p. 44]. The inner containment consists of 30 to 40 mm thick steel and has a diameter of 56 m for all KWU PWRs built after the NPP Biblis-A [31], [46]. As seen in Figure B.2, it houses the RCS, the steam generators, the spent fuel pool and the crane. In case of a loss-of-coolant accident (LOCA), separated parts could damage important components of the cooling circuit and are therefore enclosed by reinforced concrete [46]. The inner containment is designed to withstand a pressure increase of up to 5.7 bar, induced by a LBLOCA. It is assumed that the total content of the RCS leaks into the structure and that the increase of pressure is caused primarily by the decay heat of the core. [31] The safety philosophy of western countries regarding nuclear power, including Western Germany, was very different to the approach of the Soviet Union. With the beginning of the commercialization of nuclear power, the implementation of a containment was required for a licensing permit in countries like the USA and Western

Germany. Gen II NPPs originating from the Soviet Union, e.g. the RBMK and the early VVER-440 designs, did not feature a containment, nor was it formally required in the deployed countries. The Chernobyl disaster in 1986, inter alia, highlighted the importance of the lacking containment structure, as large quantities of radioactive material were released directly into the environment and spread over large parts of the former Soviet Union and Europe.

3.2. Active Safety Systems

In case of a transient condition, respectively an accident, the reactor protection system (RPS) initiates various active safety systems in order to minimize the resulting consequences and radiological danger [31]. With the second generation of NPPs, a variety of active systems were designed and implemented to prevent and mitigate DBAs, as well as other transient and accident conditions. These systems are effective when they are available, but because of their active nature, it is rather difficult to ensure the availability and functionality at all times over the course of the operational lifetime, especially when combined stress situations are considered. The residual risk, in German termed "Restrisiko", of a severe accident with a large release of activity has preoccupied engineers in the USA and Western Germany since the early days of the commercial use of nuclear power in the 1970s. In order to quantify the involved risks with a probabilistic risk assessment and analysis approach, the German risk studies stage A and B were composed. Based on the results of these studies, which were published in 1979 and 1990 respectively, various improvements of safety related systems were undertaken [46, pp. 674]. The discourse about the risk of nuclear power has not been limited to the scientist community, but was also widely discussed in the German public. After the Fukushima Daiichi accident in 2011, the discourse once more gained immense momentum and, as mentioned in section 1.3, partially contributed to the shutdown and phasing out of German nuclear power.

3.2.1. Containment Isolation

In case of an accident, the containment is designed to ensure that no radioactivity from the primary circuit is released beyond the containment [31], [37]. As described by Möller [31], in case of a LBLOCA it is assumed that the water inside the RPV evaporates and induces an increase of pressure inside the inner containment layer. In order to prevent radioactivity to propagate beyond this layer, all lines penetrating the inner containment and annulus have to be isolated via isolation valves. Lines that are necessary for emergency core cooling and residual heat removal, as well as lines with a small diameter, are exempt. The isolation of the containment is initiated by the RPS if the criteria for emergency core cooling are met. [31] Generally, the containment isolation can be divided into three areas, differing in the systems which are being isolated [31]:

- General containment isolation: This includes auxiliary systems that penetrate the inner containment.
- Isolation of the nuclear ventilation system.
- Isolation of the volume control system.

3.2.2. Reactor Scram

In case of an emergency, the reactor can be shut down rapidly by inserting all control rods into the core [31]. This emergency shutdown of the reactor is also termed "SCRAM" in the technical terminology. The control rods absorb neutrons, which are needed for the nuclear chain reaction, rendering the core subcritical [2]. As described in subsection 2.1.2, the control rods are grouped into multiple RCCs. These RCCs are inserted into the core from the top by their associated drive shaft and drive mechanism, which operates electromagnetically [35]. If an emergency shutdown of the reactor is initiated, the electrical circuit of the drive mechanism is interrupted. Thus, the control rods are not held in place anymore and fall into the core by gravitational force. The SCRAM is automatically actuated if certain

[2]The neutron multiplication factor k describes the chain reaction of a finite system and is characterized by the six-factor formula $k = k_\infty P_{FNL} P_{TNL} = \eta f p \epsilon P_{FNL} P_{TNL}$. For a subcritical reaction, this characteristic value is smaller than one ($k < 1$) [48, pp. 83-84].

parameters like the neutron flux density, thermal reactor power, fluid level in the steam generator and pressurizer deviate from their set boundary values. In order to maximize the reliability and availability of this system, the direct current signal feeding the control rod coils can be interrupted by three redundant and diverse systems. [31] Long-term reactivity control of the core is achieved by the injection of boric acid into the RCS [35]. KWU-PWRs feature a dedicated boric acid system, which is also designed to counteract the coolant loss in case of a leakage in the RCS, following an impact from an external source [31].

3.2.3. Emergency Power Supply

The emergency power supply of safety systems is ensured by two separate systems: The emergency power system and the emergency feedwater power system. Each arrangement is four times redundant and physically separated. For the emergency power system, this is achieved by four relays, each being coupled with their dedicated diesel generator. The supplied safety systems include the ECCS and residual heat removal system (RHRS), the EFWS, the additional boric acid system as well as different coolant pumps. [31]

3.2.4. Emergency Core Cooling System

After the reactor has been shutdown, the produced heat from the radioactive decay has to be removed in order to prevent the fuel and core from being damaged [29]. Right after shutdown, the heat is transferred to the secondary system via the steam generator. At a temperature of around 120 °C, the decay heat is not sufficient to generate steam. Therefore, the ECCS and RHRS take over the task of cooling the core as well as the fuel in the spent fuel pool. In case of a LOCA, the RPS activates the ECCS, which consists of different injection systems. [31] These separate systems comprise high to low pressure injection trains to counteract different pressure variations in the RCS following primary breaks. The pumps of these systems direct borated water from a dedicated refueling water storage tank into the primary circuit. For LBLOCAs and therefore a large pressure drop in the RCS, the cold leg accumulators are designed to supply large amounts of borated water from their tanks. This system works passively, without the need of electrical power, by a nitrogen gas bubble that is enclosed on top of the tank. Additionally, the RHRS, which acts as a low-pressure system, is actuated. This system also pumps water from the refueling water storage tank into the RCS. At some point, the two tanks are drained and the RHRS pumps the water from the containment sump through the RHRS heat exchanger, which extracts the remaining heat and directs it back to the core. [29] As described by Kok [35], the design goal of the RHRS is to reduce the temperature of the primary coolant to 60 °C within 20 hours of shutdown initiation. This system is also used when the plant is shutdown for refueling or service operations [35]. There are dedicated pool pumps for the spent fuel pool that circulate the water through the RHRS heat exchanger and back to the pool [29].

3.2.5. Emergency Feedwater System

Following an incident affecting the feedwater/steam circuit, e.g. when the main feedwater system does not supply the necessary feedwater to the steam generators, the RPS actuates the emergency feedwater system (EFWS) [31], which supplies water from the condensate storage tank [29]. This ensures that the heat from the RCS can be removed via the steam generators. The steam can then be directed to the main condenser by the steam dump system, or it can be released directly to the atmosphere. [29]

3.2.6. Hydrogen Monitoring and Control System

Following a LOCA, the supplied coolant from the emergency systems reacts with materials of the reactor core and the containment [31]. The ionizing radiation from the core, fuel in the spent fuel pool, as well as other radioactive materials that are released into the containment cause radiolysis of water, which according to Möller [31, p. 107] can be described by the following formula: $H_2O \longrightarrow H + OH$. It is furthermore explained that if the temperature of the fuel cladding, which is typically made of a zirconium alloy, reaches around 1200 °C, the greatest contribution of hydrogen production results from the zirconium-water-reaction: $Zr + H_2O \longrightarrow ZrO_2 + 2\,H_2$ [31, p. 107]. The formed hydrogen accumulates in the containment, and the danger of a detonation or deflagration arises if the H_2-concentration exceeds 4% [31]. As described in [31], [35], the probability of such an event also depends on the air moisture, as a higher concentration of steam in the containment atmosphere is unfavorable for the ignition of the explosive mixture. These exothermic reactions can damage the containment as well as other safety systems, and therefore pose a serious threat to the integrity of the present safety barriers. In order to detect and measure the H_2-concentration, the hydrogen detection system measures the concentration via sensors that are placed at multiple locations inside the containment. The hydrogen control system consists of a ventilation system, that circulates the air inside the containment and therefore prevents high local hydrogen concentrations, and hydrogen recombiners. [31] For this purpose, both passive autocatalytic and thermal recombiners are commonly used in order to reduce the H_2-concentration by controlled recombination with oxygen to water [31], [35]. As mentioned by Kok [35], for a present-generation PWR 64 catalytic heaters are placed inside the containment, although this number depends on the technical requirements and general layout of the plant. Furthermore, Kok [35] describes the critical H_2-concentration for an explosive mixture at 20% [35, p. 79], when dry air is present. It has to be mentioned, that this value deviates if the steam content in the containment atmosphere is higher.

3.3. Reactor Protection System

As described in [49], the reactor protection system (RPS) is an additional safety layer that is designed to constantly monitor important operational parameters and other safety-relevant information in order to initiate safety measures following accident or transient conditions. The RPS works self-sufficiently without the need of active intervention by an operator. If necessary, the RPS actuates the active safety systems which were discussed in this chapter [49].

4. Safety Concepts of Generation III Reactors

Generation III NPPs are a further development of Gen II reactors. The implemented evolutionary design improvements in the areas of safety and economics are based on the gained operational experience from the previous generation. This subsequent development includes, inter alia, fuel technology, thermal efficiency, safety systems that rely heavily on a passive operating principle and a standardized and simplified design. The majority of Gen III reactors is represented by Advanced Light Water Reactors (ALWRs). [8] The Advanced Power Reactor 1400 (APR-1400) is an ALWR that has been developed from 1992 until 2002 by the Korea Electric Power Corporation (KEPCO) [50]. The APR-1400 is based on the Korean Standard Nuclear Power Plant (KSNP)/OPR-1000 [50], which originated from the System 80+ by Combustion Engineering [51]. Major design requirements of the Gen III reactor include [52]:

- Capacity: 4000 megawatt-thermal (MWth) (rated thermal output)
- Plant lifetime: 60 years
- Availability: average of $\geq$ 90%
- Unplanned trip: $\leq$ 0.8/year
- Safety requirements:
  - Core damage frequency < 10^{-5}/RY
  - Containment failure frequency < 10^{-6}/RY
  - Occupational radiation exposure < 1 man-Sievert/RY
  - Thermal margin of 10% (up to 15%)
  - Station blackout coping time of a minimum of 8 hours

Country	Site	Unit	Gross MWe	Status
Korea	Shin Kori	3	1488	Operational
Korea	Shin Kori	4	1494	Operational
Korea	Shin Kori	5	1400	Under construction
Korea	Shin Kori	6	1400	Under construction
Korea	Shin Hanul	1	1400	Under construction
Korea	Shin Hanul	2	1400	Under construction
UAE	Barakah	1	1417	Operational
UAE	Barakah	2	1400	Operational
UAE	Barakah	3	1400	Under construction
UAE	Barakah	4	1400	Under construction

Table 4.1.: Summary of APR-1400 NPPs (operational or under construction) [27], [53]

4.1. Nuclear and Safety Systems

The primary system (see Figure C.1) comprises the RPV and two coolant loops: Each loop consists of one hot leg and two cold legs. There is one steam generator per loop, and there are two RCPs connected to the two cold legs at each side. [54] There is one pressurizer connected to one of the two hot legs via the surge line, as seen in Figure C.1. In order to reduce the frequency of reactor trips during transient conditions and to prolong the operator response time after a total loss of feedwater (LOFW) accident, the pressurizer and the steam generator have been enlarged, as described in [54]. There are pilot operated safety relief valves (POSRVs) installed at the pressurizer, which are according to Lee, Kim, and Suh [55] more reliable than common safety valves and can be controlled remotely following a transient or accident condition with a pressure increase. The core contains 241 fuel assemblies, each consisting of 16x16 fuel elements (236 fuel rods) with an average enrichment of 2.6% U-235. The refueling cycle amounts to 18 months and the thermal margin of the core has been increased to 15%. [54] According to [56], up to one-third of the core can be loaded with MOX-fuel. The safety design is selected to fulfill the regulatory and licensing criteria demanded by the countries part of the target market, and even exceed those requirements[1], in order to safeguard the public health and the investment into the plant. The latter is achieved by an increased design margin, which e.g. allows the plant to stay operational following a small-break LOCA, as well as various safety systems designed to prevent and mitigate accidents and their consequences. [54]

4.1.1. Safety Injection System

The safety injection system (SIS) (see Figure C.2) is designed to inject water directly into the core in order to remove decay heat following a LOCA. The SIS consists of four mechanical trains and two electrical divisions. For each train, there is one safety injection pump (SIP) and one safety injection tank (SIT) with an integrated passive fluidic device (FD). Every train works independently and the four dedicated safety injection lines are directly connected to the direct vessel injection nozzles, through which borated water taken from the in-containment refuelling water storage tank (IRWST) is delivered into the core. [54] The use of sufficiently borated water is of high significance, as the insertion of moderating water could lead to a rapid reactivity insertion accident with potentially catastrophic consequences [57, p. 82]. The FD works based on vortex flow resistance and regulates the flow rate of the borated water passively, in order to optimize the water injection rate following a LOCA. This eliminates the need for a low-pressure SIP. The different operation modes used in older NPPs can be replaced by one operation mode (safety injection mode). [54] The residual decay heat removal after a reactor shutdown is provided by a different, dedicated system called the shutdown cooling system (SCS), along with the EFWS. The SCS is mainly used to maintain the temperature of the RCS for refueling operations, and it can also be used to cool down the RCS in case of a small-break loss-of-coolant accident (SBLOCA). [32] The simplified design of the SIS and the separation of the initial emergency cooling by the SIS and the residual heat removal by the independent SCS increases the reliability

[1] Generally, the licensing requirements differ from country to country, but are relatively similar in countries with high safety standards. The International Atomic Energy Agency (IAEA) provides with the publication of the “Safety Standard Series”, inter alia, guidance for national regulatory institutions. These guidelines are especially helpful for countries, which lack extensive experience in the field of nuclear energy/power.

and availability of the related safety systems compared to the arrangement used in Gen II KWU plants, where the RHRS is part of the low-pressure injection system of the ECCS (see subsection 3.2.4). It also reduces the required maintenance effort, as the complexity of the system is significantly reduced.

4.1.2. In-Containment Refueling Water Storage Tank

In order to minimize the probability of the refueling water storage tank being damaged by external forces, it is located inside the containment and is a toroidal cylindrical tank arranged along the containment wall with a volume of 2470 m^3 (see Figure C.2) [56]. The IRWST serves as the normal refueling water storage and also supplies borated water to the SIS and other safety systems like the containment spray system (CSS), safety depressurization and vent system (SDVS), cavity flooding system (CFS) and the external reactor vessel cooling system (ERVCS). The injected water from the SIS can be redirected to the IRWST through four sumps. [54], [56]

4.1.3. Emergency Feedwater System

The EFWS takes on the same task compared to the reference system in the Gen II plant (see subsection 3.2.5): It ensures that there is enough feedwater supplied to the steam generators for the purpose of removing the decay heat from the RCS. When the design is compared to previous generation systems, it is noticeable that there has been a further emphasis on increased reliability. As described in [56], the enhanced reliability is achieved by using a two-divison system, one for each steam generator. A division consists of two trains, of which one uses a 100% capacity motor-driven pump, and the other a 100% capacity turbine-driven pump. The feedwater is supplied from the condensate storage tank, although each division has a dedicated emergency feedwater storage tank to increase redundancy and availability of the system. [56]

4.1.4. Containment Spray System

The CSS is designed to reduce the temperature and pressure inside the containment after a LOCA or MSLB by spraying borated water from the top of the containment. Two independent trains deliver the water taken from the IRWST through dedicated containment spray pumps. These pumps can be used as backup pumps for the SCS in order to increase redundancy and reliability. [50]

4.2. Systems designed for Severe Accidents

Safety systems and measures that are designed for severe accidents (which exceed DBAs) can be divided into the following two categories [50], [56]:

Severe accident prevention: This includes the increased design margin, e.g. a higher thermal margin, as well as an increased capacity of the pressurizer and the steam generator. It also comprises the integration of reliable engineered safety features (ESFs), e.g. the SIS, IRWST, SCS, CSS and SDVS. [50], [54] The SDVS provides a countermeasure for high-pressure core melt scenarios by depressurizing the RCS [50]. This scenario was not considered for the Gen II KWU plants, but gained more attention after the Three Misle Island Unit 2 (TMI-2) accident. In case of a high-pressure core melt, the RPV could fail and lead to direct containment heating (DCH).

Severe accident mitigation: These systems are designed to limit the possible consequences and the damage caused by a major accident [50], and in particular to prevent early containment failure [56]. They, inter alia, include the robust containment, the hydrogen mitigation system, the CFS and the ERVCS [50], [54]. As mentioned in both [50], [56], for the event of a late containment failure, an evaluation of the probabilistic safety goal combined with the cost of integrating the associated mitigation systems indicates if the realization of such systems is economically justifiable and expedient. Compared to Generation II NPPs, the deployment of the CFS and the ERVCS increases the probability that the containment does not fail in case of a severe accident. Securing the integrity of the RPV and consequently the containment presents an integral aspect in safeguarding the principle of barriers and thus limiting the possibility of a radioactive release beyond these boundaries.

4.2.1. Plant Layout and Containment

As described in [54], the plant layout of the APR-1400 is characterized by a twin-unit arrangement (see Figure C.3). The containment is enclosed by the auxiliary building, which houses the safety systems and is divided into four quadrants. In each of these sections there is one SIP installed. This minimizes the susceptibility of the relevant systems to external forces like a plane crash and other destructive events, e.g. flooding or fire. The emergency diesel generators are also physically separated. [54] The containment and auxiliary building are built on a common basemat in order to strengthen the buildings in case of an earthquake (safe shutdown earthquake of 0.3 g) [56]. The containment structure can be divided into an outer cylindrical containment made of steel-lined, post-tensioned concrete with a diameter of 45.72 m and an inner containment made of reinforced concrete, which surrounds the RCS and additionally functions as a biological shield. The inner and outer containment structures are designed to withstand external threats and forces, e.g. a plane crash, missile attack, as well as internal pressure caused by e.g. a LOCA or a MSLB. [50], [54] When comparing the accident scenarios and conditions taken into account with the APR-1400, described in [32, pp. 37-40], it can be recognized that many scenarios of DBAs and partially even beyond-DBAs were essentially already taken into account with the Gen II KWU reactors. This can be explained by the fact that these scenarios have to be considered in order to meet the national licensing requirements, which were relatively rigorous in Western Germany since the first days of nuclear power. With the help of safety analyses of the APR-1400 under different transient and accident conditions, the systems designed to deal with the prevention and mitigation of catastrophic events (beyond-DBAs) have been upgraded accordingly. These installations include the containment building, the hydrogen mitigation system, the CFS and the ERVCS. [32] The systems designed to deal with DBAs have also been further developed with an emphasis on passive working principles and simplicity, in order to increase availability and reliability, as well as to reduce the cost of the deployed ESFs and safety systems.

4.2.2. Reactor Cavity Design and CFS

In order to mitigate the consequences of a severe accident involving a core meltdown leading to the formation of corium [50] and consequently to the failure of the RPV [58], the reactor cavity is specifically designed to contain the corium in the core debris chamber and to prevent DCH through the convoluted cavity design and the cavity flooding system (CFS) [56]. The CFS works passively and comprises two trains which connect the IRWST to the reactor cavity via the hold-up volume tank (HVT) (see Figure C.4). In case of an accident, two isolation valves placed at each line are opened and the water flows by gravity from the IRWST to the cavity. [56] As described in [50], the resulting steam from the cooling of the core debris is then condensed by the CSS. The CFS prevents interaction between the molten corium and concrete. It furthermore averts the formation of combustible gases and reduces fission product releases. [50], [56] The CFS is intended to be used as a backup for the ERVCS in case the in-vessel retention of the corium has not been successful and the RPV has failed [56].

4.2.3. External Reactor Vessel Cooling System

The ERVCS is an active system ensuring the in-vessel retention of the corium in case of a severe accident involving a core meltdown. The system consists of two trains (see Figure C.4): The first train utilizes the shutdown cooling pump in order to fill the reactor cavity up to the level of a hot leg. The second one consists of a boric acid makeup pump supplying enough water to keep the cavity flooded, as the decay heat evaporates the initial coolant. [50], [56] According to [50], probabilistic safety assessments of PWRs strongly indicate that the retention of the corium inside the RPV and therefore ensuring the integrity of the RPV presents a critical role in safeguarding the intactness of the containment. Further evaluations cited in [50] suggest that the best accident management strategy would consist of utilizing the ERVCS along with the SIS.

5. Safety Concepts of Generation III+ Reactors

Although the distinction between Gen II, Gen III and III+ reactors is arbitrary [59], they can be differentiated and categorized by their design goals and features. The development of Gen III+ plants started in the 1990s. Their design is very similar to Gen III reactors, with further emphasis on safety systems that rely on passive operating principles like natural convection cooling and gravity driven injection of water rather than solely on active safety elements. [8] The advanced designs also incorporate stronger protection against external forces, e.g. aircraft impacts, by further reinforcing the outer containment [47]. In order to deal with severe accidents (beyond-DBAs), these reactors feature unique safety systems and designs which were developed based on the experience accumulated through the decade-long operation of LWRs combined with extensive safety analyses. The aim is also to further minimize any grave radiological consequences of severe accidents. Two systems of this generation will be discussed in this chapter more thoroughly. Different Generation III+ reactors generally vary in their technical design, but the safety systems are based on similar passive working principles in order to ensure the highest achievable availability and reliability.

5.1. EPR

The EPR was designed by AREVA NP and is based on the German Konvoi and French N4 reactor designs. The EPR has a thermal power output of 4500 MWth and gross power of 1750 MWe, which translates to a net output of around 1630 MWe. [59] The system features a four-loop design with four steam generators and four RCPs [33], as well as a core that can be fully loaded with MOX-fuel [59]. The safety systems of the EPR aim to eliminate consequences for the environment following a severe accident by preventing early and late containment failures with different design approaches [60], of which the depressurization device (see subsection 5.1.1), the core catcher (see subsection 5.1.2) and the double containment (see subsection 5.1.3) will be discussed. The deployment of the incorporated safety systems results in a featured core melt frequency of under 10^{-5}/RY, as well as in a frequency concerning the early primary containment failure or the release of large amounts of activity of under 10^{-6}/RY [33]. Currently, there are several EPR projects finished or under construction: Taishan 1 & 2, located in China, are operational since 2018 and 2019 respectively [61]. In Finland, Olkiluoto-3 reached criticality at the end of 2021 and is expected to be connected to the grid in 2022 [62], while one reactor in France, Flamanville-3, is still under construction [63]. The high fuel burn-up of 65 GWd/t, high thermal and net efficiency of 37% and 36% respectively, combined with an average availability of 92% over the projected operational lifetime of 60 years should improve fuel and economic efficiency [59], although the economic viability of the EPR remains in question as construction projects, especially in Europe, were associated with substantial construction time and cost overruns [63].

The capacity of main components of the RCS has been increased compared to previous-gen four-loop reactors [33]. This can be partially attributed to the high thermal output of 4500 MWth. As described in [33], the design margin of main components of the RCS has been also increased in order to reduce the number of trips caused by normal operating transients, to extend the reactor operator response time and to minimize the general stress on the components. Most safety systems are constructed in a four-train layout [33], further increasing redundancy and reliability. The fourfold layout can also be ascribed to the increased size and complexity of the RCS. The most important safety systems can be seen in Figure 5.1: The IRWST is similar in design and function as the discussed reservoir of the APR-1400. The core melt spreading area is part of the "EPR core catcher", discussed in subsection 5.1.2. The double-wall containment with a dedicated ventilation system will be described in subsection 5.1.3.

Following the TMI-2 accident in 1979, in which insufficient cooling of the core led to a partial meltdown, more awareness of this scenario and the possible consequences arose: Following a partial or total meltdown of the

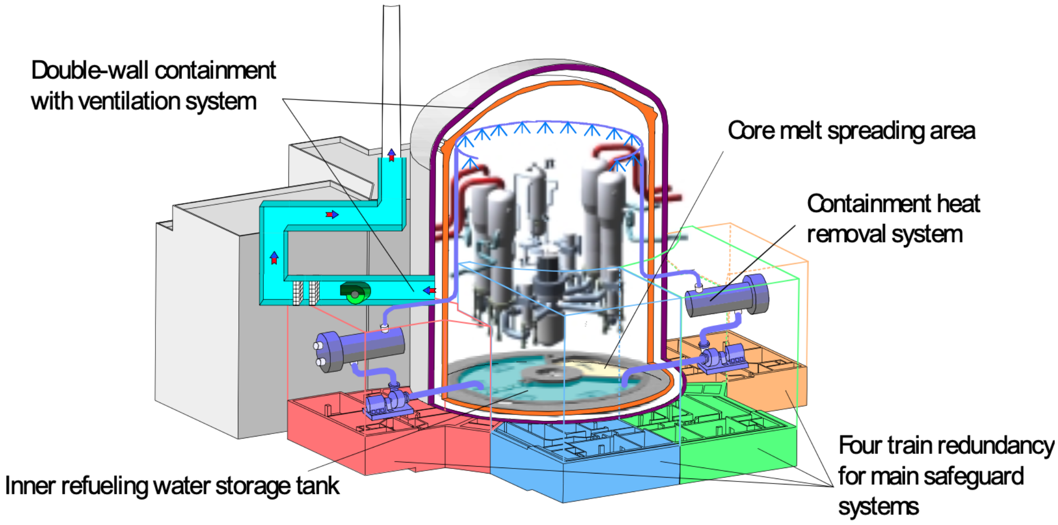

Figure 5.1.: Main safety systems of the EPR [64, Fig. 1]

reactor core, the formed corium could lead to a high-pressure core melt resulting in DCH, or inter alia, to a low-pressure core melt penetrating the containment basemat. [57] These new findings demanded the design and implementation of safety systems counteracting such severe accident scenarios, as the RPV and consequently the containment could fail. Because of the high power rating of the EPR, a different design approach had to be selected than for the APR-1400.

5.1.1. Depressurization System

The depressurization system (see Figure D.1) is a safety system designed to prevent the severe accident scenarios of a high-pressure melt-through of the RPV ($p > 15$-20 bar) or a steam generator tube rupture (SGTR), which could lead to DCH or a containment bypass. It reduces the pressure of the RCS in order to ensure that the structural integrity of the relevant parts is not endangered. It comprises three pressure relief valves (named tandem safety valves in Figure D.1), which are part of the pressurizer and provide a total discharge rate of 900 tons per hour, as well as two separate valves, which form the depressurization device seen on the right side of Figure D.1. This device can operate in a feed-and-bleed mode for the case of total LOFW, as well as in an emergency blowdown mode to eliminate a high-pressure core melt scenario. All valves are connected to the pressurizer relief tank via the letdown line. The system is also designed to prevent the distribution of corium particles in the containment atmosphere, which could lead to DCH. [57]

5.1.2. Core Catcher

In order to prevent basemat penetration by the corium in a low-pressure core melt accident scenario, a different main design strategy was selected than for the APR-1400: Because of the high thermal rating of the reactor, the in-vessel retention of the corium by external vessel cooling, which is used in the APR-1400 (see subsection 4.2.3), is expected to be very unlikely. Therefore, a "core catcher" has been implemented, which is designed to retain, cool and consequently stabilize the molten core debris outside the then-failed RPV. [60] This core catcher can be compared regarding its function and design to the reactor cavity design of the APR-1400, which also features a core debris chamber and a CFS. In contrast to the EPR core catcher, the used system in the APR-1400 serves only as a backup system for the ERVCS, for the unlikely case that the RPV fails. The EPR core catcher consists of a melt plug, located at the lower head of the RPV, and a spreading compartment, which is connected to the melt plug via a discharge channel (see Figure D.2). Both the reactor pit and the spreading compartment feature a top layer of sacrificial concrete, which temporarily holds back the corium for the purpose of accumulating all the core debris and furthermore making its behavior more predictable [60]. The reactor pit and the melt discharge channel are coated with a protective layer. The separation of the spreading compartment from the reactor pit combined with the used surface materials ensures adequate heat transfer and protection from both thermal loads and mechanical stress by the failed RPV. The corium is cooled in the spreading compartment by water, which is supplied from the IRWST. It first flows through cooling channels integrated under the thick steel plate, and afterwards into the spreading compartment. This process takes place passively by gravitational force until an equilibrium is reached and the supply of cooling water and residual heat removal is assumed by the containment heat removal system (CHRS) (see Figure 5.1). The cooling water is subsequently evaporated by the decay heat and afterwards condensed by sprays. The CHRS works actively and ensures that the pressure inside the containment is limited, as well as that the integrity of the structure is maintained in the long-term. [57], [60]

5.1.3. Double Containment Design

The containment of the EPR features a double-wall concrete building, consisting of an inner pre-stressed concrete structure with a metal sealing liner, and an outer reinforced concrete wall. In order to prevent uncollected leakage via peripheral buildings or the annulus, a dynamic confinement, consisting of the annulus ventilation system (AVS), combined with a static confinement is installed. The latter is established through the leak-tightness of the inner concrete wall and the isolation of penetrating lines. [57] If nevertheless a leakage past the inner containment would occur, the subatmospheric pressure inside the annulus combined with the extraction and filtration by the AVS ensures that any radioactive emission can be safely managed and subsequently discharged via the chimney (see Figure 5.1) [64]. In order to ensure that even in the considered event of a low-pressure core melt and failure of the RPV no direct leakage into the environment takes place, the leakage of the inner containment has to be smaller than 0.3% per day of the volume that it encloses. For the outer containment, this value has to be smaller than 1.5% per day. [57] For the purpose of preventing hydrogen deflagration, there are passive autocatalytic recombiners installed [33]. When the containment is compared to the structure used in KWU-plants (see subsection 3.1.1), the similarities regarding the used double-wall layout combined with a ventilated annulus and filter system can be accentuated. The EPR furthermore improves the containment design by regarding both LOCAs and core melt accidents as a design basis for the concerned structure [57, p. 308], which was not considered with KWU-plants. Therefore, the AVS is required to ensure that there is a negative pressure in the annulus under all regarded accident conditions, including core melt scenarios [57].

Country	Site & Unit	Status	Grid Connection
China	Sanmen 1	Operational	2018-06-30
China	Sanmen 2	Operational	2018-08-24
China	Haiyang 1	Operational	2018-08-17
China	Haiyang 2	Operational	2018-10-13
USA	Vogtle 3	Under construction	-
USA	Vogtle 4	Under construction	-
USA	Summer 2	Abandoned	-
USA	Summer 3	Abandoned	-

Table 5.1.: Summary of AP1000 NPPs [65], [66]

5.2. AP1000

The Advanced Passive 1000 (AP1000) is a two-loop PWR developed by Westinghouse based on the smaller AP600 and the older System 80+ design [59], [67]. Final US design certification was awarded by the U.S. Nuclear Regulatory Commission (NRC) in 2005 [59]. The AP1000 has a thermal capacity of 3400 MWth and a gross power output of 1250 MWe, which equates to a net output of around 1100 MWe [34]. As of April 2022, there are four operational units in China (see Table 5.1). In the USA there are two units being constructed at the Vogtle Electric Generating Plant. Originally, the construction of two units at the Virgil C. Summer Nuclear Generating Station began in 2013, but the project was abandoned after Westinghouse filed for bankruptcy protection in 2017 due to losses from nuclear power construction projects, which amounted to $9 billion USD. The two projects in the USA were subject to controversial discussions because of their large time and cost overruns. [67] The construction of Vogtle units 3 and 4 started in 2013, and the first grid connection of both reactors is expected not before 2023 [26].

5.2.1. General Safety Design

The AP1000 incorporates extensive passive design elements and safety systems, including the passive containment cooling system (PCCS) and the passive core cooling system (PXS), which comprises the passive residual heat removal system (PRHRS) [43]. These systems are designed to remove the decay heat from the RCS and ensure that the integrity of the containment is maintained following a postulated DBA. The safety-related systems and components achieve this based on their passive working principle, without the need for electrical power or additional components with moving parts, like coolant pumps. Furthermore, no intervention by plant personnel is needed for up to 72 hours. [34] For the severe accident scenario of a core meltdown, a similar design strategy to the APR-1400 has been selected: The in-vessel retention of molten core debris and moreover the integrity of the containment is ensured by flooding the reactor cavity with borated water taken from the IRWST, combined with the depressurization of the RCS [34]. In contrast to the APR-1400, the AP1000 does not feature an additional ex-vessel core retention and cooling system.

5.2.2. Passive Containment Cooling System

The PCCS integrates an innovative design in order to reduce the containment pressure and remove the heat during various accident scenarios. This is done by harnessing natural forces and thus eliminating the need for any major active components. The accident scenarios include DBAs like LOCAs and MSLBs [43], as well as severe accident scenarios like a core meltdown [34]. As described in [47], the containment consists of an inner steel vessel with a thickness of 44 mm and an outer reinforced concrete structure. The steel containment vessel serves as a heat sink, which transfers the heat from the RCS and therefore the inner containment to the outside by natural convection. The air leaves the space between the inner and outer containment structure via a discharge channel, located at the center top (see Figure 5.2). In order to increase the circulation and decrease the temperature of the air, a gravity drain water tank sprays water driven by gravity on top of the steel containment vessel. It is positioned at the top of the outer

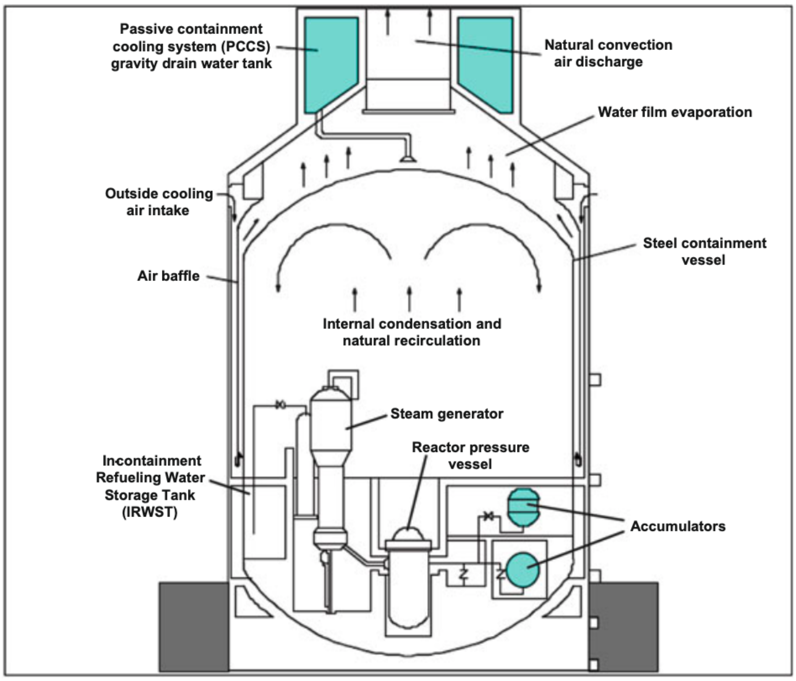

Figure 5.2.: PCCS of the AP1000 [47, Fig. 3.12]

containment and encloses the discharge channel with its toroid shape (see Figure 5.2). [34] Inside this inner steel vessel, the hot air condenses at the steel surface because of the resulting temperature difference. The formed water is then redirected back into the IRWST. This process provides natural recirculation and convection, and thus the heat transfer is also possible without any active components. [47]

5.2.3. Passive Core Cooling System

The PXS (see Figure 5.3) is designed to provide emergency core cooling and residual heat removal during different accident scenarios, including a double-ended rupture of a main coolant line, or a loss of secondary side heat removal. This is achieved by integrating a PRHRS, as well as a safety injection and depressurization system, which uses accumulators and core makeup tanks in order to passively inject borated water provided by the IRWST directly into the RPV. The automatic depressurization system reduces the pressure of the RCS to 1.8 bar (0.18 MPa) by venting into the IRWST. This pressure reduction has to be achieved so that borated water can be supplied to the core for long-term cooling, which is driven by gravity and taken from the IRWST. The IRWST is located above the level of the hot legs. [34] In case of a LOFW, a SGTR or a MSLB accident, the passive residual heat removal heat exchanger (PRHR HX) takes over the residual heat removal instead of the secondary side system [43]. The hot and cold leg of one RCS loop are connected to the PRHR HX, which utilizes the IRWST as a heat sink. After one hour, the water inside the IRWST starts to boil due to the absorbed decay heat and the steam is subsequently released into the inner containment. It then condenses at the surface of the inner steel containment and flows back into the IRWST by gravity. [34]

Because of the utilization and integration of the outlined passive safety systems, the AP1000 provides advantages in the fields of accident management and mitigation in regard to the operator response time and dependence on active elements (and thus reliability and availability of safety systems) when compared to other Generation III/III+ PWRs with similar power ratings, such as the APR-1400: In case of a DBA, the necessary coolant and boration for emergency core cooling and decay heat removal is provided by the plant inventory for up to 30 days. For long-term accident mitigation, no further adjustment and action by plant staff is needed for three days. [34] When compared to the APR-1400 and furthermore to other energy sources like renewables, the economic competitiveness of the AP1000 on the open market remains in question, especially as the construction time and cost of the units being built in the USA keep rising. The general increase of these two factors during the construction of ALWRs can be partially explained by the increased complexity of Gen III and especially Gen III+ systems. Because of the rigorous licensing criteria concerning new reactors in countries like France and the USA, the demanded implementation of advanced safety designs has increased the intricacy of NPPs substantially. Another aspect which contributes to the construction problems is that countries, including the USA and France, have not built and commissioned new reactors since decades. These problems could therefore be attributed to the loss of practical knowledge and to the lack of experience building complex structures like nuclear power reactors.

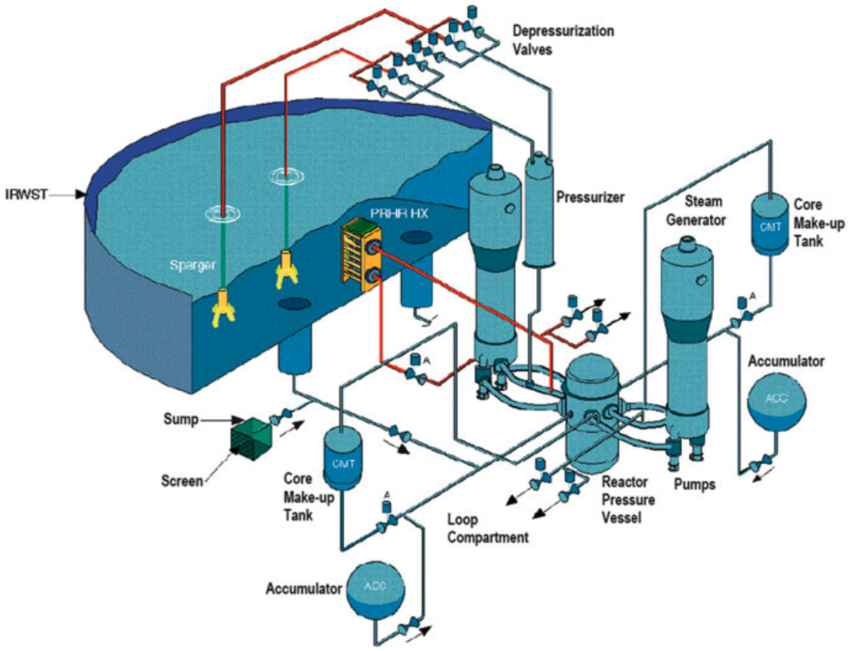

Figure 5.3.: Passive core cooling system of the AP1000 [47, Fig. 3.13]

6. Conclusion

With the Generation II KWU plants, the "Defense in Depth" approach to nuclear safety has been incorporated successfully: Both passive and active safety systems ensure that in case of the regarded DBAs, mainly LOCAs and LOFW accidents, major radiological consequences can be prevented or mitigated. The use of a double-layer containment with a ventilated annulus combined with the containment isolation and hydrogen control offers advantages in containing the release of radioactive material in case of a severe accident compared to Gen II NPPs, which do not feature these safety structures and systems. The active safety systems, including the ECCS, the EFWS as well as the containment isolation, present a balanced net of safety in order to counteract DBAs, especially LBLOCAs and LOFW accidents. The emergency power supply of these systems is designed by taking into account both redundancy and physical separation. A weakness of the regarded active safety systems is that because of their active working principle and their relative high complexity, they can be vulnerable in combined stress situations. With the Generation III APR-1400, design improvements were made in the areas of safety and economics. The latter is achieved with a reduced complexity of major systems and with an increased design margin. The design and operation modes of the ECCS have been simplified and the function of the ECCS is replaced by the SIS, while the residual heat removal is performed by a separate system, the SCS. These changes increase the reliability and availability of the systems compared to the Generation II KWU plants. The IRWST is located inside the containment for better protection against external forces and provides borated water to the safety systems. Additionally, a CSS is installed inside the containment for the purpose of reducing the containment pressure and temperature after an accident. For the purpose of severe accident mitigation, the strategy of in-vessel retention of the molten corium has been selected to minimize the possibility of a containment failure and major radioactive release. The

ERVCS aims to cool the molten core by flooding the reactor cavity, which should subsequently prevent the failure of the RPV. The CFS is designed as a backup for the ERVCS, in case the in-vessel retention has been unsuccessful. The molten corium is retained by the convoluted cavity design and cooled by the CFS. Compared to the Generation II KWU plants, major advancements were made in the mitigation of core melt accidents: The SDVS is able to reduce the pressure of the RCS to prevent a high-pressure core melt, while the ERVCS aims to cool and retain the molten corium inside the RPV. For the possibility of a RPV-failure, additional provisions were made with the robust cavity design and the CFS. The Gen III+ EPR features several design approaches in order to prevent early and late containment failures and thus any grave radiological release following a severe accident. The EPR is equipped with a depressurization system, which reduces the pressure of the RCS to prevent a high-pressure core melt or a SGTR, which could lead to a containment bypass or DCH. Because of its high thermal output of 4500 MWth, a different severe accident strategy has been selected for the EPR than for the APR-1400: For the possibility of a low-pressure core melt, the in-vessel retention by external cooling is expected to be very unlikely. Therefore, the EPR features a core catcher for the purpose of ex-vessel cooling and retention of the corium. It can be compared to the CFS in the APR-1400. The corium is stabilized in the EPR core catcher by a sophisticated geometrical layout and passive cooling of the spreading compartment by water supplied from the IRWST. The initial cooling is driven by gravitational force, and is subsequently assumed by the CHRS. The supplied water then evaporates and is condensed by sprays. The CHRS ensures the pressure-limitation and structural integrity of the containment in the long-term. The double-wall containment with the AVS of the EPR is similar in its layout to the containment used in Gen II KWU plants. The containment design of the EPR has been upgraded in order to ensure that no significant uncollected leakage past both containment layers occurs for both LOCAs and core melt accidents. The AP1000 follows a more drastic and innovative design route, in order to utilize natural forces like gravity and natural convection for the most important safety systems and thus minimize the need for major active components and action by plant personnel. The containment design incorporates the PCCS, which enables the removal of the decay heat from the RCS by natural convection and gravity following DBAs as well as core melt accidents. The PXS consists of the PRHRS and the safety injection and depressurization system in order to provide emergency core cooling and residual heat removal in various accident scenarios. The injection of borated water is driven by gravity. The PRHR HX utilizes the IRWST as a heat sink in order to fulfill the necessary cooling in case the secondary side heat removal is not available. For the severe accident scenario of a core melt, the strategy of in-vessel retention through external reactor vessel cooling has been selected. The utilized system is similar to the ERVCS used in the APR-1400. The AP1000 does not feature an ex-vessel retention and cooling system as a backup. The deployment of the discussed passive safety systems of the AP1000 provides advantages regarding the self-sufficiency of these arrangements. Furthermore, these advantages result in an enhancement regarding the prevention and mitigation of both DBAs and severe accidents. There is a clear need for a higher share of low-carbon energy sources in the global energy mix. With the rising construction costs and construction time of Generation III and especially Generation III+ reactors, nuclear power has to prove its economic competitiveness on the open market. The construction projects of the EPR and AP1000 in Europe and the USA have deepened these concerns, as they have been over-budget and over-due. At the same time, the successful commission of APR-1400s in the UAE and Korea, as well as EPRs and AP1000s in China have demonstrated that Gen III/III+ nuclear power projects can be practically realizable on a tighter budget and timeframe and thus be economically feasible. This therefore indicates that nuclear power would be a viable source of low-carbon energy for the future.

Appendix

Appendix A.

Pressurized Water Reactor Systems and Components

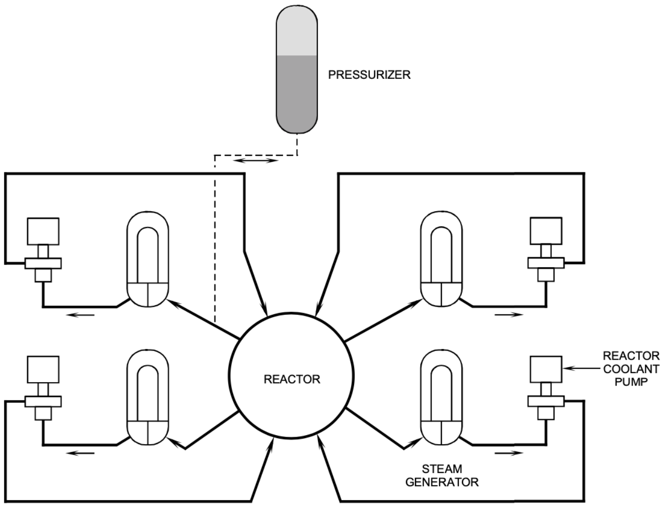

Figure A.1.: Design-Layout of the reactor coolant system [29, p. 3]

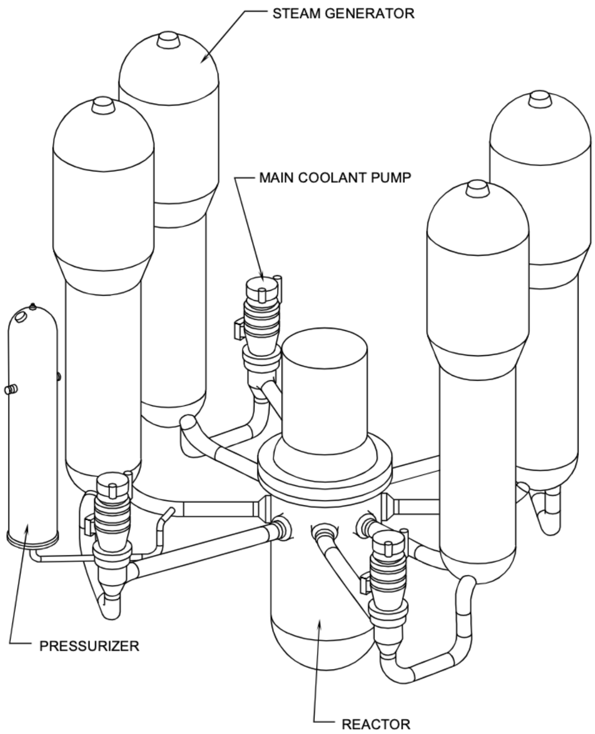

Figure A.2.: Reactor coolant system of a four-loop Westinghouse plant [29, p. 6]

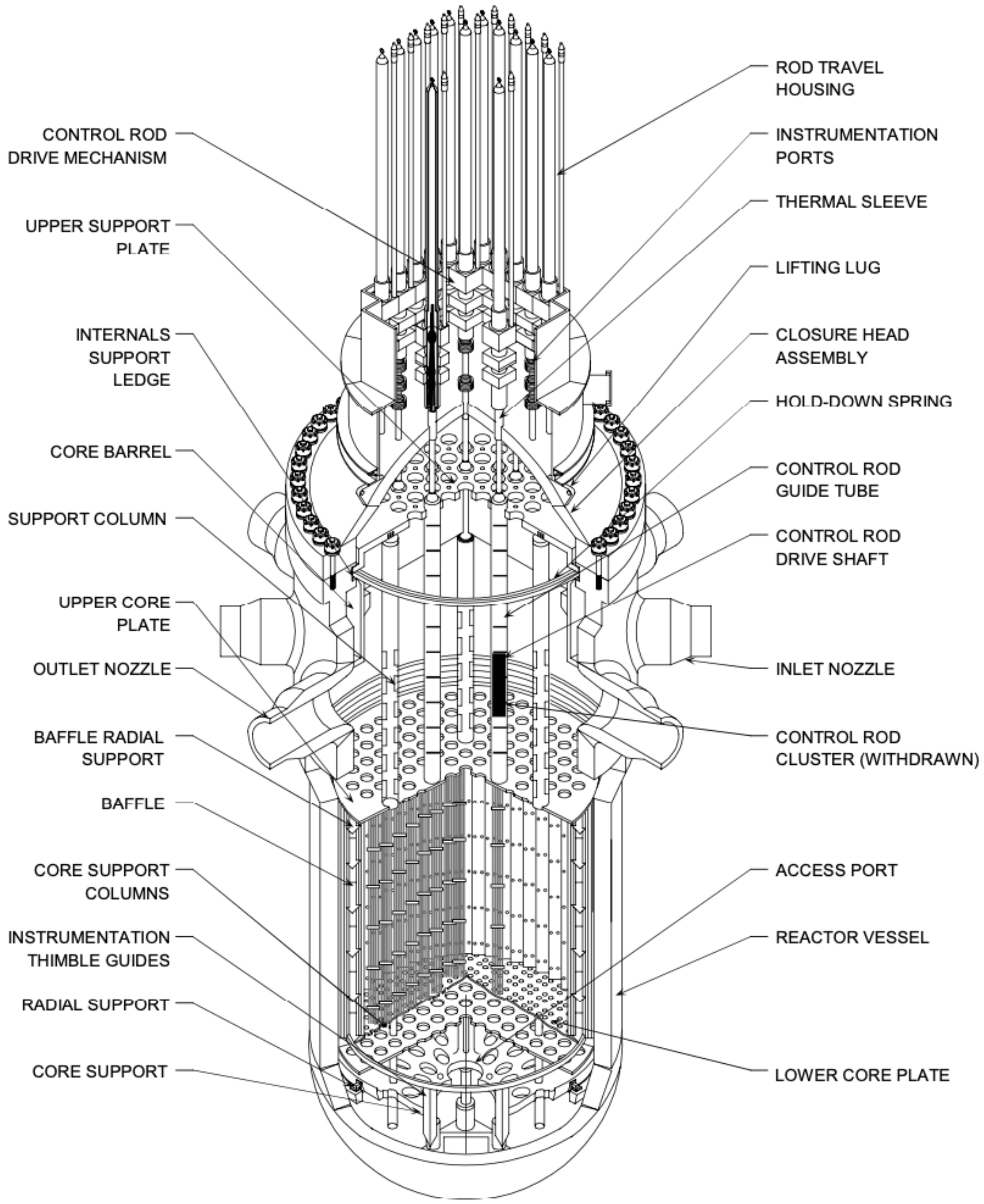

Figure A.3.: Cutaway view of the reactor pressure vessel [29, p. 10]

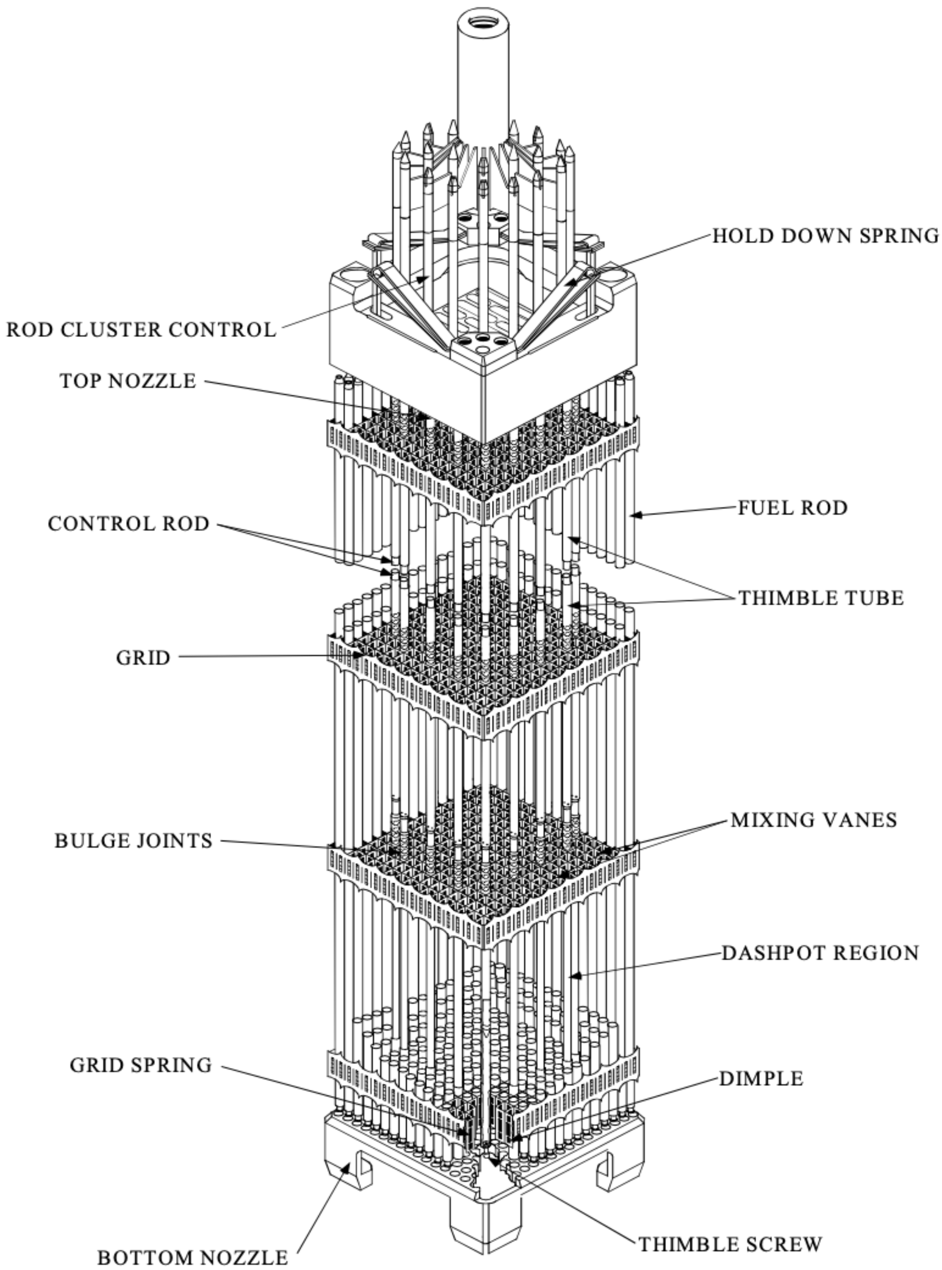

Figure A.4.: Major components of a PWR fuel assembly [68, p. 11]

Appendix B.

Safety Concepts of Generation II Reactors

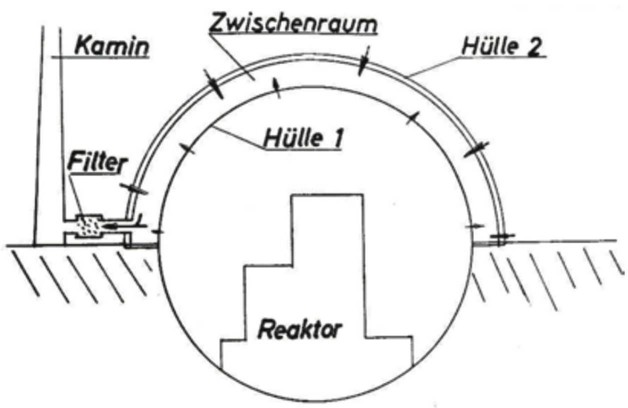

Figure B.1.: Design principle of the double containment used in German KWU plants [46, Fig. 6.10]

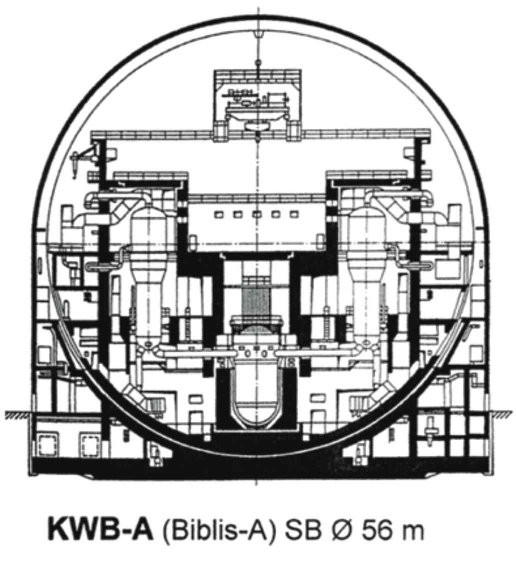

Figure B.2.: Containment building of the NPP Biblis-A [46, Fig. 6.87]

Appendix C.

Safety Concepts of Generation III Reactors

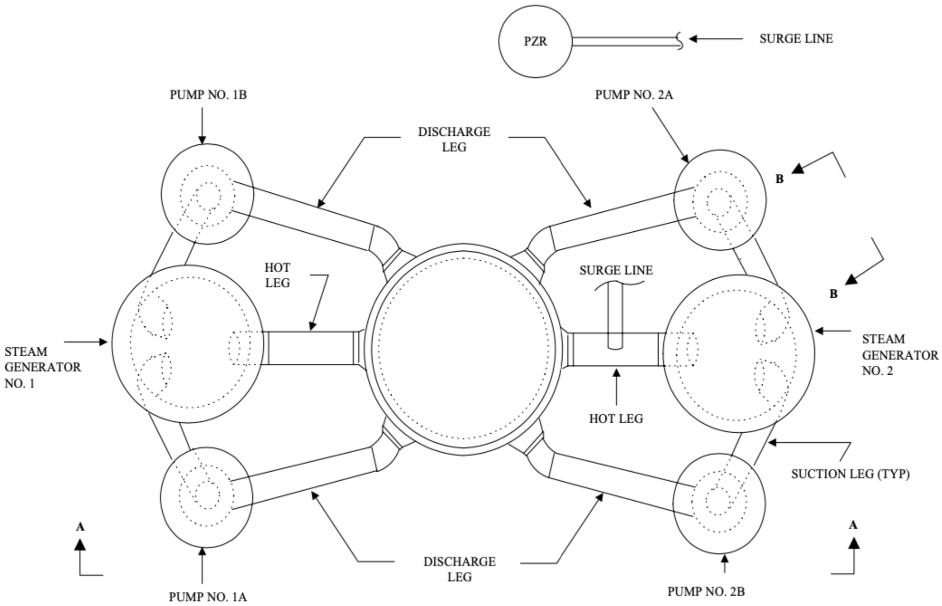

Figure C.1.: Design-Layout of the RCS of the APR-1400 [54, Fig. 1]

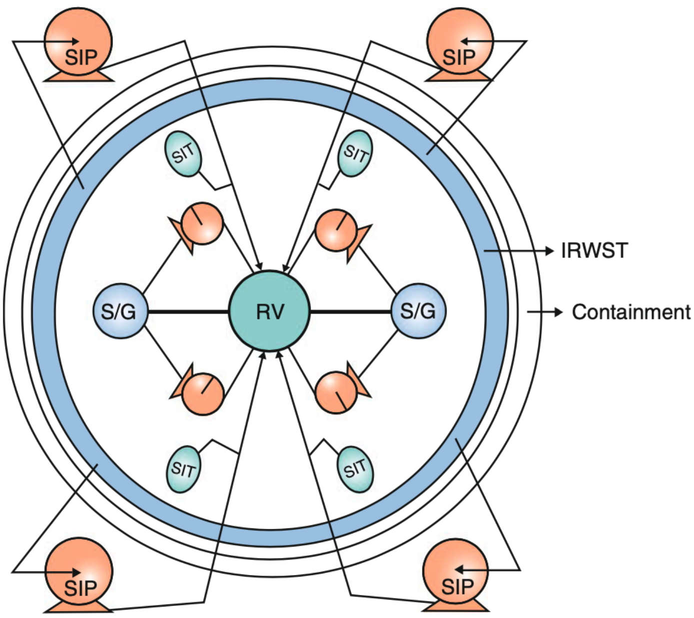

Figure C.2.: Safety injection system of the APR-1400 [69, p. 2221]

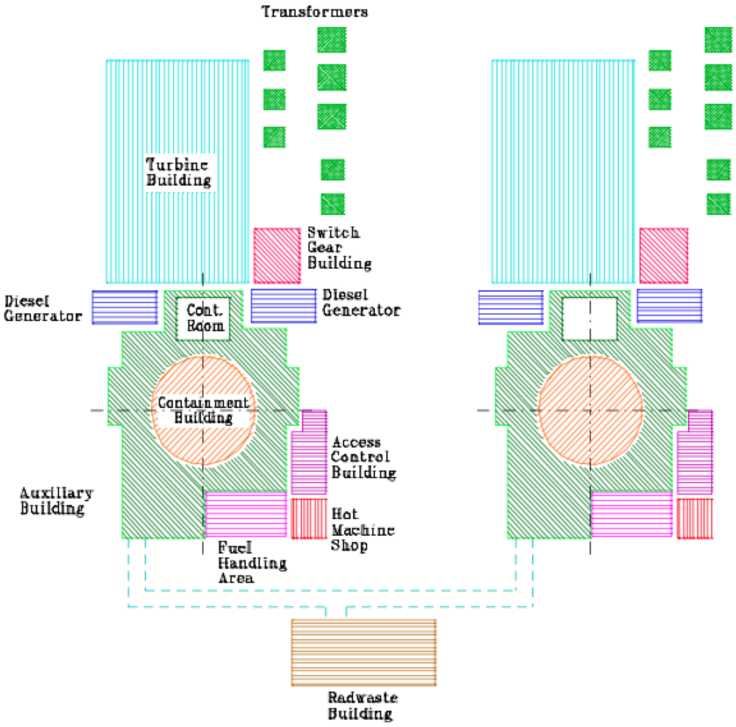

Figure C.3.: Plant layout of the APR-1400 [54, Fig. 6]

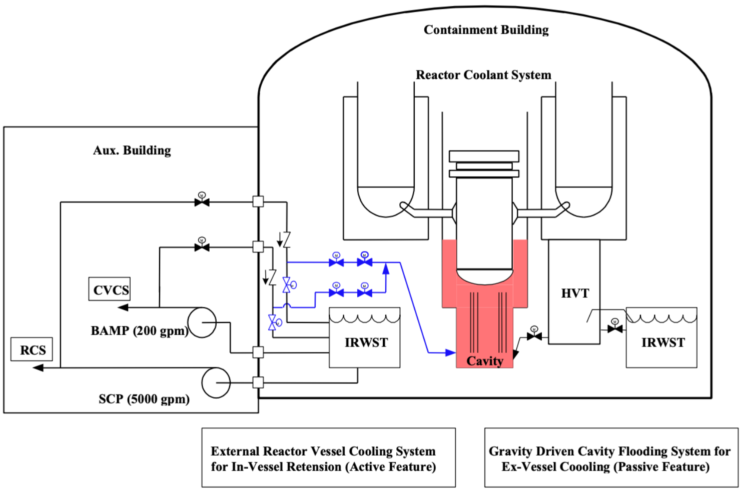

Figure C.4.: ERVCS and CFS of the APR-1400 [32, Fig. 18]

Appendix D.

Safety Concepts of Generation III+ Reactors

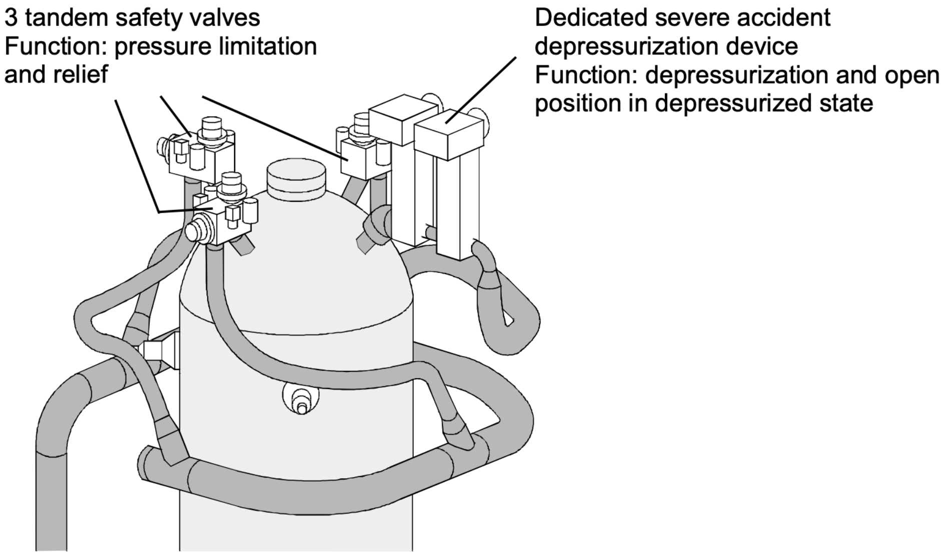

Figure D.1.: Depressurization system of the EPR [64, Fig. 3]

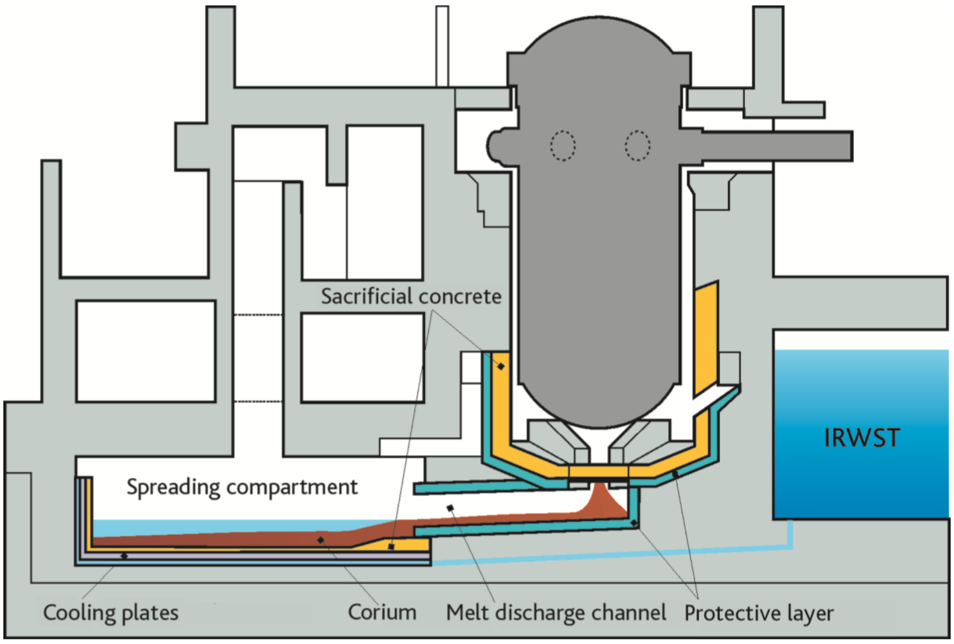

Figure D.2.: Cutaway view of the EPR core catcher [57, Fig. 4.7]

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