Advanced Review

Environmental characteristics of the current Generation III nuclear power plants

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The environmental impact of electricity generation sources can be generically characterized by their use of natural resources, the thermal pollution they cause, and their emissions of chemical pollutants and radionuclides. Nuclear power plants generate concern regarding their radioactive emissions, which are often poorly understood. A presentation is made of the known data on the environmental impact of pre-Gen III nuclear power plants during normal operation and as a consequence of severe accidents. The radiation doses received by the public and exposed workers as a consequence of nuclear power are compared with the ones because of natural radioactivity and medical applications. The characteristics of the new Gen III reactors which will bring significant improvements to the environmental impact of nuclear power generation are discussed in detail. © 2013 John Wiley & Sons, Ltd.

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INTRODUCTION

Nuclear power plants (NPPs) based on fission reactors are used to generate electricity since the early 1950s. The experimental breeder reactor (EBR) in Arco, United States, generated electricity for the first time on December 1951, illuminating four light bulbs.[1] Although the EBR was not connected to the electrical grid, this path was quickly followed. More than 500 power reactors were meanwhile connected to the grid. Figure 1 shows the evolution of the total number of power reactors connected to the electrical grid until the end of 2011, as well as the total number of operating power reactors, with data from the PRIS[2] database of the International Atomic Energy Agency (IAEA). The accumulated operating experience currently exceeds 14,000 reactor·year.[3] Figure 1 also indicates the years of occurrence of the Three Mile Island (TMI), Chernobyl, and Fukushima accidents, the three major events of this type.[4]

Most NPPs in operation today were built in the 1970s and 1980s and are considered as Generation II type, as they are based on the experience gained with the Generation I plants built in the early days. Generation III NPPs started being built in the early 1990s. They are based on advanced designs featuring improved safety and economics, whereas Generation III+ plants include further developments. We will refer to Generation III and III+ plants simply as Gen III plants in this review.

Gen III plants present a set of distinctive characteristics:[5–7]

- A simpler and more rugged design, making the reactors easier to operate and less vulnerable to operational disturbances.
- Significant use of passive safety features that require no active controls and rely on natural phenomena.
- Reduced probability of occurrence of accidents involving core melting.
- New mitigation measures in case of core melt accidents, to reduce significantly the impact

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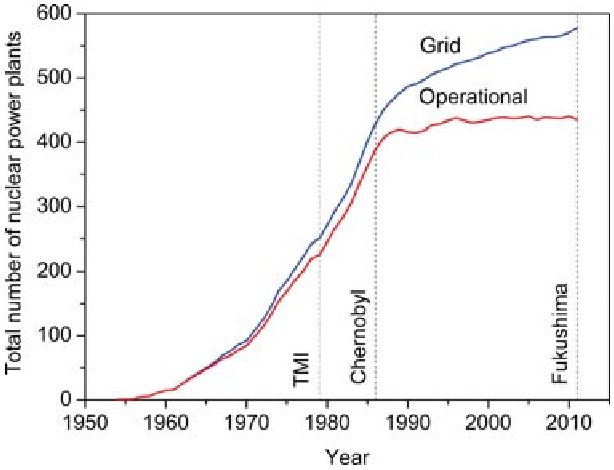

FIGURE 1 | Evolution of the total number of power reactors connected to the electrical grid and of the total number of operating power reactors until the end of 2011.

of such accidents to the environment and to the public.

- Resistance to the impact of a large aircraft.
- Standardized designs, able to reduce licensing and construction time, as well as capital cost.
- Longer time interval between refueling, resulting in a higher availability.
- Higher burn-up to increase fuel use and reduce the amount of waste produced.
- Longer operating lifetime, of 60 years, already from design.

The environmental impact of electricity generation sources can be generically characterized as follows:[8] (1) use of natural resources, (2) thermal pollution, (3) emission of chemical pollutants, and (4) emission of radionuclides. The emission of radionuclides is probably the less understood and more feared component of the environmental impact of NPPs, even if the emissions from coal power plants are also significant.[9,10] We will address the environmental impact of Gen III NPPs in two groups: nonradiological aspects and radiological aspects.

ENVIRONMENTAL IMPACT: NONRADIOLOGICAL ASPECTS

The vast majority of today's fission reactors operate in an open fuel cycle, in which uranium is mined, enriched up to 3–4% in U-235, partially fissioned in the reactor and stored after that. In such a cycle, a generic 1.0 GW_e NPP working 8000 h per year requires about 180 t of natural uranium, which is transformed into about 30 t of enriched uranium, out of which approximately only 1 t will be fissioned.[11]

Gen III reactors make a better use of natural resources than their predecessors. The average discharge burn-up rate achieved in a pre-Gen III pressurized water reactor (PWR) was approximately 30 MWd/kgU in the 1980s, whereas most Gen III PWR will attain 50–60 MWd/kgU,[7] mostly because of the progress made in manufacturing the fuel elements. This burn-up increase also allows increasing the time between refueling operations, thus improving the availability of the NPPs. Recent studies show that there is little economic gain in increasing the burn-up above 60 MWd/kgU with the current fuel cycles.[12] On the other hand, the currently used UO_2-based fuel undergoes structural changes during the irradiation in the reactor and these do not allow the burn-up to increase significantly above 50–60 MWd/kgU.[13] Further improvements are only expected with Generation IV reactors, using different fuels and fuel cycles, where burn-up values in excess of 100 MWd/kgU are anticipated.[14]

The increase in the attained fuel burn-up values translates directly in the production of less high-level radioactive waste. As most countries with NPPs use an open fuel cycle, the fuel elements are considered as waste once they are removed from the core of reactor. Besides the approximately 30 t of irradiated fuel (high level waste), a 1 GWe NPP generates yearly 150–300 m^3 of other wastes, mostly low level waste.[15]

One must refer that NPPs require significantly less natural resources than fossil plants. A 1 GW_e coal power plant working 8000 h per year requires approximately 4×10^6 t per year.[16] Even if the natural uranium for the 1 GW_e mentioned above would be obtained from a 0.1% low grade ore, there would still an advantage factor of 20 for nuclear; some uranium deposits have ores with more than 20% U_3O_8,[17] and thus the average advantage factor is higher.

The thermal pollution issue is common to all thermal plants, regardless of whether the heat source is nuclear fission or combustion of fossil fuels. Steam–electric power plants heat water in pressurized boilers to produce steam that is used to spin turbines and generate electricity. The steam is condensed back to water after passing through the turbines and this water is returned to the boiler. The condenser uses large quantities of additional water, pumped from a water body, to condense the steam. The pumped water gains a significant amount of heat in the process. For a power plant with 33% thermal efficiency, about 67% of the heat produced by the power plant has to be disposed of, e.g., 2000 MW for a 3000 MW thermal plant (1000 MW_e). The heat in the coolant is reduced by releasing it into the atmosphere or by returning it to a water body, usually the one from which it was originally extracted. The difference between the final and initial temperatures of the water used in the condenser is usually in the range of 5–25°C.[18] Such temperature increases have the potential to alter the biological integrity of the water body ecosystem.[19] Thermal pollution tends to be somewhat more severe for NPPs, as their power is normally higher than the one of conventional thermal plants. When it is not possible to return the coolant directly to the water body where it was extracted, closed-cycle or recirculating cooling systems can be used. These use vertical cooling towers in which the effluent either falls from a height so that evaporative cooling occurs in air or passes through a radiator in which cooling occurs with no direct contact of the effluent with air (thus reducing evaporative loss of water) or incorporate ponds in which heated discharges cool sufficiently before being reused.

Fossil plants pollute the environment with sulfur-containing products, carbonaceous particulates, fly ash, carbon dioxide, and some heavy metals. NPPs do not emit these substances during production. On a global perspective, considering the whole fuel cycle, it is nevertheless possible to assign greenhouse gas emissions for NPPs, considering that conventional sources were used during uranium extraction and processing, as well as during the construction and dismantling of the NPP. The reported range of emissions for nuclear energy over the lifetime of a plant is from 1.4 g of carbon dioxide equivalent per kWh (g CO_2 equiv./kWh) to 288 g CO_2 equiv./kWh, with a mean value of 66 g CO_2 equiv./kWh.[20]

ENVIRONMENTAL IMPACT: RADIOLOGICAL ASPECTS

Naturally occurring radionuclides of terrestrial origin, also called primordial radionuclides, are present in the environment and even in the human body. Only those radionuclides with half-lives comparable with the age of the Earth, and their decay products, are found in significant quantities. The three most important primordial radionuclides are K-40, Th-232, and U-238.[9] In addition, high-energy cosmic ray particles incident on the earth's atmosphere also contribute to human exposure. The average worldwide exposure to natural radiation sources is 2.4 mSv per year.[21] However, the range of individual doses is wide. It is estimated that about 65% of the population have exposures between 1 and 3 mSv, about 25% have exposures less than 1 mSv, and 10% have exposures greater than 3 mSv.[21]

The second largest contribution to radiation exposure is from medical procedures. The use of ionizing radiation for medical diagnosis is widespread throughout the world, albeit with significant variations from country to country. These exposures are characterized by doses typically in the range 0.1–10 mSv per examination. The average worldwide exposure because of diagnostic medical examinations is 0.4 mSv per year, whereas in developed countries this average value is 1.2 mSv per year.[22]

The third and less important source of exposure derives from releases of radioactive materials to the environment from several activities, practices and events involving radiation. Hu et al.[23] have recently reviewed the sources of anthropogenic radionuclides in the environment from nuclear weapons programs, nuclear weapons testing, NPPs, uranium mining and milling, commercial fuel reprocessing, geological repository of high-level nuclear wastes that include radionuclides which might be released in the future, and nuclear accidents. Figure 2 shows radioactivity evaluations in various areas throughout the world made by Hu et al.[23] Several events are considered: (1) atmospheric and underground weapon tests in Polynesia and Nevada Test Station (NTS), (2) areas contaminated by the weapons programs in the former USSR (Mayak, Techa River, Lake Karachai, Kyshtym accident) and in the United States (Hanford), (3) spent fuel in an abandoned nuclear powered ship (Lepse), (4) the radiological accident of Goiânia, Brazil, (5) the TMI and Chernobyl nuclear accidents, and (6) the maximum activity in reactor spent fuel which was planned for storage in the geological repository of the Yucca Mountain Project (YMP). The U.S. Environment Protection Agency (EPA) drinking water standards for tritium and alpha particles are shown on the same scale, at different times and only for relative comparison.

The testing of nuclear weapons in the atmosphere, from 1945 to 1980, was the most significant cause of exposure to man-made environmental sources of radiation.[24] It has been calculated that the world average annual effective dose reached a peak of 150 μSv in 1963, which had decreased to about 5 μSv in 2000, from residual radionuclides in the environment.[22]

During NPP normal operation, the intact cladding of the fuel represents an almost perfect barrier to the release of radionuclides from the fuel to the coolant, with the exception of some tritium.[25] Nevertheless, small amounts of fission products are still found in the coolant, mostly from small defects in the cladding of fuel rods, which happen on average in 1 out of every 10^5 rods.[26] Three categories of

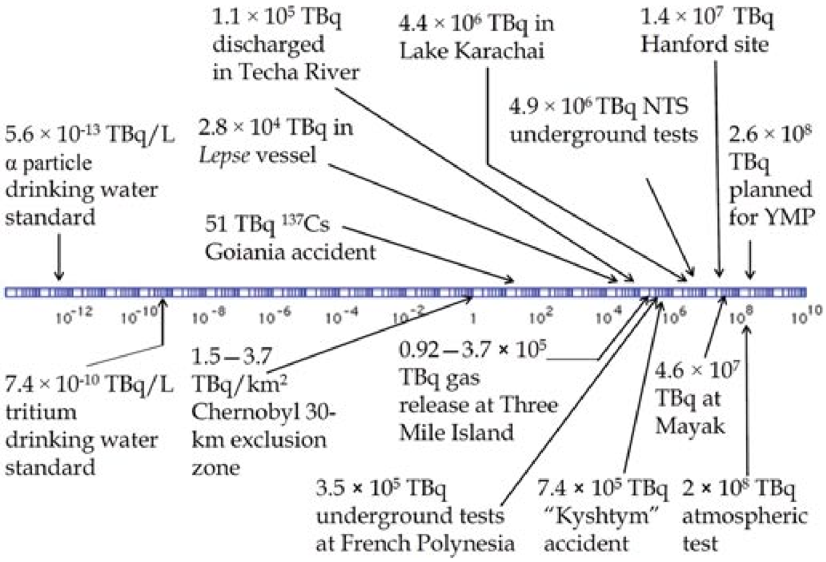

FIGURE 2 | Radioactivity evaluations in various areas throughout the world. (Reproduced with permission from Ref 23. Copyright 2008, Elsevier.)

radionuclides are found in the primary coolant because of small defects in rods: noble gases, iodine isotopes, and cesium isotopes.[25] The most important noble gases are the fission products Xe-133 and Xe-135 in a PWR, whereas the composition of the noble gas releases is more varied in a boiling water reactor (BWR), where most krypton and xenon radionuclides are included. The main contributors to the total iodine activity are the I-131 and I-133 fission products. The fission product Cs-137 is always the major cesium isotope in the coolant (as in the fuel) and is found together with Cs-134, which is produced from the neutron activation of the stable fission product Cs-133. The normalized release rates of noble gases and of tritium are generally in the interval of 1–100 TBq/GW/year, whereas the ones of I-131 and of particulate radionuclides are generally in the interval of 1–10 GBq/GW/year.[27] The average annual dose because of the operation of NPPs is 0.2 μSv per year.[22] NPPs are often designed to give a dose of no more than 0.05 mSv/year (about 2% of the natural background) to the most exposed nonworker group.[28]

Figure 3 shows the importance of the different components of the average worldwide radiation exposure from natural radiation, medical applications and anthropogenic radionuclides, where it is clear that the last component is the smallest one.

The doses to exposed workers in the nuclear industry are also relatively low. A recent study involving nearly 600,000 workers employed in 154 facilities in 15 countries, with data going back to the 1940s, has revealed an average yearly exposure of 1.5 mSv/year per worker.[29]

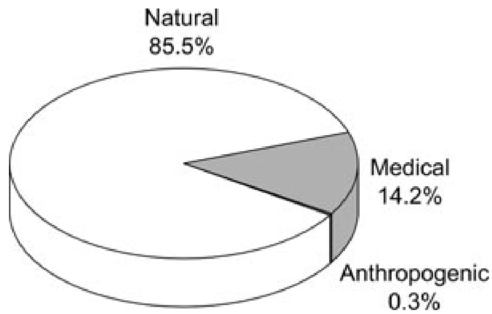

FIGURE 3 | Average worldwide radiation exposure from natural radiation, medical applications, and anthropogenic radionuclides.

When severe accidents occur in NPPs, the environmental contamination and exposures may become significant. The radiation exposure that the public may receive is determined to a large extent by the amount of fission products that escape or are released from the containment. This amount depends in turn upon the quantity of fission products that are released from the fuel and are transported into the containment, the fractional deposition and removal by engineered systems within the containment, and the fractional leakage from the containment.[30]

Even in cases where a significant meltdown occurs, not the whole core inventory is released, as the chemical behavior of the elements plays a determinant role. The inventory of fission products, activation products, and transuranic elements in the core includes more than 800 nuclides, but not all are radioactive. If the stable isotopes and the ones with half-lives less than about 26 min (as they will not be

TABLE 1 | Elements in the Core Inventory Grouped by Chemical Behavior

Group	Name	Elements in Group
1	Noble gases	Xe, Kr
2	Halogens	I, Br
3	Alkali metals	Cs, Rb
4	Tellurium group	Te, Sb, Se
5	Barium, strontium	Ba, Sr
6	Noble metals	Ru, Rh, Pd, Mo, Tc, Co
7	Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
8	Cerium group	Ce, Pu, Np

TABLE 2 | Relative Importance to Early Bone Marrow Dose of the Radioactive Isotopes Found in the Core of a 3412 MW PWR after a Significant Release

Group	Activity (Ci)	Main Contributor(s) to Dose	Relative Importance to Dose (%)
1	1.66×10^8	Krypton	12.8
2	7.75×10^8	Iodine	30.8
3	2.31×10^7	Cesium	6.9
4	2.15×10^8	Tellurium	33.6
5	3.54×10^8	Strontium, Barium	6.2
6–8	3.23×10^9	–	7.4

significant in terms of risk analysis) are removed, then only 54 nuclides remain.[31] These can be conveniently divided in eight groups,[32] on the basis of similarity of chemical behavior, as shown in Table 1.

Table 2 shows the relative importance to early bone marrow dose (4 h exposure) of the radioactive isotopes found in the core of a 3412 MW PWR after a significant release,[33] where iodine and tellurium isotopes are responsible for approximately two-third of the dose, even if they account only for 20% of the total activity. In general, the iodine isotopes and the tellurium isotopes which decay to iodine are responsible for most of the short-term dose, whereas Cs-137 is responsible for most of the dose integrated over periods of the order of several years.[31]

The increase in fuel burn-up for Gen III reactors, when compared with previous generations, leads to some changes in the inventory of fission products, activation products, and transuranic elements in the core. A similar situation happens when considering the use of MOX fuel (containing a mix of uranium and plutonium oxides). Nevertheless, detailed simulations of practical core configurations showed that these inventory differences have a small impact in accident scenarios.[34,35]

RADIOACTIVE RELEASES DURING ACCIDENTS IN NPPS

Three accidents are recorded in the history of commercial NPPs: TMI in 1979, Chernobyl in 1986, and Fukushima in 2011. According to the International Nuclear Event Scale (INES) of the IAEA, TMI was a level 5 accident (or an accident with wider consequences). The Chernobyl and Fukushima accidents were level 7 events (major accidents), as the radioactive releases to the atmosphere exceeded a quantity equivalent to several tens of thousands of TBq of I-131 (10,000 TBq corresponds to approximately 0.3 MCi).[36] The characteristics of the accidents and their radiological consequences were significantly different.

The radioactive releases from the TMI accident were minimal. The noble gas release was estimated at about 10 MCi, mostly as Xe-133, whereas the iodine release was estimated at only 18 Ci, mostly as I-131.[31] The resulting radiation doses to members of the public were small. The average likely and maximum whole-body gamma-doses for individuals in a 5 mile area around TMI, during the 10 days after the accident, were determined to be 0.09 and 0.25 mSv, respectively.[37] Several public health studies were made since the accident.[37–41] The most recent study, on a cohort of 32,000 people, confirms previous conclusions that there is no consistent evidence that the radioactivity released during the TMI accident had a significant impact on the mortality through 1998.[41]

The releases from the Chernobyl accident were the largest occurring thus far. The total release from the accident is estimated at approximately 320 MCi, including 48 MCi of I-131 and 2 MCi of Cs-137.[42] The accident caused many severe radiation effects almost immediately. Of 600 workers present on the site during the early morning of April 26, 1986, 134 received high doses (0.7–13.4 Gy) and suffered from radiation sickness. Of these, 30 died within a few months of the accident. The average doses to those persons most affected by the accident were about 100 mSv for 240,000 recovery operation workers, 30 mSv for 116,000 evacuated persons, and 10 mSv during the first decade after the accident to those who continued to reside in contaminated areas.[22] Several countries outside the ex-USSR were affected by the accident. Doses in European countries were at most 1 mSv in the first year after the accident with progressively decreasing doses in subsequent years. The dose over a lifetime was estimated to be 2–5 times the first-year dose. These doses are comparable with an annual dose from natural background radiation and are, therefore, of little radiological significance. The exposures were highest in the local areas surrounding the reactor, but low-level exposures could be estimated for the European region and for the entire northern hemisphere. In the first year following the accident, the highest regionally averaged annual doses in Europe outside the former USSR were less than 50% of the natural background dose and subsequent exposures decreased rapidly. The worldwide annual per caput effective dose due this accident was 2 μSv in the year 2000.[22]

Several studies on the health effects were made following the accident.[42–47] About 4000 cases of thyroid cancer in exposed children and adolescents were diagnosed in the 1990–2002 period. However, the rapid increase in detected thyroid cancers suggests that some of it at least is an artifact of the screening process.[43] Thyroid cancer is usually not fatal if diagnosed and treated early. So far no increased risk of leukemia has been observed in children, liquidators, and in the general population. Increases in a number of nonspecific detrimental health effects other than cancer have been observed in liquidators and residents, but these are difficult to interpret, because of a lack of baseline for comparison.[43]

The releases from the Fukushima accident were significantly lower than the ones from the Chernobyl accident. The Nuclear Safety Commission of Japan estimated that 4 MCi of I-131 and 0.3 MCi of Cs-137 were released.[48] Details on this estimate were later given by Chino et al.[49] During the initial recovery operations about 30 workers received doses in excess of 100 mSv but below the 250 mSv limit set by the Japanese authorities for emergency workers. The exposure to the public was relatively small in part because the prevailing winds blew much of the radioactive releases toward the ocean, and because of the protection actions—evacuation within a 20 km radius and sheltering within 30 km—that were promptly taken.[50] The estimated committed equivalent dose for an adult in the first year after the accident in the most affected areas of the Fukushima prefecture in the interval of 1–50 mSv, whereas in the rest of Japan it is in the interval of 1–10 mSv.[51]

The consequences outside Japan were limited. Airborne radioactivity was transported through the Pacific to North America and afterwards to Europe. Figure 4 shows the result of a model of plume transport from Japan to U.S. territories, obtained by the National Atmospheric Release Advisory Center (NARAC) at Lawrence Livermore National Laboratory.[52] Colored particles represent airborne radioactivity, with different colors indicating different days of potential releases from Japan. The highest concentrations measured in aerosols were those of particulate I-131, at levels below 5 mBq/m^3 in the United States.[53] and in Europe,[54] which have no radiological impact. The World Health Organization estimated the committed equivalent dose for an adult in the first year outside Japan to be below 0.01 mSv.[51]

PREVENTION AND MITIGATION OF SEVERE ACCIDENTS

The potential hazards of NPPs were recognized very early and features to prevent, contain, and otherwise protect the environment and the public were applied from the outset.[55] As in other industries, the design and operation of NPPs aims to minimize the likelihood of accidents and to mitigate their environmental consequences when they occur.

The concept of defence-in-depth (DID) is fundamental to the safety of nuclear installations. The first priority for DID is to prevent accidents and the second priority is to limit the potential consequences of accidents and prevent any evolution to more serious conditions. The rationale for the priority is that provisions to prevent deviations of the plant state from well-known operating conditions are generally more effective and more predictable than measures aimed at mitigation of the consequences of such a departure, because the plant's performance generally deteriorates when the status of the plant or a component departs from normal conditions.[56] DID relies on having multiple independent protections against the occurrence of accidents and their progression, in such a way that, should one of these protections fail, at least another is present whose failure is independent from the operation of the previous one. This is achieved in practice by means of four successive barriers that prevent the release of the radioactive fission products: (1) the fuel matrix, (2) the cladding of the fuel, (3) the primary coolant boundary, and (4) the containment building. The application of DID helps to ensure that the three basic safety functions—control of the reactivity, removal of heat from the fuel in the core, and confinement of the radioactive fission products—are preserved.[5] Figure 5 illustrates the typical arrangement of the four barriers.

The importance of the barriers is more obvious during severe accidents. In most such cases, the first two barriers are progressively breached and part of the core inventory is released to the containment. The

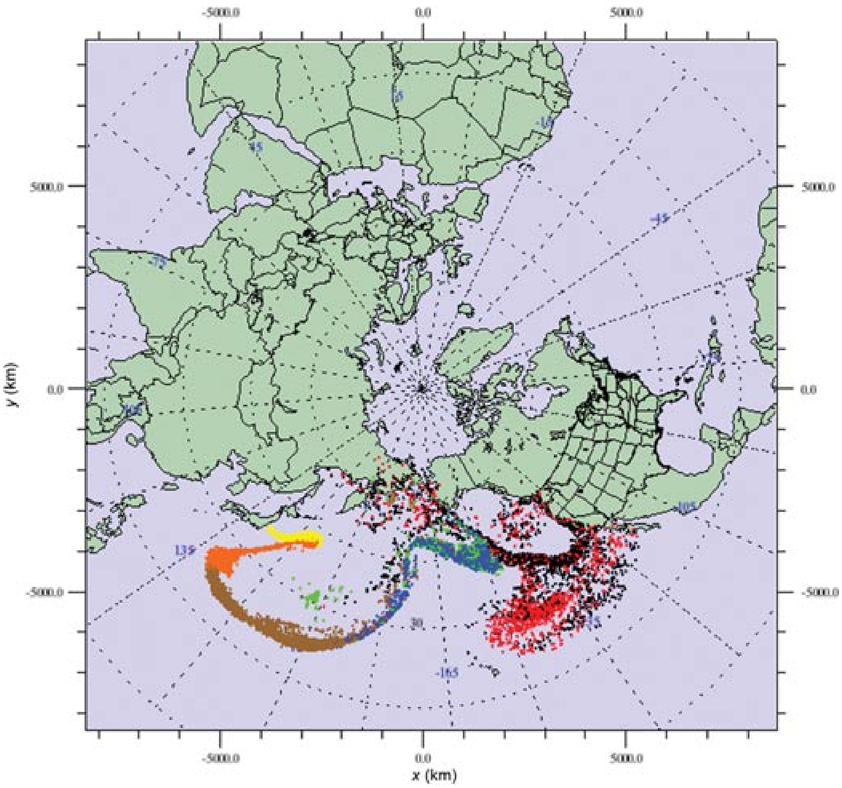

FIGURE 4 | Model of plume transport from Japan to U.S. territories as a result of the Fukushima accident. Colored particles represent airborne radioactivity, with different colors indicating different days of potential releases from Japan. Graphic provided courtesy of the U.S. Department of Energy's National Atmospheric Release Advisory Center (NARAC) at Lawrence Livermore National Laboratory (LLNL). Inclusion of this graphic does not imply endorsement of the conclusions of this paper by NARAC/LLNL.

containment building represents the final barrier and avoiding the release of radioactivity to the environment relies upon maintain its integrity.

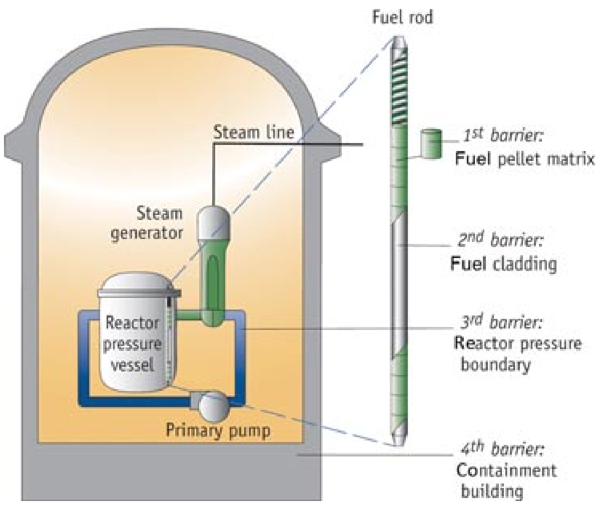

FIGURE 5 | Typical barriers confining radioactive materials in nuclear power plants. (Reproduced with permission from OECD/Nuclear Energy Agency (2003), Nuclear Energy Today, Nuclear Development, OECD Publishing. http://dx.doi.org/10.1787/9789264103306-en.)

The safety features implemented in NPPs evolved significantly with successive generations. Gen II designs were conservatively designed against design basis accidents. The TMI accident was the driving force to further prevent severe accidents and to mitigate their consequences by understanding and improving the barriers to major releases. A key recommendation in the early 1990s was that severe accidents beyond the existing design basis should be systematically considered and addressed during the design phase of new reactors.[57]

Gen III designs have an increased reliance in passive systems, when compared with designs of previous generations. The use of passive systems circumvents the eventual disruption of external sources of electricity, cooling water, and other essential supplies following an extreme event. Several Gen III designs provide for the physical presence of large thermal capacity heat sinks available to cool the reactor core without depending on the availability of externally powered pumps, relying rather on cooling by natural convection, radiation, and conduction. When valves are required for the activation of passive safety systems, they are generally 'fail safe', as they require power to stay in their normal, closed position, and loss of power causes them to open; their movement is made using stored energy from compressed gas, batteries, or springs.

The IAEA issued its 'Basic Safety Principles for Nuclear Power Plants' in 1988 and revised it in 1999.[58] The IAEA recommends that the core damage frequency (CDF) value for advanced designs should not exceed 1×10^{-5} events per reactor·year, which is a factor of ten lower than the U.S. Nuclear Regulatory Commission (NRC) requirement for the CDF of current plants. This recommendation has been widely adopted both by utilities and manufacturers for new NPPs.[59] The CDF value taken as representative for current plants is 5×10^{-5} events per reactor-year, even if this is subjected to large variations by design and country.[60]

To guarantee that a core melt does not necessarily lead to a large radioactive release from the reactor, the IAEA recommended that severe accident management and mitigation measures are used to reduce by a factor of at least ten the probability of large off-site releases requiring short term off-site response.[58] The definition of large release is not universal. It can be defined either as absolute magnitude of activity and isotope released, e.g., 100 TBq of Cs-137 or as relative magnitude, e.g., 1% of the core inventory of Cs-137 from an 1800 MWt BWR.[61] Most vendors specify for their designs a large release frequency (LRF) or, sometimes, a large early release frequency (LERF). In contrast to the relatively moderate differences in the criteria used to specify the values of CDF, there is both a considerably larger variation in the L(E)RF limits, and very different answers to the question of what constitutes an unacceptable release.

Gen III NPPs include provisions to cool and contain the corium, i.e., the molten fuel-structure mixture resulting from a core melt, to prevent large (early) radioactive releases from the reactor, following a severe accident. Cooling the corium is essential, because the release of fission products and the generation of noncondensable gas stop as the melt/debris temperature drops below approximately 1000°C.[62]

The cooling and contention of the corium are achieved using ex-vessel or in-vessel structures. An ex-vessel structure, also designated core catcher, adds an additional barrier that aims at limiting and restricting the consequences of an accident with core melting to the immediate vicinity of the plant. As this requires an intact confinement, it is necessary to avoid an attack of the molten core on the containment base material. An ex-vessel core catcher increases the surface-to-volume ratio of the melt after its release from the reactor pressure vessel and allows for the effective quenching and stabilization of the melt before it can attack the structural concrete. The corium in the core catcher can be cooled passively or actively. The deliberate interaction with sacrificial materials, usually concrete or oxide materials, on a first layer helps to cool the corium and to keep it liquid over a wider temperature range, so that it spreads efficiently. The use of nonlimestone aggregate concrete (so called basaltic concrete) minimizes further production of carbon-based noncondensable gases, such as CO and CO_2, which could contribute to an eventual failure of the containment.

The main objective of in-vessel retention of the corium is to maintain the reactor pressure vessel as a barrier against the release of fission products, by preventing a melt-through of the vessel. This is achieved by flooding the reactor cavity and transferring the decay heat from the corium on the lower head of the pressure vessel to the water surrounding the vessel.[63] This heat transfer must be efficient so that the vessel wall maintains its structural properties and is able to support the mechanical load that results from the weight of the corium and the lower head, and from a possible pressure increase inside the vessel. The main advantage of this type of corium retention scheme is the fact that all ex-vessel phenomena are avoided, such as direct containment heating, corium-concrete interactions, and eventual steam explosions.[62]

OVERVIEW OF CURRENT GEN III NUCLEAR REACTORS

Table 3 shows the main Gen III designs constructed, under construction, or undergoing licensing procedures,[6,64] in the categories of BWR, PWR, and pressurized heavy-water reactors (PHWR), by alphabetical order of their abbreviation.

The first Gen III reactor that went into commercial operation was an advanced boiling water reactor (ABWR) as unit 6 of the Kashiwazaki–Kariwa NPP (Japan) in 1996. Compared with its predecessor BWR designs, the ABWR featured reactor internal recirculation pumps, fine-motion control rod drives, an improved emergency core cooling system (ECCS), as well as an advanced control room, using digital and fiber optic technologies.[65]

The use of internal pumps instead of external excludes large pipe ruptures at or below the elevation of the core and is a key factor to keep the core completely and continuously flooded for the entire spectrum of design basis loss of coolant accident (LOCA). The CDF (internal events) for the ABWR is 1.6×10^{-7}

TABLE 3 | Generation III Fission Reactors in Operation, Construction, or Licensing

Reactor[1]	Developer(s)	Net Electric Output (MW_e)	Type[2]	Status
ABWR	General Electric, Toshiba, Hitachi	1315	BWR	Start of operation in Japan, 1996
ESBWR	General Electric Hitachi	1333	BWR	Design certification on-going in the United States
AES-92	Gidropress	1000	PWR	Start of operation in India, 2013[3]
AP1000	Westinghouse	1117	PWR	In construction. Start of operation in China, 2013[3]
APR-1400	Korea Hydro & Nuclear Power	1350	PWR	In construction. Start of operation in South Korea, 2013[3]
APWR	Mitsubishi	1600	PWR	In construction. Start of operation in Japan, 2016[3]
EPR	Areva	1600	PWR	In construction. Start of operation in China, 2013[3]
ACR-1000	Atomic Energy of Canada	1082	PHWR	Design certification on-going in Canada. Start of operation in 2016[3]
EC6	Atomic Energy of Canada	690	PHWR	Design certification on-going in Canada

[1] AP1000 is a trademark of Westinghouse Electric Company, LLC; APR1400 is a trademark of Korea Hydro & Nuclear Power Company; EPR is a trademark of the Areva Group; ACR-1000 and EC6 are registered trademarks of Atomic Energy of Canada Limited.
[2] BWR, boiling water reactor; PWR, pressurized water reactor; PHWR, pressurized heavy water reactor.
[3] Expected.

events per reactor·year,[66] nearly an order of magnitude lower than General Electric's BWR/6 older design.[67] The ABWR also features a basaltic floor with passive cooling features that will terminate the flow of corium in the event of a core melt.[68] The LRF for the ABWR is 2×10^{-8} events per reactor·year.[66]

The ABWR is certified or licensed in three countries: the United States, Japan, and Taiwan. The NRC issued a final rule certifying the ABWR design in May 1997.[69] The current certification in the United States expires in 2012 and Toshiba submitted a renewal application in late 2010. The compliance assessment with the European utility requirements (EUR) was successfully completed in December 2001.[70] Four ABWR units are in operation in Japan (Kashiwazaki–Kariwa-6 and 7, Hamaoka-5, and Shika-2), two are under construction in Taiwan (Lungmen-1 and 2), and one unit is under construction in Japan (Shimane-3).

The economic simplified boiling water reactor (ESBWR) is a Gen III design, with a thermal power of 4500 MW and net electric power of 1535 MW_e (gross 1600 MW_e),[71] which took several technological features from the ABWR. The core is made shorter than conventional BWR plants to reduce the pressure drop over the fuel and improve natural circulation.[72,73] The ESBWR has no recirculation pumps, external or internal, thereby greatly increasing design integrity and reducing overall costs.

Figure 6 shows the general arrangement of the passive safety system of the ESBWR.[74] It incorporates four redundant and independent divisions of a passive gravity driven core cooling system (GDCS), an automatic depressurization system (ADS), and a passive containment cooling system (PCCS). In case of a LOCA, the ADS depressurizes the reactor vessel and the GDCS injects sufficient water to maintain the fuel cooled. Heat removal and inventory addition are also provided by an isolation condenser system (ICS) and the standby liquid control system (SLCS). The reactor pressure vessel has no external recirculation loops or large pipe nozzles below the top of the core region. This, together with a high capacity ADS allowed the incorporation of an ECCS driven solely by gravity, not needing any pumps. The water source needed for the ECCS function is stored in the containment upper drywell, with sufficient water to insure core coverage to 1 meter above the top of active fuel as well as flooding the lower drywell. The PCCS heat exchangers are located above and immediately outside of containment. There is sufficient water in the external pools to remove decay heat for at least 72 h following a postulated design basis accident, and provisions exist for external makeup beyond that, if necessary. The ESBWR is also equipped with an ex-vessel core catcher, which uses thick concrete and a passive cooling system to prevent escape of the corium from the containment.[75]

As a result of these safety systems, there is an increase in the calculated safety performance margin of the ESBWR over earlier BWRs. The CDF of the ESBWR is currently the lowest of all Gen III designs, at 2.8×10^{-8} events per reactor·year for initiating events occurring during power operation, and at 3.36×10^{-8} events per reactor·year when the plant is shutdown (in both cases the CDF values include internal events, plus fire and flood).[76] The design certification review of the ESBWR by the NRC was started in 2005, with revision 6 of the design control documents having been submitted in August 2009. Step 2 of the 'Generic Design Assessment' in the United Kingdom was completed during 2008.[75] The

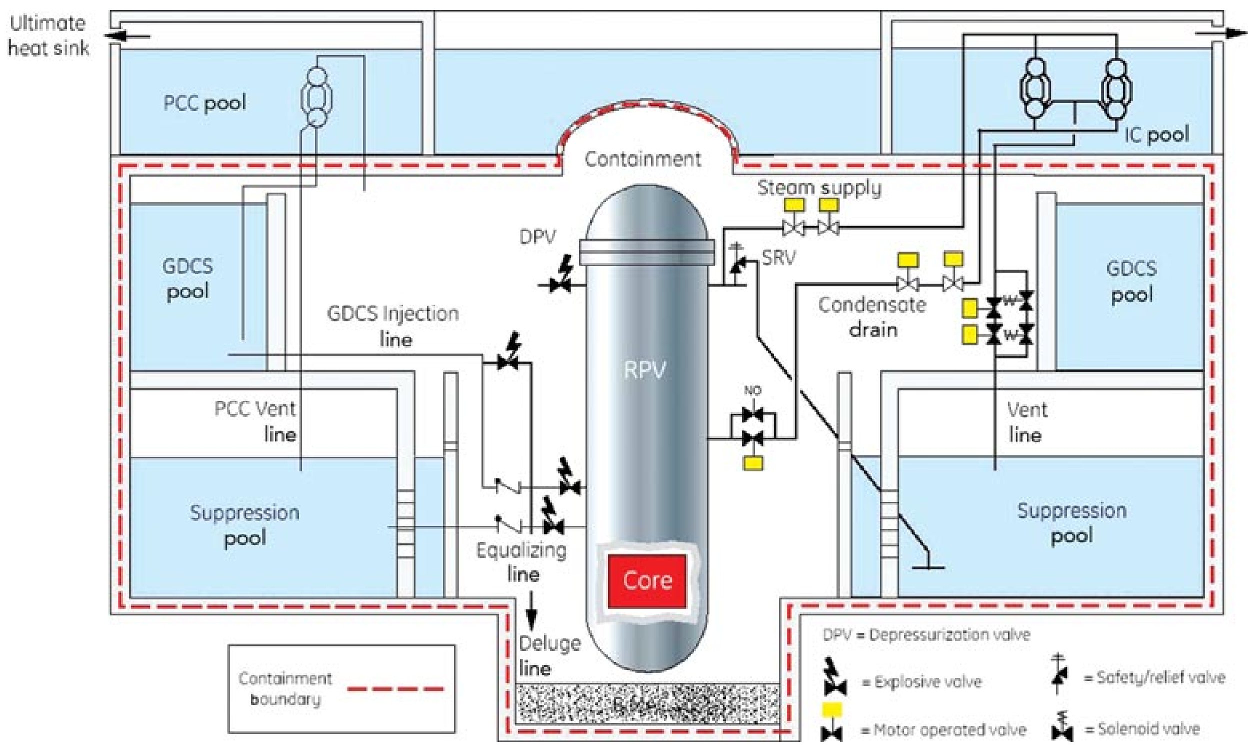

FIGURE 6 | Passive safety features of the economic simplified boiling water reactor (ESBWR). (Reproduced with permission from Ref 74. Copyright 2009, Elsevier.)

NRC received Combined License (COL) requests for the construction of five ESBWR units in the United States, as of March 2012.[77]

The AES-92 is an advanced PWR designed by Gidropress (Russia) with 1000 MW_e net electric output (1068 MW_e gross electric output), also designated NPP-92 or V-392.[59,78] It is based on the well-known VVER-1000 (from 'Vodo-Vodyanoi Energetichesky Reactor', literally translatable as water–water energetic reactor). The AES-92 uses a combination of active and passive safety systems. Its coolant system includes an accumulator in each loop as a passive part of the emergency core cooling system.[79] Each accumulator stores 50 m^3 of borated water which is automatically injected if the primary circuit pressure falls during a LOCA. An ex-vessel core catcher is provided as mitigation measure, similar to the one of the previous AES-91 design installed in China.[80] The sacrificial material contains gadolinium oxide, a neutron absorber, in its composition so that the molten mass will remain sub-critical.[81]

The compliance assessment of the AES-92 with the EUR was successfully completed in June 2006.[78] Two AES-92 units, in its variant V-412 are expected to start operating in India (Kudankulam-1 and 2) in 2012–2013.[82] The AES-92 design, in its variant V-466B was planned for installation in the Belene power plant in Bulgaria (two units) and will now be installed in the Kozloduy power plant.[83] The V-466B design has a planned lifetime of 60 years, whereas the earlier AES-92 variants were planned for 40 years. The calculated CDF for the AES-92 in Bulgaria is $<6.1 \times 10^{-7}$ events per reactor·year, whereas the LERF is $<1.77 \times 10^{-8}$ events per reactor·year.[84]

Gidropress also developed the AES-2006 (also designated VVER-1200), with a thermal output of 3200–3300 MW and net electric output of approximately 1200 MW_e. The inner vessel diameter of the AES-2006 will be 10 cm larger than the one of the AES-92, to decrease the neutron fluence in the RPV.[85] The AES-2006 (V-392M) will have a lifetime of 60 years, as the AES-92 in its variant V-466B.[86] The first two units of the AES-2006 are planned for the Novovoronezh II plant in Russia in 2014.[87] Russia signed an agreement with India in early 2010 which includes eight new VVER reactors, of the AES-92 or AES-2006 designs.[82]

The AP1000 is an advanced passive PWR developed by Westinghouse (United States). It is based on the AP600, a Gen III design approved by the NRC in 1998, for which no units were built. It has 1117 MW_e net electric output (1200 MW_e gross).[88] The reactor vessel is the same as that for a standard Westinghouse three-loop plant, with nozzles adjusted to accommodate the two loops of the new design. The safety systems include passive safety injection, passive residual heat removal, and passive containment cooling. The passive safety systems have typically three times less remote valves than active systems and contain no pumps. This type of design is less expensive to build than a conventional PWR because of a significant reduction in the number of pipes, wires, valves, and associated components. In the event of a core melt, the operator can flood the reactor cavity space immediately surrounding the reactor vessel with water to submerge the reactor vessel; this cooling is sufficient to prevent molten core debris in the lower head from melting the steel vessel wall and spilling into the containment. The CDF of the AP1000 is 3.0×10^{-7} events per reactor·year for initiating events occurring during power operation, and 2.1×10^{-7} events per reactor·year while shutdown; in both cases the CDF values include internal events, plus fire and flood.[89] The LRF of the AP1000 for internal events at power is 1.86×10^{-8} events per reactor·year and for internal events during low power and shutdown is 2.05×10^{-8} events per reactor·year.

The AP1000 was certified by the NRC in December 2005.[90] Westinghouse applied for an amendment in May 2007, addressing *inter alia* the effects of the impact of a large commercial aircraft, which was ruled favorably by the NRC in December 2011.[91] Outside the United States, the AP1000 successfully passed all steps of the analysis of compliance with the EUR in June 2006,[70] as well as Step 4 of the generic design assessment in the United Kingdom in December 2011,[92] and Phase 1 of the pre-project design review by the Canadian Nuclear Safety Commission (CNSC) in January 2010.[93] The construction of four AP1000 units in China started in 2009 (Sanmen-1 and 2, Haiyang-1, and 2).[94] The NRC received COL requests for the construction of fourteen AP1000 units in the United States, as of March 2012.[77]

The APR1400 is a two-loop PWR with 4000 MW thermal power and 1390 MW_e net electric output (1450 MW_e gross),[59] developed in South Korea from the optimized power reactor OPR1000, which is based on the System 80+ design of Combustion Engineering (United States). The APR1400 is equipped with a combination of active and passive safety measures,[95] including an in-vessel corium retention system.[96] The CDF for internal initiating events was estimated at 2.3×10^{-6} events per reactor·year, and for external events it is 4.4×10^{-7} events per reactor·year, including fire and flood induced events.[97] The LERF for internal initiating events was estimated at 7.0×10^{-8} events per reactor·year, and for external events it is 1.2×10^{-8} events per reactor·year, including fire and flood induced events.[97] The APR1400 was certified by the Korean Institute of Nuclear Energy in 2003. The construction permit for Shin-Kori-3 and 4, which are the first APR1400 plants in Korea, was issued in April of 2008; commercial exploitation of these units is expected in 2013–2014.[98] A Korea Electric Power Corporation (KEPCO)-led consortium won a tender in early 2010 to build four APR-1400 units in the United Arab Emirates. The first of the four units is scheduled to begin providing electricity to the grid in 2017, with the three later units being completed by 2020.[99]

The advanced pressurized water reactor (APWR) was developed by Mitsubishi (Japan). It is a PWR with 4451 MW thermal output and 1538 MW_e gross electric output.[59] The APWR features several design enhancements including a neutron reflector, improved efficiency, and improved safety systems. It uses a combination of passive and active safety systems.[100] The CDF from at power internal events, fire and flood events is 4.4×10^{-6} per reactor·year and from low power and shutdown events is 2.0×10^{-7} per reactor·year. The LRF from at power internal events, fire and flood events is 6.1×10^{-7} per reactor·year and from low power and shutdown events is 2.0×10^{-7} per reactor·year. The total CDF and LRF are therefore 4.6×10^{-6} per reactor·year and 8.1×10^{-7} per reactor·year, respectively.[101]

The European pressurized reactor or evolutionary pressurized reactor (EPR) is a PWR from Areva. It is an evolution of the French N4 and German Konvoi reactors. It is one of the largest reactors available, with a thermal power of 4300 MW and net electric output of 1600 MW_e (gross 1720 MW_e).[102]

The expected gain in overall fuel efficiency in the EPR compared with its N4 predecessor is 22% (in natural uranium) for an equivalent electricity generation.[103] A reduction of 15% in the quantity of plutonium present in the fuel assemblies at the end of the fuel cycle is also expected.[103] Thus, the EPR will have a more efficient use of natural uranium resources, a better use of irradiated fuel in the reactor, and a significant reduction in the long lived radioactive waste.

The EPR features four independent cooling systems, an extra cooling and containment area in the bottom to catch the molten core if a core meltdown should occur, and a containment building that can withstand a direct crash by a large airplane.[104]

The basic concept of the EPR for corium stabilisation is its spreading into a large lateral compartment, followed by flooding, quenching and cooling

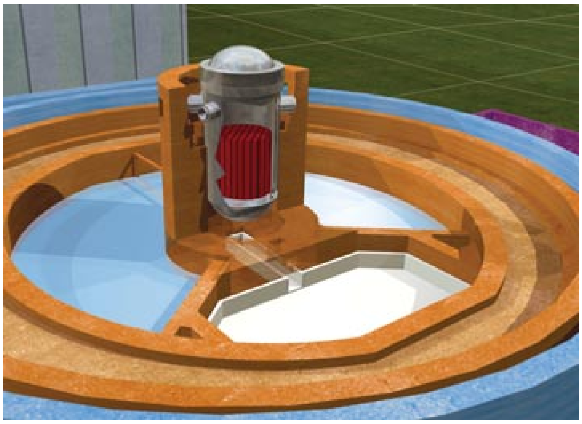

FIGURE 7 | Three-dimensional view of the pressure vessel of the European pressurized reactor (EPR), its corium spreading area, and the nearby in-containment refuelling water storage tank (IRWST). (Reproduced with permission of Teollisuuden Voima Oyj.)

with water from the top, drained passively from an internal reservoir, the in-containment refueling water storage tank (IRWST). Figure 7 shows a three-dimensional view of the RPV of the EPR, its corium spreading area, and the nearby IRWST. Figure 8 shows the main components of the EPR melt stabilization concept, in a simplified design.[104]

In the EPR, the corium is initially retained in the reactor pit and is only discharged into the spreading compartment after most of the core inventory is accumulated. This strategy has the advantage of achieving a spatial separation of the functions to withstand the thermo-mechanical loads during reactor pressure vessel failure and to transfer the melt into a coolable configuration. The spatial separation leads to a simplification of the design of the retention device and preserves the freedom to replace it by an alternative solution if necessary.[105]

The CDF for initiating events occurring during all power states of the EPR is 6.1×10^{-7} events per reactor·year, whereas the LRF is 3.9×10^{-8} events per reactor·year.[106] The EPR successfully passed all steps of the analysis of compliance with the EUR in December 1999, and June 2009 (revision B).[70] The construction license of the first EPR, in Finland (Olkiluoto-3), was granted in February 2005, whereas the construction of the second EPR, in France (Flamanville-3), was authorized in 2007.[107] The EPR passed Step 4 of the 'Generic Design Assessment' in the United Kingdom in December 2011.[108] The application for design certification by the NRC was submitted in late 2007.[109] Besides the two EPR units under construction in Europe, there are also two more under construction in China (Taishan-1 and 2). The NRC received COL requests for the construction of four EPR in the United States, as of March 2012.[77]

The Advanced CANDU Reactor (ACR-1000) with thermal output of 3200 MW and gross electric output of 1165 MW_e was developed by Atomic Energy of Canada Limited (AECL). In contrast with previous CANDU models, the ACR-1000 uses low enriched uranium instead of natural uranium and the coolant is light water instead of heavy water, which is retained only as moderator. The new design simplifies the complex system of cooling pipes running

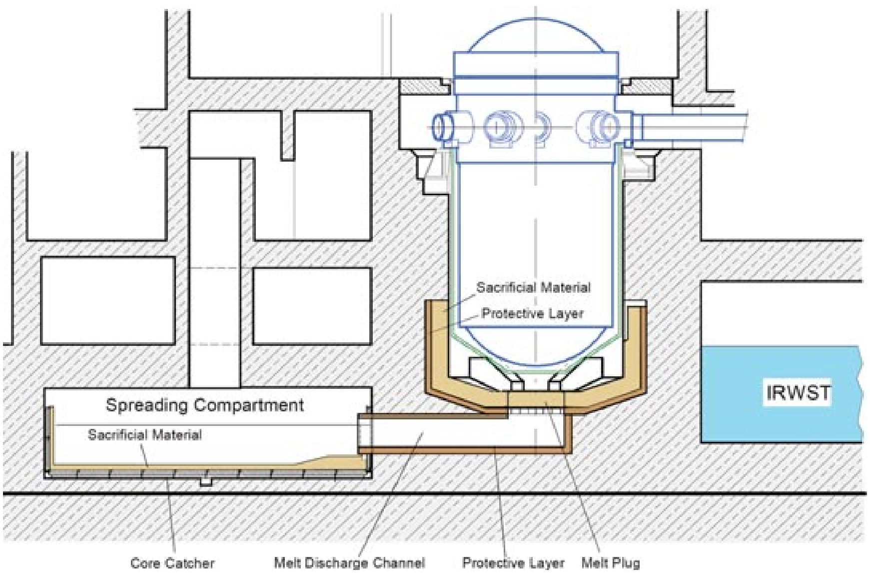

FIGURE 8 | Main components of the European pressurized reactor (EPR) core catcher. (Reproduced with permission from Ref 104. Copyright 2004, Elsevier.)

TABLE 4 | Summary of Safety-Related Parameters of Selected Generation III Power Plants

Reactor[1]	Type[2]	Core Damage Frequency (Events/Reactor·Year)	Large Release Frequency (Events/Reactor·Year)	Mitigation of Severe Accidents
ABWR	BWR	1.6×10^{-7}	2.5×10^{-8}	Core catcher
ESBWR	BWR	6.2×10^{-8}	2×10^{-9}	Core catcher
AES-92	PWR	$<6.1 \times 10^{-7}$	$<1.8 \times 10^{-8}$	Core catcher
AP1000	PWR	5.1×10^{-7}	3.9×10^{-8}	In-vessel retention
APR-1400	PWR	2.7×10^{-6}	8.2×10^{-8}	In-vessel retention
APWR	PWR	4.6×10^{-6}	8.1×10^{-7}	Core catcher
EPR	PWR	6.1×10^{-7}	3.9×10^{-8}	Core catcher
ACR-1000	PHWR	1.8×10^{-7}	8×10^{-9}–8×10^{-8}	In-vessel retention
EC6	PHWR	$<1 \times 10^{-6}$	$<1 \times 10^{-7}$	In-vessel retention

[1] AP1000 is a trademark of Westinghouse Electric Company, LLC; APR-1400 is a trademark of Korea Hydro & Nuclear Power Company; EPR is a trademark of the Areva Group; ACR-1000 and EC6 are registered trademarks of Atomic Energy of Canada Limited.
[2] BWR, boiling water reactor; PWR, pressurized water reactor; PHWR, pressurized heavy water reactor.

through a more compact core[110] and uses new alloys for the piping to guarantee a lifetime of 60 years.[111] The use of enriched uranium is expected to result in operational savings. The ACR-1000 will have a small negative coolant void coefficient,[110] in contrast with earlier designs, which feature a small positive void coefficient,[112] although they include both inherent and engineered systems that compensate for this undesirable characteristic.

In general, the progression of severe core damage accidents in CANDU reactors is slow because the fuel is surrounded by a large quantity of water, which acts as a heat sink to remove the decay heat, and the mechanical deformation mechanism leading to disassembly of the core is creep, which is a slow process.[113] In the unlikely event that the moderator cooling also fails, the fuel channels would sag and collapse as the moderator boils off, but the core debris would still be contained within the calandria vessel as long as it remains cooled on the outside by the reactor vault water.[114]

The target CDF value for the ACR-1000 for at-power internal events and shutdown events, is 1.8×10^{-7} events per reactor·year, whereas the target LRF value is in the range 8×10^{-9}–8×10^{-8} events per reactor·year.[110] The CNSC has completed Phase 2 of the 'Pre-Project Design Review' of the ACR-1000 in 2009 and found no barriers to license the reactor.[110] The ACR-1000 design also successfully completed step 2 of the Generic Design Assessment in the United Kingdom in 2008.[115] The first ACR-1000 is expected to be operational in Canada around 2016.

AECL also developed the Enhanced CANDU 6 (EC6), closer to the CANDU-6 (Generation II) plants built by AECL in China (Qinshan III-1 and III-2). The EC6 is a heavy-water moderated and heavy-water cooled pressure tube reactor, with 2084 MW thermal output and 740 MW_e gross electric output.[116] The

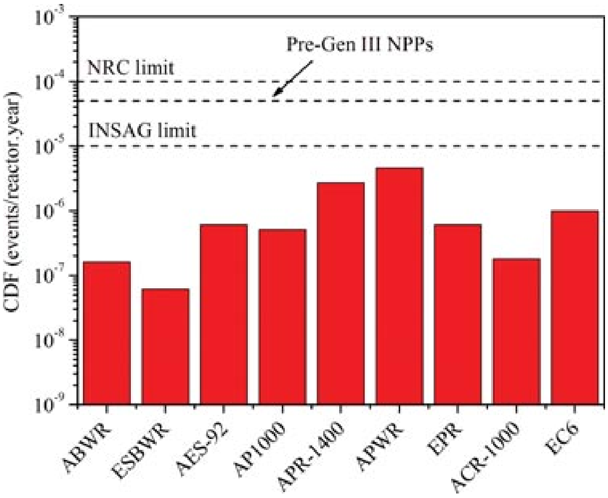

FIGURE 9 | Core damage frequency (CDF) of Gen III nuclear power plants. The NRC limit for the CDF of current plants, the typical CDF value of pre-Gen III plants, and the INSAG limit for the CDF of new reactor designs are also shown for comparison.

target operating life for the EC6 is 60 years (whereas the one of CANDU6 is 30–40 years), with some critical components being replaced around mid-life. The CNSC has completed Phase 2 of the Pre-Project Design Review of the EC-6 in 2012 and found no barriers to license the reactor.[117]

Table 4 presents a summary of the safety-related parameters of the Gen III power plants considered above. Figure 9 shows a plot of the CDF of these Gen III power plants, together with the NRC limit for the CDF of current plants, the typical CDF value of pre-Gen III plants, and the INSAG limit for the CDF of new designs. When compared with pre-Gen III plants, it is clear that Gen III plants present a significant reduction in the probability for severe accidents by a factor of 10–100. This reduction is further magnified when one takes into account that typically only 1 out of 10 accidents will result in a significant off-site release.

CONCLUSIONS

Gen III NPPs started being built in the early 1990s and will dominate the market in the coming decades. The environmental impact of nuclear power is largely determined by the radioactive releases in case of severe accidents. Following the accidents at TMI and Chernobyl, the nuclear power industry developed a broad range of advanced designs with improved safety and more stringent safety objectives and requirements, including a significant reduction of the probability for core melt accidents and consequent releases into the environment. At same time, Gen III NPPs make a better use of natural resources and generate less waste than their predecessors.

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REFERENCES

1. Stacy SM. *Proving the principle: a history of the Idaho national engineering and environmental laboratory 1949–1999*. DOE/ID-10799. Idaho Falls: Idaho Operations Office of the Department of Energy; 2000, 64–73.
2. International Atomic Energy Agency, Power Reactor Information System (PRIS). Available at: <http://www.iaea.org/programmes/a2>. (Accessed November 23, 2012).
3. World Nuclear Association. *Safety of nuclear power reactors*. Information Paper 6. London: WNA; October 2012. Available at: <http://www.world-nuclear.org>. (Accessed November 23, 2012).
4. Marques JG. Safety of nuclear fission reactors: learning from accidents. In: Lehr JH, Krivit SB, Kingery TB, eds. *Nuclear Energy Encyclopedia: Science, Technology and Applications*. Hoboken: John Wiley & Sons; 2011, 127–149. doi:10.1002/9781118043493.ch15.
5. Frisch W, Gros G. Improving the safety of future nuclear fission power plants. *Fusion Eng Des* 2001, 56–57:83–93. doi:10.1016/S0920-3796(01)00238-1.
6. Hore-Lacy I. *Nuclear Energy in the 21st Century*. 2nd ed. London: World Nuclear University Press; 2010, 42–51. ISBN 978-0-9550784-1-5.
7. Marques JG. Evolution of nuclear fission reactors: third generation and beyond. *Energ Convers Manage* 2010, 51:1774–1780. doi:10.1016/j.enconman.2009.12.043.
8. Vujic J, Antic DP, Vukmirovic Z. Environmental impact and cost analysis of coal versus nuclear power: the U.S. case. *Energy* 2012, 45:31–42. doi:10.1016/j.energy.2012.02.011.
9. Cooper JR, Randle K, Sokhi RS. *Radioactice Releases in the Environment*. Chichester: John Wiley & Sons; 2001. ISBN 0-471-89924-0.
10. Papastefanou C. Escaping radioactivity from coal-fired power plants due to coal burning and the associated hazards: a review. *J Environ Radioact* 2010, 101:191–200. doi:10.1016/j.jenvrad.2009.11.006.
11. Lamarsh JR, Baratta AJ. *Introduction to Nuclear Engineering*. 3rd ed. Upper Saddle River: Prentice Hall; 2001, 187. ISBN 0-201-82498-1.
12. Nuclear Energy Agency. *Very High Burn-ups in Light Water Reactors*. Paris: NEA; 2006. ISBN 92-64-02303-8.
13. Rondinella VV, Wiss T. The high burn-up structure in nuclear fuel. *Mater Today* 2010, 13:24–32.
14. US Department of Energy Research Advisory Committee and the Generation IV International Forum. A technology roadmap for generation IV nuclear energy systems. GIF-002–00. DOE and GIF, 2002. Available at: <http://gif.inel.gov/>. (Accessed November 23, 2012).
15. Massachusetts Institute of Technology. *The Future of Nuclear Fuel Cycle*. Cambridge: MIT; 2010. ISBN 978-0-9828008-1-2.
16. Massachusetts Institute of Technology. *The Future of Coal*. Cambridge: MIT; 2007. ISBN 978-0-615-14092-6.
17. McKay AD, Miezitis Y. *Australia's Uranium Resources, Geology and Development of Deposits*. AGSO—*Geoscience Australia, Mineral Resource Report 1*. Canberra: Australian Department of Industry, Science & Resources; 2001. ISBN 0-642-46716-1.

18. Kennedy VS. *Thermal Pollution. Encyclopedia of Energy*. Vol. 6. Burlington: Elsevier Academic; 2004, 79–89. doi:10.1016/B0-12-176480-X/00416-2.
19. Vallero DA. Thermal pollution. In: Letcher TM, Vallero DA, eds. *Waste: A Handbook for Management*. Burlington: Elsevier Academic; 2011, 425–443. doi:10.1016/B978-0-12-381475-3.10028-2.
20. Sovacool BK. Valuing the greenhouse gas emissions from nuclear power: a critical survey. *Energy Policy* 2008, 36:2950–2963. doi:10.1016/j.enpol.2008.04.017.
21. United Nations Scientific Committee on the Effects of Atomic Radiation. *UNSCEAR 2000 Report to the General Assembly, Volume I, Annex B: Exposures From Natural Radiation Sources*. New York: United Nations Publishing; 2000, 140.
22. United Nations Scientific Committee on the Effects of Atomic Radiation. UNSCEAR *2000 Report to the General Assembly*. Vol. I. New York: United Nations Publishing; 2000.
23. Hu QH, Weng JQ, Wang JS. Sources of anthropogenic radionuclides in the environment: a review. *J Environ Radioact* 2010, 101:426–437. doi:10.1016/j.jenvrad.2008.08.004.
24. United Nations Scientific Committee on the Effects of Atomic Radiation. UNSCEAR *2000 Report to the General Assembly, Volume I, Annex C: Exposures to the Public from Man-Made Sources of Radiation*. New York: United Nations Publishing; 2000, 158–180.
25. Neeb KH. *The Radiochemistry of Nuclear Power Plants with Light Water Reactors*. Berlin: Walter de Gruyter & Co.; 1997, 137–144. ISBN 3-11-013242-7.
26. Olander D. Nuclear fuels—present and future. *J Nucl Mater* 2009, 389:1–22. doi:10.1016/j.jnucmat.2009.01.297.
27. Till JE, Grogan HA. *Radiological Risk Assessment and Environmental Analysis*. Oxford: Oxford University Press; 2008, 46. ISBN 978-0-19-512727-0.
28. Farrell AE. Environmental impacts of electricity. In: *Encyclopedia of Energy*. Vol 2. Burlington: Elsevier Academic; 2004, 165–175. doi:10.1016/B0-12-176480-X/00504-0.
29. Vrijheid M, Cardis E, Blettner M, Gilbert E, Hakama M, Hill C, Howe G, Kaldor J, Muirhead CR, Schubauer-Berigan M, et al. The 15-country collaborative study of cancer risk among radiation workers in the nuclear industry: design, epidemiological methods and descriptive results. *Radiat Res* 2007, 167:361–379.
30. Silverman L, Morrison DL, Ritzman RL. Fission product behavior and retention in containment systems. In: Thomson TJ, Beckerley JG, eds. *The Technology of Reactor Safety*. Vol. 2. Cambridge: The MIT Press; 1973, 619–697.
31. Eisenbud M, Gesell T. *Environmental Radioactivity from Natural, Industrial, and Military Sources*. 4th ed. San Diego: Academic Press; 1997, 244–265. ISBN 978-0-12-235154-9.
32. Nuclear Regulatory Commission. *Accident Source Terms for Light-Water Nuclear Power Plants*. NUREG-1465, Washington: NRC; 1995, 10.
33. Alpert DJ, Chanin DI, Ritchie LT. *Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions*. SAND85–2575, NUREG/CR-4467. Albuquerque: Sandia National Laboratories; 1986, 13.
34. Ashbaugh SG, Wagner KC, Longmire P, Gauntt RO, Goldmann AS, Powers DA. *Assessment of Severe Accident Source Terms in Pressurized-Water Reactors with a 40% Mixed-Oxide and 60% Low-Enriched Uranium Core Using* MELCOR *1.8.5*. SAND2008–6665. Albuquerque: Sandia National Laboratories; 2008.
35. Powers DA, Leonard MT, Gauntt RO, Lee RY, Salay M. *Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or* MOX *Fuel*. SAND2011–0128. Albuquerque: Sandia National Laboratories; 2011.
36. International Atomic Energy Agency. INES *The International Nuclear and Radiological Event Scale User's Manual*. Vienna: IAEA; 2008.
37. Talbott EO, Youk AO, McHugh, KP, Shire JD, Zhang A, Murphy BP, Engberg RA. Mortality among the residents of the Three Mile Island accident area: 1979–1992. *Environ Health Persp* 2000, 108:545–552.
38. Hatch MC, Beyea J, Nieves JW, Susser M. Cancer near the Three Mile Island nuclear plant: radiation emissions. *Am J Epidemiol* 1990, 132:397–412.
39. Hatch MC, Wallenstein S, Beyea J, Nieves JW, Susser M. Cancer rates after the Three Mile Island nuclear accident and proximity of residence to the plant. *Am J Public Health* 1991, 81:719–724.
40. Wing S, Richardson D, Armstrong D, Crawford-Brown D. A reevaluation of cancer incidence near the Three Mile Island nuclear plant: the collision of evidence and assumptions. *Environ Health Persp* 1997, 105:52–57.
41. Talbott EO, Youk AO, McHugh-Pemu, KP, Zborowski JV. Long-term follow-up of the residents of the Three Mile Island accident area: 1979–1998. *Environ Health Persp* 2003, 111:341–348.
42. United Nations Scientific Committee on the Effects of Atomic Radiation. *Exposures and Effects of the Chernobyl Accident,* UNSCEAR *2000*. Vol. 2: Effects, Annex J. New York: United Nations Publishing; 2000, 456–457.
43. World Health Organization. *Health Effects of the Chernobyl Accident and Special Health Care*

Programmes. Geneva: WHO; 2006. ISBN 978-92-4-159417-2.

44. Nuclear Energy Agency. *Chernobyl: Assessment of Radiological and Health Impacts*. Paris: NEA; 2002.
45. Rahu M. Health effects of the Chernobyl accident: fears, rumours and the truth. *Eur J Cancer* 2003, 39:295–299. doi:10.1016/S0959-8049(02)00764-5.
46. International Atomic Energy Agency. *Environmental Consequences of the Chernobyl Accident and Their Remediation: Twenty Years of Experience*. Vienna: IAEA; 2006. ISBN 92-0-114705-8.
47. Bebeshko VG, Bazyka DA, Buzunov V. Chernobyl: current situation of non-cancer diseases. *Int Congr Ser* 2007, 1299:54–59.
48. INES rating on the events in Fukushima Dai-ichi Nuclear Power Station by the Tohoku district—off the Pacific ocean earthquake (press release). Tokyo: Nuclear and Industry Safety Agency (NISA); April 12, 2011. Available at: <http://www.nisa.meti.go.jp/english/files/en20110412-4.pdf>. (Accessed June 1, 2012).
49. Chino M, Nakayama H, Nagai H, Terada H, Katata G, Yamazawa H. Preliminary estimation of release amounts of ^{131}I and ^{137}Cs accidentally discharged from the Fukushima Daiichi Nuclear Power Plant into the atmosphere. *J Nucl Sci Technol* 2011, 48:1129–1134. doi:10.1080/18811248.2011.9711799.
50. Boice JD. Radiation epidemiology: a perspective on Fukushima. *J Radiol Prot* 2012, 32:N33–N40. doi:10.1088/0952-4746/32/1/N33.
51. World Health Organization. *Preliminary Dose Estimation from the Nuclear Accident after the 2011 Great East Japan Earthquake and Tsunami*. Geneva: WHO; 2012. ISBN 978-92-4-150366-2.
52. Lawrence Livermore National Laboratory. *Livermore Responds to Crisis in Post-Earthquake*. S&TR January/February 2012. Livermore: LLNL. Available at: <https://str.llnl.gov/JanFeb12/JanFeb12.html>. (Accessed November 23, 2012).
53. Diaz Leon J, Jaffe DA, Kaspar J, Knecht A, Miller ML, Robertson RGH, Schubert AG. Arrival time and magnitude of airborne fission products from the Fukushima, Japan, reactor incident as measured in Seattle, WA, USA. *J Environ Radioact* 2011, 102:1032–1038. doi:10.1016/j.jenvrad.2011.06.005.
54. Carvalho FP, Reis MC, Oliveira JM, Malta M, Silva L. Radioactivity from Fukushima nuclear accident detected in Lisbon, Portugal. *J Environ Radioact* 2012, 114:152–156. doi:10.1016/j.jenvrad.2012.03.005.
55. Haskin FE, Camp AL, Hodge SA. *Perspectives on Reactor Safety*. NUREG/CR-6042, Rev. 1. Washington: NRC; 1997.
56. International Nuclear Safety Advisory Group. *Defence in Depth in Nuclear Safety*. INSAG-10, STI/PUB/1013. Vienna: IAEA; 1999, 4–13. ISBN 92-0-103295-1.
57. International Atomic Energy Agency. *Development of Safety Principles for the Design of Future Nuclear Power Plants*. TECDOC-801. Vienna: IAEA; 1995. ISSN 1011-4289.
58. International Nuclear Safety Advisory Group. *Basic Safety Principles for Nuclear Power Plants*. 75-INSAG-3 Rev. 1, STI/PUB/1082. Vienna: IAEA; 1999, 11–12. ISBN 92-0-102699-4.
59. International Atomic Energy Agency. *Status of Advanced Light Water Reactor Designs 2004*. TECDOC-1391, Vienna: IAEA; 2004. ISBN 92-0-104804-1.
60. Werner WF, Hirano M, Kondo S, Johanson G, Lanore JM, Murphy JA, Schmocker U. Results and insights from level-1 probabilistic safety assessments for nuclear power plants in France, Germany, Sweden, Switzerland and the United States. *Reliab Eng Syst Safety* 1995, 48:165–179. doi:10.1016/0951-8320(95)00016-U.
61. Bengtsson L, Holmberg JE, Rossi J, Knochenhauer M. *Probabilistic Safety Goals for Nuclear Power Plants*; Phases 2–4/Final report. NKS-226. Roskilde: NKS; 2011. ISBN 978-87-7893-296-9.
62. Sehgal BR. Stabilization and termination of severe accidents in LWRs. *Nucl Eng Des* 2006, 236:1941–1952. doi:10.1016/j.nucengdes.2006.03.040.
63. Seiler JM, Tourniaire B, Defoort F, Froment K. Consequences of material effects on in-vessel retention. *Nucl Eng Des* 2007, 237:1752–1758. doi:10.1016/j.nucengdes.2007.03.007
64. World Nuclear Association. *Advanced nuclear power reactors*. Information Paper 8. London: WNA. April 2012. Available at: <http://www.world-nuclear.org>. (Accessed November 23, 2012).
65. Mehta HS, Pappone DC. New generation of BWRs. In: Rao KR, ed. *Companion Guide to the* ASME *Boiler and Pressure Vessel Code*. Vol. 3. 3rd ed. doi:10.1115/1.802717.ch58.
66. Nuclear Regulatory Commission. *Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design*. NUREG-1503, Chapter 19. Washington: NRC; 1993.
67. Nayak AK, Sinha RK. Role of passive systems in advanced reactors. *Prog Nucl Energ* 2007, 49:486–498. doi:10.1016/j.pnucene.2007.07.007.
68. General Electric Hitachi. *Advanced Boiling Water Reactor Fact Sheet*. GEA-14576E. GE Hitachi, 2008. Available at: <http://www.gepower.com>. (Accessed November 23, 2012).
69. Federal Register, 62 FR 25800. May 12, 1997.
70. Berbey P. *Status and Near-Term Works on the* EUR *Document—Possible Use by Third Parties*.

International Conference on Opportunities and Challenges for Water Cooled Reactors in the 21st Century, IAEA-CN-164–3S01, Vienna: IAEA; 27–30 October 2009. Available at: <http://www.europeanutility requirements.org>. (Accessed November 23, 2012).

71. GE-Hitachi Nuclear Energy. ESBWR *Design Control Document Tier 2*. Chapter 1, 26A6642AD, Revision 4, GEH; 2007.
72. Hinds D, Maslak C. Next generation nuclear energy: the ESBWR. *Nucl News* 2006, 49:35–40.
73. International Atomic Energy Agency. *Status of Advanced Light Water Reactor Designs 2004*. TECDOC-1391. Vienna: IAEA; 2004, 207–231. ISBN 92-0-104804-1.
74. Saha P, Gamble RE, Shiralkar BS, Fitch JR. Applicability of small-scale integral test data to the 4500MWt ESBWR loss-of-coolant accidents. *Nucl Eng Des* 2009, 239:956–963. doi:10.1016/j.nucengdes.2008.12.006.
75. Health and Safety Executive. *Conclusions of the Fundamental Safety Overview of the* ESBWR *Nuclear Reactor (Step 2 of the Generic Design Assessment Process)*. HSE-GDA/003 2008/41278. London: HSE; March 2008. Available at: <http://www.hse.gov.uk>. (Accessed November 23, 2012).
76. Health and Safety Executive. *New Reactor Build:* GEH ESBWR *Step 2 PSA Assessment*. London: HSE; 2008. Available at: <http://www.hse.gov.uk>. (Accessed November 23, 2012).
77. Nuclear Regulatory Commission. *Combined License Applications for New Reactors*. Washington: NRC; 2012. Available at: <http://www.nrc.gov/reactors/new-reactors/col.html>. (Accessed November 23, 2012).
78. Ermakov Y, Rousselot O. EUR volume 3 AES 92 subset. European Utility Requirements Seminar 2007, Nice, May 15, 2007. Available at: <http://www.europeanutilityrequirements.org>. (Accessed November 23, 2012).
79. Agrawal SK, Chauhan A, Mishra A. The VVERs at KudanKulam. *Nucl Eng Des* 2006, 236: 812–835. doi:10.1016/j.nucengdes.2005.09.030.
80. Khabensky VB, Granovsky VS, Bechta SV, Gusarov VV. Severe accident management concept of the VVER-1000 and the justification of corium retention in a crucible-type core catcher. *Nucl Eng Technol* 2009, 41:561–574.
81. Dombrovskii LA, Mineev VN, Vlasov AS, Zaichik LI, Zeigarnik YA, Nedorezov AB, Sidorov AS. In-vessel corium catcher of a nuclear reactor. *Nucl Eng Des* 2007, 237:1745–1751. doi:10.1016/j.nucengdes.2007.03.009.
82. World Nuclear Association. *Nuclear power in India*. Information Paper 53. London: WNA; September 2012. Available at: <http://www.world-nuclear.org>. (Accessed November 23, 2012).
83. World Nuclear Association. *Nuclear power in Bulgaria*. Information Paper 87. London: WNA; October 2012. Available at: <http://www.world-nuclear.org>. (Accessed November 23, 2012).
84. International Atomic Energy Agency. *Status Report 93*—VVER–*1000 (V-466*B*)*. Vienna: IAEA; 2011. Available at: <http://aris.iaea.org>. (Accessed November 23, 2012).
85. Dragunov YG, Ryzhov SB, Denisov VP, Mokhov VA. Prospects for development of VVER-type pressurized light-water reactor installations. *Therm Eng* 2007, 54:343–347.
86. Mokhov V, Trunov N. VVER *Reactors: Clean and Reliable Source of Energy in the Past and in the Future*. International Conference on Opportunities and Challenges for Water Cooled Reactors in the 21st Century, IAEA-CN-164–10KS, Vienna: IAEA, October 27–30, 2009.
87. World Nuclear Association. *Nuclear power in Russia*. Information Paper 45. London: WNA; May 2012. Available at: <http://www.world-nuclear.org>. (Accessed November 23, 2012).
88. Schulz TL. Westinghouse AP1000 advanced passive plant. *Nucl Eng Des* 2006, 236:1547–1557. doi:10.1016/j.nucengdes.2006.03.049.
89. Health and Safety Executive. *Step 3 Probabilistic Safety Analysis of the Westinghouse AP1000*. London: HSE; 2009. Available at: <http://www.hse.gov.uk.>. (Accessed November 23, 2012).
90. Federal Register, 71 FR 4464, January 27, 2006.
91. Federal Register, 76 FR 82079, December 30, 2011.
92. Office for Nuclear Regulation. *Summary of the Detailed Design Assessment of the Westinghouse Electric Company* LLC AP*1000 Nuclear Reactor (Step 4 of the Generic Design Assessment process)*. ONR-GDA-SR-11-002 Revision 0. Bootle: ONR; 2011. Available at: <http://www.hse.gov.uk.>. (Accessed November 23, 2012).
93. Westinghouse AP1000 completes phase 1 of Canadian pre-project design review (press release). Cranberry Township: Westinghouse Electric Company. January 2010. Available at: <http://westinghousenuclear.mediaroom.com.>. (Accessed November 23, 2012).
94. World Nuclear Association. *Nuclear power in China*. Information Paper 63. London: WNA; May 2012. Available at: <http://www.world-nuclear.org.>. (Accessed November 23, 2012).
95. Chu IC, Song CH, Cho BH, Park JK. Development of passive flow controlling safety injection tank for APR1400. *Nuclear Engineering and Design* 2008, 238:200–206.
96. Kang KH, Park RJ, Kima SK, Suh KY, Cheung FB, Remped JL. Simulant melt experiments on

performance of the in-vessel core catcher. *Nucl Eng Des* 2007, 237:1803–1813. doi:10.1016/j.nucengdes.2007.07.002.

97. Kim SB, Oh SJ. Severe accident mitigation features of the APR1400. In: Saito T, Yamashita J, Ishiwatari Y, Oka Y, eds. *Advances in Light Water Reactor Technologies*. New York: Springer; 2011. ISBN 978-1-4419-7100-5.
98. World Nuclear Association. *Nuclear power in Korea*. Information Paper 81. London: WNA; November 2012. Available at: <http://www.world-nuclear.org.>. (Accessed November 23, 2012).
99. World Nuclear Association. *Nuclear power in the United Arab Emirates*. Information Paper 132. London: WNA; September 2011. Available at: <http://www.world-nuclear.org.>. (Accessed November 23, 2012).
100. Tujikura Y, Oshibe T, Kijima K, Tabuchi K. Development of passive safety systems for next generation PWR in Japan. *Nucl Eng Des* 2000, 201:61–70. doi:10.1016/S0029-5493(00)00261-2.
101. Mitsubishi Heavy Industries Ltd. *Design Control Document for the* US-APWR. Chapter 19, MUAP-DC019, Revision 3. Tokyo: MHI; 2011.
102. Teollisuuden Voima Oyj. *Nuclear Power Plant Unit Olkiluoto 3*. Helsinki: TVO, 09/2008. Available at: <http://www.tvo.fi.>. (Accessed November 23, 2012).
103. Areva NP and EDF. *Pre-construction safety report, Chapter 11.4, outputs for the operating installation*. UKEPR-0002–113 Issue 04. Paris and Montrouge: Areva NP and EDF; 2011. Available at: <http://www.epr-reactor.co.uk/.>. (Accessed November 23, 2012).
104. Fisher M. The severe accident mitigation concept and the design measures for core melt retention of the European Pressurized Reactor. *Nucl Eng Des* 2004, 230:169–180. doi:10.1016/j.nucengdes.2003.11.034.
105. Bittermann D, Krugmann U, Azarian G. EPR accident scenarios and provisions. *Nucl Eng Des* 2001, 207:49–57. doi:10.1016/S0029-5493(00)00425-8.
106. Health and Safety Executive. *New Reactor Build:* EDF/AREVA EPR *Step* 2 PSA *Assessment*. London: HSE; 2008. Available at: <http://www.hse.gov.uk.>. (Accessed November 23, 2012).
107. Health and Safety Executive. *New Reactor Build* GDA *Step 2 Summary of Overseas Regulatory Assessments*. London: HSE; March 2008 Available at: <http://www.hse.gov.uk.>. (Accessed November 23, 2012).
108. Office for Nuclear Regulation. *Summary of the Detailed Design Assessment of the Electricité de France* SA *and* AREVA NP SAS UK EPR *Nuclear Reactor (Step 4 of the Generic Design Assessment Process)*. ONR-GDA-SR-11-001 Revision 0. Bootle: ONR; December 2011. Available at: <http://www.hse.gov.uk.>. (Accessed November 23, 2012).
109. Nuclear Regulatory Commission. *Design Certification Application Review - US EPR*. Washington: NRC; 2012. Available at: <http://www.nrc.gov/reactors/new-reactors/design-cert/epr.html.>. (Accessed November 23, 2012).
110. Atomic Energy of Canada Ltd. ACR-*1000 Technical Description Summary*. AECL 10820-01372-230-002, Revision 1. Mississauga: AECL; 2010. Available at: <http://canteach.candu.org.>. (Accessed November 23, 2012).
111. Tapping RL. Materials performance in CANDU reactors: the first 30 years and the prognosis for life extension and new designs. *J Nucl Mater* 2008, 383:1–8. doi:10.1016/j.jnucmat.2008.08.030.
112. Snell VG, Howieson JQ. Chernobyl—a Canadian perspective. Mississauga: AECL; 1999. Available at: <http://canteach.candu.org.>. (Accessed November 23, 2012).
113. Mathew PM. Severe core damage accident progression within a CANDU calandria vessel. MASCA Seminar 2004, Aix-en-Provence, June 10–11, 2004.
114. Meneley DA, Blahnik C, Rogers JT, Snell VG, Nijhawan S. *Coolability of Severely Degraded* CANDU *Cores*. AECL-11110. Mississauga: AECL; 1995.
115. Health and Safety Executive. *Conclusions of the Fundamental Safety Overview of the ACR-1000 Nuclear Reactor (Step 2 of the Generic Design Assessment Process)*. HSE-GDA/001 2008/41268. London: HSE; 2008. Available at: <http://www.hse.gov.uk.>. (Accessed November 23, 2012).
116. International Atomic Energy Agency. *Status Report 68—Enhanced* CANDU *6 (EC6)*. Vienna: IAEA; 2011. Available at: <http://aris.iaea.org.>. (Accessed November 23, 2012).
117. Canadian Nuclear Safety Commission. *Phase 2 Executive Summary: Pre-Project Design Review of* CANDU *Energy Enhanced* CANDU *6 Reactor*—EC6. Ottawa: CNSC; 2012.

