[EHNUR WP 4]

ADVANCED NUCLEAR POWER PLANT CONCEPTS AND TIMETABLES FOR THEIR COMMERCIAL DEPLOYMENT

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EXECUTIVE SUMMARY

Most currently operating nuclear power plants are Generation II reactors (except for a few remaining Generation I units and a few Generation III units). Generation III and Generation III+ nuclear power plant concepts are widely recognized to be significant improvements over Generation II reactor designs. Both Generation III designs (standardized designs safer than Generation II) and Generation III+ designs (standardized designs safer than Generation II and with the expectation of greater economy of scale) are available for immediate deployment.

The absolute minimum schedule for a Generation III or III+ nuclear power plant project is 10 years from feasibility study to completion of startup testing. Such a schedule is only achievable by: (a) an experienced utility, (b) with the reactor sited at an existing nuclear power plant site, and (c) with a design for which first-of-a-kind engineering (FOAKE) is complete. Under other circumstances (e.g. a utility new to nuclear generation, a greenfield site, a utility in a country without significant nuclear infrastructure, a nuclear power plant design where FOAKE has not yet been accomplished), the schedule would extend from fifteen to seventeen years and perhaps more.

Within the 2030 time horizon of the ENHUR project, there are a number of advanced reactor designs available for immediate deployment that could be licensed, constructed, and placed in operation in time contribute to electricity generation by the year 2030. These designs are:

- Eight advanced pressurized water reactors (PWRs) – AP1000, APR-1400, APWR, ATMEA1, EPR, VVER-1000 AES-91, VVER-1000 AES-92, and VVER-1200/491. As of June 2013, two units of VVER-1000 AES 91 were in operation, and two units of VVER-1000 AES 92 were nearing operation. The first units of AP1000 and EPR were also nearing operation. Units of the AP1000, APR-1400, EPR, and VVER-1200/491 designs were under construction in June 2013.

- Five boiling water reactors (BWRs) – GE-Hitachi ABWR, ESBWR, Toshiba EU-ABWR, KERENA, and Toshiba US-ABWR. As of June 2013, there were four ABWRs in operation, and two ABWRs were under construction.

- Two pressurized heavy water reactors (PHWRs) – ACR-1000 and CANDU EC-6. There have been no orders as of June 2013 for either of these designs.

- Three small modular reactors – CAREM-25, KLT-40S, and SMART, all PWRs. As of June 2013, there was one unit of CAREM-25 and two reactors (on one barge) of KLT-40S under construction.

- One Generation IV Very High Temperature Reactor (HTR-PM). As of June 2013, there were two HTR-PM modules under construction.

There are also five remaining Generation II reactor designs that were still (as of June 2013) under construction and for which plans exist to construct additional plants of these designs): BN-800 fast breeder reactor, the CNP-300 and CNP-600 PWRs, the CPR-1000 PWR, the PHWR-700, and the OPR-1000 PWR.

It is possible, although not very likely in view of the nominal duration of 17 years and the minimum to maximum range of 13-33 years for a nuclear power plant construction project (from feasibility study to commercial operation), that a few additional reactor designs with near-term deployment possibilities (2015-2020) could be finished in time to contribute to electricity generation by the year 2030. Plants with such designs would have to be ordered by 2015 -2020 in order to be able to be completed and online by 2030 using the absolute minimum schedule constraints (experienced utility, existing nuclear power plant site, standard design with FOAKE complete, and design certification by the regulatory authority).

There are an increasing number of advanced reactor designs that may become available in time to generate electricity after 2030. Generation IV design concepts are still being studied, and except for a few prototype units, Generation IV reactors are not expected to begin operation until 2040 or thereafter.

Nuclear fusion, although promising as a source of electricity, has no chance of producing electricity before 2030, and only a small chance of producing electricity on a commercial scale before 2050.

CONTENT

LIST OF FIGURES

LIST OF TABLES

INTRODUCTION

This chapter of the EHNUR report identifies the advanced nuclear power plant designs that either are available for immediate deployment or that are expected to become available for deployment within the 2030 time frame of EHNUR. This chapter provides input to Work Package 6 and Work Package 10.

This chapter of the EHNUR report answers the following questions[2]:

- What is the current (June 2013) situation with nuclear power plants in operation? (Chapter 1)
- What are the stages in the design of a nuclear power plant? (Chapter 2)
- How long does it take for a nuclear power plant to be constructed and placed in operation? (Section 4)
- What are advanced reactors? (Section 5)
- How do advanced reactors compare with existing nuclear power plants? (Section 6)
- How does the duration of nuclear power plant construction project affect the potential of advanced reactors to be deployed in time to start producing electricity before 2030? (Section 7)
- What advanced reactor designs could be deployed in time in order to impact on electricity generation by the year 2030? (Section 8)
- What advanced reactor designs might become available for deployment in the near term (before 2020) and could begin producing electricity before 2030? (Section 9)
- Based on current (2013) knowledge, what advanced reactor designs might become available to be deployed after 2020, and could begin producing electricity before 2050? (Section 10)
- What are the potential advantages and detriments of small modular reactors? (Section 11)
- What is the deployment horizon for Generation IV advanced reactor designs? (Section 12)
- What are the potential advantages and detriments of Generation IV nuclear power plant technologies? (Section 13)
- What is the deployment horizon for nuclear fusion technology on a commercial scale? (Section 14)
- To which reactor generation do these designs properly belong? (Section 15)
- What advanced reactor designs appear to have been abandoned? (Section 16)

A cautionary discussion on nuclear power plant cost estimates is provided as Section 17. Section 18 presents the conclusions and recommendations of the Chapter. Section 19 provides bibliographic citations to the references cited in this Chapter.

This Chapter of the EHNUR report is supplemented by detailed Fact Sheets on the advanced reactor designs identified in Table 7 as being capable of deployment in time to contribute to electricity generation by 2030, consistent with the EHNUR report time horizon.

[2] This Chapter of the EHNUR report is dedicated to five individuals whose work related to inherently safe reactors and passive safety systems has had much to do with my original and continuing interest in advanced nuclear power plant technology: (1) the late Dr. Alvin M. Weinberg, (2) Dr. Charles W. Forsberg, (3) Robert D. Pollard, (4) Dr. Gordon R. Thompson, and (5) Dr. Michael Golay. The influence of these five experts led to a paper that I presented at the First MIT conference on next generation nuclear power technology in 1990 (Sholly, 1990b). I would also like to dedicate this chapter to the numerous unnamed colleagues and acquaintances who over the years have endured, contributed to, or at least tolerated my all-too-frequent digressions into discussions about advanced reactor designs. I thank you one and all. I would also like thank my wife, who had to put up with my late hours at the computer during the preparation of this Chapter.

METHODOLOGY

The objective of this Chapter of the EHNUR report is to identify prospective advanced nuclear power plant designs that either are available for immediate deployment or that could become available within the time horizon (2030) of EHNUR. Design documentation on advanced nuclear power plant designs and design concept was identified in publicly available literature. Extensive use was made of presentations by reactor vendors at meetings sponsored by the International Atomic Energy Agency (IAEA). A wide variety of published literature and so-called "gray literature" has been used in this chapter.

LIMITATIONS

The timing of availability of the various advanced reactor designs and design concepts is based on current (as of June 2013) literature. It can be expected that dates of design availability for construction will change over time between June 2013 and December 2030. Design features of the advanced reactor designs are also based on current (as of June 2013) literature. It can be expected as well that design features and design parameters may change as the designs are completed, and first-of-a-kind-engineering is completed in anticipation of the start of construction.

1 BACKGROUND

Chapter 1 of the WP4 report provides a snapshot picture of the nuclear power plants operating and those under construction as of June 2013 (Chapter 1.1). The development of the nuclear industry over the past forty years is also briefly discussed (Chapter 1.2). The stages in the design of a nuclear power plant are identified in Chapter 1.3, along with estimates of the duration of these design stages. Finally, Chapter 1.4 answers the important questions of how long it takes to site, design, and construct a nuclear power plant, and then bring it into operation.

1.1 STATUS OF OPERATING UNITS AND UNITS UNDER CONSTRUCTION

In June 2013, the International Atomic Energy Agency (IAEA) Power Reactor Information System (PRIS) identified 434 nuclear power reactors <u>in</u> <u>operation</u> and 69 units under <u>construction</u>. This listing included 50 units in Japan – only two of which were actually operating. The other 48 units in Japan were either in outages at the time of the Fukushima Daiichi accidents in March 2011 and had not been restarted, or were shut down for inspection, refueling, and maintenance in the fifteen months following the accident (i.e., by May 2012) and had likewise not been restarted.

<u>Realistically</u> therefore, IAEA should only show 386 units <u>in</u> <u>operation</u>, with 48 reactors in long-term shutdown[3]. Restart of some of the Japanese units is in question due to damage sustained in the March 2011 earthquake and tsunami (Fukushima Daiichi 5 & 6; Fukushima Daini 1-4; Onagawa 1-3), due to damage sustained when five tonnes of sea water contaminated Hamaoka 5 (Reuters, 2012), and due to the discovery of an active fault under the Tsuruga nuclear power plant in 2012 (Tsuruga 1 & 2). Completion of construction of the Mitsubishi APWR units at Tsuruga as Units 3 & 4 is also in doubt for the same reason.

As of June 2013, nuclear power plants were operating in 31 IAEA Member States plus Taiwan. Nuclear power plants were formerly operated in Italy, Kazakhstan, and Lithuania. According to the IAEA's list of 114 nuclear power plant units planned for construction as known at the end of 2011 and units under construction at the end of 2011, only Vietnam is (officially) planned to join this list (IAEA, 2012c)[4]. In mid-2012, there were 132 nuclear power units operating in the EU (ENSREG, 2013), and

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3 Indeed, the 2013 Canadian Nuclear Factbook issued by the Canadian Nuclear Association acknowledged this reality (CNA, 2013), and the IAEA itself so re-classified the status of the Japanese units on 16 January 2013, and then reversed itself completely two days later, stating that the change in status was the result of a "*clerical error*") (IAEA, 2013c). The conclusion in this Chapter that only 388 units are <u>in</u> <u>operation</u> is consistent with IAEA's own definition of "*Long-Term Shutdown*" (IAEA, 2013b), which states that a reactor is considered to be in long-term shutdown "*if it has been shut down for an extended period (usually several years) without any firm recovery schedule at the beginning but there is the intention of re-starting the unit eventually*". No restart schedules for the Japanese units exist. In mid-2012 TEPCO announced plans to restart the seven units at the Kashiwazki-Kariwa nuclear plant, with the first unit to come back online in April 2013 and the next six to follow suit over a 17 month period. On 1 April 2013, TEPCO announced that this would not take place (WSJ, 2013). In early March 2013, an executive for French nuclear reactor vendor Areva predicted that six units in Japan would restart in 2013, and that two-thirds of the shutdown units would eventually be restarted within several years (Bloomberg, 2013). On 5 March 2013, however, the Nuclear Regulation Authority in Japan said (AFP, 2013), "*Nothing has been decided as it's impossible for us to predict how many reactors can reopen this year before new safety measures are announced. Even if some reactors clear our safety screening, there will be additional procedures ahead, including getting the approval of local residents.*"

4 The membership and observers status of countries in the International Framework for Nuclear Energy Cooperation (IFNEC, formerly GNEP) is perhaps one place to look to identify other countries potentially interested in nuclear power plants. There were 32 members in October 2012, which included fourteen states not currently operating nuclear power plants (Australia, Bahrain, Estonia, Ghana, Italy, Jordan, Kenya, Kuwait, Lithuania, Morocco, Oman, Poland, Senegal, and the United Arab

another 5 units operating in Switzerland (which is surrounded by Member States of the European Union).

According to the Nuclear Energy Institute (NEI), operating nuclear power plants provided 12.3% of electricity generation worldwide in 2011 (NEI, 2013a)[5]. As of the end of 2012, the cumulative operating experience worldwide with commercial nuclear power plants amounted to about 15,000 reactor-years.

Of the 69 units identified by the IAEA as under construction in June 2013, almost half were Generation II reactors instead of advanced reactor designs in Generation III or Generation III+[6]. The remaining 34 units under construction as of March 2013 were Generation III (ABWR, APR-1400) and Generation III+ designs (AP1000, EPR, and various advanced VVER designs), as well one Generation IV high temperature gas-cooled prototype facility[7].

Emirates). Of these fourteen states, Italy and Lithuania formerly had nuclear power plants, and a four-unit plant is being constructed in the United Arab Emirates. Jordan and Poland are actively considering nuclear power plant construction.

There were 31 observer countries in IFNEC in October 2012. Twenty-one of these 31 observer states (Algeria, Bangladesh, Chile, Egypt, Georgia, Greece, Indonesia, Latvia, Malaysia, Moldova, Mongolia, Nigeria, the Philippines, Qatar, Saudi Arabia, Singapore, Tanzania, Tunisia, Turkey, Uganda, and Vietnam) did not have nuclear power plants in operation. Of these 21 states, Turkey and Vietnam have active nuclear power plant construction programmes (IFNEC, 2012).

Of the 32 states with operating nuclear power plants in October 2012, three of these states (India, Iran, and Pakistan) were not represented in IFNEC as either participants or observers. There was also one state (Belarus) without an operating unit, but with two units under construction. Austria is not a participant or an observer state in IFNEC.

[5] Note that the percentage contribution to total electricity generation by nuclear power has declined from a high of 16% in 2005 to the 2011 value of 12.3% (15% in 2007; 14% in 2008-2009; 12.8% in 2010). Data for 2012 was not available at the time this chapter was written, although it is expected to be less than in 2011 because of the additional nuclear power plant shutdowns in Germany (eight units, with a combined capacity of 8336 MWe, were permanently shut down in May 2011) and Japan (in May 2011, only 17 out of 50 reactors in Japan were operating, all but two units were shut down in 2012); two units at the Ohi nuclear power plant in Japan were restarted in July 2012.

[6] The 34 Generation II units under construction as of April 2013 were:

- The Siemens Pre-Konvoi PWR at Angra Unit 3 in Brazil;
- The Siemens heavy water reactor at Atucha Unit 2 in Argentina;
- The Beloyarsk Unit 4 fast breeder reactor (BN-800) in Russia;
- Two CNNC 600 MWe units at Changjiang in the People's Republic of China;
- Two CNNC 300 units at Chasnupp in Pakistan;
- Seventeen CPR-1000 units in the People's Republic of China (Fajgchenggang 1 & 2, Fuqing Units 1-4, Hongyanhe Units 1-4, Ningde Units 2-4, and Yangjiang Units 1-4) (see the discussion in Section 11, herein, for the reasons why EHNUR classifies this design as Generation II);
- The Kursk Unit 5 RBMK in the Russian Federation;
- Four PHWR-700 units in India (Kakrapar Units 3 & 4, and Rajasthan Units 7 & 8);
- The VVER-440/213 units at Mochovce 3 & 4 in Slovakia;
- The VVER-1000/320 units at Rostov 3 & 4 in Russia;
- The prototype fast breed reactor (PFBR) in India (see the discussion in Section 11, herein, for the reasons why EHNUR classifies this design as Generation II);
- The Shin Wolsung Unit 2 PWR in the Republic of Korea (OPR-1000); and
- The Watts Bar Unit 2 four-loop PWR in the United States.

[7] Shidao Bay, in the People's Republic of China, is a two-module HTR-PM pebble bed high temperature gas-cooled reactor, graphite moderated and helium-cooled, with two 250 MWt modules producing 200 MWe with a single turbine that began construction on 9 December 2012 (Dong, 2011).

About 80% of all operating nuclear power plants are either pressurized water reactors (PWRs, about 60%, or boiling water reactors (BWRs, about 20%), and the remaining 81 reactors (20%) are one of five other types (IAEA, 2012c):

- Forty-seven pressurized heavy water reactors, PHWRs (located in Argentina, Canada, the People’s Republic of China, the Republic of Korea, and Romania) (11% of the total);
- Sixteen carbon dioxide cooled, graphite moderated advanced gas-cooled reactors, AGRs (all located in the United Kingdom) (3.7% of the total);
- Fifteen boiling light water cooled, graphite moderated reactors, RBMK (all located in the Russian Federation) (3.4% of the total);
- Two fast breeder reactors (one in India – the 40 MWt Fast Breeder Test Reactor – and one in the People’s Republic of China – the 25 MWe Experimental Fast Reactor) (0.46% of the total); and
- One remaining Generation I carbon dioxide cooled graphite moderated MAGNOX reactor (located in the United Kingdom) (0.23% of the total.

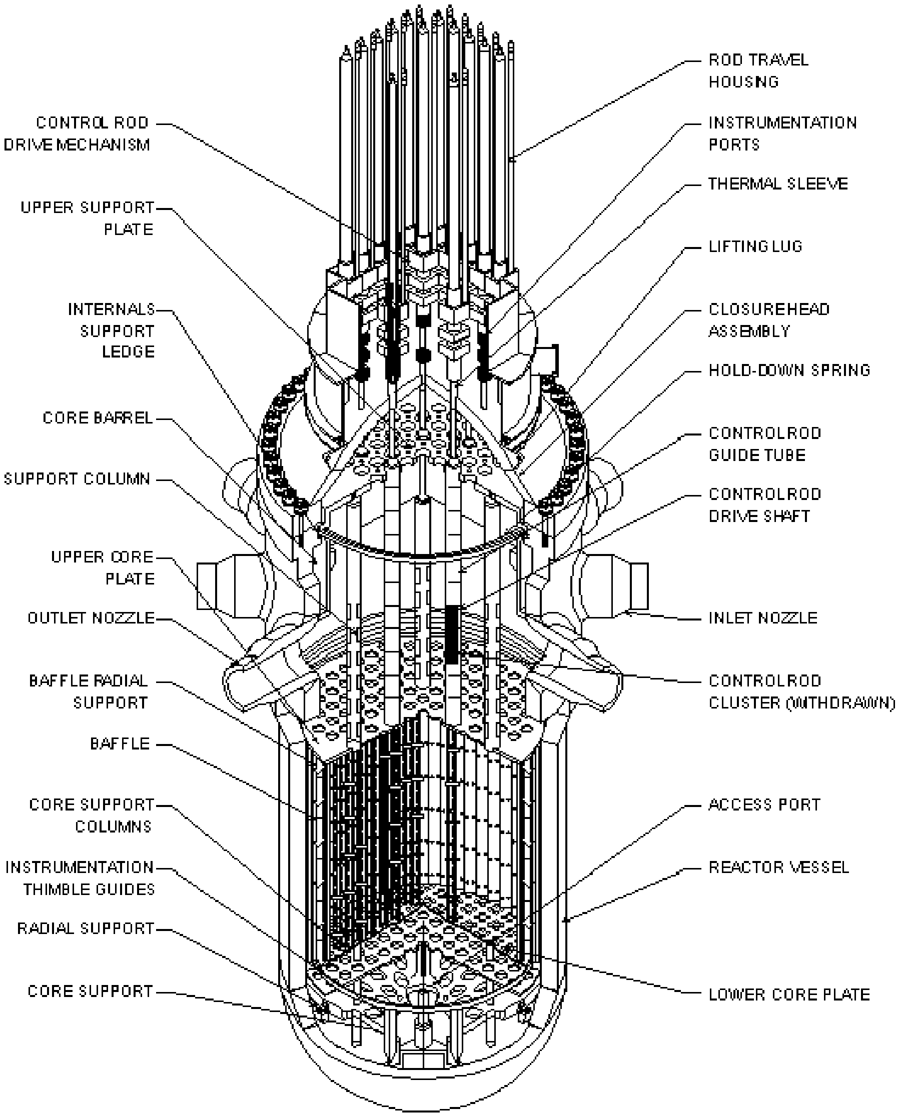


FIGURE 1: PWR REACTOR PRESSURE VESSEL. [SOURCE: WIKIPEDIA & US NRC]

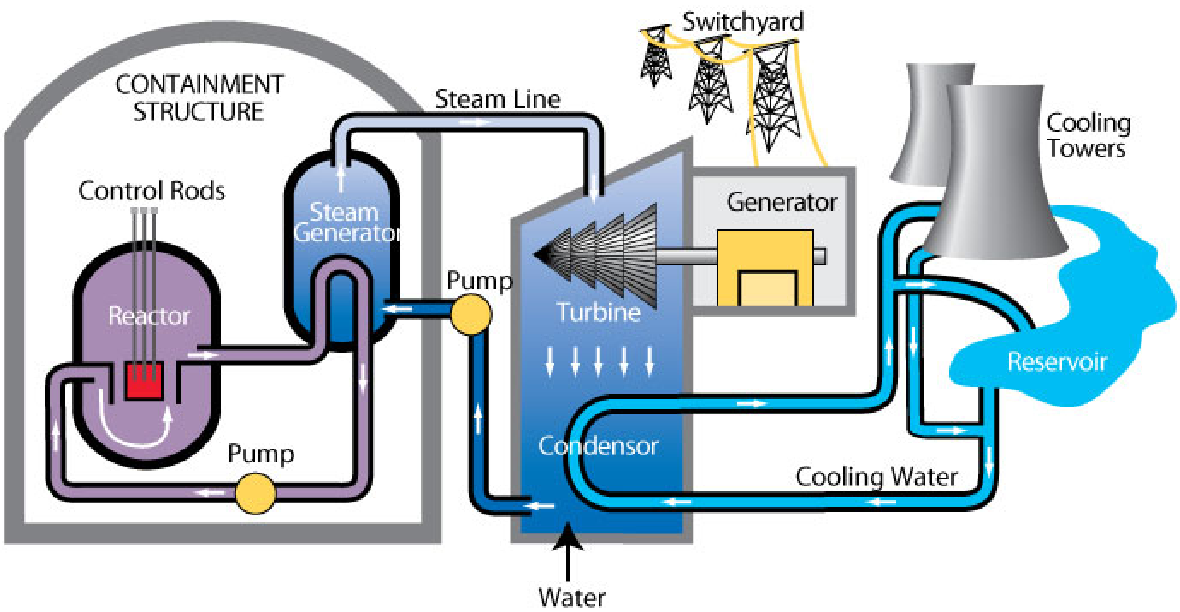


FIGURE 2: PRESSURIZED WATER REACTOR CONCEPT. [SOURCE: WIKIPEDIA & TVA]

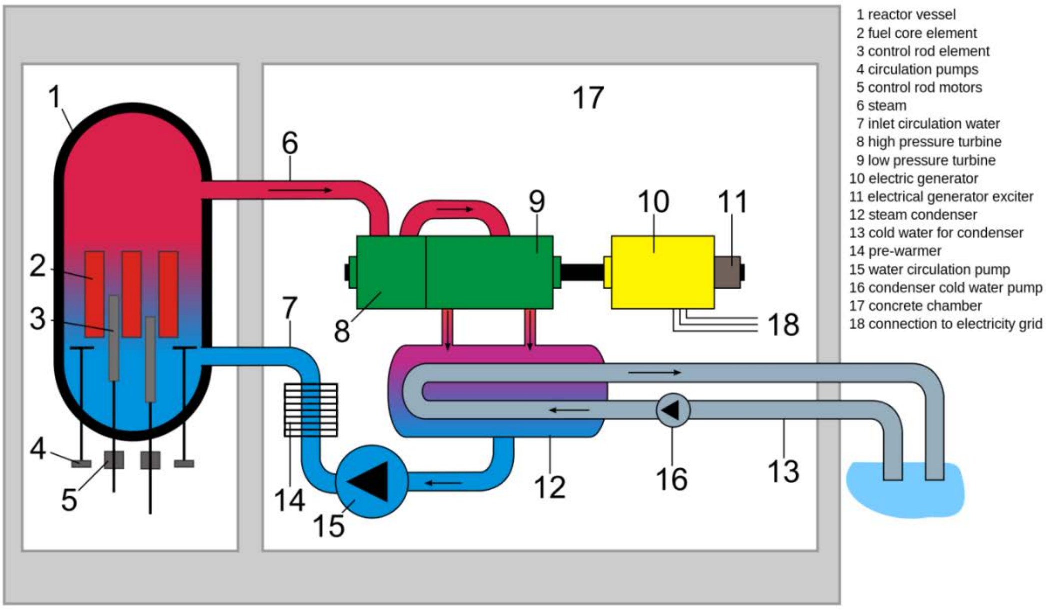
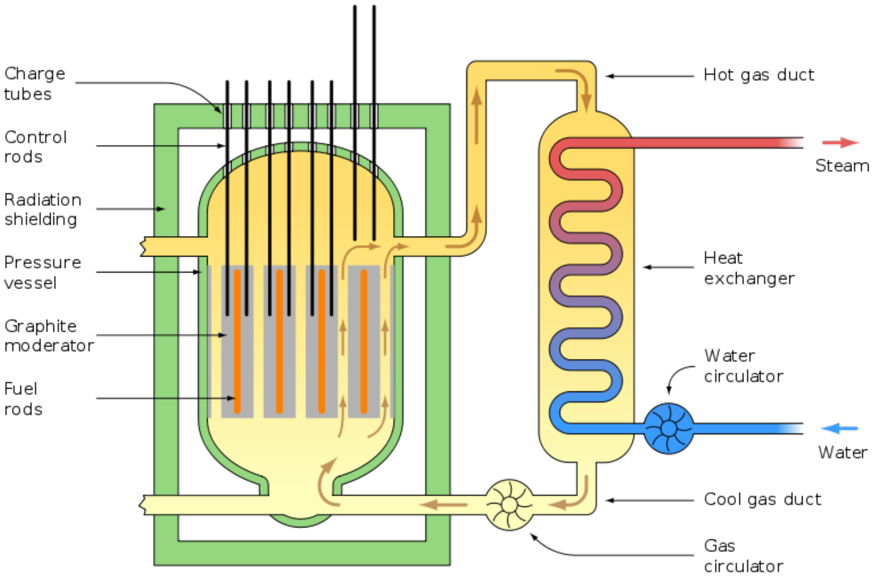


FIGURE 3: BOILING WATER REACTOR CONCEPT. [SOURCE: WIKIPEDIA]



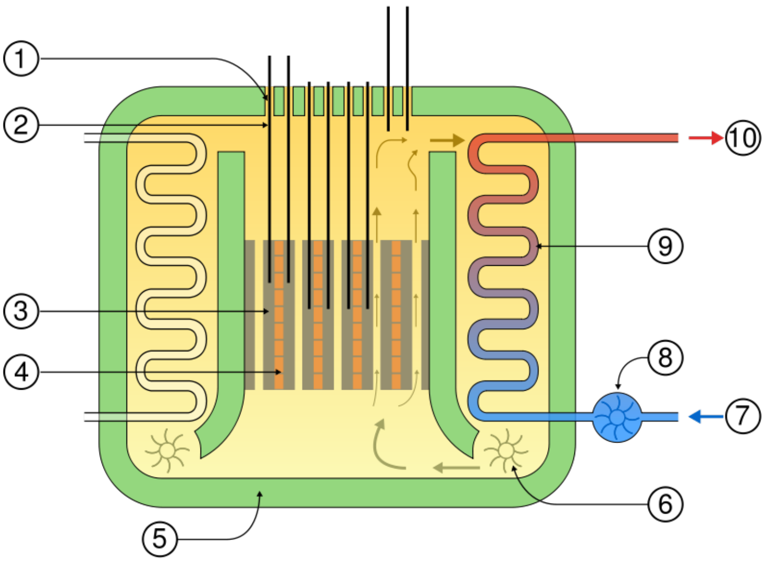


FIGURE 4: GAS-COOLED REACTOR CONCEPTS. (MAGNOX-ABOVE, AGR-BELOW) [SOURCE: WIKIPEDIA]

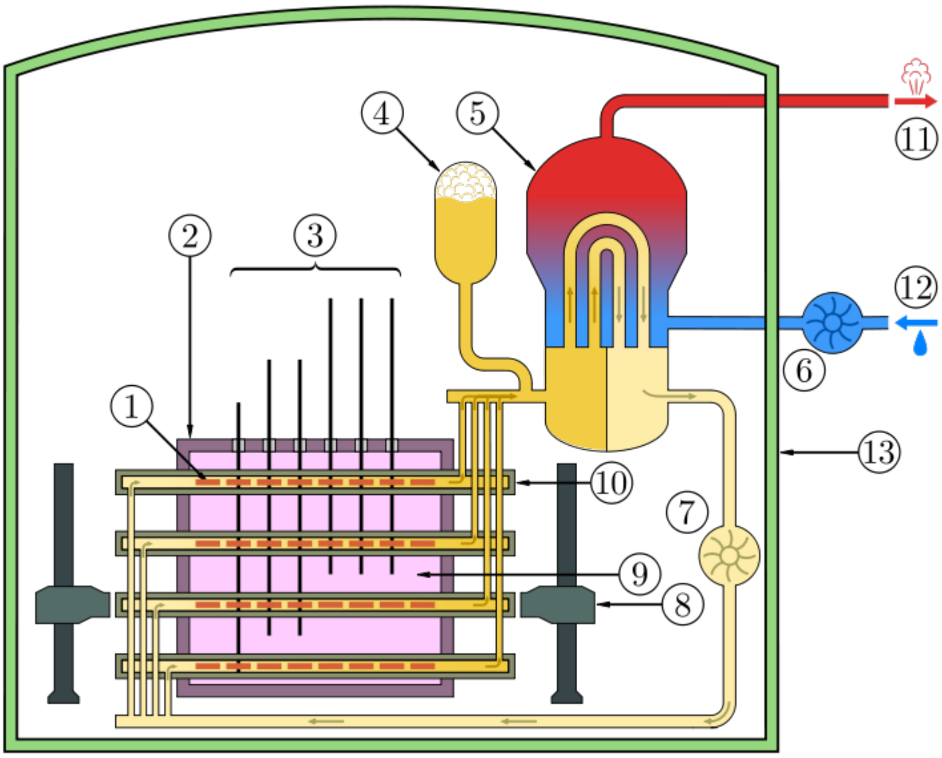


FIGURE 5: PHWR REACTOR CONCEPT. [SOURCE: WIKIPEDIA]

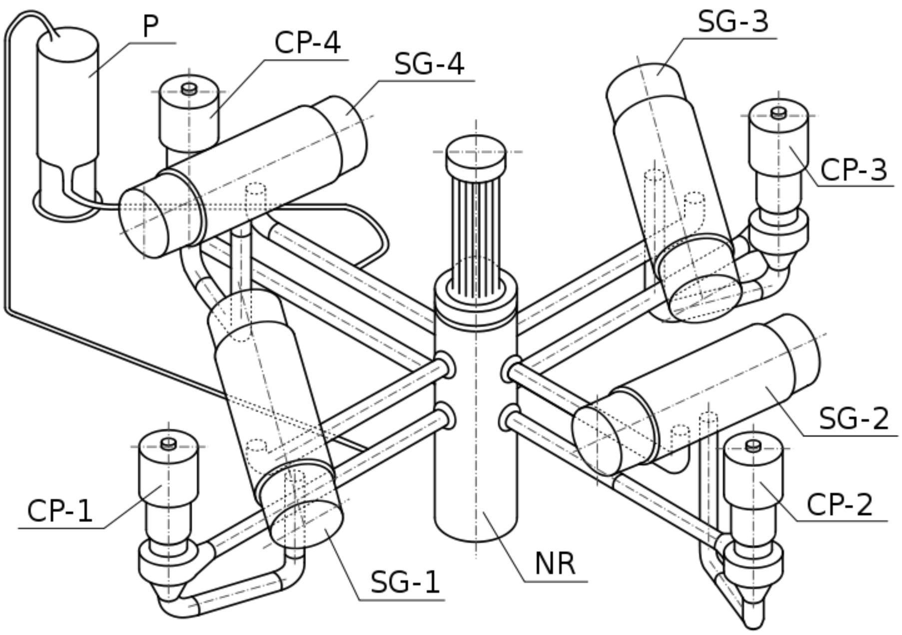


FIGURE 6: VVER REACTOR CONCEPT– VVER PRIMARY SYSTEM. [SOURCE: WIKIPEDIA]

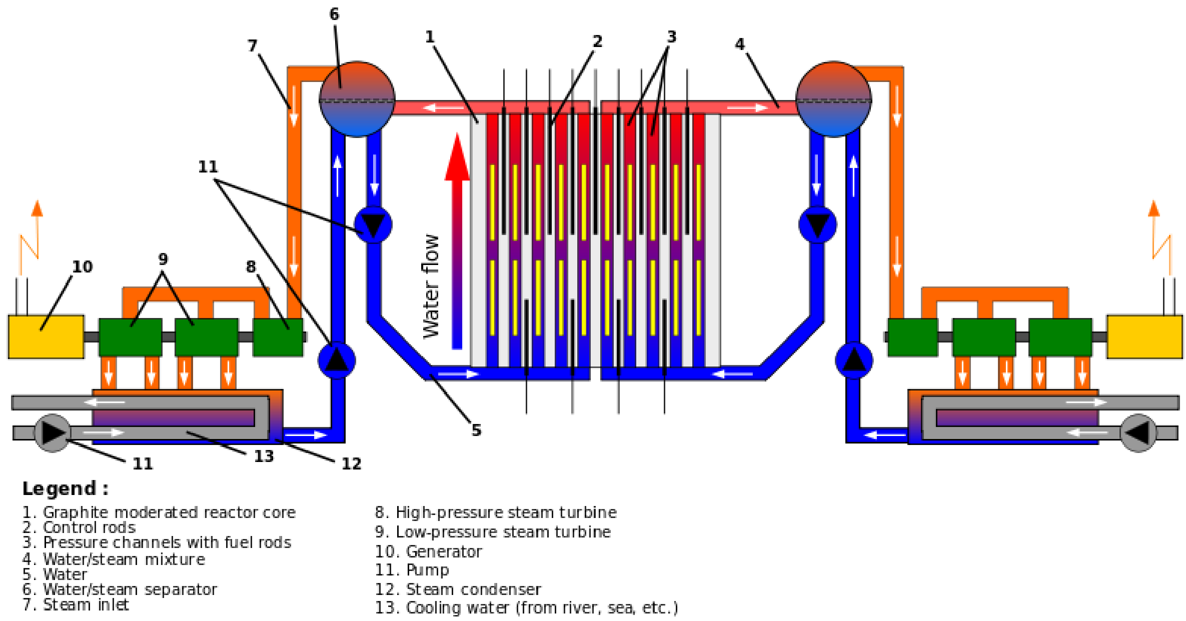


FIGURE 7: RBMK REACTOR CONCEPT. [SOURCE: WIKIPEDIA]

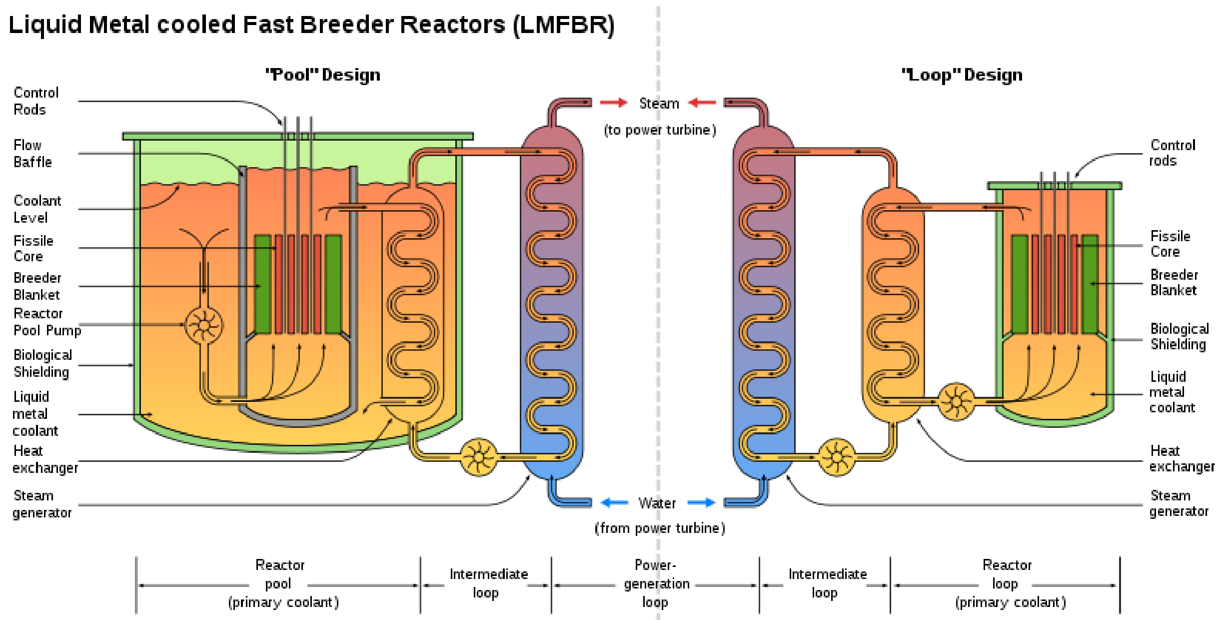


FIGURE 8: FAST REACTOR CONCEPT. POOL & LOOP TYPE FBR. [SOURCE: WIKIPEDIA]

1.2 DEVELOPMENT OF THE NUCLEAR INDUSTRY OVER THE PAST 40 YEARS

There has been major consolidation in the nuclear industry since the 1960s and 1970s, and a number of then familiar names have disappeared. For example, in the 1960s and 1970s there were a dozen or so reactor vendors:

- Atomic Energy of Canada, Ltd. (AECL) (in October 2011, AECL licensed the PHWR technology to Candu Energy, Inc., a wholly owned subsidiary of SNC-Lavallin);
- Allgemeine Elektrizitäts-Gesellschaft (AEG) (In 1969, AEG and Siemens merged their nuclear power plant activities, forming Kraftwerk Union, KWU);
- Allmänna Svenska Elektriska Aktiebolaget (ASEA) (ASEA merged with Brown, Boveri & Cie, BBC, in 1988 to form Asea Brown Boveri, ABB);
- Atomstroyexport (Atomstroyexport was made part of the Russian State corporation Rosatom in 2007);
- Babcock & Wilcox Companies (B&W went bankrupt in 2000; B&W emerged from bankruptcy and two McDermott International, Inc., companies merged to form The Babcock & Wilcox Companies, which in 2010 was spun off as a separate company;
- Brown Boveri & Cie (see above under ASEA);
- Combustion Engineering (CE was acquired by ABB in 1989 to form ABB Combustion Engineering, which subsequently sold its nuclear business to BNFL in 2000);
- Framatome (Areva was formed when Framatome ANP merged with Siemens' nuclear business in 2001; Siemens sold all of its shares of Areva NP in 2009);
- General Electric (GE and Hitachi are operating as joint ventures, GE-Hitachi in the United States, and Hitachi-GE in Japan and worldwide);
- Hitachi (see immediately above);
- Kraftwerk Union (KWU) (see AEG above; KWU was reintegrated into Siemens in 1987);
- Mitsubishi Heavy Industries;
- Siemens (Siemens left the nuclear business after the Fukushima Daiichi nuclear accidents in 2011);
- Toshiba (see immediately below); and
- Westinghouse (British Nuclear Fuels Limited was a nuclear energy and nuclear fuels company formerly owned by the UK government. BNFL purchased Westinghouse Electric Company from CBS in 1995. In 2000, BNFL purchased the nuclear businesses of ABB. In 2006, BNFL sold its nuclear business to Toshiba).

In 2013, are now only seven large reactor vendors operating internationally:

- Areva;
- CANDU Energy, Inc.;
- China National Nuclear Corporation (CNNC);
- General Electric & Hitachi (GE-Hitachi and Hitachi-GE);
- Mitsubishi Heavy Industries (which also has a joint project with AREVA called ATMEA);
- Toshiba & Westinghouse; and
- Rosatom (Atomstroyexport).

Since the 1970s, there has also been consolidation of utilities operating nuclear power plants in the United States and in Europe. For example:

- Electricité de France (EdF) purchased British Energy Group PLC, which is now known as EDF Energy. EdF now operates 73 nuclear units in France and the United Kingdom.
- GDF Suez was formed in 2008 by the merger of Gaz de France and Suez. GDZ Suez then purchased 70% of International Power in 2010 (and the remaining 30% by 2012), creating the

world's largest independent utility company. GDF Suez owns Electrabel and Tractebel, and operates seven nuclear units in Belgium and owns states in the Chooz and Tricastin plants in France.
- Exelon was formed by the merger of PECO Energy Company and Unicom (which owned Commonwealth Edison). Exelon then merged with the Constellation Energy Group (which itself had previously merged with FPL Group, Inc.). Exelon also acquired AmerGen. Exelon now operates seventeen nuclear units in the United States.
- E.ON operates four nuclear units in Germany and three units in Sweden. E.ON is now the largest investor-owned utility in the world.
- RWE operates the Emsland and Gundremmingen B & C nuclear units in Germany.
- Vattenfall was increased by acquisitions, and now operates Forsmark (three BWRs) and Ringhals (one BWR and three PWRs) units in Sweden (), and has a 20% stake in the Brokdorf nuclear power plant in Germany.
- Entergy (formerly Middle South Utilities) has acquired a number of nuclear power plants throughout the United States by purchase (operating eleven units), and also acquired TLG Services, Inc., and currently performs decommissioning services for 90% of US nuclear power plants and all of the Canadian nuclear power plants.
- ENEL acquired Slovenske Elektrarne and Endesa, and is now operating eleven nuclear units in Slovakia and Spain, with two more under construction (Mochovce Units 3 & 4).

There has also been consolidation of architect engineering concerns that were designing nuclear power plants and managing nuclear power plant construction. For example:

- Kellogg, Brown & Root, and C.F. Braun are now all part of KBR, which used to be part of Halliburton but was spun off as a separate company.
- Scientech absorbed NUS Corporation, Nuclear Energy Services (NES), EGS, Anatech, Enertech, and Target Rock, among others, before itself being acquired by Curtis Wright Flow Control Solutions Company in 2007.
- EBASCO (which designed nuclear power plants in Mexico, Taiwan, and the United States) was sold to Raytheon, and merged with Raytheon subsidiary United Engineers & Constructors.
- Gibbs & Hill, which designed nuclear power plants in Brazil, Italy, and the United States, has apparently left the nuclear business.
- CANATOM, which designed Point Lepreau, and AECL's nuclear business have been acquired by SNC-Lavalin.
- Gilbert Associates, which designed nuclear power plants in the United States, acquired Commonwealth Associates (forming Gilbert/Commonwealth, Inc.), and was then sold to Parsons Corporation, which subsequently became WorleyParsons.
- URS has over the years acquired John A. Blume and Associates, the Woodward-Clyde Group, Dames & Moore Group, EG&G Technical Services, and The Washington Group International (which itself had earlier acquired Raytheon which – as noted above – had already absorbed EBASCO and United Engineers & Constructors).
- AMEC was formed in 1982 from the amalgamation of Leonard Fairclough & Son with the William Press Group. AMEC then acquired NNC (earlier National Nuclear Corporation), Geomatrix Consultants, Inc., BCI Engineers and MACTEC (which itself had earlier acquired Harding Lawson Associates and Environmental Science & Engineering). Then AMEC acquired Law Engineering & Environmental Services, and Serco Group plc's nuclear technical services business.
- Stone & Webster was acquired by The Shaw Group, which was itself recently acquired by Chicago Bridge & Iron.

If it appears to be a complicated picture, that is because this <u>is</u> so. The end result, however, is not complicated – there is less competition in the nuclear power plant design and construction business in 2013 compared with the 1960s and 1970s.

It is useful to note, as observed by Rothwell (Rothwell, 2012) that many currently operating nuclear power plants were constructed under rate-of-return regulation. This refers to a utility regulatory structure in which utilities building power plants (nuclear or otherwise) are permitted to start charging ratepayers for the costs of construction before the plant is placed in operation, and in which the utilities are guaranteed a fixed rate of return on their prudent investments. Rate-of-return regulation was common in the era before deregulation, but it is much less common in 2013. Without guaranteed rate-of-return and the ability to charge ratepayers for construction expenditures in progress, nuclear power plant construction poses a greater risk to utilities due to its higher construction cost (compared with fossil-fired electric generation).

In countries where the utility building nuclear power plants is government-owned, the situation will be different since the cost of construction may be government financed and the utility could receive a guaranteed price for its electricity. In such cases, the need to borrow funds to support construction was not an issue in the past.

In many cases, however, future nuclear power plants will be built in <u>deregulated</u> environments[8]. In principle, this <u>should</u> put pressure on reactor vendors to reduce the cost of nuclear power plant construction (Rothwell ,2004), and reactor vendors in the 1990s and early 2000s responded to this with very low <u>claimed</u> construction costs ($1200-$2000 per kWe installed). By 2010, however it was not clear that this is actually occurring in practice. Overnight construction costs advertised by nuclear power plant vendors in the 1990s and early 2000s have been replaced by overnight construction cost estimates in the range of $5000-$6000 per kWe installed in many countries. The exceptions seem to be projects in India, the People's Republic of China, and the Russian Federation where specific market forces are at work that are not widely present. Even in these countries, however, overnight construction costs are often in the range of or exceeding $2000 per kWe installed[9].

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8 The difference between regulated and deregulated markets is well described by researchers at the Jülich Research Center in Germany (Hake, Kupitz & Pesch, 2010): "*In regulated markets, utilities calculate with full costs and have small planning risks since their revenues are well known. In deregulated markets, the market price is determined by the marginal costs of the most expensive power plant, and the profitability of a power plant depends on the spread between its costs and the market price, and therefore on the power plant portfolio of a market.*" In addition, "*Most existing nuclear power plants were built in regulated markets in which the customer base and income of the utility was guaranteed. Deregulated markets are very different. There is an international capital market with strong competition for financing in other technologies. In addition, with nuclear power plants in such a market there are high specific investment costs and a long payback period compared to competing technologies. In addition, the time of completion, the magnitude of construction costs, and the timing of the start of a revenue stream are often underestimated. Nuclear power plants are paying a 3-5% risk premium on capital costs compared with competing technologies.*"

9 The following overnight costs have been identified:

- In 2007, the UK Department of Trade & Industry estimated the overnight cost of an EPR in the United Kingdom at between $1700/kWe and $3200/kWe, with a central value of $2500/kWe (IEA, 2010).
- In 2010, IEA/NEA estimated the overnight costs of nuclear power plants to be commissioned in 2015 to be between $1600/kWe and $5900/kWe, with a central estimate of $4100/kWe (IEA, 2010a).
- The original 2007 overnight cost estimate for the EPR at Flamanville 3 was €2060, but rose in a year to €2500/kWe (IEA, 2010).
- Estimated overnight costs for three advanced nuclear power plant designs in the United Arab Emirates in 2009 were $2900/kWe for APR1400, $2900/kWe for EPR, and $3600/kWe for ESBWR (IEA, 2010).
- IEA/NEA estimated overnight EPR costs in Belgium ($5383/kWe) and France at Flamanville ($3860/kWe). Eurelectric also provided an estimate of $4724/kWe. Estimated overnight costs for ABWR were estimated for Japan ($3009/kWe), and EPRI contributed a generic estimate for ABWR of $2970/kWe. For AP1000 in the People's Republic of China, the overnight cost estimate was $2302/kWe. The overnight cost estimate for a VVER-1150 (we presume this refers to VVER-1200) was $2933/kWe. In the Republic of Korea, overnight cost estimates were provided for the OPR-1000 ($1876/kWe) and the APR-1400 ($1556/kWe). Finally, for the United States, a Generation III+ plant overnight cost estimate of $3382/kWe was provided without specifying the precise design (IEA/NEA, 2010a).

1.3 THE STAGES IN THE DESIGN OF A NUCLEAR POWER PLANT

The IAEA Safety Glossary (IAEA, 2007b) defines design as "*the process and result of developing a concept, detailed plans, supporting calculations and specifications for a facility and its parts*". For the purposes of the EHNUR report, the facility that we are addressing is a nuclear power plant[10]. A nuclear power plant design goes through a series of stages as described in the following paragraphs.

The first stage of nuclear power plant design is a concept description, which is usually a memorandum, a conference paper, a presentation at a conference or seminar, or a brief report in which the basic idea and goals of the design are described. There may be a few calculations and sketches, and usually basic data are provided. Development and testing needs for the design concept are also discussed. There may be rough estimates of cost and schedule considerations also provided (IAEA, 1997b). Two examples of concept descriptions are identified here by way of example (ARC, 2010a; ARC, 2010b; Lee et al., 2013).

The second stage of a nuclear power plant design is the conceptual design. In the conceptual design key components and layout drawings, brief descriptions of key components and systems, and identification and preliminary analysis of relevant incidents and accidents and how they are handled by the design are discussed (IAEA, 1997b). The concept description identifies technologies and systems that could be used to fulfill the safety functions required of the reactor design, and to evaluate the merits and demerits of alternative design features. Conceptual design includes high-level descriptions of system function, and descriptions of how the proposed design achieves functional goals, safety goals, defense-in-depth, and diversity. The conceptual design of the Next Generation Nuclear Plant (NGNP) in the United States (the ANTARES gas-cooled reactor from AREVA) is identified here as an example (AREVA, 2009).

Not all nuclear power plant designs progress directly from the concept description to the conceptual design. Some designs go through an intermediate stage referred to as pre-conceptual design in which the concept description is expanded in order to ascertain whether there is sufficient justification for proceeding to the conceptual design stage. Three examples of pre-conceptual design are identified here by way of example (ANL, 2006; INL, 2007; Mays et al., 2004).

The third stage of a nuclear power plant design is the basic (generic) design (also referred to as the preliminary design)[11]. System descriptions are provided for all plant systems, safety analyses

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- IEA/NEA also estimated the overnight costs of the CPR-1000 Generation II reactor in the People's Republic of China at $1748/kWe to $1763/kWe (IEA/NEA, 2010a).

[10] The IAEA Safety Glossary describes design in the broader context of a facility, which can include nuclear facilities, irradiation installations, uranium mines, radioactive waste management facilities, spent nuclear fuel reprocessing plants, and others. The IAEA's Fundamental Safety Principles document (IAEA, 2006b), which serves as the basis for the development of Safety Requirements and Safety Guides, uses the general terminology facilities and activities as encompassing any human activity that may cause people to be exposed to radiation risks arising from naturally occurring or artificial sources of radioactivity. Readers interested in the IAEA Safety Standards development process can consult a brief IAEA brochure (IAEA, 2009) or a much more detailed description of the process used to develop the standards (IAEA, 2013s). A complete listing of all IAEA Safety Standards applicable to nuclear power plants was available at http://www-ns.iaea.org/standards/documents/default.asp?s=11&l=90&sub=10 as of June 2013.

[11] As an illustration, the time required from the beginning of the conceptual design until the completion of the basic design was four years for the ATMEA1 reactor (Pirson, 2010). It should be noted that this was a case of a joint venture where both parties involved (AREVA and Mitsubishi Heavy Industries) both had a clear idea of where they were going with the design, and a clear financial incentive to complete the design as rapidly as possible. The design of the APR-1400 (which involved creation of a new design based on an uprating of the OPR-1000 plus other design changes) took from December 1992 until February 1999 (six years and three months) to proceed through the conceptual design and basic design stages (Seo 2011).

needed for design approval are completed, and licensing documents for design certification are prepared. In addition, procurement specifications and documentation for major components, systems, and structures, and itemized cost estimate and master scheduled are prepared. The basic design consists of marketing and licensing files, in total at least tens of files (IAEA, 1997b). In the basic design stage, initial technology choices are made for safety and support systems, a basic layout of the plant is set forth. All plant systems and structures are described in more detail. Basic design also includes the first significant effort at defining the overnight costs of the nuclear island (and usually the turbine island as well).

The fourth stage of a nuclear power plant design is the final (generic) design (also sometimes called the detailed design). The final design is completed in detail in order that it could be used as the basis for review by the national nuclear regulatory authority (NRA), or as the basis for a bid prepared by the reactor vendor in response to a request for tender from a utility wishing to build a nuclear power plant. Nearly complete documentation (thousands of files) is prepared for the design, including a complete construction schedule, manufacturing and procurement specifications, and commissioning specifications (IAEA, 1997b). At this stage of the design, some nuclear regulatory authorities will entertain applications from the reactor vendor for some sort of generic design review and approval; this can also take place at the basic design stage in some cases. Such reviews are done, for example, in France, Japan, the Republic of Korea, the United Kingdom (Generic Design Approval, GDA), and the United States (Design Certification, DC).

The fifth stage of nuclear power plant design intended for series construction is first-of-a-kind engineering (FOAKE). FOAKE requires completion of the standard or generic design to the point required to be ready for procurement. FOAKE includes the following items (NEA 2000)[12]:

- Functional studies.
- Elaboration of technical specifications for ordering plant equipment.
- The general layout of the power block.
- Detailed design of civil engineering of standard buildings.
- Detailed design of equipment.
- Detailed design for piping and cabling.
- Creation of testing and commissioning procedures.
- Preparation of operating documents.
- Safety studies.
- Qualification of equipment and facilities.

The sixth and last stage of nuclear power plant design is the site-specific and utility-specific design ready for construction. At this stage, the final design with FOAKE is adapted to site-specific requirements and utility/owner preferences. This is the design of an otherwise standard nuclear power plant design that is constructed at a specific site for a specific utility (which has specific requirements for power plants generally or nuclear power plants specifically that it wants taken into

[12] The amount of work done in FOAKE is considerable, and can cost several hundred million dollars for a design. For example, the expected FOAKE costs for the ESBWR design was a rough estimate of $300 million (year 2000 dollars), spread over five years. FOAKE for the AP600 design were $190 million (and produced 12,000 design documents, a three dimensional model of the entire plant, and a detailed 36-month construction schedule. FOAKE expenditures for the AP1000 design were estimated at $303 million spread over five years (year 2000 dollars). FOAKE expenditures on PBMR were estimated at $200 million. FOAKE costs for GT-MHR were estimated by the vendor at approximately $300 million (DOE, 2001). Engineering costs required in order to bring a Generation III+ plant through design certification and FOAKE were estimated in a University of Chicago study at about $800 million (Rosner & Goldberg, 2011).

account in the plant design, especially insofar as balance-of-plant systems are concerned). The generic or certified design is adapted to site-specific conditions, and a final safety analysis is performed (including deterministic and probabilistic analyses) (IAEA, 1997b).

Site-specific requirements, which are part of owner's costs, include (suggested in part by IAEA, 2007d):

- <u>Balance-of-plant system and structure design</u>

  – Main steam system outside the containment, including the turbine, generator, and condenser.
  – Circulating water system (this provides cooling water to the condenser), including provisions for avoiding flooding of the turbine hall in case of a rupture in the circulating water system or the condenser).
  – Main feedwater system, which provides feedwater to the steam generators (in PWRs) or to the reactor vessel (in BWRs) during power operation.
  – The emergency feedwater system in PWRs (which provides feedwater to the steam generators when the main feedwater system is not available), including the demineralized water storage tanks and the protection of the system and the storage tanks against external hazards.
  – Essential and non-essential service water systems (essential service water provides for cooling of safety-related plant equipment and heat exchangers, such as the diesel generators and the residual heat removal system heat exchangers).
  – The fire protection system, including motor-driven and diesel-driven fire protection water pumps, the fire protection system water storage tank, and the protection of both the system and the tanks against external hazards).
  – The site security and access control systems, including an emergency power source dedicated to these systems.
  – Emergency power AC supply sources (normally diesel generators or gas turbine generators), and provisions for they fuel supply as well as the protection of the emergency generators and their fuel supplies against external hazards. Emergency DC power supply sources, including batteries (and their required discharge capacity) and inverters to convert AC power to DC power to charge the batteries.
  – Radioactive waste collection, processing, and temporary storage systems (pending offsite shipment for disposal).
  – Instrument air systems (both safety-related and non-safety-related).
  – Heating, ventilation, and air conditioning systems, including air intake systems and their protection against airborne radioactivity (in case of an accident), chemical contamination, and smoke from fires.
  – Such normal provisions as drinking water; sanitary (sewage) systems; storm drainage; road, rail, and harbour access as required; as well as spare parts warehousing.

- Onsite switchyard for incoming (offsite power) and outgoing (electricity produced at the plant) electricity, and the connection of the grid to the onsite switchyard.

- The system used to transfer heat from the power plant to the ultimate heat sink during normal operation (e.g., once-through cooling, natural draft cooling towers, mechanical draft cooling towers, or hybrid fan assisted natural draft cooling towers).

- Flooding protection of plant buildings (consideration of probable maximum precipitation and probable maximum flooding, and a margin of safety against uncertainties in these parameters).

- Provisions for onsite dry spent fuel storage (including space considerations and the type of dry storage system), if any.

- Systems related to use of heat for district heating, desalination, or industrial heating, if any.

- Provisions for an access building, a hot machine shop, dressing and dress-out facilities for contamination control, radiation protection including portal monitors, sanitary facilities, cafeteria, etc.

- Provisions for onsite storage of chemicals and gases needed for plant operation (e.g., hydrogen for generator cooling and hydrogen addition to primary coolant, nitrogen for containment inerting for boiling water reactor containments, and chemicals for treatment of service water and wastewater).

- In-plant and external communications systems (including connections to a siren-based emergency alert system in the plant area, if applicable).

An example of how long the design process can last is provided by the EPR construction at the Olkiluoto site in Finland. The project to design the EPR started in 1992. The basic design was completed five years later in 1997. A design optimization phase started in 1998 and lasted until 2000. The detailed design was started in 2000 (Bernstrauch, 2000), the same year that the initial application for construction of Olkiluoto Unit 3 in Finland was made. A turnkey project contract was signed between TVO and AREVA. The design was reviewed and approved for construction by STUK (the Finnish nuclear regulatory authority), and was finally approved by the Finnish cabinet in 2005, thirteen years after the work on the design was started.

1.4 HOW LONG DOES IT TAKE FOR A NUCLEAR POWER PLANT TO BE CONSTRUCTED AND PLACED IN OPERATION?

Assuming 400 GWe (40,000 MWe) of nuclear generation in 2020 (compared with 3 (for illustrative purposes), for a class of nuclear power plants, such as advanced reactors with near term – 2015-2020 – to contribute 10% of electricity generation from nuclear power would require construction and operation by 2030 of ten percent of 400 GWe, or 40 GWe of net capacity. If one considers units in the 1500 MWe class, this would require construction and operation of more than 25 such units. If one considers a small modular reactor with a net capacity of 300 MWe, then achieving 40 GWe of capacity would require construction and operation of more than 130 modular units.

In general, we have concluded that a class of nuclear power technology must have a completed design and be ready for construction not later than 2015 to 2020 in order to have any impact (even a modest impact) on electricity production by 2030 (and even this would require simultaneous construction of 25 or more units). This statement is based on the following considerations.

Nuclear power plant projects project follow a number of stages (IAEA, 2012b)[13]:

[13] With an organization of 2300 people, it is perhaps not surprising that IAEA managers do not all speak with the same voice. One IAEA manager presentation in 2011 (Rogner, 2011) identified the following longer nuclear power plant schedule (compare with IAEA, 2012b):

- Planning, Infrastructure Development, Design & Licensing – 5-20 years.
- Site Preparation – 3-5 years (can run in parallel with Planning, Infrastructure Development, Design & Licensing).
- Construction – 5-10 years.
- Total implied duration – 10-30 years.

- Feasibility study (12 months nominal – range 3-18 months).[14]

  The feasibility study can be eliminated or significantly shortened (to perhaps 3 months) where an experienced utility is proposing to build a standard design at an existing site.

- Detailed site survey, site evaluation and selection (IAEA, 2003b), and environmental impact assessment (24 months nominal – range 7-51 months).[15,16]

  This stage can be shortened where an experienced utility is proposing to build a standard design at an existing site. At an existing site, the utility would not need to perform the site survey and site selection. This would limit the duration of this stage to the preparation of an environmental impact assessment (EIA) using data normally collected as a matter of course as well as information on a standard design that should mostly be available in design and environmental documentation that was submitted in the course of standard design approval. Recent EIA documentation suggests that an EIA could be prepared by an experienced utility in about 7-24 months considering compliance with the Espoo Convention requirements (UNECE, 2009)[17].

- Preparation by the utility of bid specifications (6 months nominal – range 4-12 months).[18]

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14 A feasibility study for a nuclear power plant in Thailand, conducted by an experienced contractor (Burns & Roe) recently took 17 months (October 2008 to May 2010) (Patchimpattapong, 2010). A KEPCO presentation indicated that it required 12-18 months for the feasibility study as well as 6-12 months for a pre-feasibility study (KEPCO, 2012).

15 An IAEA publication in the Nuclear Energy Series has different durations (IAEA, 2012d):

- Site survey, 9-12 months.
- Site selection and assessment, 12-36 months (of which 12 months is required to measure relevant data).
- Total duration of detailed site survey, site selection, and environmental impact assessment (not covered in IAEA 2012d) is 21-48 months.

16 The siting process for the Fennovoima plant at Hanhiviki required 51 months, beginning in mid-2007 and lasting until 5 October 2011 when Fennovoima chose the Hanhiviki site. KAERI training guidance suggests allowing 18 months for site qualification (KAERI NTC). A study prepared for the U.S. Department of Energy, employing the NRC's Early Site Permit (ESP) process suggested allowing 12-15 months to prepare and then submit the ESP application to NRC, and allowing 20-25 months for NRC review and issuance of the ESP (Dominion 2002).

17 The Visaginas EIA procedure in Lithuania required a total of 24 months (Pöyry, 2008). The EIA procedure for the Fennovoima project at Hanhikivi, Finland, required 12.5 months (MEE, 2009b; Fennovoima, 2008; Fennovoima, 2009). Fennovoima submitted an application to the government for a Decision-in-Principle on 14 January 2009 (Fennovoima, 2009); the government made a positive decision on the application 6 May 2010, and the Parliament ratified this decision on 1 July 2010. The EIA procedure for TVO's Olkiluoto Unit 4 required nine months (TVO, 2007; MEE, 2007a). The EIA procedure for the planned Loviisa Unit 3 project lasted only seven months (Fortum, 2007; MEE, 2007b; Fortum, 2008). These experiences suggest a range of 7 to 24 months for EIA preparation and approval.

18 Another publication by the IAEA nuclear energy organization shows phase and durations for the bidding process for a nuclear power plant with schedule breakdowns as follows (IAEA, 2011b):

- Preparation of bid invitation specifications by the project sponsor, 4-12 months.
- Bid preparation by vendors, 6-9 months.
- Bid evaluation by the project sponsor, 6-12 months.
- Contract negotiations, contract closure & signature, 4-6 months.

Compared with the above suggested duration for preparation of bid specifications of 6 months, this reference suggests that the range is 4-12 months. For preparation of bids, evaluation of bids, and successful contract negotiations ending in contract signature, the above reference suggests 15 months, while the reference in this footnote suggests a range from 16-27 months. In total, this reference in this footnote suggests a range of 20-39 months, compared with the 21 months indicated above.

The timing immediately above in this footnote was consistent with the Olkiluoto Unit 3 project (request for tender prepared in 4 months, tender opened for quotations and bids submitted in 6 months, and evaluation of bids and contract signing 9

- Issuance by the utility of a request for tender from vendors followed by receipt and evaluation of the resulting bids, negotiations and contract signing with successful bidder (15 months nominal – range 15-27 months).[19, 20]

- Preparation and submission of licensing documentation (by the utility, the architect-engineer, and the reactor vendor; including safety analysis, environmental impact assessment, and probabilistic safety assessment)[21], review of the licensing application by the nuclear regulatory authority, and issuance by the authority of an authorization allowing construction (30 months nominal – range from 30-70 months).[22,23]

  A recent example is provided by the UK review of the application for a site license for Hinkley Point C and the Generic Design Approval of the EPR design which is planned for the site. The first step of the GDA review of EPR started in July 2007, and was completed in December 2012 – a 67-

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months) (Leverenz, 2004; Patrakka, 2004). A guidance document preparedby the US Agency for International Development for the construction of a new nuclear power plant in Armenia recommended allowing 5 months for bid specification preparation, and 15 months to conduct the tender and complete contract negotiations (US AID, 2008) – a total of 20 months, comparable to the total of 21 months suggested above and on the next page.

[19] ČEZ Group spent 38 months preparing the Temelín Units 3 & 4 Tender Documentation. The Tender was issued on 31 October 2011, and allowed 8 months for reply by pre-qualified vendors (i.e., by 2 July 2012). ČEZ expects to announce the winning bidder and sign the contract in late 2013 (ČEZ Group, 2011). Assuming that this schedule is kept, this would represent a total of 63 months, compared with the combined total of 21 months nominal and a range of 19-41 months cited above.

[20] Other IAEA guidance estimates that the total time needed for the bidding process may require 20-40 months from the start of preparation of bid specifications through evaluation of bids, and to contract negotiations and contract signing (IAEA, 2011b).

[21] This documentation can be quite extensive. The documentation for the construction license application (December 2010) for the first nuclear station in the United Arab Emirates – documentation that amounted to about 9,000 pages (ENEC 2012). The documentation originally submitted by Westinghouse for AP1000 design certification amounted to about 11,000 pages (more were ultimately submitted due to document revisions and responses to questions from the US NRC), and the NRC's safety evaluation report on this documentation amounted to about 2,400 pages (Westinghouse 2007). The submittal by Ontario Power Generation for construction of two new units at the Darlington site amounted to more than 14,000 pages (OPG 2009).

[22] This duration does not account for the longer review times associated with the U.S. Nuclear Regulatory Commission design certification program (NRC design certification review have lasted as short as 46 months and as long as 116 months) (Rothwell, 2010). The 30 month period for preparation and review of licensing documentation is for single or multiple units at a single site. The duration in the text above is considered by the EHNUR project to be very optimistic. The EHNUR project is at a loss to explain how this duration could have been offered in a document written by experts from reactor designers, operating utilities, and consultants to these organizations. The reported duration consists of 12 months for preparation of the Preliminary Safety Analysis Report (PSAR) and 18 months for its review by the nuclear regulatory authority. The EHNUR project believes that both of these durations are optimistic (indeed, the Nuclear Energy Institute, NEI, shows 24 months to develop documentation for design certification and early site approval), and 27-48 months for the NRC to review it (NEI, 2013b). Schedules can sometimes be shorter than projected in the IAEA report; for example, the preliminary safety analysis and probabilistic safety assessment for the Olkiluoto Unit 3 project were reviewed in 12 months by the nuclear regulatory authority STUK before a favorable decision on construction was issued (STUK, 2005). The standard guidance in Canada places the duration at the 30 months suggested by IAEA above (CNSC, 2008).

[23] A very recent World Nuclear Association (WNA) report indicates that preparation of the license application requires 12-48 months for a multi-step licensing procedure, and 12-24 months for a one-step licensing procedure. (The longest application preparation times of 36-48 months were for Germany and Ukraine, both of which lack design certification processes.) For review and approval of the application by the nuclear regulatory authority, WNA reports 12-40 months for a construction permit application and 6-36 months for an operating license application. For the U.S. system (Combined Operating License or COL), WNA reported durations of 60 months for a first application and 24 months for subsequent COL applications for the same design (WNA, 2013d).

month duration (ONR, 2012). During the course of this process, NNB GenCo applied for a nuclear site license for Hinkley Point C in July 2011, and the license was granted in November 2012 – an 18-month duration (but running concurrently with the GDA review) (Gibson, 2013).

- Site preparation activities, including excavation (18 months nominal – range 10-36 months).[24]

- Construction and system/building turnovers (48 months nominal – range 36-54 months (based on advanced reactor vendor ***projections*** for serial units)[25].
  The OECD's Nuclear Energy Agency (NEA) offers different ranges for different regions. NEA cites 48-72 months (4-6 years) for the People's Republic of China and the Republic of Korea; 72-94 months (6-8 years) in Europe (NEA, 2012a).

- Conduct of startup testing, including warranty run (3 months nominal – range 3-15 months).[26]

- Declaration of commercial operation of the unit (official end to the construction phase).

The nominal duration from the above is 207 months (17 years). The range from minimum to maximum is 157-392 months (about 13-33 years)[27]. In fact, no particular project will achieve either all of the minimum durations or all of the maximum durations, so that any particular project – absent extenuating circumstances – should fall between 13 and 33 years at a new site.

If a nuclear power plant construction project is executed at an existing site, the first two steps (feasibility study and site selection) could be significantly shortened; making extensive use of existing environmental data (which should normally be collected as a routine matter at sites with operating nuclear power units), the activities that in the first two steps are projected to require 36 months could be accomplished in 10 months (minimum schedule), shaving 26 months off the schedule

[24] This is consistent with the 18-month site preparation time projection for the Canadian licensing process (CNSC 2008). It is also consistent with the expectation of 18-36 months for site preparation identified by Horizon Nuclear Power for the Oldbury site in the United Kingdom (Horizon, 2009). A project schedule for a new nuclear power plant in Armenia indicates that site preparation will require 10 months (US AID, 2008). In a separate publication, IAEA reported a duration of 20 months for site preparation and excavation for Ulchin Units 5 & 6 in the Republic of Korea (IAEA, 2008a).

[25] The OECD's Nuclear Energy Agency (NEA) has recently indicated that actual construction duration (presumably from the first pour of concrete on safety-related structures to the loading of fuel or start of operation) ranges from four to eight years (48-96 months) (NEA, 2012a). The four ABWRs in Japan were constructed from 1996-2006, and took between 36.9 and 43.2 months (from first concrete to fuel load (Hitachi, 2012). Generation III and III+ construction projects in Finland (EPR), France (EPR), and Taiwan (ABWR) have not gone nearly as well, while AP1000 and EPR projects in the People's Republic of China were reported (as of March 2013) to be on schedule. The above range of 36-54 months represents projected on-schedule construction durations from the reactor vendors, and does not account for schedule slippages of any kind, nor does it account for the typically longer schedules of first-of-a-kind units. In 2010, the IEA estimated construction durations at a minimum of 40 months, a typical duration of 60 months, and a conservative duration of 72 months (IEA 2010). Olkiluoto Unit 3 began construction in mid-2005 and remained under construction in April 2013 with the likelihood of startup in 2016 – eleven years. Construction of Flamanville Unit 3 began in 2007, and was still underway in April 2013 with the likelihood of startup in 2016 – nine years. In contrast, the Taishan EPRs in the People's Republic of China appeared as of April 2013 to be running a much shorter duration. Construction began in 2009, and the first unit was on schedule to start operation at the end of 2013 – 4 years.

[26] Durations from fuel load to commercial operation vary more widely than the IAEA guidance would indicate. For example, GE-Hitachi reported durations of 9-11.2 months for five ABWR projects in Japan between 1996 and 2011 (GE-Hitachi, 2011a). Another IAEA publication shows 10 months from fuel to commercial operation, rather than the 3 months shown above (IAEA, 2008a).

[27] Visagino Atominė Elektrinė (VAE) forecasts 14-16 years for the Visaginas ABWR project (Glevinaskas, 2012). A period of 10-15 years is forecast for the Bangka site in Indonesia (BATAN, 2010). The process for a new nuclear unit in Slovenia is forecast to last 12-15 years (Cimeša et al., 2009).

indicated above. This would place the nominal project duration at 181 months – 15 years. The minimum duration would be unchanged; the maximum duration would be reduced to 323 months (27 years).

The IAEA has frequently cited a range from 10-15 years for nuclear power plant project duration from feasibility study to startup testing. The IAEA has indicated that even an experienced utility in a country with a well-developed regulatory system, and constructing a unit at an existing site would have a difficult time in performing the above activities in less than 10 years (IAEA, 2007a)[28]. Other IAEA guidance cites 12 years as being required (IAEA, 2011e). In any event, these estimates appear to be based on smooth transitions with no problems and no delays.

A real-world example that things can go wrong in an advanced reactor project is provided by the Olkiluoto Unit 3 project. TVO submitted its EIA in August 1999. The application for a Decision-in-Principle was made on 15 November 2000. The Finnish nuclear regulatory authority STUK completed its preliminary safety assessment in February 2001. The Finnish government made a Decision-in-Principle on 17 January 2002. Parliament approved the Decision-in-Principle on 24 May 2002. TVO submitted an application for a construction license on 8 January 2004. The Finnish government issued the construction license on 17 February 2005 (MTI, 2005). Construction was anticipated to require four years, and the request for and granting of an operating license was anticipated to require one year. The real duration until the construction license was issued was August 1999 to January 2004, 53 months. The expected duration for construction and operation was another 60 months (five years) (MTI, 2004). This would have resulted in duration of 113 months from EIA submittal until commercial operation – about nine and half years. If one takes the minimum durations from feasibility study through EIA and licensing document preparation of 59 months (above), the total project duration would be 172 months – a little over 14 years.

However, as of April 2013, Olkiluoto Unit 3 remained under construction (estimated at 75% complete in February 2013), and construction was not expected by the utility to be completed until 2016 (WNN, 2013). This would result in project duration of 197 months, or almost 16.5 years (assuming startup testing in January 2016), from EIA submittal to commercial operation. Again, adding in the minimum durations of activities preceding the submittal of licensing documentation raises the total project duration to 256 months – 21 years (within the range of 13-27 years for a new unit at an existing site, identified above). Bear in mind that the Olkiluoto Unit 3 project involves an experienced utility (TVO, which operates Olkiluoto Units 1 & 2), and one of the three or four largest reactor vendors (AREVA) in terms of the number of units constructed worldwide, and an experienced and well-established regulatory authority as well. This project was begun as a first-of-a-kind EPR construction project (it will in all likelihood not be the first EPR completed, as the Taishan project in the People's Republic of China appears to be set for operation in 2014).

The duration required for nuclear power plants discussed above can be compared with the duration from the start of planning to the completion combustion turbine gas-fired plants (CCGTs) completed in 2-2.5 years, and to the construction duration for fossil fueled plants in the United States – 8-10 years (Walden, 1991).

[28] On the other hand, the Canadian nuclear regulatory system predicts a total duration of 108 months for the total duration from the application for a construction permit to the issuance of an operating license (CNSC, 2008).

2 ADVANCED REACTORS AND ISSUES RELATED THERETO

2.1 WHAT ARE ADVANCED REACTORS?

Advanced reactors are evolutionary nuclear power plant designs with enhanced safety and risk characteristics compared with Generation II reactors. The IAEA's International Nuclear Safety Group (INSAG) has issued guidance that requires advanced reactors to be a factor of ten lower in core damage frequency and large early release frequency (INSAG 1992a).

There are certain trends in reactor design that collectively distinguish Generation III and III+ reactors (Bilbao y Léon 2012):

Cost Reduction

- Standardization and series construction;
- Improved construction methods to shorten the duration of construction;
- Modularization and factory fabrication of modules;
- Design features for longer service lifetime (60 years with the possibility of 80 years with refurbishment);
- Fuel cycle optimization (longer intervals between refueling, less spent fuel per GWh generated);
- Either economy of scale (large units) or affordability (in the total cost sense, for small modular reactors).

Performance Improvement

- Establishment of user design requirements (EPRI URD & EUR)[29];
- Development of highly reliable components and systems, including so-called "smart" components;
- Improving the technology base for reducing over-design;
- Further development of PSA methods and databases (and iterative use of PSA as part of the design process);
- Development of digital instrumentation and control (I&C);
- Development of computer based techniques;
- Development of systems with higher thermal efficiency and expanded applications as part of the basic design (non-electrical applications, such as desalination of sea water, delivery of industrial heat or steam, district heating, and hydrogen production).

Commercially deployed nuclear fission power plants and advanced nuclear power plant design concepts are often described as fitting into one of four generations of designs (see Figure 9).

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[29] The EPRI Utility Requirements Document (URD) was developed by American utilities to provide a complete statement of utility requirements for advanced light water-cooled reactors (LWRs) to be built in the United States. The European Utility Requirements (EUR) plans a similar role for advanced LWRs to be built in Europe. Both the URD and EUR can be used by utilities as part of the basis for requesting design proposals from reactor vendors.

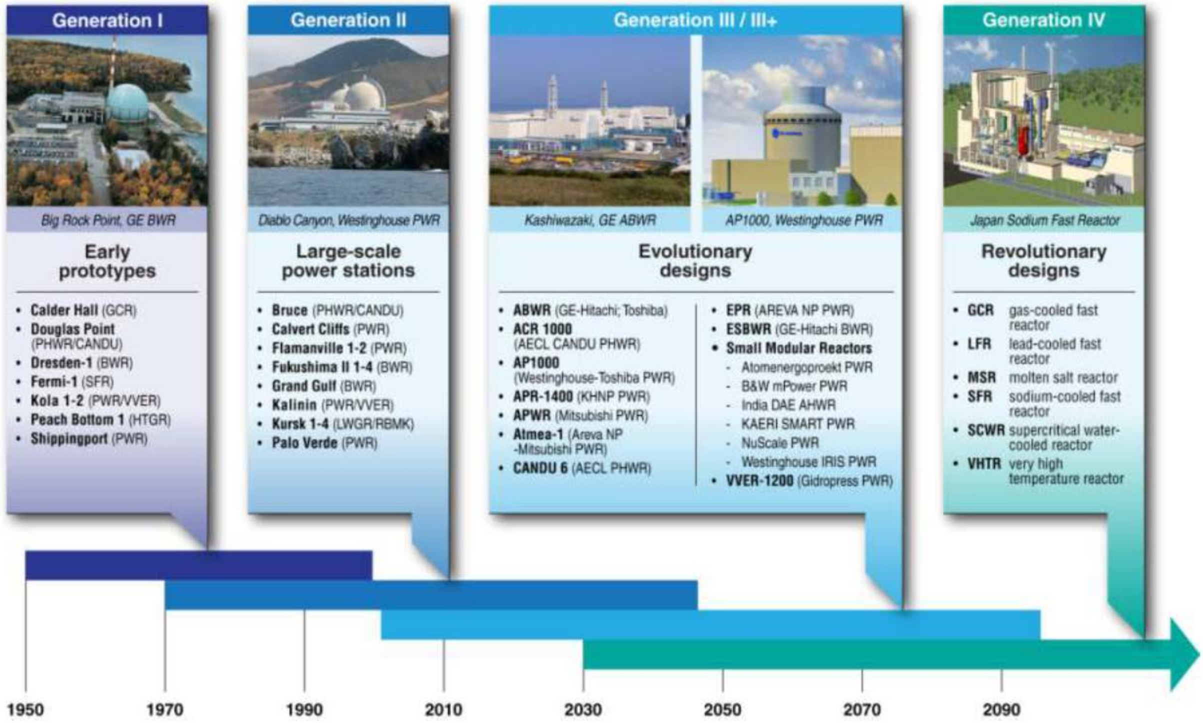

FIGURE 9: ILLUSTRATION OF NUCLEAR POWER - PLANT DESIGN GENERATIONS. [SOURCE: BILBAO Y LÉON 2012]

By far most operating nuclear power plants are Generation II (existing designs). In order to continue with the discussion of advanced reactor designs, it is important to identify the five categories of nuclear plants:

Generation II Designs

Generation II reactor designs originated in the 1970s-1990s, following the initial operation of Generation I prototypes. Typical Generation II reactors included pressurized water reactors from Framatome (now AREVA), Siemens, Combustion Engineering (subsequently absorbed into Westinghouse), Westinghouse, Babcock & Wilcox, and VVERs by OKBM and Atomstroyexport, boiling water reactors from General Electric and Asea Atom, and Advanced Gas-Cooled Reactors (AGRs) in the U.K. All but 11 of the currently operable 437 nuclear power reactors are Generation II designs, with seven Generation III reactors in operation[30] and four remaining Generation I plants continuing in operation[31].

30 The seven Generation III reactors in operation as of April 2013 were four ABWRs in Japan and three advanced VVER units in India and the People's Republic of China.

31 The four remaining Generation I units are Kola Units 1 & 2 in Russia and Metzamor Unit 2 in Armenia, all of which are VVER-440/230 units (Metzamor has upgraded seismic design, and is formally designated VVER-440/270); and the last of the MAGNOX units (Wylfa Unit 1) which is due to be shut down in 2014 following transfer of partially used fuel from the adjacent Unit 2 reactor, which was shut down at the end of 2012. MAGNOX fuel is no longer being manufactured. Kola Units 1 and 2 are licensed to operate until 2018 and 2019. Metzamor Unit 2 is expected to be operated until a new power reactor can be built on the site, the current estimate for which is the start of operation in 2020.

Note that the designation of the three VVER units as Generation I was made by OKB Gidropress, the original plant designer (Mokhov 2010) and not by the EHNUR project (although we agree with the designation). Generation I units formerly operating in Bulgaria and the Slovak Republic (six VVER-440/230 units at Bohunice and Kozloduy) were shut down as part of the accession of Bulgaria and Slovakia to the European Union.

Generation III Designs

Generation III reactor designs incorporate improved fuel technology, improved thermal efficiency, passive safety systems, standardized design, nominal 60-year plant lifetimes, improved capabilities to manage severe accidents, and reduced core damage frequencies (CDF) and large release frequencies (LRF), as calculated by probabilistic safety assessment (see Section 5.3, below), compared with Generation II designs[32].

Generation III+ Designs

Generation III+ reactor designs promise further improvements in safety and reductions in risk compared even with Generation III designs, as well as improved economics (normally involving a larger power output taking advantage of perceived economies of scale).

Generation IV Designs

Generation IV reactor designs are being researched to provide significant improvements in sustainability, safety, reliability, economics, proliferation resistance, and physical protection compared with Generation II, Generation III, and Generation III+ technologies.

Small Modular Reactors (SMRs)

Small modular reactors are advanced designs generally less than 350 MWe net (some as low as 10 MWe net) intended to be deployed in smaller increments than the large Generation III and Generation III+ designs, and intended to become available before the Generation IV designs. Although the IAEA officially considers anything less than 300 MWe to represent a small reactor, as a practical matter anything less than 500 MWe is considered to be a small reactor. (Generation III and Generation III+ designs provide generating capacities in the 700-1700 MWe range.)

2.2 TO WHICH GENERATION OF REACTORS DO THE FOLLOWING DESIGNS CORRECTLY BELONG?

During the course of the EHNUR project, questions arose about the correct classification of nuclear power plant generations for certain of the reactor designs considered in the project. These questions are addressed below.

ATMEA 1 PWR (France & Japan Joint Adventure)

France's ambassador to Jordan, AREVA, and Mitsubishi all refer to ATMEA1 as Generation III+, but it is decidedly not so, as a quick comparison with Generation III+ PWRs clearly shows why:

- Generation III+ PWRs – double containment; ATMEA1 – single containment;

32 Note regarding passive safety systems – Generation II nuclear power plants use mostly "*active*" safety systems. Such systems are active in the sense that they require electrical or mechanical operation on command, and typically require both AC power (for motor and valve operation) and DC power (for control and actuation, and for valve operation). According to WNA (WNA 2013f), "*Inherent or passive safety depends only on physical phenomena such as convection, gravity or resistance to high temperatures, not on functioning of engineered components, but these terms are not properly used to characterize whole reactors.*" Both active and passive systems require as a minimum parallel redundant trains of equipment. Safety systems that have both active trains and passive trains performing the same safety function are referred to as hybrid safety systems.

- Generation III+ PWRs – core damage frequency 3×10^{-7}/a or less; ATMEA1 core damage frequency – 10 times less than existing PWRs (average core damage frequency of existing conventional PWRs is 5×10^{-5}/a, so this would make ATMEA1's core damage frequency about 5×10^{-6}/a (a factor of 10 or more higher than Generation III+ PWRs);

The only similarity of ATMEA1 to Generation III+ PWRs is the presence of a core catcher (but this is also supplied in other Generation III PWRs, such as the VVER-1000 AES 91 and AES 92 designs – which by the way have double containments, unlike ATMEA1 which has a single containment. Although ATMEA1 has passive autocatalytic recombiners (PARs), so do many Generation II PWRs, especially in Europe. Although AREVA and Mitsubishi state that the ATMEA1 safety systems with passive features, in reality the only passive feature in the design is the advanced accumulators. Generation II PWRs also have accumulators. ATMEA1 has digital control systems. So do the other Generation III PWRs, and in fact many Generation II PWRs have replaced their original control systems with digital technology. AREVA and Mitsubishi claim higher thermal efficiency of 36% and then contrast it with 33% for existing PWRs. This true, but Generation III PWRs are also more thermally efficient than existing PWRs.

Compared with Generation III+ PWRs, ATMEA1 is either the worst of its class as a GeneratIon III+ design, or it is really Generation III. We find that ATMEA1 shares far more in common with Generation III PWRs like APR1400, Mitsubishi APWR, or the VVER AES 91 and AES 92 units. Consequently, we conclude that ATMEA1 is Generation III technology, not Generation III+.

BN-800 FAST BREEDER REACTOR (Russian Federation)

The BN-800 is a MOX-fueled fast breeder reactor under construction at the Beloyarsk site in Russia, designed by OKBM Afrikantov. (In 2009, Russia sold two BN-800 reactors to the People's Republic of China. Construction of the first of the two BN-800 units is scheduled for 2013 at Sanming, and the two units are scheduled to go into service in 2019 and 2020. A second phase of two additional BN-800 units is foreseen to start construction at the same site in 2015.) BN-800 is a pool type reactor in which the reactor, coolant pumps, and intermediate heat exchangers are all located within the reactor vessel in a common pool. The reactor is a sodium-cooled fast reactor.

The design of the BN-800 was started in 1983, and was revised in 1987 after the Chornobyl Unit 4 accident and again in 1993 based on new safety guidelines. In 1993, the power level was also increased by 10% to 2100 MWt/880 MWe net. The reactor is planned to be finished in 2014 (WNA, 2013c).

The BN-800 has a 40 year design service life (Generation III & III+ units have a 60-year design service life). The estimated CDF for BN-800 is 7×10^{-6}/a; Generation III, III+, and IV reactors have calculated or designed CDFs of 1×10^{-6}/a or less. The design basis earthquake ground acceleration for the BN-800 is only 0.1g; Generation III & III+ reactors are nearly all designed for 0.3g (and those that aren't are designed for 0.25g). The external explosion shock front pressure for 1 second is 10 kPa for BN-800; for the VVER-1200, the shock front pressure is 30 kPa for 1 second. The BN-800 is Generation II technology.

PROTOTYPE FAST BREEDER REACTOR (PFBR) (India)

The Prototype Fast Breeder Reactor in India is a 1253 MWt/470 MWe net sodium cooled MOX fueled fast reactor. The reactor is designed for a 40-year service lifetime; Generation III & III+ reactors are designed for service lives of 60 years. The PFBR includes a very limited core catcher (designed for the debris from seven fuel assemblies due total blockage of one assembly, in which this assembly and the six surrounding assemblies are assumed to melt; the core consists of 181 fuel assemblies. The design incorporates both active and passive decay heat removal systems. The PFBR is designed for a seismic acceleration of 0.2g; Generation III & III+ reactors are nearly all designed for 0.3g (and those that aren't are designed for 0.25g). On balance, it is considered that the PFBR is Generation II technology.

CPR-1000 PWR (People's Republic of China)

The CPR-1000 reactor has been designed by China Guangdong Nuclear Power Company (CGNPC), and is so far only available in the People's Republic of China (PRC). CGNPC claims a special Generation II+ status for the CPR-1000. This claim has been investigated in EHNUR and found to be a misnomer. The principal advantage of the CPR-1000 design seems to be cost (compared with Generation III and III+ PWRs), and secondarily that it is a design that is increasingly supplied by indigenous Chinese companies.

The CPR-1000 design is based on Units 5 & 6 of the Gravelines nuclear power station in northern France (Lau, 2011). These units at Gravelines are from the CPY/CP1 class of AREVA 900 MWe PWRs (ASN, 2010). Construction started on Gravelines Units 5 & 6 in October 1979. Gravelines Unit 5 was first connected to the grid on 28 August 1984; Gravelines Unit 6 was first connected to the grid on 1 August 1985[33]. The CPR-1000 has a design capacity of 2895 MWt and a net electrical capacity of 1021 MWe (with house loads of 65 MWe) (Subki, 2012a). This is designed to produce a net efficiency of 35.1%, which is better than most Generation II PWRs.

The CPR-1000 design has an intended 40-year design life (plus a 20-year service life extension), which is the same as existing Generation II PWRs The refueling outage interval is planned at 18 months, also similar to many currently operating Generation II PWRs.

The CPR-1000 design has the refueling water storage tank located outside the containment, like Generation II PWRs. Generation III and III+ PWRs have the refueling water storage tank inside the containment, where it is protected from external hazards.

The CPR-1000 has a seismic design basis of 0.2g PGA horizontal ground acceleration. Nearly all Generation III and Generation III+ designs (with the exception of the VVER-1200) have a seismic design basis of 0.3g PGA horizontal ground acceleration (VVER-1200 has a seismic design basis of 0.25g PGA). Indeed, a 0.3g PGA seismic design is required for certification to the EPRI Utility Requirements Document (URD); for the EUR, it is 0.25g PGA.

The CPR-1000 has an expected lifetime capability factor of 87%, which is similar to many Generation II PWRs, but less than for Generation III and III+ designs. All Generation III and Generation III+ designs have expected lifetime capability factors of 90-95%.

The CPR-1000 containment is a single pre-stressed concrete design with a steel liner. The containment shell thickness is 0.9 meters. The containment design leak rate is 0.3 volume percent per day at the design pressure of 0.52 MPa. The free volume of the CPR-1000 containment is 49,400 m^3 (17 m^3/MWt, similar in this respect to the AREVA EPR and the Westinghouse AP1000, but less than the ATMEA1 at 23.8 m^3/MWt).

Except for being equipped with a core catcher, and having somewhat better net thermal efficiency, there is little to distinguish the CPR-1000 design from Generation II reactors, and we conclude that the CPR-1000 is a Generation II reactor design.

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33 To CGNPC's credit, they got the CPR-1000 units turbine halls correctly oriented (i.e., perpendicular orientation; Gravelines has the turbine halls parallel to the reactor buildings, exposing the reactor buildings to turbine missile hits in the event of a turbine failure).

PHWR-700 (India)

The PHWR-700 design is the latest domestically (Indian) nuclear power plant design from Nuclear Power Corporation of India, Ltd. (NPCIL). The EHNUR project concludes that while the design is an improvement from the previous series of Indian PHWRs, it has not yet reached the standards of contemporary Generation III and Generation III+ designs. A comparison between the PHWR-700 (NPCIL, 2011; Muktibodh, 2011) and the Generation III CANDU EC6 this (see Table 10) illuminates several important points:

- Although the PWHR-700 is a larger reactor (2166 MWt vs. 2084 MWt), it has larger house loads than the EC6 (70 MWe vs. 50 MWe).

- The net thermal efficiency of the PWHR -700 at 29% is very low, even compared with Generation II reactor designs. The thermal efficiency of the EC6 is 33.1%.

- The CDF target for PHWR-700 is a factor of 10 higher than for EC6, and is more typical of what is expected from Generation II reactors. Most Generation III and III+ have much lower CDFs than 1×10^{-5}/a.

- The design containment leak rate of 1% volume per day for the PHWR-700 is rather high – five times higher than the design containment leak rate for EC6.

- While both the PWHR-700 and EC6 have passive decay heat removal systems, the PWHR-700 system requires replenishment in 6 hours, whereas the EC6 system has a duration of 168 hours before replenishment is needed.

- The containment design pressure for the PHWR-700 is very low (0.16MPa) compared with the EC6 containment design pressure (0.5 MPa).

- The system of providing emergency makeup water to the secondary side of the steam generators to continue heat removal from the heat transport system is active in the case of the PHWR-700, whereas in the EC6 the system is passive.

- The design service life of the PHWR-700 is 40 years (typical of Generation II units), whereas for the EC6 the design service life is 60 years (typical of Generation III and III+ units).

On balance, we consider that PHWR-700 is a Generation II reactor design.

2.3 WHICH ADVANCED REACTOR DESIGNS APPEAR TO HAVE BEEN ABANDONED, AND FOR WHAT REASONS?

The EHNUR project has identified six advanced reactor designs that appear to have been abandoned by their vendors. These abandoned designs are very briefly identified below:

- AP600 (Westinghouse) – This Generation III design was granted Design Certification by the US Nuclear Regulatory Commission in December 1999. Nevertheless, there were no orders for the AP600 and the AP600 design was replaced by the larger AP1000 reactor. This illustrates the difference between Generation III and Generation III+ reactors.

- BWR 90+ (ABB Atom/Westinghouse) – The BWR 90+ was a Generation III boiling water reactor that was successfully reviewed again the European Utility Requirements. There were no orders placed for this design.

- Process Inherent Ultimate Safety (PIUS) – The PIUS passive PWR concept was developed by ABB Atom (Sweden) and Oak Ridge National Laboratory (United States) (IAEA, 1997b). The design placed the primary coolant system in a passively cooled, borated water pool (3300 m^3) that is located in a 7-meter thick pre-stressed concrete reactor vessel. The design was advocated by Dr. Alvin Weinberg and his coworkers in the Second Nuclear Era project (Hannerz, 1983; Forsberg & Reich, 1991; Weinberg et al., 1984). No orders were placed for the PIUS design, and the design was abandoned in 1996 (Slater, 2005). Another version of PIUS had three PIUS reactors, an interim fuel storage rack, and three spent fuel racks all located underwater within the pre-stressed concrete containment (ORNL, 1986).

- SAFR (Sodium Advanced Fast Reactor)(Rockwell International) – The SAFR concept was advanced in conceptual design by Rockwell International. The US Nuclear Regulatory Commission published a pre-application review of SAFR (NRC, 1991b).

- Simplified Boiling Water Reactor (SBWR) (General Electric) - The Simplified Boiling Water Reactor (SBWR) was intended as a 600 MWe class Generation III+ reactor with passive safety features and simplified construction and operation. SBWR would have been a natural circulation design. General Electric submitted SBWR design documents to the U.S. Nuclear Regulatory Commission, but then withdrew them in1996. The SBWR design effort was used in part as the basis for the ESBWR design.

- System 80+ PWR (Combustion Engineering, now Westinghouse) – The System 80+ PWR was granted Design Certification by the US Nuclear Regulatory Commission. There were no orders placed for this design. However, in the Republic of Korea, the System 80+ design served as the basis for the OPR-1000 design, and ultimately for the APR1400.

2.4 NUCLEAR POWER PLANT EFFICIENCY

Efficiency is a measure of the electricity generated compared with the thermal energy produced in the reactor core. Every nuclear power plant consumes electricity as well as producing it. The internal plant electricity consumption is referred to as "*house loads*" (sometimes called "*normal station loads*"), and represents electricity produced that never reaches the grid because it is essentially consumed where it is produced.

The EHNUR project does not regard it as technical correct to represent plant efficiency as MWe gross/MWt, because this does not represent the house loads correctly. Rather, efficiency is more correctly represented by MWe net/MWe. The reason why this is so is best illustrated by the extreme examples of early nuclear fusion plant concepts in which the difference between the generated electricity and the house loads is very large (hundreds of megawatts). In this chapter of the EHNUR report and in the related fact sheets, efficiency is presented as net thermal efficiency, calculated as MWe net/MWt.

2.5 DETERMINISTIC SAFETY ASSESSMENT (SAFETY ANALYSIS) AND PROBABILISTIC SAFETY ASSESSMENT (PSA)

Most Generation II nuclear power plants were licensed under a licensing regime which considered design basis accidents, and the criteria under which the design basis accidents were assessed were deterministic. Frequencies of accident initiating events were considered only coarsely in the deterministic context (e.g. within frequency bands of one or two orders of magnitude, such as anticipated operational occurrences).

Probabilistic safety assessment (PSA) – also called probabilistic risk assessment or PRA in the United States[34] – was first done on a large scale in the 1975 WASH-1400 Reactor Safety Study (NRC, 1975). PSA in safety review and licensing activities came into use after the Three Mile Island Unit 2 accident in 1979, and especially after the Chornobyl Unit 4 accident in 1986.

The 1975 WASH-1400 study was followed within a decade by the issuance of a methodology book for fault tree analysis (NRC, 1981) and by a PRA Procedures Guide (NRC, 1983) which had been developed jointly by the American Nuclear Society (ANS) and the Institution of Electrical and Electronics Engineers (IEEE).

In December 1990, the NRC published what was effectively an update to WASH-1400. After a several-year period which saw the release of two drafts (February 1987, which drew 800 pages of comments, and June 1989) of the document, the NRC issued NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, a three-volume report (Vol. 3 was published in January 1991) (NRC, 1990).

In October 1989, the NRC issued Generic Letter 88-20 (Supplement 1) (NRC, 1989a), which required all of its nuclear power plant licensees to prepare an Individual Plant Examination for Severe Accident Vulnerabilities (IPE). NRC also issued submittal guidance for the IPEs (NRC, 1989b). This requirement was met by licensees preparing a Level 1 PSA (estimation of core damage frequency).

This was followed in September 1995 by Supplement 5 to Generic Letter 88-20 (NRC, 1995) that required all nuclear power plant licensees to perform a similar examination for external events called an IPEEE[35], guidance for which had been issued in 1991 (NRC, 1991a). In most cases, a Level 2 PSA (analysis of accident progression, containment leakage or failure, estimation of source term magnitude and their frequency) was performed, although many licensees used a non-probabilistic analysis method for earthquakes known as a Seismic Margin Analysis (SMA). In December 1997, the

[34] PSA and PRA are often considered to be identical, since the methods and data are the same. There is a more subtle difference, however, in the focus of the study. The PRA is looking for indication of what risk is posed by the facility. The PSA is looking for an indication of how safe the facility is. This is an example of an issue of the sort "*is the glass half empty or half full*".

[35] As a matter of disclosure, the principal author of the Work Package 4 report performed a number of reviews of the IPEEE seismic submittals (including the IPEEEs for Catawba Units 1 & 2, Indian Point Units 2 & 3, Limerick Units 1 2, Nine Mile Point Units 1 &2, Pilgrim, and Three Mile Island Unit 1) for the NRC under subcontract to Energy Research, Incorporated (ERI). He also submitted (together with Jim Harding, a colleague at MHB Technical Associates), extensive public comments on the first draft of NUREG-1150, and also submitted comments on the second draft of NUREG-1150.

NRC published NUREG-1560, a three-volume summary of the IPE results (1997). This was followed in April 2002 with the publication of the two-volume NUREG-1742 report, which identified perspectives gained from the IPEEE program (NRC, 2002). Finally, and most recently, the NRC released NUREG-1935, which updates severe reactor accident progression and consequence analysis to the 2012 state-of-the-art (NRC, 2012b; NRC, 2012c).

A considerable PSA research and analysis program was also conducted in Germany, leading to the publication of the German Risk Study for the Biblis PWR (Phase A) in 1980 (GRS, 1980), followed by Phase B of the study in 1989 (GRS, 1989). This was followed by a PSA of German BWRs in 1993 (GRS, 1993a; GRS, 1993b). Two PSA methodology guides were issued by Bundesamt für Strahlenschutz (BfS) (Berg, 2004; BfS, 2005). A further study of risks posed by PWRs in Germany was issued in 2002 (GRS, 2002).

In 1990, IRSN (the technical support organization for the French nuclear regulatory authority) issued a pioneering analysis of nuclear power plant risk during shutdown conditions at the Paluel nuclear power plant. Within two years, there was an international meeting on the subject of shutdown PSA, and an increasing number of plants performing such analyses.

In the 1990s, the IAEA issued guidance documents for Level 1 PSA (IAEA, 1992a), Level 2 PSA (IAEA, 1995a), Level 3 PSA (IAEA, 1996a), and human reliability analysis in PSA (IAEA, 1996b). In 2010, these documents were superseded (IAEA, 2010a; IAEA, 2010b)[36].

In 2006, IAEA published a report providing guidance for determining the quality of PSAs for nuclear power plant applications (IAEA, 2006c). A number of nuclear regulatory authorities (NRAs) have published PSA guidance (AERB, 2007; CNSC, 2005; ENSI, 2009a; ENSI, 2009b; ONR, 2009; PNRA, 2010; STUK, 2003).

In 2002, the American National Standards Institute (ANSI) and the American Society of Mechanical Engineers (ASME) issued ANSI/ASME RA-S-2002, a consensus standard on how to perform Level 1 PSAs. This standard was updated in 2008 (ASME/ANS RS-S-2008, part 4 of which concerns external events (ASME/ANS, 2008). In 2007, ANSI and the American Nuclear Society (ANS) issued ANSI/ANS-58.21-2007(ANSI/ANS, 2007), a consensus standard on the methodology to be used in external events PSA. The Nuclear Energy Institute published an NRC-approved guideline on PSA peer review in 2000, revised in 2006 (NEI, 2006).

The first PSAM conference (Probabilistic Safety Assessment and Management) was held in 1991, and is held about every two years since then. The Society for Risk Analysis (SRA, http://www.sra.org/) was formed in 1980. The European Safety and Reliability Association (ESRA) was formed in 1986. A regional SRA organization, SRA – Europe, was founded in 1987, and began holding annual meetings in 1997. The Nordic PSA Group (NPSAG) was founded in 2000 by nuclear utilities in Finland and Sweden (http://www.npsag.org/home/). In March 2007, the Western European Nuclear Regulators' Association published a PSA Explanatory Note (WENRA, 2007) emphasizing the complementary roles of deterministic and probabilistic safety assessment.

Some countries and regulatory authorities have laws or regulations that require PSAs to be performed. Both the original version of the IAEA safety requirements document on design (IAEA, 2000) as well as the 2012 revised version of the IAEA Safety Standard on design of nuclear power plants (2012a) require the performance of a PSA as a complementary method of analysis together with deterministic safety analyses for nuclear power plants. Quantitative safety goals in a number of

36 Again as a matter of disclosure, the principal author of the Work Package 4 was a member of the IAEA Nuclear Safety Standards Committee (NUSSC) that approved SSG-3 and SSG-4, as well as SSR-2/1, also mentioned above.

countries[37] specify acceptance values for CDF and either LERF or LRF (large release frequency), or specify a band of acceptability (Bengtsson et al., 2010). Quantitative safety goals are included in the EPRI Utility Requirements Document (URD) and the European Utility Requirements (EUR) for new nuclear power plants. Quantitative safety goals were identified by INSAG in the 1988 (INSAG, 1999).

The result of all this PSA activity is that the performance of and reliance on Probabilistic Safety Assessments is fully ingrained in the safety culture of the nuclear industry (NEA, 2012b). Essentially all of the nuclear power plants in the world have been the subject of a least a Level 1 PSA (estimation of core damage frequency), and many have been the subject of a Level 2 PSA (analysis of accident progression and calculation of source term magnitudes and their frequency). Level 3 PSAs (accident consequence analysis and risk estimation) are much less common, although they are required in The Netherlands.

The point here is not that PSAs are a panacea (they are not – they are complementary to deterministic safety analysis), nor that the bottom line results of PSAs (such as CDF, LERF, or LRF) should be accepted uncritically (they should not – users of PSA results should be aware of both the limitations and the uncertainties of PSAs). Some possible sources of severe accidents are explicitly excluded from PSAs, such as malevolent acts (cyber-attacks, electromagnetic pulse from nuclear weapon detonation, sabotage, terrorism, and warfare)[38], and external hazards assumed to be low frequency (such as atmospheric bolide explosions, meteorite impacts, uncontrolled satellite re-entry, and glaciation). As a recent report noted (Keystone Center, 2007)[39], "*PRAs are only as good as the data, models, and assumptions on which they are based.*" PSAs have scope issues and PSA results are associated with uncertainty (Khatib-Rahbar, 2011; UCS, 2000; WENRA, 2007)[40]. The point is rather that PSAs are part and parcel of the safety analysis of currently operating nuclear power plants, and are fully integrated into the design and safety analysis processes for advanced nuclear power plant designs.

Annex 3 to this Chapter of the EHNUR report provides core damage frequency results for PSAs for both advanced reactors and a sampling of Generation II reactors (for comparison). Where scope limitations are known, these are noted. It should also be noted that the reported values are a mix of mean and point estimate values, and cannot be compared with one another without understanding the scope and arithmetic meaning of the numbers. The results in Annex 3 are offered solely as an

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37 These countries include Canada, the Czech Republic, Finland, Hungary, Japan, the Netherlands, the Republic of Korea, Russia, Slovakia, Sweden, Switzerland, the United Kingdom, and the United States.

38 A 2007 NRC report proposed a probabilistic approach to evaluating such hazards; see Section 6.7 in (NRC, 2007).

39 As a matter of disclosure, the author of the current Chapter of the EHNUR report knows or has met several members of the Joint Fact-Finding authors (Peter Bradford, Thomas Cochran, Victor Gilinsky, and Sonny Popowsky), and worked with one (James Harding) for several years. As well, the author knows or has met several of the experts who contributed to the Joint Fact-Finding effort (John Ahearne, James K. Asselstine, Robert J. Budnitz, and Michael W. Golay).

40 Uncertainties in Level 1 PSA are primarily aleatory in nature (parameter uncertainties), due to unpredictable variations. Expert knowledge cannot reduce these uncertainties, but it can help to quantify them. Uncertainties in Level 2 PSA are primarily epistemic (model uncertainties) in nature, due to incomplete knowledge. These uncertainties can be narrowed over time. Of course, there are examples of both types of uncertainties in Level 1 and Level 2 PSAs – the previous statements are expressions of general trends, not absolutes. Both Level 1 and Level 2 PSAs have completeness uncertainties.

The recognition of PSA uncertainties and scope questions has led to the publication of several standards and guides for PSA quality and for assuring PSA adequacy (ASME, 2005; NRC, 2007 – Annex F; NRC, 2009a; NRC, 2012d; Zhu, 2004).

indication of the ranges of CDFs for Generation III and III+ reactors compared with CDFs for Generation II plants.

As the March 2011 severe accidents at Fukushima Daiichi Units 1-4 made clear, the scope of currently available PSAs needs to be expanded to make the PSAs as complete as possible. It is no longer considered acceptable simply to analyze accidents initiated at full power from internal events. From now on, a full scope PSA requires consideration of (see, for example, ASME, 2012; Lyubarskiy, Kuzmina & El-Shanawany, 2011; Siu et al., 2013):

- <u>Internally-initiated accidents at full power and shutdown (including refueling and times when containment integrity is not available during shutdown).</u>

  Internally initiated accidents include fires and internal flooding, heavy load drops, loss of offsite power, loss of coolant accidents, containment bypass accidents, and a variety of accidents initiated by events other than loss of offsite power. Indeed, even a normal plant trip with no initiating equipment failure has been shown by past PSAs to contribute to CDF, and should therefore be considered.

- <u>Externally initiated accidents due to natural phenomena hazards and man-made hazards, at power operation and during shutdown (including refueling and times when containment integrity is not available during shutdown).</u>

  Usually external events are "screened", and only the initiators surviving screening are subject to a detailed analysis. Screening procedures must be tightened – arguments amounting to "hand waving" are no longer acceptable. Seismic events must be analyzed by probabilistic safety assessment – it is not adequate to do a seismic margin analysis (SMA), both because those analyses are incomplete and because it does not cost much more to do a seismic PSA than it does to perform an SMA. Among the external natural phenomena hazards and external man-made hazards that must be considered are external flooding, seismic phenomena, sand & dust storms, volcanic phenomena, high winds & storms, external fires, aircraft crash, turbine missiles, near-site & on-site transportation accidents, tsunami, extreme waves (rogue waves, storm surge), wind-driven missiles, upstream and downstream dam failures, electromagnetic interference (EMI), external explosions, biofouling of cooling and service water intakes, groundwater, drought, ice formation, temperature extremes in the air and cooling water, corrosive gas release, and lightning[41].

- <u>For multiple unit nuclear power plants (or units at adjacent sites), the potential for an external hazard to cause simultaneous accidents at more than one unit.</u>

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[41] There are long lists of external hazards that can potentially affect nuclear power plants contained in a number of publications (De Martinville & Herviou, 2010; Fleming, 2012; Glöckler, 2010; IAEA, 2002a; IAEA, 2003c; IAEA, 2003d; IAEA, 2010c; IAEA, 2011h; IAEA, 2012e; Modarres, 2010; NEA, 2009:17-19; NRC, 1983:Table 10-1; Patrakka, 2002). Readers are urged to use extreme caution concerning the limitations of so-called screening methods for eliminating external hazards (particularly as advocated in NRC 1983). Such screening methods were originally developed for Generation II nuclear power plants with CDFs between 1×10^{-5}/a and 1×10^{-4}/a. Screening events for Generation III and III+ units with internal event CDFs between 1×10^{-8}/a and 1×10^{-6}/a much be approached much more cautiously since, for example, an external hazard with a mean frequency of 1×10^{-6}/a and a conditional probability of causing core melt of 0.1-0.01 could turn out to be an important or even dominant contributor to overall CDF. If such an event were screened, risk would be unknowingly underestimated. This remains a problem for Generation III and III+ plants because there are methodological and data issues for many external hazards (e.g., estimation of flooding frequencies at the one in a million or one in 100 million frequency), and because there are external hazards at the 1×10^{-8}/a and 1×10^{-6}/a internal events CDF range of Generation III and III+ plants that are normally screened for Generation II plants (such as meteorite impacts, bolide explosions, supervolcanoes, etc.).

While this aspect of external hazards was known prior to March 2011, the Fukushima Daiichi nuclear power plant accidents illustrated vividly the potential for external hazards to cause simultaneous accidents at more than one unit (indeed, at more than one site). In this case, four reactors were destroyed and many others damaged in some respect by a megathrust earthquake and the resulting tsunami.

- For multiple unit nuclear plants (or units at adjacent sites), the potential for an accident at one unit to cause problems at an adjacent unit.

  There were examples of this problem during the March 2011 Fukushima Daiichi accidents, not only due to the general nature of multi-unit accidents but also due to the fact that control rooms were shared (in pairs) as were turbine halls. Shared control rooms are present at other nuclear power plants, as are shared auxiliary buildings and shared essential service water buildings (these are just examples).

- A complete and detailed uncertainty analysis must be performed.

  It is important that a thorough uncertainty analysis accompany a PSA. The uncertainty analysis must represent aleatory (stochastic) uncertainties and state-of-knowledge (epistemic) uncertainties. The uncertainty analysis must carefully document exclusions from the PSA and evaluate their possible impact on the results. Finally, the uncertainty analysis must also carefully consider assumptions made in the analysis in order to evaluate whether the assumptions could have an important influence on the PSA results (either individually or collectively).
  There are several reports that provide more details on PSA uncertainty analyses (Fleming, 2003; EPRI, 2008; NRC, 2009b).

2.6 CAUTIONARY DISCUSSION REGARDING ADVANCED REACTOR COST ESTIMATES (THE DIFFERENCE BETWEEN OVER-NIGHT AND ALL-IN ESTIMATES)

One must necessarily be cautious about reported cost estimates for nuclear power plants, especially those made by reactor vendors as part of sales campaigns and public relations efforts. If one sees nuclear power plant cost estimates in 2013 (or later) in the range of $1000-$3500 per kilowatt of installed capacity, one should view such estimates with considerable skepticism and within the proper framework[42]. The World Nuclear Association specifically cautions that such cost ranges are

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42 In 2007, the IAEA published Considerations to Launch a Nuclear Power Programme (IAEA, 2007a). This document created widespread confusion about the cost of nuclear power plants. The IAEA stated that a value of $1500-$2000 per kWe installed was "*indicative of current costs*", and that with efforts to reduce capital costs for future designs the cost could be in the range of $1000-$1500 per kWe installed. This was wildly optimistic, as subsequent events clearly showed. Even the contemporary (2007) cost of the turnkey contract on Olkiluoto Unit 3 (signed in December 2003) was for a cost of €3.5 billion for 1600 MWe net (€2187 per kWe installed, or about $2690 per kWe using the Dollar to Euro exchange rate for December 2003). But what is worse, the head of IAEA's Planning and Economic Studies Section in the Department of Nuclear Energy was stubbornly holding on – even as late as mid-2012 – to the idea that the capital costs of nuclear power were $1800/kWe installed (Rogner, 2012). This was an indefensible assertion when it was first made in 2007, and it was completely absurd in 2012 unless one was speaking about India, the People's Republic of China, or Russia where conditions are quite different from countries in the rest of the world where nuclear power plants were being built.

Similarly, the contract to construct the first two AP1000 units in the People's Republic of China was worth $5.3 billion, or $2372 per kWe installed. The overnight costs for pairs of AP1000 units for construction in the United States ranged from $2444 to $5358 per kWe installed for projects for which costs were estimated in 2008 (Turkey Point 3 & 4, Level 1 usually just for engineering, procurement, and construction of the nuclear unit – so-called EPC cost estimates (WNA, 2013h), and do not account for owners' costs or construction financing. In addition, sometimes vendor cost estimates are for the nuclear island and the turbine island only (and sometimes only the nuclear island), and do not consider all the other systems and structures needed (and supplied by the owner of the facility) so that the nuclear island and turbine island can operate.

First, early vendor cost estimates for standard nuclear power plant designs are, for the most part, not based on actual construction experience with the design in question (an exception is the ABWR, for which four units have already been built and are in operation in Japan). Second, such estimates are invariably so-called "overnight" cost estimates[43] which presume immediate purchase of all plant systems, components, and commodities in the morning of the first day of construction, and the completion of the construction of the plant before the next morning – i.e., overnight. Such cost estimates are completely unrealistic – it takes years (at least three and often more) to build a nuclear power plant, and during this three or more years of inflation and escalation will increase the total cost of construction[44]. Third, the vendor cost estimates and overnight cost estimates do not account for all of the costs associated with constructing a nuclear power plant, conducting cold and hot functional testing, loading fuel, conducting startup testing and a warranty power run (typically 100 hours at full power), and declaration of commercial operation. Some of the costs typically not included in vendor and utility overnight cost estimates are the following:

- Interest charges on funds borrowed to purchase plant systems and components and commodities (such as steel rebar, concrete, cabling, etc.), and on funds borrowed to pay construction workers, legal and technical personnel who must prepare licensing documentation (such as safety analysis reports, probabilistic safety assessments, and air & water pollution control permit applications), quality assurance and quality control personnel, etc.

- Owner's costs[45], including:

& 2, W.S. Lee 1 & 2, Bellefonte 1 & 2, Virgil Summer 2 & 3, and Vogtle 1 & 2). "All-in" cost estimates for these projects (which include not only overnight costs but owner's costs, escalation, contingency, and interest on funds borrowed to support construction) ranged from $9.3-14 billion, or $4163 to $6267 per kWe installed. Only projects in India, the People's Republic of China, and Russia have been (as of March 2003) coming in at even close to the range cited by the IAEA in 2007. The IAEA's 2007 estimate was wildly optimistic even compared with other contemporary estimates for the U.S. market (June 2007 Keystone Center study estimated an overnight cost of $2950/kWe, and Moody's Investor Services in October 2007 estimated costs of $5000 to $6000 per kWe installed, including escalation and financing, and acknowledged that this was "*only marginally better than a guess*").

43 As noted by Geoffrey Rothwell at Stanford, overnight cost estimates are based on the assumption that money has no time value. The reality is quite different. Rothwell converted an overnight cost estimate of $3,538 per kWe installed (2007 dollars) to escalated dollars assuming a 2012 construction start and a 2018 construction completion. The final escalated cost for twin 1,100 MWe units (ABWRs) was estimated at $9.87 billion (Rothwell, 2010). The latter number is $9.78 billion divided by 1.1 million kWe installed, or $8,972 per kWe installed – 2.5 times greater than the overnight cost.

44 Inflation is the change in prices of a basket of goods and services overtime, generally at the national level. Escalation is the change in the prices of specific commodities and services (such as steel, cement, copper cable, construction labor, etc.). The increased in the costs and in the offered bid prices have to be estimated and considered by the owner of the nuclear power plant, and this increase must be taken into account when the total financing requirements for the project are established (IAEA, 2011b).

45 A report from the University of Chicago indicates that owner's costs include costs for the owner's agent/engineer, licensing and project development costs, project management and oversight, owner's contingency, administration building and security, site facility transportation upgrades and sute improvements, interconnect and switchyard upgrades, spare parts, the initial nuclear fuel core, banking and legal fees, state permitting, property tax, sales tax, working capital, and transmission line costs (Rosner & Goldberg, 2011).

- Land purchased on which to construct the nuclear power plant.
- Infrastructure and administration buildings.
- Project management.
- Connections of the nuclear island and turbine hall with cooling water systems (e.g., essential service water, circulation water, and cooling towers if required).
- The plant switchyard and its connections to the grid (both two or more sources of offsite power coming into the plant, as well as outgoing distribution of power produced by the nuclear power plant).
- Systems and structures related to plant security; such security is needed during both construction (especially once nuclear fuel is received at the plant site) and operation.
- Systems and structures related to radiological protection.
- Hiring and training the plant workforce (construction work force, security, radiological protection, operational work force, maintenance personnel, operators, etc.).
- Purchase and installation of a plant-specific simulator.
- The writing and validation of a variety of different types of procedures (including industrial safety/occupational health & safety procedures, maintenance procedures, radiological protection procedures, security procedures, normal operating procedures, alarm response procedures, emergency operating procedures, and severe accident management guidelines), as well as training plant and construction staff on these procedures.
- Quality assurance (QA) and quality control (QC) costs.
- Normal non-nuclear needs, such as sanitary systems, drinking water, refuse and waste disposal, snow removal equipment, etc.
- Contingency.
- Escalation.
- Taxes.
- The cost of oversight by nuclear regulatory authorities, which is in some cases (especially in the United States) charged to the owner of the plant instead of coming out of general government revenues.

• Costs related to long-term storage of spent fuel or reprocessing of spent fuel and corresponding storage of vitrified high level waste (pending construction of a high level waste repository).

• The cost of the first reactor core of fresh fuel, as well as additional fuel assemblies for one or two fuel reloads (to be used during the first one or two refueling outages).

The costs not included in the vendor overnight costs easily amount to several billion dollars or more, and are also associated with interest charges on money borrowed until the plant begins operation and income from electricity sales can be used in part to finance repayment of the loans.

In 1953, US Admiral Rickover wrote a memo in which he distinguished between what he referred to as academic reactors (or what the author of this Chapter of the EHNUR report has heard called paper-moderated reactors) and practical (real) reactors under construction. Academic designs are simple, small, cheap, and can be built very quickly. Admiral Rickover characterized practical reactors as under construction, behind schedule, complicated, requiring a lot of development, being very expensive and taking a long lead time to build (Rosner & Goldberg, 2011). The vast majority of the advanced reactors described in this Chapter can fairly be characterized as paper-moderated reactors – they have not yet been built.

A final note – the World Nuclear Association has reported cost per kilowatt hour of generation in terms of gross electrical capacity (WNA, 2013h). This is misleading since the gross electrical capacity fails to account for so-called house loads – electricity that never reaches the grid but instead is consumed in the plant that produces the power in the first place. House loads benefit the electricity customers only indirectly in the sense that if the house loads were not consumed the plant could not operate. The direct beneficiary of house loads, however, is the operating utility which effectively gets this electricity for free because it charges its customers for the nuclear fuel and other related costs involved in generating the electricity. Nuclear power plants vary considerably in the size of house loads, from 30 MWe gross to 120 MWe gross. House loads clearly make a difference, and reflecting them in the cost per kWe of capacity makes no sense. The sensible calculation is the plant cost (whether one is talking about overnight costs or all in costs) divided by the net capacity in kWe installed.

Under current (early 2013) market conditions, the cost of a 1000-1700 MWe net nuclear power plant should be expected to be of the order of €6 billion (or more) per unit (perhaps somewhat less per unit when units are built in two-unit blocks and can share some structures and costs)[46].

46 The Nuclear Energy Institute (NEI) has published a "myths and facts" document that acknowledges that the cost of a large reactor is $6 billion to $10 billion (about €4.6 billion to €7.7 billion) (NEI, 2012). According to its website description (accessed 24 March 2013) the Nuclear energy Institute (NEI) is the "*policy organization of the nuclear energy and technologies industry*". NEI has over 350 member organizations in fifteen countries. NEI's membership includes AREVA, Babcock & Wilcox, GE Hitachi Nuclear Energy, General Atomics, Korea Hydro & Nuclear Power Company, Mitsubishi Heavy Industries, Toshiba America Nuclear Energy Corporation, and Westinghouse – all of which are vendors for advanced reactor designs. The President of B&W Modular Nuclear Energy, Inc., the President and CEO of GE Hitachi Nuclear Energy, the President & CEO of Holtec International, the President & CEO of Mitsubishi Nuclear Energy Systems, Inc., and the CEO of AREVA Inc. (all of which companies are vendors for advanced reactor designs) were members of the NEI Board of Directors as of March 2013.

3 HOW DO ADVANCED REACTORS COMPARE WITH GENERATION II NUCLEAR POWER PLANT DESIGNS?

Compared with currently operating Generation II reactors, Generation III and III+ reactor technologies appear to offer certain general advantages:

- Improved containments, with about half of Generation III and Generation III+ designs incorporating double containments, as well as core debris management systems (including passive reactor cavity flooding systems and core catchers), and explicitly designed protection against large aircraft crash consequences.

- Correctly oriented turbines in most cases (perpendicular to the reactor instead of parallel to the reactor to avoid turbine missile hits on the containment and auxiliary building)[47].

- A design service life of 60 years (compared with 30-40 years with operating Generation II reactors[48]).

- Higher (projected) lifetime reactor online availability of 90-95% (compared with the average worldwide for operating Generation II reactors of 78.4%, with a range from 67.4% to 91.6%)[49].

- Generally higher peak ground acceleration design basis for seismic events – typically 0.25g PGA to 0.3g PGA for Generation III and III+ designs[50], compared with 0.1g PGA to 0.2g PGA for most operating plants[51].

- Generally lower predicted likelihood of severe accidents, with core damage frequencies (CDF) as calculated in probabilistic safety assessments (PSAs[52]) ranging from 4.2×10^{-7}/a at the low end to

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47 Exceptions to this, with adversely oriented turbines are the EC-6 CANDU design, the GE-Hitachi ABWR (Kashiwazai-Kariwa Units 6 & 7, Shimane), and the Toshiba ABWR (Higashidori Unit 1, planned Visaginas units in Lithuania), all of which still orient the turbine parallel to the reactor building. On the other hand, the Lungmen ABWR units which are under construction in Taiwan do not have the unfavorable turbine orientation problem with respect to the reactor building.

48 It has to be noted in this context that while operating Generation II reactors had originally designated 30-40 service lives, many such reactors have had their operating lifetimes extended to 50-60 years. This is particularly true in the United States, where the Calvert Cliffs Units 1 & 2 reactors were the first to be granted 60 year operating licenses in 2000. Since then, over 80% of operating reactors in the U.S. have been granted a 20-year license extension to 60 years (NRC, 2012a). Operating reactors in the Russian Federation have also been granted 15-25 year life extensions. The World Nuclear Association has observed that license extension is "*now common*" (WNA, 2012). It also has to be noted that the U.S. Department of Energy held a workshop on so-called "*life-after-60*" considerations, and DOE's Light Water Reactor Sustainability Program is developing a technical basis for an operating life of 80 years for Generation II reactors in the United States (http://energy.gov/ne/nuclear-reactor-technologies/light-water-reactor-sustainability-lwrs-program).

49 The data for currently operating reactors (which are mostly Generation II) are from the International Atomic Energy Agency (IAEA) Power Reactor Information System (PRIS) as of March 2013.

50 The exception is the ESBWR, which is apparently designed for 0.5g PGA.

51 There are exceptions; for example, the San Onofre and Diablo Canyon plants are designed for higher PGAs due to their locations with respect to faults in the regions surrounding the plant sites.

52 Probabilistic safety assessments (PSAs) are a structured approach to identify weaknesses in the design and operation of a nuclear power plant, and to evaluate and compare potential options for remedying any such weaknesses, and to identify accident scenarios together with their likelihoods and impact on the plant and environment. Three levels of PSA are recognized: (1) Level 1 PSA identifies accident scenarios and their frequency of occurrence; (2) Level 2 PSA

2.5×10^{-6}/a for advanced PWRs at the high end, and ranging from 8×10^{-8}/a at the low end (ESBWR and KERENA) to 1.7×10^{-7}/a for at the high end for advanced BWRs (GE-Hitachi ABWR). These Generation III and III+ PWR & BWR CDFs can be compared to the average for Generation II units of about 2×10^{-5}/a to 5×10^{-5}/a (a range cited by EPRI and the European Commission), and with a range from 1×10^{-6}/a to 1.7×10^{-4}/a for existing plants. The Generation III and III+ CDFs can also be compared to the CDF of 1.3×10^{-5}/a for the most recent Generation II Chinese CPR-1000 units) (Bo, 2011) (see Annex 3 to this chapter).

- Lower expected rates of production of low-level radioactive waste than operating Generation II units.

Notwithstanding these advantages, there are some potential detriments of Generation III and III+ designs available for immediate deployment:

- The large unit size of Generation III and III+ reactors in theory limits the grid size to which they can be deployed. For the range of net generating capacity from 1082 MWe (VVER-1200, ACR-1000) to 1650 MWe (APWR, EPR), and considering that no single unit should be more than 10-15% of the grid capacity, the smallest grid capacity for which such a unit could be added would be about 10-16 GWe (considering a single unit at 10% of grid capacity), or 15-27 GWe (considering a single unit at 15% of grid capacity). Yet countries with grid capacities from 2.6-8.3 GWe are planning on adding two 1000-1200 MWe PWRs at a single site (Bangladesh, Jordan, and Belarus), and countries with grid capacities between 15.2-23.3 GWe are planning on adding four or eight 1000-1200 MWe PWRs (United Arab Emirates & Vietnam) (Subki, 2012b).

- Orientation of turbines in multi-unit plants typically is such that the units are parallel to one another (see Figure 10, which illustrates an adverse turbine orientation, and Figure 11, which shows correctly oriented turbines that are nonetheless parallel). Although this may be due constructability issues and the size of the site, it has the unfortunate effect of placing the turbine halls of adjacent units at risk in the case of a turbine failure of one unit that produces external

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examines accident progression and its effects on the containment system, as well as any resulting source term (release of radioactive material to the environment); and (3) Level 3 PSA is an assessment of offsite accident consequences. The IAEA has published safety standards for Level 1 PSA (IAEA, 2010a) and Level 2 PSA (IAEA, 2010b).

While there is no means to assure that PSAs are complete in a formal way, the aim of PSAs is to be as complete as possible given the current state of knowledge. Consistent with the current state-of-the-art, a comprehensive PSA looks at accidents with the reactor at power, accidents during shutdown, and accidents during refueling (when the reactor vessel and spent fuel pool are connected). A state-of-the-art PSA also looks at accidents occurring in the spent fuel pool (including the time varying hazard arising over the period from a current refueling outage to the next such outage), including so-called "*full core discharge*" in which the entire contents of the reactor core are discharged to the spent fuel pool in order to more efficiently accomplish outage tasks or to make modifications in the reactor vessel internal structures.

The accident initiating events examined in a state-of-the-art PSA include internal events (various transients and loss-of-coolant accidents or LOCAs), external man-made hazards (such as fires, turbine missiles, and aircraft crash), and external natural phenomena hazards (such as earthquakes, flooding, and severe storms). As the 2011 Fukushima accidents made clear, a nuclear power plant PSA also needs to look at the potential for multiple concurrent accidents, as well as the influence of an accident at one or more units of a multi-unit plant on other units at the site (be they operating, shut down, or in refueling).

The IAEA safety requirements for design of nuclear power plants (IAEA, 2012a), and for their commissioning and operation (IAEA, 2011a) both require that PSAs be performed. PSAs are also reviewed as part of periodic safety reviews (PSRs) of nuclear power plants (IAEA, 2003a), which are conducted in many countries (but ironically not in the United States, which has the largest population of operating nuclear power plants (103 out of the world total of 437, 23.5% of the operating reactors).

missiles (turbine blades or turbine rotors). In a deregulated market, this is a risk borne by the operating utility, which may lose billions of Euros in electricity sales until replacement equipment can be ordered, fabricated, delivered to the site, and installed. Equipment replacement is another expense from such accidents.

- Some Generation III and Generation III+ designs still use a single containment (like most Generation II reactors): (a) CANDU ACR-1000 & EC-6, (b) GE-Hitachi ABWR, (c) Toshiba ABWR (Japan and US designs), and (d) AREVA KERENA.

  Other Generation III and III+ designs have double containments, with a secondary containment to protect the primary containment from external hazards. The double containment designs

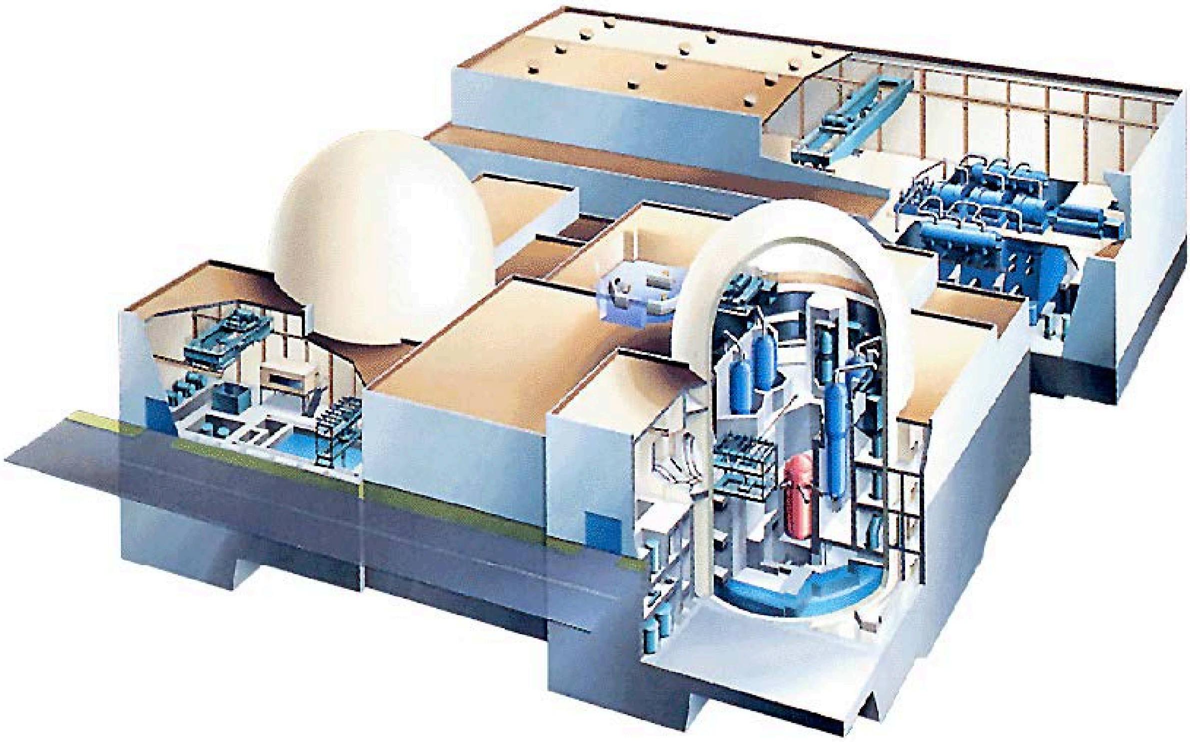

FIGURE 10: ILLUSTRATION OF ADVERSE TURBINE ORIENTATION. (SUBKI 2012A, SLIDE 3 OF 87, B, AND SLIDE 35 OF 87)(NOTE THAT THERE IS NO INDICATION OF COPYRIGHT OR PROPRIETARY DESIGNATION ON THE PERTINENT PAGES OF THE REFERENCED SOURCE.)

FIGURE 11: ILLUSTRATION OF PARALLEL TURBINE HALLS. (SUBKI 2012A, SLIDE 3 OF 87, B, AND SLIDE 35 OF 87)(NOTE THAT THERE IS NO INDICATION OF COPYRIGHT OR PROPRIETARY DESIGNATION ON THE PERTINENT PAGES OF THE REFERENCED SOURCE.)

typically have an air filtration system releasing to the plant stack which can provide additional protection against containment leakage in a severe accident[53].

Generation III and III+ designs with double containments are: (a) AREVA EPR, (b) Toshiba EU-ABWR, (c) Westinghouse AP1000, (d) OKB Gidropress VVER-1000 AES 91 & AES 92, (e) OKB Gidropress VVER-1200, and (f) GE-Hitachi ESBWR.

The ATMEA1 design (from a consortium of AREVA and Mitsubishi) and the Mitsubishi APWR have a single containment with a partial double containment covering the area where containment penetrations are located. (This is similar to the Generation II design for the Sizewell B PWR in the United Kingdom.)

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53 We do not count the BWR Mark I and Mark II units among the units with double containments, although General Electric (and possibly Hitachi and Toshiba as well) refers to the reactor building as a secondary containment. As the Fukushima accidents so clearly showed, this is not the case – the reactor building structure is a confinement only. The only BWR Mark I unit that we have so far identified as having a double containment is the Mühleberg plant in Switzerland.

A number of currently operating Generation II plants also have double containments (there are others):

- PWRs with ice condenser containments (10 units in Finland, Japan, and the US. D.C. Cook Units 1 & 2 are not included as double containments because they have single steel-lined reinforced concrete containments) (AEP, 2011).
- Doel Units 1-4 and Tihange Units 2 & 3in Belgium.
- Siemens PWRs (11 units) in Brazil, Germany, the Netherlands, and Spain.
- Siemens BWR72 (Gundremmingen B&C) in Germany.
- AREVA PWRs (24 units, 1300 MWe & 1450 MWe).
- NPPs in Switzerland (5 units).
- PWRs with steel primary containments and concrete secondary containments (2 units).

- Most of the Generation III and III+ designs still have the spent fuel pool outside the containment in non-hermetic structures (lacking a steel liner, lacking combustible gas control systems, and lacking adequate filtration systems). (The problem with such an arrangement was illustrated by the Fukushima Daiichi accidents in March 2011.) When the spent fuel pool is inside the containment, releases of radioactivity to the environment from a severe spent fuel pool accident can be reduced or perhaps even avoided (of course, there are potential consequences for the reactor arising from heavy contamination the containment due to a spent fuel pool severe accident). Generation III and III+ designs with the spent fuel pool inside containment are the AREVA KERENA BWR, the Toshiba US- and EU-ABWRs, and the OKB Gidropress VVER-1000 AES 91, the VVER-1000 AES 92, and the VVER-1200 designs. Generation III and III+ designs with double containments are: (a) AREVA EPR, (b) Toshiba EU-ABWR, (c) Westinghouse AP1000, (d) OKB Gidropress VVER-1000 AES 91 & AES 92, (e) OKB Gidropress VVER-1200, and (f) GE-Hitachi ESBWR.

  The ATMEA1 design (from a consortium of AREVA and Mitsubishi) and the Mitsubishi APWR have a single containment with a partial double containment covering the area where containment penetrations are located. (This is similar to the Generation II design for the Sizewell B PWR in the United Kingdom.)

4 HOW DOES THE DURATION OF SITING, AND CONSTRUCTON AFFECT THE POTENTIAL OF ADVANCED REACTORS TO BE DEPLOYED IN TIME TO START PRODUCING ELECTRICITY BEFORE 2030?

Based on Section 3, above, we have assumed that the advanced reactor technology in question must be deployed before 2020 in order to have a realistic chance of impacting electricity generation by 2030. This presumes the minimum project duration is 10 years. This is a potentially optimistic assumption in terms of the number of advanced reactor models that might contribute to electricity generation by 2030 since it assumes the minimum feasible schedule. The duration from feasibility study to start of commercial operation of ten years appears to be an absolute minimum even for an experienced utility in a country with significant existing nuclear power-related infrastructure, and using a pre-approve standard Generation III or III+ reactor design. The 10-year minimum duration presumes no problems in licensing, no problems in construction, and no delays anywhere along the process.

The shortest nuclear power plant construction duration of which the EHNUR project is aware is for the Qinshan III pressurized heavy water reactors. The contract effective date was 12 February 1997, first concrete was poured on 8 June 1997, Unit 1 was in service on 31 December 2002, and Unit 2 was in service on 24 July 2003 (Khan, 2011). Thus, the total duration from contract effective date until commercial operation was 70 months for Unit 1 and 77 months for Unit 2. Note that these durations do not include site evaluation and environmental impact assessment, preparation of bid specifications, issuance of tender, evaluation of tenders, or the contract negotiations that led to the contract effective date sited above. Considering the minimum schedules for these activities of 29 months (see Section 3), the total project durations might have been 99 months for Qinshan Unit 1 and 106 months for Unit 2 – that is, about 8-9 years. Construction durations shorter than 10 years can be achieved, as demonstrated by the Qinshan III project, but such durations are for Nth-of-a-kind units at an existing site – not for first-of-a-kind units at a new ("greenfield") site.

The minimum 10-year schedule cited previously presumes that the experienced utility will site the new generating unit at or adjacent to an existing nuclear power plant site, and will take full advantage of site banking[54] and design certification (advance approval by the nuclear regulatory authority of a standard nuclear power design before a specific construction project is identified) in order to shorten the schedule.

For a new utility entrant in the commercial nuclear power business, even in a country with significant existing infrastructure, the duration would be expected to be longer than ten years. For a new utility entrant in the commercial nuclear power business in a country with only modest nuclear power plant-related infrastructure, the duration from feasibility study to commercial operation would clearly take longer than the minimum schedule cited above, and would be expected to exceed the fifteen year mark.

The above perspective implies that advanced reactor designs available for immediate deployment will have the best chance of impacting electricity generation by 2030 since this would allow a 17-year

54 Site banking refers to advance approval of a prospective nuclear power plant site by the national nuclear regulatory authority in advance of a specific nuclear power plant construction project. In the United States, site banking corresponds to the Early Site Permit (ESP) process. ESPs in the United States are valid for 10-20 years from the date of issuance, and can be renewed for an additional 10-20 years.

period in which to bring such units into operation and generating electricity. Nuclear power plant projects based on advanced reactor technology that is already approved (or is being reviewed with the expectation of approval by the national nuclear regulatory authority concurrently with the start of activities leading to the construction of a new nuclear power plant) could – if started almost immediately – be completed and on line by about 2023-2025. The latest that a nuclear power plant project could be initiated would be in 2020 – the 10-year minimum schedule would have such a plant coming on line in 2030. Such near-term designs constructed on the minimum 10-year total project schedules would just be completed in 2030. Recall, as stated previously, that the EHNUR project horizon is 2030.

Advanced reactor designs with near-term availability will have the next best chance of impacting electricity generation by the year 2030, though possibly only a few, if any, of these reactors could be licensed and built and put into operation before 2030. Further, as a group, the contribution of these units to total nuclear generation in 2030 would be small. The contribution to electricity generation from such projects would be expected – in the absence of a "crash program" with massive short-term investments – to make only a small percentage contribution to the percentage of electricity generated by nuclear power by 2030 since the first units would only be coming on line (at the earliest) in the 2025-2030 time frame. Advanced reactor designs with potential long-term availability (that is, available for construction after 2022) have essentially no chance of impacting electricity generation by 2030.

These perspectives limit the number of designs for which detailed information is required for EHNUR. Advanced reactor designs that will not have an impact on electricity generation by 2030 are addressed in summary fashion herein[55].

In 2010, the IEA and NEA issued a nuclear energy roadmap that forecast nuclear power plant capacity growing from the current 372 GWe to between 475 and 500 GWe by 2020 (IEA/NEA, 2010b). This increase could only have been met by nuclear power plants under construction or soon entering construction.

By April 2013 when this Chapter of the EHNUR report was written, it was clear that the IEA/NEA roadmap projection was not going to happen. In April 2013, 68 nuclear power plants with a total combined net electrical capacity of 65.5 GWe were under construction. If one assumes that all 68 nuclear power plants under construction in April 2013 were to be completed by 2020, this would increase total world net installed nuclear capacity of 372.6 GWe in April 2013 to 438.1 GWe in 2020. Several factors will work against achieving this amount of nuclear generation by 2020:

- In all likelihood, some of the 68 reactors under construction as of April 2013 will not be completed by 2020 (this is clear from the history of construction durations to date during the last 10 years).

55 Readers who are interested in more details on advanced nuclear power plants or on nuclear power programs in various countries can refer to the following general references: (1) International Atomic Energy Agency (IAEA) Country Nuclear Power Profiles, available at http://www-pub.iaea.org/MTCD/Publications/PDF/CNPP2012_CD/pages/index.htm in the 2012 edition at the time this report was written; (2) similar but more frequently updated Country Profiles are issued by the World Nuclear Association (WNA) and could be obtained (at the time this report was written) at http://www.world-nuclear.org/info/Country-Profiles/; or (3) readers interested in a somewhat detailed discussion of the various advanced reactor designs can access information at the IAEA's Advanced Reactors Information System, http://www.iaea.org/NuclearPower/aris/.

- Some existing operating units will be shut down by 2020. (Exemplifying this are the 23 shutdowns for decommissioning that occurred in the last five years[56], removing 16.4 GWe from the grid.)

- Some units that were originally planned to start construction in 2010 or later, and be completed prior to 2020, have since been cancelled (such as the pair of ABWRs that were planned to be Fukushima Daiichi Units 7 & 8, which were cancelled by TEPCO in April 2011, but were originally planned to commence operation in 2016 and 2017).

- There are 50 nuclear power units in Japan that are considered by IAEA to be "operational". Only two of them (Ohi Units 3 & 4) are actually operating. All but one of the Japanese reactors had been shut down by 27 March 2012 (only 19 units were operating as of 16 May 2011; by 5 September 2011, the number of operating units had dropped to 11, and by 27 January 2012 this had dropped to 3 units operating), and the last operating unit (Tomari Unit 3) stopped operation on 5 May 2012. It is an open question how many of these 48 units will be eventually restarted, but it is likely that some of the Japanese reactors will be designated for decommissioning. The 48 non-operating units in Japan should be considered to be in long-term shutdown until they are returned to operation or designated for decommissioning. At one point, IAEA removed the 48 non-operating Japanese units from its list of operating reactors, but then they were soon restored to the list of operating plants, and all operating plants (not just those in Japan) were redesignated "operational" instead of operating.

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56 In the five-year period between April 2008 and April 2013, the following units were shut down for decommissioning (data from IAEA PRIS):

- Crystal River Unit 3, 860 MWe net, closed on 5 February 2013.
- Gentilly Unit 2, 645 MWe net, closed on 28 December 2012.
- Wylfa Unit 2, 550 MWe net, closed on 25 April 2012.
- Oldbury A1, 300 MWe net, closed on 29 February 2012.
- Biblis A, 1146 MWe net, closed on 6 August 2011.
- Biblis B, 1178 MWe net, closed on 6 August 2011.
- Brunsbüttel, 770 MWe net, closed on 6 Au870+gust 2011.
- Isar Unit 1, 870 MWe net, closed on 6 August 2011.
- Krümmel, 1260 MWe net, closed on 06 August 2011.
- Neckarwestheim Unit 1, 805 MWe net, closed on 6 August 2011.
- Phillipsburg Unit 1, 864 MWe net, closed on 6 August 2011.
- Unterweser, 1230 MWe net, closed on 6 August 2011.
- Oldbury A2, 300 MWe net, closed on 30 June 2011.
- Fukushima Daiichi Unit 1, 439 MWe net, closed 19 May 2011.
- Fukushima Daiichi Unit 2, 760 MWe net, closed on 19 May 2011.
- Fukushima Daiichi Unit 3, 760 MWe net, closed on 19 May 2011.
- Fukushima Daiichi Unit 4, 760 MWe net, closed on 19 May 2011.
- Phenix, 233 MWe net, closed on 01 February 2010.
- Ignalina Unit 2, 1185 MWe net, closed on 31 December 2009.
- Hamaoka Unit 1, 516 MWe net, closed on 30 January 2009.
- Hamaoka Unit 2, 814 MWe net, closed on 30 January 2009.
- Bohunice V1 Unit 2, 408 MWe net, closed on 31 December 2008.
- Pickering Unit 3, 508 MWe net, closed on 31 October 2008.

5 WHAT ADVANCED REACTOR DESIGNS COULD BE DEPLOYED IN TIME IN ORDER TO PRODUCE ELECTRICITY BEFORE 2030 – SOURCES OF INFORMATION AND GENERAL CONSIDERATIONS

In order to identify advanced reactor design candidates, the EHNUR project made use of internal expert knowledge as well as reference to a number of sources:

- International Atomic Energy Agency (IAEA) web sites for:
  - The Advanced Reactor Information System (ARIS) http://www.iaea.org/NuclearPower/aris/;
  - The Power Reactor Information System (PRIS);
  - Meetings and publications of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO);
  - Meetings and publications of the IAEA's Division of Nuclear Power;
  - Meetings and publications of the IAEA's Nuclear Power Technology Development section of the Division of Nuclear Power, including the Technical Working Groups on Fast Reactors (TWG-FR), Gas Cooled Reactors (TWG-GCR), Advanced Technologies for Light Water Reactors (TWG-LWR), and Advanced Technologies for Heavy Water Reactors (TWG-HWR).
- Ux Consulting Company LLC, Small Modular Reactor List http://www.uxc.com/smr/uxc_SMRList.aspx.
- Information Papers from the World Nuclear Association, including:
  - Plans for New Reactors Worldwide, updated March 2013;
  - Advanced Nuclear Power Reactors, updated 19 March 2013; and
  - Small Nuclear Power Reactors, updated March 2013.
- The Generation IV International Forum.
- The U.S. Nuclear Regulatory Commission websites on Advanced Reactors and New Reactors.
- The United Kingdom Office for Nuclear Regulation's Generic Design Assessment (GDA) web site.

Two additional considerations referenced in the context of designs that are available for immediate deployment are design certifications by national nuclear regulatory authorities and compliance with the European Utility Requirements (EUR, http://www.europeanutilityrequirements.org/) issued by a consortium of European utilities for the European power generation market, or the Utility Requirements Document (URD, http://urd.epri.com/) issued by the Electric Power Research Institute (EPRI).

The following designs have been evaluated for conformance with the EPRI URD (Pirson, 2010):

- Combustion Engineering (now Westinghouse) System 80+;
- GE-Hitachi ABWR;
- GE-Hitachi ESBWR;
- Westinghouse AP600; and
- Westinghouse AP1000.

The EUR has certified compliance with its requirements documents for the following designs (Pirson, 2010):

- ABB (now Westinghouse) BWR 90 (June 1999);
- AREVA EPR (December 1999 and June 2009).
- GE-Hitachi ABWR (December 2001);
- KERENA (SWR-1000, the predecessor design) (February 2002);
- VVER AES 92 (June 2006);

- Westinghouse AP1000 (June 2006); and
- Westinghouse European Passive Plant (December 1999).

The results of the assessment of advanced reactor designs available for immediate deployment are provided in Table 1, an extended table that shows not only the summary list, but includes several pages of notes on each of the designs. Nineteen advanced reactors are identified on Table 1[57]. Fact Sheets have been prepared for each of the advanced reactor designs in Table 1. Additionally, Table 1 identifies six remaining Generation II designs that were in April 2013 under active construction and for which additional units were planned[58].

Figure 12 shows a cutaway drawing of the AREVA EPR design, an example of an advanced pressurized water reactor. Figure 13 shows a cutaway drawing of the GE-Hitachi ABWR, an example of an advanced boiling water reactor. Figure 14 shows a concept for the KLT-40S barge-based floating nuclear power plant. Figure 15 shows a cutaway drawing of the CPR-1000, a Generation II design that was still under construction in April 2013, the first unit of which had just gone into operation in the People's Republic of China.

Additional details on the reactor designs available for immediate deployment, including comparisons, are provided in Table 2 through Table 6.

Some initially planned Generation III and III+ projects have already been cancelled by the utilities involved:

- In September 2009, Exelon filed a COL application with the NRC to construct a two-unit nuclear station called Victoria County Station. The project was cancelled in August 2012.
- In September 2008, construction was renewed at the Belene site in Bulgaria, with the intention of completing two units as VVER-1000/446B (AES92) Generation III reactors. The project was cancelled in March 2012.
- In September 2007, NRG Energy filed a combined operating license (COL) application with the US Nuclear Regulatory Commission to construct two Toshiba ABWRs at the South Texas plant site. In April 2011, NRG announced it was cancelling the project.
- In August 2007, Energy Alberta Corporation filed an Application for a License to Prepare Site with the Canadian Nuclear Safety Commission for two ACR-1000 units at Lac Cardinal (near Peace River) planned to be commissioned by 2017 (Alberta, 2009a; Alberta, 2009b). Three months later, Bruce Power purchased Energy Alberta Corporation, and proposed expanding the project to four units. Bruce Power later withdrew the application in January 2009, and proposed a new site about 60 km from La Cardinal. This project was cancelled by Bruce Power in December 2011 (Bruce Power, 2011).

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57 The EHNUR project ranking of SMRs available for immediate deployment is in agreement with that of the IAEA, except for the SMART integral PWR. IAEA has the SMART design designated as available for near-term deployment (Subki, 2012b). EHNUR considers that in view of the SMART Standard Design Approval in July 2012, this design is available for immediate deployment.

58 Every year, the IAEA issues a "Nuclear Technology Review" during its General Conference. Part of that document is a discussion of advanced designs for nuclear power plants. Referring to that discussion in 2002, it is perhaps noteworthy that a number of the designs listed in Table 1 were already discussed 10 years ago (ABWR, APR-1400, APWR, CAREM, CPR-1000, EPR, ESBWR, KERENA, and SMART). In addition, several of the designs listed in Table 7 and Table 8 were also mentioned (ABWR-II, AHWR, BREST-300, GT-MHR, PBMR, KALIMER, and SVBR-100) (IAEA, 2002b).

FIGURE 12: GENERATION III+ EPR PWR (CUTAWAY VIEW). [SOURCE: (SUBKI 2012A, SLIDE 29 OF 87)](NOTE THAT THERE IS NO INDICATION OF COPYRIGHT OR PROPRIETARY DESIGNATION ON THE PERTINENT PAGE OF THE REFERENCED SOURCE.) (NOTE THE TRUCK AT THE JUST TO THE LEFT OF CENTER NEAR THE BOTTOM OF THE DRAWING, PROVIDING SOME SENSE OF THE SCALE OF THE POWER PLANT.)

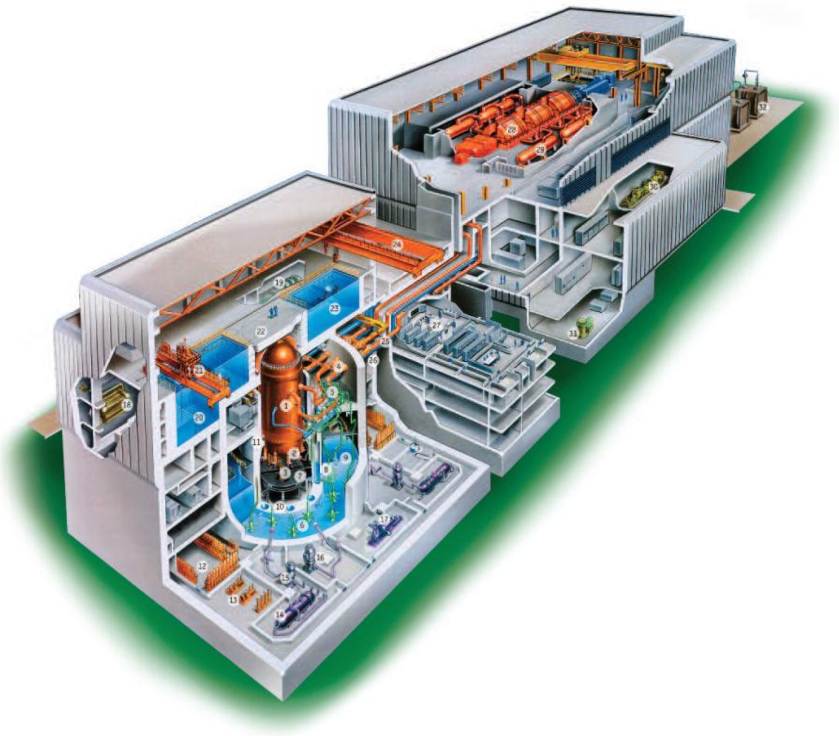

FIGURE 13: GE-HITACHI ABWR (CUTAWAY VIEW). [SOURCE: (SUBKI 2012A, SLIDE 56 OF 87) (NOTE THAT THERE IS NO INDICATION OF COPYRIGHT OR PROPRIETARY DESIGNATIONON THE PERTINENT PAGE OF THE REFERENCED SOURCE.)

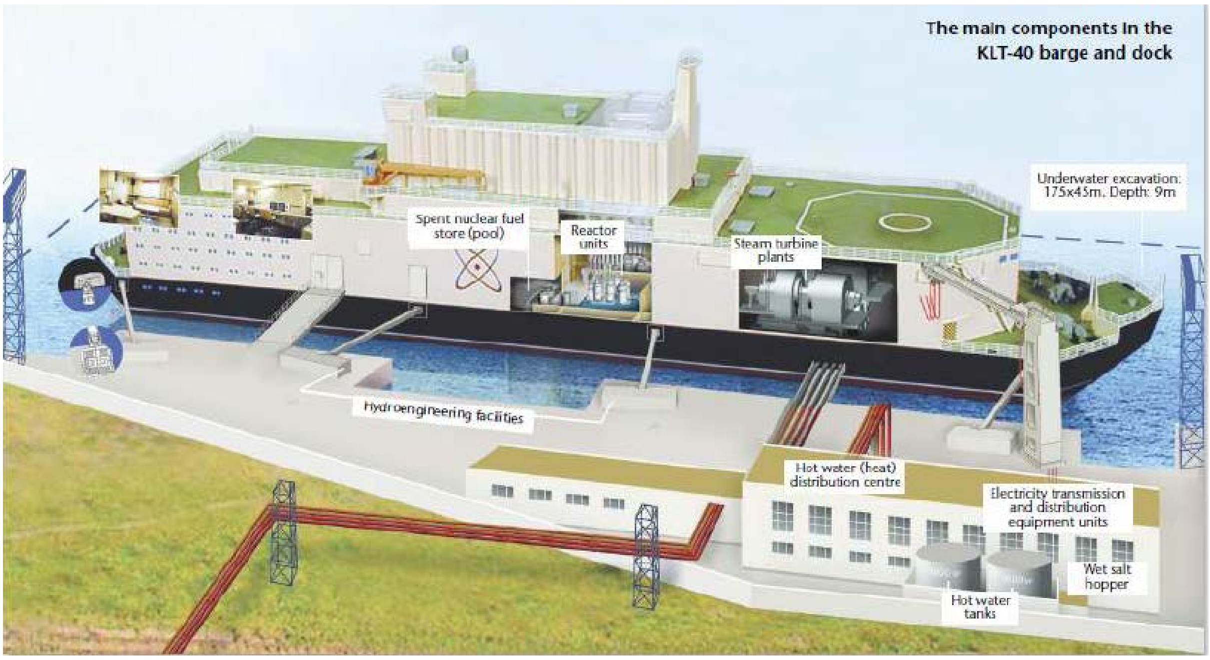

FIGURE 14: KLT-40S SMR (FLOATING NUCLEAR PLANT). [SOURCE: LEÓN 2012; NO INDICATION OF COPYRIGHT PROTECTION AT THE SOURCE PAGE OF THE DOCUMENT]

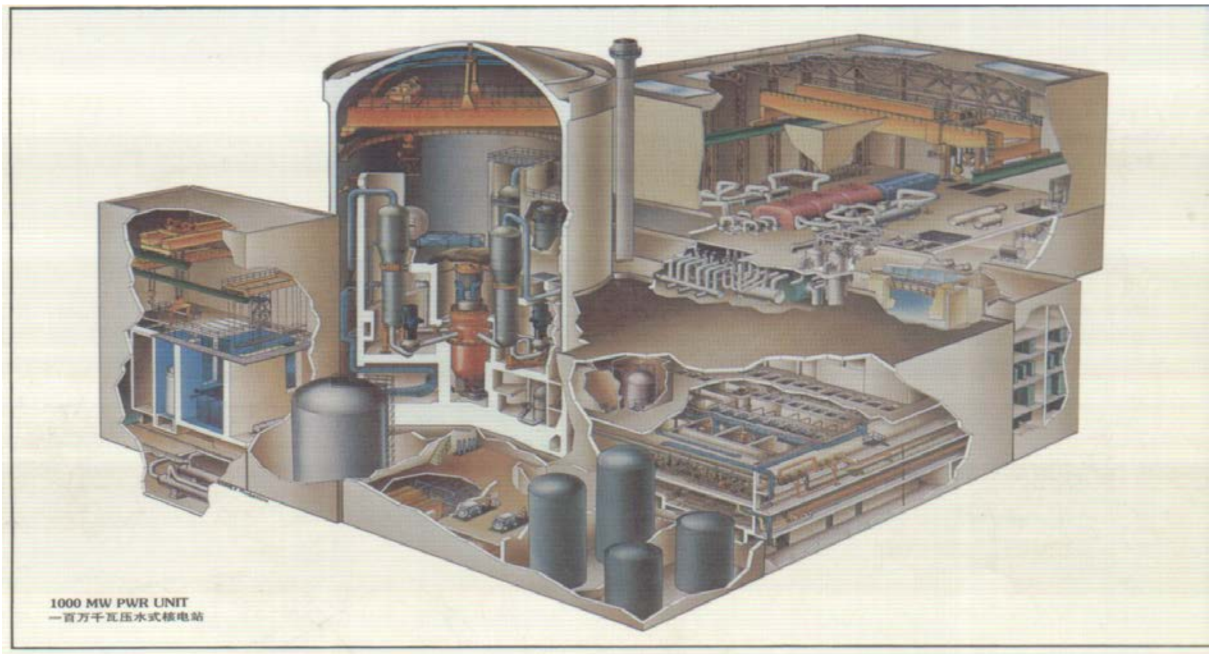

FIGURE 15: GENERATION II CPR-1000 PWR DESIGN (CUTAWAY VIEW). [SOURCE: (SUBKI 2012A, SLIDE 17 OF 87) (NOTE THAT THERE IS NO INDICATION OF COPYRIGHT OR PROPRIETARY DESIGNATION ON THE PERTINENT PAGE OF THE REFERENCED SOURCE, UNLESS IT IS IN CHINESE WHICH REGRETTABLY THE AUTHOR OF THIS CHAPTER CANNOT READ.)

- In August 2006, Bruce Power wrote to the Canadian Nuclear Safety Commission that it intended to start the regulatory process for a site for 4 nuclear power reactors totaling about 4 GWe on the existing Bruce Nuclear Site. Guidelines for the preparation of an Environmental Impact Statement for the project were issued in August 2008. Bruce Power withdrew its application in July 2009. The project was cancelled by Bruce Power in August 2012.
- In 1999, the Pebble Bed Modular Reactor (PBMR) project was initiated to build a 165 MWe Generation IV high-temperature gas cooled reactor module. In 2003, the South African government approved construction of a smaller 110 MWe prototype at the existing Koeberg site, to be put into operation in 2014. Construction of 24 PBMRs was foreseen by 2030 (Carre et al., 2009). With schedules slipping, the South African government stopped funding the development of the PBMR technology in 2010.

6 WHAT ADVANCED REACTOR DESIGNS ARE AVAILABLE FOR NEAR-TERM DEPLOYMENT (BEFORE 2020)?

There are pending advanced reactor designs that while they are not available for immediate deployment, might nonetheless become available for commercial deployment before 2020, and which might have some chance of producing electricity before 2030. In order to do so, however, the designs would have to be available for deployment by 2020, and then promptly constructed and place into operation. Even if this occurs, absent a "crash program", it is likely that the percentage contribution to nuclear power plant electricity generation by 2030 will be small (of the order of a few percent) for the same reason identified in Section 4 herein.

We characterize such reactors as potentially available in the "near term" (i.e., likely to become available for commercial deployment in the 2015-2020 time frame). The advanced reactors in this category are identified in Table 7, which provides a summary listing, and then several pages of notes on the various designs.

7 BASED ON CURRENT (MID-2013) INFORMATION, WHAT ADVANCED REACTOR DESIGNS COULD BECOME AVAILABLE FOR DEPLOYMENT AFTER 2020 IN ORDER TO PRODUCE ELECTRICITY BEFORE 2050?

There are quite a number of advanced reactor designs, small modular reactor designs, Generation IV reactor concepts, and nuclear fusion technology which are considered to be unlikely to be available for widespread commercial deployment before 2030. These designs can be characterized as long-term designs, and are identified in Table 8. The inclusion of a number of SMRs in this table is consistent with expectations that SMRs could be ready for licensing after 2020 (Rosner & Goldberg, 2011).

The reasons that the long-term designs will not be available in time to impact electricity production before 2030 are varied. Most often this has to do with either an early design stage (design concept description or conceptual design as of 2013) or the need to solve important technical issues before the design can be finalized.

8 WHAT ARE THE POTENTIAL ADVANTAGES AND DETRIMENTS OF SMALL MODULAR REACTORS?

Under IAEA definitions, small reactors are those with a net rating of 300 MWe or less. Many small reactors are intended to be built into larger power plants in a modular fashion, and are referred to as small modular reactors or SMRs. Generally, such reactor units can be factory fabricated and shipped to the power plant site by road, rail, or barge.

The potential advantages of such small reactors are:

- Plant safety systems tend to be passive, resulting in an expectation of a relatively low core damage frequency. Often, the passive systems are complemented by a one or two train active system; such combinations of active and passive components performing the same safety function are referred to as hybrid safety systems.

- The reactor and main safety systems are most often intended for below grade installation, which reduces external hazard risks for many types of hazards (a possible exception is external flooding).

- The reactor designs tend to be simpler than full size (900-1700 MWe) designs with fewer components and less cabling required.

- The reactors can generally be factory fabricated, transported by heavy truck or rail (or ship), and then relatively quickly installed at the plant site.

- The small reactors are flexible. They can more easily support remote areas of the electrical grid, and are more easily integrated into countries where the electrical grid size would have problems supporting full size nuclear units.

- The reactors can provide process heat for manufacturing and industrial requirements, and can support more modest sized desalination systems than would be required for larger population centers.

- Electrical capacity additions can be in smaller increments to match the availability of construction funds, and to match smaller increments in electricity demand growth.

According to a U.S. Department of Energy presentation to IAEA in 2012, the light water-cooled SMRs are expected to be available for deployment in the 5-10 year time frame. SMRs cooled by liquid metal or gas are expected to be available in the 10-15 year time frame. More advanced designs should not be expected before 15 years or more (Reister, 2012).

9 WHAT IS THE DEPLOYMENT HORIZON FOR GENERATION IV ADVANCED REACTOR DESIGN CONCEPTS, AND WHAT ARE THE POTENTIAL ADVANTAGES AND DETRIMENTS OF SUCH CONCEPTS?

The Technology Roadmap issued in December 2002 for Generation IV nuclear energy system deployment stated, "*The objective for Generation IV nuclear energy systems is to have them available for international deployment about the year 2030, when many of the world's currently operating nuclear power plants will be at or near the end of their operating licenses.*" (DOE, 2002) Specifically, the Generation IV Technology Roadmap estimated industrial deployment of the Sodium-cooled Fast Reactor (SFR) technology by 2015; industrial deployment of the Very High Temperature Reactor (VHTR) technology by 2020; and industrial deployment of Gas-cooled Fast Reactor (GFR), Lead-cooled Fast Reactor (LFR), Molten Salt Reactor (MSR) and Super Critical Water-cooled Reactor (SCWR) technologies by 2025.

By 2009, it was clear that this was not going to happen. The IEA/NEA nuclear power roadmap issued in 2010 forecasts completion and operation of demonstration units for Generation IV technologies between 2020 and 2040, with construction and operation of commercial-scale Generation IV nuclear power plants from 2040-2050, and there seems to be general agreement on this time frame (De Santi, 2009; IEA/NEA, 2010; Lee & Taylor, 2010; Riou, Verdaerde & Mignot, 2009).

It is possible that a VHTR prototype (HTR-PM) will begin operation before 2020 (startup was forecast in March 2013 for 2016) (WNA 2013k). It is unlikely that any other Generation IV prototype will begin operation before 2020[59].

The Generation IV International Forum (GIF) was created in the year 2000, and formalized the next year with the signing of the GIF Charter document. Current membership in the Forum and the year the GIF Charter was signed by each includes Argentina (2001), Brazil (2001), Canada (2001), EURATOM (2003), France (2001), Japan (2001), the People's Republic of China (PRC, 2006), the Republic of Korea (South, 2001), Russia (2006)South Africa (2001), Switzerland (2002), the United Kingdom (2001, currently inactive), and the United States (2001).

GIF has established eight goals for Generation IV systems (GIF, 2008):

- Generate energy sustainably, and promote long-term availability of nuclear fuel.
- Minimize nuclear waste and reduce the long term stewardship burden.
- Excel in safety and reliability.
- Have a very low likelihood and degree of reactor core damage.
- Eliminate the need for offsite emergency response.
- Have a life cycle cost advantage over other energy sources.
- Have a level of financial risk comparable to other energy projects.
- Be a very unattractive route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

GIF is pursuing six reactor technologies. Of the six Generation IV technologies (VHTR, LFR, SFR, SCWR, GFR, and MSR), only three are being pursued as of early 2013: (1) the VHTR design, (2) the

[59] As of April 2013, the schedule for startup of the AREVA ANTARES VHTR prototype as part of the US Next Generation Nuclear Plant (NGNP) was scheduled for 2021 (WNA, 2013e). The World Nuclear Association projects startup of a prototype SFR in France, possibly at Marcoule, in 2020 (WNA, 2013l), but considering that neither the plant site nor the power level of the reactor had been chosen as of April 2013, this schedule appears to be unachievable.

SFR design, and (3) the LFR design. The next Generation IV concept likely to be developed after these three is the SCWR type reactor. The GFR (gas-cooled fast reactor) and MSR (molten salt reactor) are lagging well behind the other four Generation IV technologies in terms of development.

Gas-Cooled Fast Reactor (GFR)

The Gas-Cooled Fast Reactors (GFRs) are high temperature gas-cooled reactors featuring a fast neutron spectrum (i.e., no moderator) and a closed fuel cycle for breeding of fissile fuel and management of actinides. To date, the reference concept involves use of helium as a coolant for a 2400 MWt reactor and use of a Brayton cycle gas turbine (no steam production; direct use of the helium coolant to run a gas turbine). GFRs are now intended to be fast breeder reactors – i.e., they would produce more fuel than they consume by conversion of fertile uranium or thorium. Within the Generation IV International Forum, a GFR System Arrangement has been created with four participants (Euratom, CEA from France, JAEA from Japan, and Paul Scherrer Instiute from Switzerland).

Potential weak points of GFR technology include the high power density of the core and the lack of operating experience (Henricksson, 2012).

Lead-Cooled or Lead-Bismuth-Cooled Fast Reactor (LFR)

Three are several lead-cooled fast reactor (LFR) designs being investigated in GIF. In the US, the design is called SSTAR, a 10-100 MWe small transportable lead-cooled fast reactor, but LFR activities in the US are limited. In the European Union, the LFR design being pursued is ELFR, a 600 MWe concept, preceded by MYRRHA (an accelerator-driven [60] lead-bismuth cooled system). Commissioning of MYRRHA (a 50-100 MWt research reactor) was forecast in mid-2012 to take place around 2023[61] at a cost of approximately €960 million (SNETP, 2010). Assuming that this schedule is met, some years of successful operation would be necessary in order to serve as a basis for final design of ELFR. It is to be expected the final ELFR design would not be available until about 2030, with another eight-to-ten years (if not more) required to license the design for construction, complete construction, and begin operation of ELFR. Operation of ELFR could not reasonably be expected before about 2040. Industrial deployment of the European LFR was forecast in 2010 to take place by 2050 (SNETP, 2010).

In the Russian Federation, a small LFR (SVBR-100, 100 MWe) and a somewhat larger but still small BREST-300 (a 300 MWe design). There are plans to build pilot demonstration facilities in Russia for both the SVBR-100 and the BREST-300 designs (Alemberti, et al., 2012). There is no System Arrangement within the Generation IV International Forum for the LFR.

Potential weak points of LFR technology include the high melting point of the coolant, activation of lead and bismuth to form Polonium-210, and in case of nitride fuel use the potential for Carbon-14

[60] The World Nuclear Association has an information paper on accelerator-driven systems available on its website at http://www.world-nuclear.org/info/Current-and-Future-Generation/Accelerator-driven-Nuclear-Energy/#.UVGRdBxwp8E. A longer, more technical "*white paper*" on the subject has been co-authored by experts from national laboratories in Belgium, France, and the United States, http://science.energy.gov/~/media/hep/pdf/files/pdfs/ADS_White_Paper_final.pdf.

[61] MYRRHA is an accelerator-driven, lead-bismuth-eutectic-cooled, fast research reactor concept that is planned to be constructed at the Belgian Nuclear Research Centre in Mol. Construction is foreseen to occur from 2015-2019, and full commissioning of the facility is expected to occur from 2020-2022. The facility is planned to be operational at full power in about 2023. More details are available from the web site, http://myrrha.sckcen.be/.

contamination (Henricksson, 2012). The high melting point of lead can prove to be a problem in case of extended shutdown, as it did for Russian submarine reactors in which the coolant solidified.

Molten Salt Reactor (MSR)

For the molten salt reactor (MSR) GIF concept, only EURATOM and France are in the effort, with Russia and the US as observers, and with Japan and the PRC as occasional participants. EURATOM and France are working on the Molten Salt Fast Reactor (MSRF) concept, while Russia is pursuing the Molten Salt Actinide Recycler & Transmuter (MOSART) concept. In the US, a full size, fluoride salt cooled Advanced High Temperature Reactor (AHTR) is being investigated, along with an SMR concept referred to as SmAHTR (Boussier, 2012). There is no System Arrangement within the Generation IV International Forum for the MSR.

Potential weak points of MSR technology include irradiation of the heat exchangers (because the fuel is present everywhere in the primary system), corrosive salts, and little experience with MSRs (Henriksson, 2012).

10 WHAT IS THE DEPLOYMENT HORIZON FOR NUCLEAR FUSION TECHNOLOGY ON A COMMERCIAL SCALE?

Deuterium-tritium (D-T) nuclear fusion is the process by which light atomic nuclei in plasma are fused together to former heavier nuclei, and in the process release neutrons (needed to produce tritium from lithium in the blanket) and a large amount of energy. The attractiveness of hypothetical fusion power plants (none yet exist) arises from the following aspects:

- D-T fusion power plants would make use of abundantly available fuel (deuterium and natural lithium).
- Very limited greenhouse gas emissions (none from the power plant itself; greenhouse gas emissions are possible as a result of activities during plant construction, decommissioning, separation of deuterium from water, mining and extraction of lithium, and transport of deuterium and lithium to the power plant site).
- D-T fusion power plants would not be capable of producing a meltdown (as in fission reactors), and would also not be capable of producing a runaway reaction; any disruption of the plasma would result in inevitable shutdown of the fusion reaction.
- There is no long-lasting radioactive waste produced during D-T power plant operation.
- In the lithium breeding blanket, neutron activation produces tritium (with a nucleus of one proton and two neutrons), a radioactive substance which decays to non-radioactive Helium-3 with a half-life of 12.32 years; tritium gas released to the environment is not as serious a problem as when tritium oxidizes to tritiated water (DTO, HTO, or T_2O). Tritium is a low-energy beta emitter, which is not dangerous externally (the beta particles cannot penetrate skin), but it is a radiation hazard if inhaled, ingested, or absorbed through the skin (DOE, 2007; HPA, 2007).

The first large fusion reactor (500 MWt) is expected to be the ITER machine under construction at Cadarache in France. Completion is currently (in 2013) scheduled to be complete in 2019, with the first plasma in 2020. Another facility, called the International Fusion Material Irradiation Facility or IFMIF, is planned to be built in Rokkasho, Japan. IFMIF is an accelerator based neutron source for testing fusion materials.

Nuclear fusion design concepts are addressed briefly in Table 9. The ENHUR project considers it unlikely that a nuclear fusion prototype reactor could be constructed and placed in steady-state operation before 2050.

It should be noted, however, that based on ITER construction experience alone (i.e., before ITER operation establishes tokamak fusion technical feasibility on a large scale), the Republic of Korea (in collaboration with the US Department of Enegy's Princeton Plasma Physics Laboratory, PPPL) has embarked on a preliminary concept design for a fusion demonstration reactor called K-DEMO. The Republic of Korea plans to have the K-DEMO completed in the mid-to-late 2030s (i.e., 2035-2039) in Daejeon, under the technical leadership of the National Fusion Research Institute (NFRI). K-DEMO is aimed at producing 1000 MWt of fusion power for periods of several weeks at a time (Keeman, 2013; Princeton, 2012; Park, 2013).

Construction and operation of a commercially viable nuclear fusion power station would have to follow completion of K-DEMO construction and some period of years of operation. Considering likely design, licensing, site preparation, and construction durations, once again it seems clear that operation of a commercial nuclear fusion power plant before 2050 is not to be expected. Even the K-

DEMO path to commercial nuclear fusion power plants supports the EHNUR project conclusion that operation of a commercial fusion power plant before 2050 is unlikely[62].

The Republic of Korea may not be alone in going its own way with a demonstration fusion power plant project (DEMO). According to the Princeton Plasma Physics Laboratory of the U.S. Department of Energy, other countries (including India, Japan, and the People's Republic of China) are contemplating building their own DEMO units (Physorg, 2013; Princeton, 2012).

Officially, the path from ITER to IFMIF to a demonstration power plant (DEMO) to a commercial fusion power plant is planned to yield a commercial fusion power plant in 2050 (EFDA, 2012). Realistically, a commercial fusion power plant is unlikely to be available until 2070, and even this date is dependent on no setbacks or surprises during the preceding steps (Najmabadi, 2011). (The ITER project has seen a series of delays and in the best case is unlikely to develop the first D-T plasma until 2027-2029. Even in late 2012, the European Fusion Development Agreement, EFDA, was acknowledging that the risk exists that the baseline fusion strategy embodied in the ITER machine – a 500 MWt proof-of-principle reactor – would not be able to be extrapolated to a fusion power plant. Moreover, the only alternative to ITER-type technology – a stellarator – was characterized by EFDA as a "*possible long-term alternative*", as if commercial fusion reactor technology based on ITER was not already a long enough term[63].)

Even the most ardent fusion advocates within the European Union – EFDA – foresees in the best case commercialization by 2050 and fusion accounting for 30% of electricity production by 2100 (EFDA, 2012:5). Given the delays so far in getting ITER construction started, and the delay (already) of D-T fusion operation in ITER from 2018-2026 (at the earliest), it seems unlikely to the EHNUR project that this best case projection can be met. In any case, it is evident that nuclear fusion power plants will <u>not</u> be an important factor by 2050[64], and will not be a factor at all by 2030.

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62 A technical readiness level review (TRL) prepared for the Electric Power Research Institute (EPRI) in 2012 concluded that "*no near-term (less than 30 years) fusion options are available to the power industry*" (EPRI, 2012). The European Fusion Development Agreement (EFDA) has published a roadmap that also does not foresee commercial fusion power before 2050 (EFDA, 2012).

63 It bears noting that in November 2001 a so-called Fusion Fast Track expert meeting was held, convened under the Chairmanship of Prof. David King (at the time Chief Scientific Adviser to Her Majesty's Government Office for Science; David King was knighted in 2003) on the initiative of the President of the EU Research Council. The so-called King Panel developed a roadmap for Fusion electricity involving operation of ITER within 10 years (i.e., by 2011), design of a 2 GWth DEMO plant with operation starting by 2030, and the design of a 1.5 GWe Prototype fusion power plant with construction starting in 2040 and operating starting in 2050 (King, 2001). It is now 2013 and construction of the ITER facility has barely started.

64 The HGF Research Collaboration on Nuclear Fusion testified before the German Bundestag Committee for Educatiohn, Research and Technology Assessment in Berlin on 28 March 2001. The testimony projected that even if a demonstration fusion reactor could begin operation in 2037, five years of DEMO operation would be needed in order to begin design of the prototype fusion power plant. Construction of the prototype fusion power plant would, in this projection, begin in 2047, and the plant could be online in 2055 (HGF, 2001). However, it should be noted that this projection was premised on start of ITER construction in 2006 and beginning of experiments in 2014. In 2013, construction of ITER had not yet begun, and the projected year of the first D-T plasma was 2027-2029. The HGF projection would then take the operation of the prototype fusion power plant beyond 2060.

11 SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

The minimum schedule for a Generation III or III+ nuclear power plant project is 10 years from feasibility study to completion of startup testing. Such schedules are only achievable by an experienced utility at an existing nuclear power plant site with a standardized design for which first-of-a-kind engineering (FOAKE) is complete. Under other circumstances (e.g. a utility new to nuclear generation, a greenfield site, a utility in a country without significant nuclear infrastructure, a nuclear power plant design where FOAKE has not yet been accomplished), the schedule would extend from ten to fifteen years and more.

Within the time horizon of the ENHUR project (2030), there are a number of advanced reactor designs available for immediate deployment that could be licensed, constructed, and placed in operation in time contribute to electricity generation by the year 2030. These designs are (further identified in Tables 1-6):

- Eight advanced pressurized water reactors (AP1000, APR-1400, APWR, ATMEA1, EPR, VVER-1000 AES-91, VVER-1000 AES-92, and VVER-1200. As of April 2013, two units of VVER-1000 AES 91 were in operation, and two units of VVER-1000 AES 92 were nearing operation. The first units of AP1000 and EPR were also nearing operation.

- Five boiling water reactors (GE-Hitachi ABWR, ESBWR, Toshiba EU-ABWR), KERENA, and Toshiba US-ABWR). As of April 2013, there were four ABWRs in operation.

- Two pressurized heavy water reactors (ACR-1000 & EC-6). There have been no orders as of April 2013 for either of these designs.

- Three small modular reactors (CAREM-25, KLT-40S, and SMART). As of April 2013, there was one unit of CAREM-25 and two reactors (on one barge) of KLT-40S under construction.

- One Generation IV Very High Temperature Reactor (HTR-PM). As of April 2013, there were two modules under construction.

There are also five remaining Generation II reactor designs that were still (as of Apriil 2013) under construction and for which plans exist to construct additional plants of these designs): BN-800; CNP-300/600; CPR-1000; HWR-700; and OPR-1000.

It is possible, although not very likely in view of the nominal duration of 17 years and the minimum to maximum range of 13-33 years for a nuclear power plant construction project (from feasibility study to commercial operation), that some additional reactor designs with near-term deployment possibilities (2015-2020) could be finished in time to contribute to electricity generation by the year 2030. Plants with such designs would have to be ordered by 2015 -2020 in order to be able to be completed and online by 2030 using the shortest schedule constraints (experienced utility, existing nuclear power plant site, standard design with FOAKE complete, and design certification by the regulatory authority). These designs are identified in Table 7.

There are an increasing number of advanced reactor designs that may become available in time to generate electricity after 2030. These designs are identified in Table 8. New concepts are being advanced all of the time for Generation IV concepts. The list in this Chapter is probably incomplete, but provides a snapshot in time as of April 2013.

Nuclear fusion, although promising as a source of electricity, has no chance of producing electricity before 2030, and little chance of producing electricity on a commercial scale before 2050. Nuclear fusion concepts are briefly identified and discussed in Table 9.

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ANNEX 1 – ACRONYMS AND INITIALISMS

ABB	Asea-Brown-Boveri
ABWR	Advanced Boiling Water Reactor (GE-Hitachi & Toshiba Generation III)
AC	alternating current
ACPR+	Advanced CPR-1000 Plus (CGNPC Generation III+)
ACR-1000	Advanced CANDU Reactor 1000 MWe class (CANDU Energy Inc. Generation III+)
ACRS	Advisory Committee on Reactor Safeguards (US NRC)
ADS	accelerator driven system
AECL	Atomic Energy of Canada, Limited
AEG	Allgemeine Elektricitäts-Gesellschaft (General Electricity Company)
AERB	Atomic Energy Regulatory Board (India)
AES	Russian acronym for nuclear power plant
AFUDC	allowance for funds used during construction
AGR	Advanced Gas-cooled Reactor (Generation II)
AHTR	Advanced High Temperature Reactor
AHWR	Advanced Heavy Water Reactor
AIChE	American Insitute of Chemical Engineers
AID	Agency for International Development (United States)
ALFRED	Advanced Lead Fast Reactor European Demonstrator
ALLEGRO	European Gas Fast Reactor Demonstration Project
AMEC	AMEC plc, a British multinational consultancy, engineering and project management company
ANL	Argonne National Laboratory (US)
ANP	Advanced Nuclear Power (Framatome ANP, former name for AREVA)
ANS	American Nuclear Society
ANSI	American National Standards Institute
ANTARES	AREVA New Technology Advanced Reactor Energy System
AP600	Advanced Passive 600 MWe class (Westinghouse Generation III)
AP1000	Advanced Passive 1000 MWe class (Westinghouse Generation III+)
APR+	Advanced Passive Reactor Plus (KNHP Generation III+)
APR-1400	Advanced Passive Reactor 1400 (KHNP Generation III)
APWR	Advanced Pressurized Water Reactor (Mitsubishi Generation III+)
ARC	Advanced Reactor Corporation
ARIES-ST	Advanced Reactor Innovation and Evaluation Study-Spherical Torus
ASEA	Allmänna Svenska Elektriska Aktiebolaget (General Swedish Electric Company)
ASME	American Society of Mechanical Engineers
ASN	Autorité de sûreté nucléaire (French nuclear regulatory authority)
ASTRID	Advance Sodium Technological Reactor for Industrial Demonstration
ATMEA	Joint Venture between AREVA (France) and Mitsubishi Heavy Industries (Japan)
B&W	Babcock & Wilcox
BARC	Babbha Atomic Research Centre (India)
BATAN	Badan Tenaga Nuklir Nasional (National Nuclear Energy Agency of Indonesia)
BREST	Bystryi Reactor so Svintsovym Teplonositelem (Russian acronym for Fast Reactor with Lead Coolant)
BWR	boiling water reactor
CANDU	Canadium Deuterium Uranium (see PHWR)
CAP-1000	China Advanced Passive 1000 MWe class PWR
CAP-1400	China Advanced Passive 1400 MWe class PWR
CAREM	Central Argentina Modular Element (CNEA small modular reactor)

CDF	core damage frequency
CEA	Commissariat à l'Energie Atomique
CEO	Chief Executive Officer
CERMET	composite ceramic and metal
ČEZ	České Energetické Závody (utility)
CFR	Code of Federal Regulations (United States)
CGNPC	China Guangdong Nuclear Power Company
CNEA	Comisión Nacional de Energía Atómica (National Commission of Atomic Energy, Argentina)
CNNC	China National Nuclear Corporation
CNP	China Nuclear Power
CNPE	China Nuclear Power Engineering Company, Ltd.
CNSC	Canadian Nuclear Safety Commission
COL	Combined Operating License (United States)
CPR1000	Chinese Pressurized Reactor (CGNPC Generation II PWR)
CPY-CP1	900 MWe series pressurized water reactor, Framatome (now AREVA) (Blayais 1-4, Bugey 4 & 5, Dampiere 1-4, Gravelines 1-6)
CRIEPI	Central Research Institute of the Electric Power Industry (Japan)
CSNI	Committee on the Safety of Nuclear Installations (OECD/NEA)
CWIP	construction work in progress
DC	Design Certification (United States)
DC	direct current
DCD	Design Control Document
DCNS	DCNS S.A. (formerly the Direction Technique des Constructions Navales)
DOE	United States Department of Energy
D-T	deuterium tritium
DTO	deuterium-tritium oxide
EBASCO	Electric Bond and Share Company
EdF	Electricité de France (utility)
EC-6	Enhanced CANDU-6 (CANDU Energy Inc. Generation III)
EESTI	Estonian utility (EESTI Energia)
EFDA	European Fusion Development Agreement
EGAT	Energy Generating Authority of Thailand
EHNUR	Evaluation of a Hypothetical Nuclear Renaissance
EIA	Environmental Impact Assessment
ELFR	European Lead-cooled Fast Reactor
ELSY	European Lead System
EM^2	Energy Multiplier Module (General Atomics, Generation IV)
EMI	electro-magnetic interference
ENC	European Nuclear Society
ENEA	National Agency for New Technologies, Energy and Sustainable Economic Development (Italy)
ENEC	Emirates Nuclear Energy Corporation (utility)
ENSI	Eidgenössische Nuklearsicherheitsinspektorat (Swiss nuclear regulatory authority)
EPC	engineering, procurement, and construction
EPR	European Pressurized Reactor (Areva, Generation III+)
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor (GE-Hitachi Generation III+)
ESP	Early Site Permit (US NRC)
ESRA	European Safety and Reliability Association
ETH	Eidgenssische Technische Hochschule
EUR	European Utility Requirements
EURATOM	European Atomic Energy Community

FAO	Food and Agriculture Organization (United Nations)
FAQs	frequently asked questions
FBNR	Fixed Bed Nuclear Reactor
FBR	fast breeder reactor
FOAKE	first-of-a-kind engineering
FSAR	final safety analysis report
FUJI	(Japanese acronym)
g	acceleration of gravity (9.8 m/s^2)
GCFR	Gas Cooled Fast Reactor
GDA	Generic Design Approval (United Kingdom)
GDF SUEZ	Multinational electric utility formed by the merger of Gaz de France and Suez S.A. (since July 2008)
GWd	gigawatt-days
GE	General Electric
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GNEP	Global Nuclear Energy Partnership
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
GT-MHR	Gas Turbine Modular Helium Reactor (General Atomics, Generation IV)
GWe	gigawatts electric
HGF	Helmholtz-Gemeinschaft Deutscher Forschungszentren
HPLWR	High Performance Light Water Reactor
HTO	hydrogen-tritium oxide
HTR	High Temperature Reactor
HTR-PM	High-Temperature Reactor Pebble Bed Module
I&M	Indiana & Michigan Power Company (subsidiary of American Electric Power Company)
IAEA	International Atomic Energy Agency
ICAPP	International Conference on Advances in Nuclear Power Plants
ICTP	Abdus Salam International Centre for Theoretical Physics
IEA	International Energy Agency
IEEE	Institute of Electrical and Electronics Engineers
IFMIF	International Fusion Materials Irradiation Facility
IFNEC	International Framework for Nuclear Energy Cooperation
ILO	International Labor Organization
IMO	International Maritime Organization
IMR	Integrated Modular Water Reactor (Mitsubishi Heavy Industries Generation IV)
INL	Idaho National Laboratory (US) (formerly Idaho National Engineering & Environmental Laboratory, INEEL; and Idaho National Engineering Laboratory, INEL)
INSAG	International Nuclear Safety Group (IAEA)
INET	Institute of Nuclear and New Energy Technology (Tsinghua University, People's Republic of China)
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
INVAP	Argentina high technology company; designer of the CAREM-25 and CAREM-300 advanced reactors
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IPPE	Institute for Physics and Power Engineering (Russia)
IRIS	International Reactor Innovative and Secure (integral PWR, Generation IV)
ISFNT	International Symposium of Fusion Nuclear Technology
ITER	International Thermonuclear Experimental Reactor (formerly; now stated to mean "The Way" in Latin)

IVR	in-vessel retention
JAEA	Japan Atomic Energy Authority
JET	Joint European Torus
JSCWR	Japan Supercritical Water-cooled Reactor
K-DEMO	Korean Demonstration Fusion Power Plant
KAERI	Korea Atomic Energy Research Institute
KALIMER	Korean sodium-cooled fast reactor (KAERI, Republic of Korea, Generation IV)
KAMADO	gas-cooled fast reactor deisgn (CRIEPI, Japan, Generation IV)
KBR	Kellogg Brown & Root
KEPCO	Korea Electric Power Company (parent company of KHNP, KEPCO Engineering & Construction Company, KEPCO Nuclear Fuel Company, KEPCO Plant Service & engineering Company, ICPS, all of which are wholly owned subdiaries of KEPCO)
KERENA	Areva advanced BWR (Generation III+)
KHNP	Korea Hydro and Nuclear Power (Republic of Korea, subsidiary of KEPCO)
KLT-40S	English translation of Russian designation КЛТ40С, a small PWR designed to be built as barge-mounted floating nuclear power plants
KNSP+	Korean Nuclear Standard Plant Plus
kPa	kilopascal (1000 kPa = 1 MPa)
kWe	kilowatts electric (1000 kWe = 1 MWe)
LEADER	Lead Demonstration European Reactor
LEI	Lithuanian Energy Institute
LERF	large early release frequency
LFR	Lead-cooled Fast Reactor
LFTR	Liquid Fluoride Thorium Reactor
LLC	Limited Liability Corporation
LOCA	loss-of-coolant-accident
LRF	large release frequency
m^3	cubic meters
MAGNOX	Magnesium Non-oxidising (a gas-cooled reactor, moderated by graphite, formerly operated in the United Kingdom, Italy, and Japan (the Wylfa Unit 1 reactor, the last operating MAGNOX unit, was still operating as of April 2013, and is expected to close in 2014) (Generation I)
MARS	Multipurpose Advanced Reactor, Inherently Safe
MAST	Mega Ampere Spherical Tokamak
MEE	Ministry of Employment and the Economy (Finland)
MHR	Modular Helium Reactor
MIR-1200	Modernized International Reactor 1200 MWe (gross), a variant of the VVER-1200/491 design marketed in Europe by Atomstroyexport and SKODA JS
MOSART	Molten Salt Actinide Recycler Technology (MSR, Generation IV)
MOX	mixed oxide fuel
MPa	Megapascal
MSR	Molten Salt Reactor
MST	Madison Symmetric Torus
MTA	Institute for Particle and Nuclear Physics, Hungarian Academy of Sciences
MWd/t	megawatt-days per tonne of heavy metal
MWe	megawatts electric
MWt	megawatts thermal
MYRRHA	Multi-purpose Hybrid Research Reactor for High-tech Applications
NASA	National Aeronautics and Space Administration (US)
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NGNP	Next Generation Nuclear Plant
NGO	non-governmental organization

NIKIET	N.A. Dollezhal Research and Deveopment Institute of Power Engineering (Russia; subsidiary of Rosatom)
NNL	National Nuclear Laboratory (UK)
NPCIL	Nuclear Power Corporation of India, Limited (utility)
NPP	nuclear power plant
NPSAG	Nordic Probabilistic Safety Assessment Group
NRA	nuclear regulatory authority (generic term)
NRC	United States Nuclear Regulatory Commission
NTC	Nuclear Training Center
NUSSC	Nuclear Safety Standards Committee (IAEA)
OECD	Organisation for Economic Co-operation and Development
OKB	опытно-конструкторское бюро (Russian acronym for Experimental and Design Organization)
OKBM	Опытное конструкторское бюро машиностроения им. И. И. Африкантова, Russian acronym for OKBM Afrikantov
ONR	Office for Nuclear Regulation (UK Health and Safety Executive)
OPG	Ontario Power Generation (utility) (formerly Ontario Hydro)
OPR-1000	Optimized Power Reactor (KHNP/KEPCO design)
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory (IUS)
PAHO	Pan American Health Organization
PAR	passive autocatalytic recombiner
PBMR	Pebble Bed Modular Reactor
PDHRS	Passive Decay Heat Removal System
PESS	Planning and Economic Studies Section (IAEA)
PFBR	Prototype Fast Breeder Reactor (India)
PGA	peak ground acceleration
PHRS	Passive Heat Removal System
PHWR	pressurized heavy water reactor
PHWR-700	Pressurized Heavy Water Reactor 700 MW (NPCIL design)
PIUS	Process Inherent Ultimate Safety
PNRA	Pakistan Nuclear Regulatory Authority
PPCS	Power Plant Conceptual Study
PPPL	Princeton Plasma Physics Laboratory
PRC	People’s Republic of China
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module (GE-Hitachi, Generation IV)
PSA	probabilistic safety assessment
PSAR	Preliminary Safety Analysis Report
PRIS	Power Reactor Information System (IAEA)
PSAR	preliminary safety analysis report
PSR	Periodic Safety Review
PWR	pressurized water reactor
QA	quality assurance
QC	quality control
R&D	research and development
RBMK	Reaktor Bolshoy Moshchnosti Kanalniy (Russian acronym for "High Power Channel-type Reactor"; a boiling light water cooled reactor moderated by graphite)
RCCS	Reactor Cavity Cooling System
RDIPE	Research and Development Institute of Power Engineering (Russia)
RFX	Consorzio RFX (Italian research consortium, Padua)
RMWR	Reduced Moderation Water Reactor

Rosatom	Росатом (Rosatom Nuclear Energy State Corporation)); ARMZ Uranium Holding Company, Atomenergomash, Atomenergoprom, Atomflot, Atomspetstrans, Atomstroyexport (Атомстройэкспорт), Energoatom, Ototop All-Region Association, OKB Gidropress, and Tekhsnabexport (TENEX), TVEL, VNIAEM, and VNIPIET are all subsidiaries of Rosatom
RVACS	Reactor Vessel Auxiliary Cooling System
SAFR	Sodium Advanced Fast Reactor
SARHRS	Severe Accident Residual Heat Removal System
SBWR	Simplified Boiling Water Reactor (GE)
SCK•CEN	Studiecentrum voor Kernenergie - Centre d'Etude de l'énergie Nucléaire (Belgian Nuclear Research Centre)
SCOR	Simple Compact Reactor
SCWR	Supercritical Water-cooled Reactor (Generation IV)
SDA	Standard Design Approval
SFR	Sodium-cooled Fast Reactor
SIPRON	Protection System for the Brazilian Nuclear Program
SMA	Seismic Margins Analysis
SmAHTR	Small Advanced High Temperature Reactor (ORNL, Generation IV)
SMART	System Ingrated Modular Advanced Reactor
SMR	small modular reactor (also used by IAEA for small and medium sized reactor)
SNERDI	Shanghai Nuclear Engineering Research and Design Institute
SNETP	Sustainable Nuclear Energy Technology Platform (European Union) (established in 2007)
SNSA	Slovenian Nuclear Safety Administration
SNPTC	State Nuclear Power Technology Company (PRC)
SOARCA	State-of-the-Art Reactor Consequence Analyses (US NRC)
SRA	Society for Risk Analysis
SSAR	standard safety analysis report
SSM	Swedish Radiation Safety Authority (Strål Säkerhets Myndigheten)
SSTAR	Small Sealed Transportable Autonymous Reactor
STUK	Finnish Nuclear and Radiation Safety Authority (Finnish acronym)
SVBR-100	Svintsovo-Vismutovyi Bystryi Reaktor' (Russian acronym for lead-bismuth cooled fast reactor)
SWR	Siedewasser Reaktor (German acronym for boiling water water reactor)
T_2O	tritium oxide
TECDOC	Technical Document (IAEA)
TEPCO	Tokyo Electric Power Company
TRISO	tri-isotopic
TRL	Technical Readiness Level
TsNIIMASH	Central Research Institute of Machine Building (Russian acronym for Центральный научно-исследовательский институт машиностроения)
TVA	Tennessee Valley Authority (US government owned utility)
TVO	Teollisuuden Voima Oyj (Finnish utility)
UNECE	United Nations Environment Commission for Europe
UCS	Union of Concerned Scientists (NGO)
UCSD	University of California at Davis
ÚJD	Nuclear Regulatory Authority of the Slovak Republic (Slovak acronym)
ÚJV Rež	Nuclear Research Center, Rež (Czech Republic; subsidiary of ČEZ)
UK	United Kingdom of Great Britain and Northern Ireland
UKHPA	United Kingdom Health Protection Authority
UNEP	United Nations Environment Programme
UNITHERM	Autonomous Cogeneration Nuclear Power Plant
US	United States

URD	Utility Requirements Document (EPRI)
UxC	Ux Consulting, LLC
VAE	Visagino atominė elektrinė (Visaginas Nuclear Power Plant)
VHTR	Very High Temperature Reactor (Generation IV)
VÚJE	VÚJE a.s. (Trnava, Slovakia)
VVER	Vodo-Vodyanoi Energetichesky Reactor (Russian acronym for Water-Water Power Reactor; a type of PWR; sometimes WWER)
VVER-SKD	Supercritical Water-Cooled VVER
WENRA	Western European Nuclear Regulators Association
WHO	World Health Organization (United Nations)
WNA	World Nuclear Association (nuclear industry trade association)
WNN	World Nuclear News (published by WNA)

ANNEX 2 – DEFINITIONS OF TERMS

advanced reactor	a nuclear power reactor with an evolutionary design (compared with Generation II designs) with enhanced safety and risk characteristics
all-in capital costs	total plant capital requirements; includes overnight costs, owner's costs, escalation, and interest during construction
boiling water reactor	a nuclear power reactor in which boiling of coolant occurs in the reactor pressure vessel, with the steam resulting from boiling being directed to the turbine-generator for electricity production
breeder reactor	A nuclear reactor that produces at least as much nuclear fuel as it consumes (and typically more than this) by neutron capture in fertile material present in the core or a blanket surrounding the core. Examples of such fertile material are Uranium-238 and Thorium-232. The breeding ratio of a breeder reactor is the ratio mass of fissile material produced to the mass of fissile material consumed in a nuclear reactor. Note that unless pure fissile material is used in an inert matrix, all reactors breed fissile material, even conventional reactors such as boiling water reactors, pressurized water reactors, and pressurized heavy water reactors.
blanket	in a fast reactor, a blanket is a region within the reactor pressure vessel in which fertile material (Thorium-232 or Uranium-238) is exposted to the neutron flux from the core in order to produce fissile fuel which can be recovered after reprocessing and used to fuel a nuclear reactor
burnup	The percentage of heavy metal fissioned during the period in which the nuclear fuel is exposed during reactor operation. Burnup is normally expressed in terms of megawatt days per tonne (MWd/t) or megawatt days per kilogram (MWd/kg).
calandria	in a pressurized heavy water reactor, the calandrai is the structure that holds the pressure tubes in which the fuel assemblies are located, and which holds the heavy water moderator
condenser	the structure in which steam from the turbine is cooled and condensed by the circulating water system, with the hot condensate being returned to the reactor for further use
confinement	confinement is an enclosure around a nuclear reactor intended to reduce (but not preclude) release of radioactive materials to the environment; some Generation I and Generation II reactors do not have containments (MAGNOX, AGR, VVER-440/230, VVER-440/270, VVER-440/213), but all Generation III and III+ designs have a containment (see below)
containment	a structure around a nuclear reactor designed to withstand with minimal leakage the pressure resulting from design basis accidents (e.g. large pipe break, main steam line break), and

	which is relied upon in severe accidents to control or reduce the release of radioactive materials to the environment (the degree of success in this respect depends on the free volume of the containment, the design margin of the containment to failure, and the avoidance of contact between core debris and the containment liner)
containment bypass	a containment bypass accident is one in which the release pathway is through a pipe or pipes that pass through the containment boundary and result in a release of radioactivity outside the containment (examples include interfacing systems LOCA and steam generator tube rupture)
control rod	a rod holding neutron absorbing material that is inserted into a reactor core in order to stop the fission chain reaction and achieve reactor shutdown
coolant	the substance used to transfer heat from the reactor core, either to steam generators or directly to the turbine(s) (various coolants are possible, including light water, heavy water, helium, carbon dioxide, sodium, lead, and lead-bismuth eutectic)
core catcher	an engineered device or system intended to collect core debris once the reactor pressure vessel fails, and to sequester it in order to prevent containment failure by basemat melt-through in case of a severe accident.
decay heat	heat produced by the reactor core or by core debris after reactor shutdown due to the decay of radioactive materials
defence-in-depth	a hierarchical eployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences, and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive material and workers, members of the public, or the environment, in operational states and, for some barriers, in accident conditions (IAEA, 2007b); there are five levels of defence in depth (adapted from INSAG, 1996): o Level 1: Prevention of abnormal operation and failures; accomplished by conservative design and high quality in construction and operation; o Level 2: Control of abnormal operation & detection of failures; accomplished by control, limiting, and protection systems, and other surveillance features; o Level 3: Control of accidents within the design basis and design extension conditions; accomplished by engineered safety features and emergency operating procedures; o Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents; accomplished by complementary measures and severe accident management; and o Level 5: Mitigation of radiological consequences of significant releases of radioactive material; accomplished by off-stie emergency response and decontamination measures.

detailed design	Sometimes called detailed design & engineering (DD&E); detailed design consists of the cost of the detailed engineering of equipment and plant facilities required to being the design to a state where it can support construction; includes preparation of constructon drawings, the specification of system components, procurement engineering (including preparation of bid packages for suppliers), and general site layout (Rosner & Goldberg, 2011)
deterministic safety assessment	An analysis of nuclear power plant responses to an event, performed according to predetermined rules and assumptions. Deterministic safety assessments can be performed using conservative analytical methods/conservaticve parameter choices (typical) (without explicit uncertainty analysis) or best estimate plus uncertainty (BEPU) methods (becoming more popular).
efficiency (net)	the net generating capacity divided by the reactor thermal power
emergency core cooling system	the system or suite of systems indended to maintain reactor coolant inventory in the event of a loss of coolant accident or a transient event in which coolant is released through relief or safety valves
enrichment	the increase in the percentage concentration of Uranium-235 in material intended for use in fuel rods for a nuclear power plant; natural uranium has a concentration of about 0.7% of Uranium-235, and enrichment is used to bring this enrichment level up to 5% or less for most reactors (although some use fuel enriched to just under 20%)
external hazards	hazards not normally present in the reactor systems, and include hazards such as explosions,flooding, fire, and missiles (projectiles) occurring within the nuclear power plant, and all external man-made and natural phenomena hazards outside the nuclear power plant
fast reactor	a reactor without a moderator, running on fast neutrons (as opposed to thermal neutrons in LWRs and PHWRs); there are three types of fast reactors: breeders (which produce more fuel from a fertile blanket than they consume during operation), break-even (which produce as much fuel as they consume), and burners (which while they produce some fuel are more intended to consume existing stockpiles of plutonium)
fertile	material capable of becoming fissile, by capturing neutrons, possibly followed by radioactive decay (e.g. Thorium-232 and Uranium238)
fissile	material capable of capturing neutron and undergoing nuclear fission (e.g. Uranium-233, Uranium-235, Plutonium-239)
first-of-a-kind engineering	Usually abbreviated FOAKE, this the upfront design and engineering design work required to obtain design certification and/or a construction permit/license
graphite	a crystalline form of carbon; in very pure form, graphite is used as a moderator in some times of reactors (gas-cooled reactors and RBMK reactors)

heavy water	water containing an elevated concentration of molecules with deuterium atoms (hydrogen has one proton and no neutrons in its nucleous, while deuterium as one proton and one neutron)
house loads	the amount of electricity consumed within a nuclear power plant for the operation of plant equipment; calculated as the gross generating capacity minus the net generating capacity
large early release frequency	the total frequency of radioactive releases to the environment which would require implementation of offsite emergency measures but with insufficient time to implement them
large release	a release of radioactive materials to the environment that would require protective measures for the public that could not be limited in area or time (except in retrospect)
light water	ordinary water, as distinct from heavy water; water used in reactors is normally demineralized water
megawatt	1000 kilowatts or 1 million watts
moderator	a material such as light or heavy water or graphite used in a reactor to slow down fast neutrons by collision with lighter nuclei so as to expedite further fission
MOX (Mixed Oxide Fuel)	reactor fuel consisting of both uranium and plutonium oxides (typically about 5% plutonium)
Nth of a kind (NOAK)	nuclear power plants of the same type as the first-of-a-kind (FOAK) in which lessons learned have been incorporated, with this learning curve resulting in shorter construction durations
nuclear island	the portion of a nuclear power plant consisting of the containment and systems in other buildings required for reactor operation and safety
nuclear power plant	Any nuclear fission reactor installation constructed to generate electricity on a commercial scale. Some nuclear power plants, in addition to producing electricity, are dual use facilities that also produce materials used in the construction of nuclear weapons (e.g., Plutonium or Tritium). In some of these latter cases, the production of such materials is unavoidable (such as tritium production in PHWRs), but in other cases the production of electricity by the power plant is itself incidental to the primary purpose of the plant as a production reactor (i.e., the reactor is operated primarily to produce materials for nuclear weapons, and produces electricity as well, possibly to offset the costs of nuclear weapons material production). Examples of such facilities have included the N Reactor at Hanford (Plutonium) and Watts Bar Unit 1 (Tritium) in the United States, the MAGNOX reactors at Calder Hall and Chapelcross (Plutonium and Tritium) in the United Kingdom, the G2 and G3 reactors at Marcoule (Plutonium and Tritium) in France, and the Tomsk reactors at Seversk (Plutonium) in Russia.
practically eliminated	a condition is considered to be practically eliminated if it is physically impossible for the condition to occur, or if the condition can be considered with a high degree of confidence to be extremely unlikely to occur (no confidence level or frequency

	definition has yet been accepted for this term) (WENRA, 2009; IAEA, 2012a)
pressurized water reactor	a nuclear reactor in which the coolant is kept from boiling by a pressurizer, and in which the primary coolant is circulated through a steam generator (heat exchanger) in order to produce steam for power generation by boiling water in the secondary circuit
pressurized heavy water reactor	similar to a pressurized water reactor, except that the moderator (and often the coolant) is heavy water
probabilistic safety assessment	the engineering analysis of a nuclear power plant in which all initiating events are identified, the required system response is identified, and the frequency of occurrence of event sequences resulting in core damage is calculated (Level 1)
reactor pressure vessel	the metal container in which the reactor core and internal reactor structures are located, along with the reactor coolant being circulated
reflector	a purpose built metal structure used to reflect neutrons back into the reactor core to enhance fuel efficiency
residual heat	See decay heat.
severe accident	an event sequence in a nuclear reactor that results in severe fuel damage or fuel melting
spent fuel	used fuel assemblies removed from a reactor after use, either for eventual disposal as high level radioactive waste or for reprocessing
uncertainty	in probabilistic safety assessment (PSA), uncertainty refers the variation in PSA model outputs, and includes random (stochastic or aleatory) uncertainty (such as a pump failing to start due to a random failure), and state-of-knowledge (epistemic) uncertainty which arises from a lack of knowledge or lack of scientific understanding; state-of-knowledge uncertainties include parameter uncertainty, model uncertainty, and completeness uncertainty (which itself includes the truly unknown and unexpected, and also includes intentional exclusions from the scope of the PSA or from the model)

ANNEX 3 – PSA RESULTS FOR GENERATION III & III+ ADVANCED REACTORS AND GENERATION II REACTORS

The reported CDFs for advanced reactors are as follows (where the PSA scope is not specifically stated, ENHUR assumes the scope is limited to internal events occurring at full power):

- ABWR, 1.6×10^{-7}/a (NRC, 1994)
- ACR-1000 PHWR, 3×10^{-7}/a (Shalaby, 2010)
- AP1000, 4.6×10^{-7}/a (ONR, 2011a) (internal & external events,all plant states)
- APR1400, 2.25×10^{-6}/a (Seo, 2011)
- APWR, 1.2×10^{-6}/a (Kuroiwa, 2008)
- EC-6 PHWR, 4.6×10^{-6}/a (Pöyry, 2008)
- EPR, 7.1×10^{-7}/a (ONR, 2011b) (internal & external events, all plant states)
- ESBWR, 1.7×10^{-8}/a (Powell, 2011)
- KERENA BWR, 8.0×10^{-8}/a (AREVA, 2010)
- Toshiba US-ABWR, 9.8×10^{-8}/a (Toshiba, 2010)
- VVER-1200/491, 5.94×10^{-7}/a (BFE, 2012)
- VVER-1200/392, 6.1×10^{-7}/a (BFE, 2012)
- VVER-1200, 5.4×10^{-7}/a to 6.1×10^{-7}/a (Morozov & Tokmachev, 2011)

The internal events PSA considering at power and shutdown accidents for the Generation III VVER-1000 units at Tianwan has an outlier CDF value of 1.3×10-5/a (Bo, 2011). Due to lack of details, it is not clear what is causing such a high CDF for a Generation III design.

The reported CDFs for Generation II BWRs, PHWRs, PWRs are as follows (note that the scopes of the PSAs vary, and some results reported are mean CDF and others are point estimate; the point here is not the accuracy of the individual numbers, but rather to gain an impression of the CDF values for operating Generation II reactors):

- 1.05×10^{-6}/a – Browns Ferry Unit 2 (BWR/4 Mark I) (TVA, 2002)
- 1.9×10^{-6}/a – Current reactor minmum value (NRC, 2006)
- 1.99×10^{-6}/a – Browns Ferry Unit 3 (BWR/4 Mark I) (TVA, 2002)
- 2.1×10^{-6}/a – Pickering B (CANDU) (CNSC, 2010)
- 2.12×10-6/a – Borssele (Siemens PWR) (Ministerie EL&I, 2012)
- 2.5×10^{-6}/a – Neckarwestheim Unit 2 (Siemens Konvoi PWR) (Kröger, 2012)
- 4.7×10^{-6}/a – Angra Unit 1(Siemens PWR) (CNCN, ELETRONUCLEAR & SIPRON, 2010)
- 6.01×10^{-6}/a – South Ukraine Unit 1 (VVER-1000/302) (Bozhkoa, et al., 2009)
- 6.6×10^{-6}/a – Kola Unit 3 (VVER-440/213) (Rosatom, 2010)
- 8.8×10^{-6}/a – RBMK units in Russia, low end (Rosatom, 2010)
- 9.38×10^{-6}/a – Darlington NGS (CANDU) (includes fire, flooding & seismic) (OPG, 2012)
- 1.3×10^{-5}/a – Olkiluoto Units 1 & 2 (Asea-Atom BWR) (STUK, 2010)
- 1.5×10^{-5}/a – Dukovany Units 1-4 (VVER-440/213) (Czech Republic, 2010)
- 1.53×10^{-5}/a –Mochovce Units 1 & 2 (VVER-440/2123) (ÚJD, 2010)
- 2×10^{-5}/a – Paks Units 1-4 (VVER-440/213) (Hungary, 2010)
- 2.23×10^{-5}/a – Bohunice V2 (VVER-440/213) (ÚJD, 2010)
- 3.32×10^{-5}/a – Temelín 1 & 2 (VVER-1000/320) (Czech Republic, 2010)
- 3.5×10^{-5}/a – BN600 sodium-cooled fast reactor (Rosatom, 2010)

- 3.6×10^{-5}/a – Pickering A (CANDU) (CNSC, 2010)
- 3.6×10^{-5}/a – Current US reactors average (mean) (NRC, 2006)
- 4.03×10^{-5}/a – Point Lepreau (CANDU) (CNSC, 2010)
- 4.4×10^{-5}/a – VVER-1000/320 PWRs in Russia, low end (Rosatom, 2010)
- 4.86×10^{-5}/a – D.C. Cook (Westinghouse 4-loop PWR) (I&M, 2003)
- 6.0×10^{-5}/a – Loviisa Units 1 & 2 (VVER-440-213 with containment) (STUK, 2010)
- 6.0×10^{-5}/a – Balakovo Unit 4 (VVER-1000/320) (Morozov & Tokmachev, 2012)
- 6.4×10^{-5}/a – Bruce A Units 3 & 4 (CANDU) (CNSC, 2010)
- 8.6×10^{-5}/a – VVER-1000/320 PWRs in Russia, high end (Rosatom, 2010)
- 1.3×10^{-4}/a – Kola Unit 4 (VVER-440/213) (Rosatom, 2010)
- 1.7×10^{-4}/a – RBMK units in Russia, high end (Rosatom, 2010)
- 2.4×10^{-4}/a – Current US reactors, maximum (NRC, 2006)

ANNEX 4 – TABLES

TABLE 1: REACTOR DESIGNS AVAILABLE FOR IMMEDIATE DEPLOYMENT

Generation III and Generation III+ Pressurized Water Reactors
AP1000 (Westinghouse) (Note 1)
APR-1400 (Korea Hydro Nuclear Power) (Note 2)
APWR (Mitsubishi Heavy Industries) (Note 3)
ATMEA1 (AREVA & Mitsubishi Heavy Industries) (Note 4)
EPR (AREVA) (Note 5)
VVER-1000 AES-91 (OKBM & Atomstroyexport) (Note 6)
VVER-1000 AES-92(OKBM & Atomstroyexport) (Note 7)
VVER-1200 AES-2006 (OKBM & Atomstroyexport) (Note 8)
Generation III and Generation III+ Boiling Water Reactors
ABWR (GE-Hitachi) (Note 9)
ESBWR (GE-Hitachi) (Note 10)
EU-ABWR (Toshiba) (Note 11)
Kerena (AREVA) (Note 12)
US ABWR (Toshiba) (Note 13)
Generation III and Generation III+ Pressurized Heavy Water Reactors
ACR-1000 (CANDU Energy, Inc.) (Note 14)
EC-6 (CANDU Energy, Inc.) (Note 15)
Small Modular Reactors (SMRs)
CAREM-25 (CNEA & INVAP) (Note 16)
KLT-40S Floating NPP (OKBM Afrikantov) (Note 17)
System-Integrated Modular Advanced Reactor (SMART) PWR (KAERI) (Note 18)
Generation IV Reactors
HTR-PM 200 (INET, Tsinghua University) (Note 19)
Generation II Reactors Still Available for Deployment
BN-800 (Rosenergoatom) (Note 20)
CNP-300/CNP-600 (China National Nuclear Corporation) (Note 21)
CPR-1000 (China Guangdong Nuclear Power Company) (Note 22)
PHWR-700 (Nuclear Power Corporation of India, Ltd., NPCIL) (Note 23)
OPR-1000 (Korea Hydro Nuclear Power) (Note 24)

Notes on Table 1

Note 1 AP1000 – The detailed design of AP1000 is complete, and AP1000 units are under construction in the People's Republic of China (Sanmen & Haiyang sites) and the United States (Virgil Summer & Vogtle sites). A revised Design Certification was issued in the United States to account for aircraft crash resistance changes to the design. AP1000 was being reviewed in the United Kingdom under the Generic Design Assessment (GDA) process, but Westinghouse halted the review awaiting a U.K. customer. The AP1000 design was evaluated as complying with the EUR in 2007. Nuclear power plant construction bids were submitted for AP1000 construction in the Canada, the Czech Republic (with a decision pending in 2013), India, Lithuania, Poland, South Africa, and the United Kingdom (but did not win the tender in the UK). The AP1000 design was cited in a number of U.S. Combined Operating License applications which had not yet received NRC approval as of March 2013 (Levy; Shearon Harris; Turkey Point; William States Lee III). The Tennessee Valley Authority decided to finish construction of Bellefonte Unit 1 (a Babcock & Wilcox PWR) before deciding whether to proceed with AP1000 units at that site. As of April 2013, Westinghouse had a website dedicated to the design (http://www.ap1000.westinghousenuclear.com/)/

Note 2 APR-1400 – The APR-1400 detailed design is complete, and APR-1400 units are under construction in the Republic of Korea (Shin Kori & Shin Ulchin sites) and the United Arab Emirates (Barrakh site). The APR-1400 design was certified in the Republic of Korea in 2002, and KHNP plans to submit a Design Certification application in the United States in 2013, with a decision expected in 2017. Nuclear power plant construction bids have been submitted for APR-1400 construction in Belarus, Finland (did not get the contract), the People's Republic of China, Poland, and Turkey (did not get the contract). As of April 2013, KHNP had a website dedicated to the design (http://www.apr1400.com/).

Note 3 APWR – The APWR detailed design is complete. A Design Certification application has been submitted in the United States, with a decision expected in 2016. Nuclear power plant construction bids have been submitted for the Mitsubishi APWR in Finland, Japan (Tsuruga 3 & 4 although this project may not proceed due to the proximity of a fault that was recently evaluated as active), Turkey (did not get the contract), and the United States (Comanche Peak & North Anna sites). As of April 2013, Mitsubishi Nuclear Energy Systems had a website dedicated to the US-APWR design (https://www.mnes-us.com/us-apwr/overview) for the APWR in Japan (http://www.mhi.co.jp/atom/hq/atome_e/apwr/index.html), and for the EU-APWR (http://www.mhi.co.jp/en/nuclear/euapwr/).

Note 4 ATMEA1 – The ATMEA1 detailed design is complete. The IAEA reviewed the ATMEA1 design against SF-1, DS348, and NS-R-1. ATMEA1 construction bids have been submitted for ATMEA1 in Jordan (decision pending), Turkey (did not get the contract), and Vietnam. The ATMEA1 design has been promoted by company in Brazil, Canada, Hungary, Indonesia, Malaysia, and Slovenia. The ATMEA1 design was pre-qualified for forthcoming tender in Argentina by Nucleoeléctrica Argentina. As of April 2013, a web site was available for the design (http://www.atmea-sas.com/scripts/ATMEA/publigen/content/templates/Show.asp?P=57&L=EN), and AREVA also had an ATMEA1 web site (http://www.areva.com/EN/global-offer-418/atmea1-a-pressurized-water-reactor-for-all-networks.html).

Note 5 EPR – The detailed design is of EPR is complete, and units are under construction in Finland (Olkiluoto Unit 3), France (Flamanville Unit 3), and the People's Republic of China (Taishan site). The EPR was assessed as complying with the European Utility Requirements (EUR) December 1999. The EPR design was approved under the Generic Design Approval process in the United Kingdom, and was also approved in 2004 in France. The U.S. EPR is under Design Certification review in the United States, with a decision expected in 2014. Nuclear power plant construction bids have been submitted for EPR in Canada, the Czech Republic (decision pending), Finland (did not get the contract), India (got the contract for 2 units at Jaitapur), South Africa, the United Arab Emirates (did not get the contract), the United Kingdom (possible construction at the recently approved Hinkley C site), and the United States (two projects cancelled at Callaway & Calvert Cliffs). As of April 2013, a UK-EPR website was available (http://www.epr-reactor.co.uk/scripts/ssmod/publigen/content/templates/Show.asp?P=57&L=EN), as was an AREVA website (http://www.areva.com/EN/global-offer-419/epr-reactor-one-of-the-most-powerful-in-the-world.html),

Note 6 VVER-1000/428, AES 91 – This Generation III detailed design is complete, and two units are in operation in the People's Republic of China, and four more planned for the same site (Tianwan). The design has a double containment and a core catcher. The estimated core damage frequency for Tianwan Units 1 & 2 is 3.3×10^{-6}/a, and the estimated large release frequency estimated is 6.4×10^{-8}/a. The inner containment is a steel-lined, 1.2 meter thick, pre-stressed concrete structure, and the outer containment is a 0.6 meter thick reinforced concrete secondary containment (Ermolaev, 2012).

Note 7 VVER-1000/412, AES92 - The Generation III design is complete, and it was assessed as complying with the EUR 2007. Two units are under construction in India at Kudankulam, with Unit 1 probably starting up in 2013. The related VVER-1000/V-392B units are under construction in Ukraine at Khmelnitsky Units 3 & 4. This design also includes a double containment and a core catcher. The inner containment is a steel-lined, pre-stressed concrete structure, and the outer containment is a reinforced concrete structure (Ermakov & Rousselot, 2005; Ermakov & Rousselot, 2007). The Kudankulam design is described in some detail in a 2006 journal article (Agrawal, Chauhan & Mishra, 2006).

Note 8 VVER-1200 – There are two VVER-1200 designs, both of which are complete. The VVER-1200/491 design is an active systems design, and is under construction at Leningrad II; this is also the version being offered for construction outside Russia. The VVER-1200/392B design includes passive systems, and is under construction at Novovoronezh II and Kaliningrad in Russia, with additional units planned at Nizhny Novgorod in Russia. So far, this version is not being offered outside Russia. The VVER-1200 (although it is not clear which version, but probably VVER-1200) is planned to be constructed at the first nuclear power plant in Vietnam at Ninh Thuan. As of April 2013, Rosatom had available a 40-page document on the VVER-1200/491 design under construction at Leningrad II (Rosatom, 2011a) (http://www.rosatom.ru/wps/wcm/connect/spb_aep/site/resources/f3b59380478326aaa785ef9e1277e356/AES-2006_2011_EN_site.pdf).

The Czech-Russian consortium (Skoda JS and Atomstroyexport) offering the MIR-1200 in the Czech Republic (Temelín Units 3 & 4) is offering a VVER-1200/491 design called the MIR-1200 (Modernized International Reactor). Rusatom Overseas (the international arm of Rosatom) has announced that it may seek Design Certification in the U.S., and may request Generic Design Approval in the U.K., both for the VVER-1200/491.

Note 9 ABWR (GE-Hitachi) – The detailed design of the GE-Hitachi ABWR is complete, and the design was evaluated against the EUR in December 2001. The GE-Hitachi ABWR received Design Certification in the United States in 1997, and an application to extend the Certification was filed in 2010. Nuclear power plant construction bids submitted for GE-Hitachi ABWR construction have been submitted in Finland (the winning bid), the United Arab Emirates (did not get the contract), the United Kingdom (Hitachi purchased Horizon Nuclear Power in the U.K. in 2012, and it is reported that Hitachi would build 2 or 3 ABWRs at two sites – Oldbury & Wylfa – once the ABWR receives Generic Design Approval in the UK (Wikipedia, 2013) and in the United States. As of April 2013, GE-Hitachi had available a general description of the ABWR (GE-Hitachi, 2007).

Note 10 ESBWR – The detailed design of ESBWR is complete. ESBWR is under Design Certification review in the United States, with certification expected in 2013 or 2014. Nuclear power plant construction bids have been submitted for ESBWR construction in Finland, India, Poland, and the United States. Combined Operating License applications for ESBWR construction in the U.S. were filed for Comanche Peak Units 3 & 4, Fermi Unit 3, Grand Gulf Unit 3, North Anna Unit 3, River Bend Unit 3, Victoria County Units 1 & 2. After delays, the COL applications for Comanche Peak and North Anna were amended to select Mitsubishi APWRs. All other ESBWR COL reviews have been suspended at the utilities' requests except for Fermi Unit 3. As of April 2013, GE-Hitachi had available a general description of the ESBWR (GE-Hitachi, 2011b).

Note 11 Toshiba EU-ABWR – The detailed design for Toshiba's EU-ABWR is complete. A nuclear power plant construction bid was submitted for the Toshiba EU-ABWR in Finland (Toshiba won the bid) for the Vennovoima Hanhikivi site. The Japanese ABWRs at Kashiwazaki-Kariwa (Units 6 & 7) were designed by Toshiba.

Note 12 KERENA – The detailed design is complete. KERENA was certified as meeting the EUR requirements In February 2002. The KERENA design used the Gundremmingen B & C units as a reference plant.

The isolation condensers used for passive cooling were tested at Forschungszentrum Jülich. Nuclear power plant construction bid submitted for KERENA in Finland. As of April 2013, AREVA was maintaining a website for KERENA (http://www.areva.com/EN/global-offer-420/kerena-a-midpower-boiling-water-reactor.html).

Note 13 Toshiba US ABWR – The detailed design is complete. Toshiba US-ABWR is undergoing Design Certification review in the United States. Nuclear power plant construction bid for US-ABWR submitted in the United States. A drawing (apparently copyrighted as indicated on the main page of the website) of the Toshiba ABWR design shows an adverse turbine orientation vis-à-vis the reactor building (http://www.sargentlundy.com/images/ToshibaABWR-300dpi.jpg).

Note 14 ACR-1000 – The detailed design is complete (3187 MWt). ACR-1000 is undergoing design review in Canada, and started the Generic Design Acceptance procedure in the United Kingdom, then withdrew to concentrate on the Canadian Market. The ONR Phase 2 review was completed before the design was withdrawn from the GDA process (ONR, 2008). Nuclear power plant construction bids submitted for construction in Canada. As of April 2013, an ACR-1000 Technical Description Summary was available (AECL, 2010), and separate, shorter ACR-1000 Technical Summary was also available (AECL, 2007).

Note 15 CANDU EC6 – The detailed design for EC6 is complete. The EC6 design is undergoing design review in Canada. Nuclear power plant construction bids have been submitted for EC6 construction in Argentina, Canada, Jordan, and Turkey (did not get the contract). There are serious questions about the outage duration and cost involved in the mid-life retubing/refurbishment outage. Wolsung Unit 1 was retubed in a fixed price contract with an expected duration of 18 months; the actual outage took 27.5 months (WNN, 2011a). The Point Lepreau retubing/refurbishment was budgeted at $1.4 billion and an outage duration of 18 months; the actual duration was 56 months and the cost was greater than $3 billion. The estimated cost of the retubing/refurbishment outage for the Gentilly 2 plant was $4.3 billion. After receiving this estimate, the utility cancelled the planned refurbishment and shut down the plant for decommissioning in December 2012 (Peachey, 2013). The Embalse plant in Argentina was also scheduled for a retubing/refurbishment outage (along with a capacity increase of 35 MWe), with an estimated duration of 20 months and a cost of $1.37 billion (WNN, 2011b). The size of the CANDU 6 units is about the same, and the number of pressure tubes is identical (380). As of April 2013, CANDU Energy had available a Technical Summary for the EC6 design (CANDU Energy, 2012).

Note 16 CAREM-25 – The detailed design is complete. A prototype CAREM-25 module is under construction in Argentina adjacent to the Atucha site. CAREM-25 is a 100 MWt/27 MWe net integral PWR with the steam generators placed in the reactor vessel. The CAREM concept was introduced in 1984.

Note 17 KLT-40S – The detailed design is complete, and construction of two-unit pilot plant (barge mounted) was started in April 2007. As of April 2013, OKBM Afrikantov had available a description of the KLT-40S design (OKBM Afrikantov, Undated 1).

Note 18 SMART Integral PWR – The detailed design is complete. Standard design approval was granted in the Republic of Korea in 2012. Plant construction is under discussion. KAERI announced that it was working to make the SMART design fully passive instead of the hybrid design that was granted Standard Design Approval (SDA) in the Republic of Korea in July 2012. KAERI planned, as of early 2013, to complete the design modifications, and apply to the Nuclear Safety and Security Commission for an amended SDA (UxC, 2013). A preliminary Level 1 PSA (internal events at power) estimated a CDF of 4.7×10^{-5}/a (Cho, Lee & Kim, 2012).

Note 19 HTR-PM 200 – The detailed design of HTR-PM 200 is complete, and two modules were under construction at the Shidaowan site in the People’s Republic of China. The estimated cost of the two-module prototype plant was $476 million. If successful, the prototype plant could serve as the basis for construction of a series of these modules.

Note 20 BN-800 – The BN-800 detailed design is complete. BN-800 is a sodium-cooled fast breeder reactor under construction at the Beloyarsk site in Russia. Two additional BN-800 units are planned to be constructed in the People’s Republic of China. OKB Gidropress claims Generation III+ status for the BN-800 (Mokhov & Trunov, 2009). Based on the limited publicly available information on the design, this claim cannot be sustained, and the EHNUR project lists the BN-800 as Generation II (see Section 11 of this Chapter). The design power parameters are 2100 MWt/880 MWe. As of April

2013, Rosatom had a plant description document available (Rosatom, 2011b) as did OKBM Afrikantov (OKBM Afrikantov, Undated 2). Rosenergoatom (a subsidiary of Rosatom) had a website about the design as of April 2013 (http://www.rosatom.ru/wps/wcm/connect/rosenergoatom/belnpp_en/about/).

Note 21 CNP-300 and CNP-600 – The detailed design is complete for both CNP-300 and CNP-600. Three CNP-300 units are operating (two at Chasma in Pakistan, one at Qinshan in the People's Republic of China), and two more units are planned at Chasma. The CNP-600 units have twice the electrical capacity, and four CNP-600 units are operating at Qinshan in the People's Republic of China. Further CNP-600 units are planned at Qinshan and Changjiang. The CNP-600 units are stated by CNNC to be free of French intellectual property rights (the CNP-300 units were developed jointly by CNNC, Westinghouse, and AREVA).

Note 22 CPR-1000 – The detailed design is complete, and a number of construction projects are underway and planned in the People's Republic of China (PRC). CPR-1000 is based on Gravelines Unit 6 (France) and Ling Ao Units 1 & 2. The plant has a capacity of 1086 MWe gross and 1021 MWe net, with an 18 month refueling interval. The design is consistent with Generation II characteristics, with the exception that it includes a core catcher. The design has not been marketed outside the PRC as of April 2013.

Note 23 HWR-700 – The detailed design is complete, and under construction in India (Kakrapar Units 3 & 4; Rajasthan Units 7 & 8). The net capacity of the HWR-700 is 630 MWe. The estimated overnight cost for an HWR-700 unit is $1700/kWe according to a government statement to the Indian Parliament in August 2012 (Times of India, 2012). A Level 1 PSA has been performed on the design; the calculated CDF for internal events at power is 5.32×10^{-8}/a (Guptan, 2012). The design has not been marketed outside India as of April 2013.

Note 24 OPR1000 – This Generation II standard design (based on former Combustion Engineering System 80+ design certified by US NRC), and formerly called the Korean Nuclear Standard Plant (KNSP), is complete, with 11 units in operation and one remaining unit construction in the Republic of Korea. KHNP had, as of April 2013, a website dedicated to the OPR1000 design (http://www.opr1000.co.kr/). OPR1000 is an improved design based on the System 80+ design certified in the U.S., but it is not a Generation III design.

TABLE 2: GENERATION III AND III+ PRESSURIZED WATER REACTORS (PWRS) AVAILABLE FOR IMMEDIATE DEPLOYMENT

Design	Vendor	Type	Coolant	Net MWe	Notes
AP1000	Westinghouse	2-loop PWR	Light water	1117	Generation III+; double containment & in-vessel retention included in design; Design Certification in the U.S.; EUR certificatiojn in 2007; units under construction and planned in the People's Republic of China and US; cost for two units $11-14 billion
APR-1400	KHNP	2-loop PWR	Light water	1400	Generation III; design certified in Republic of Korea; units under construction in Republic of Korea and United Arab Emirates; cost estimated at $16.4 billion for four units in UAE (original cost estimate for Shin-Kori Units 3 & 4 was $6.3 billion in 2009 prior to start of construction)
APWR	Mitsubishi	4-loop PWR	Light water	1650	Generation III; units under construction in Japan; offered in Finland and U.S. (selected for North Anna Unit 3 and Comanche Peak Units 3 & 4); cost estimated at $9.5 for Tsuruga Units 3 & 4 in Japan (before construction start); single containment
ATMEA1	AREVA & Mitsubishi	3-loop PWR	Light water	1150	Generation III; no units under construction or in operation; offered in Argentina, Canada, Jordan & Turkey; single containment
EPR	AREVA	4-loop PWR	Light water	1650	Generation III+; units under construction in Finland, France, and People's Republic of China; offered in U.S.; EPR granted Generic Design Approval in the UK; EUR certification in July 2009; double containment and core catcher included; cost estimated to be €8-8.5 billion (originally estimated at €3-3.3 billion)
VVER-1000 AES 91 & AES 92	OKBM	4-loop PWR	Light water	933	Generation III; AES 92 EUR certification in 2007; double containment & core catcher included; 2 units in operation at Tianwan, People's Republic of China (AES 91), and 2 units nearing operation in India (AES 92); original estimated costs was $2.5-3 billion for two units, final estimated cost was $3.8 billion for two units; two additional AES 91 in India expected to cost $12 billion or more

Design	Vendor	Type	Coolant	Net MWe	Notes
VVER-1200 AES 2006	OKBM	4-loop PWR	Light water	1082	Generation III+; units under construction in Belarus, Russia, and Turkey; offered in Jordan; double containment & core catcher included; two designs, one with mostly active systems (VVER-1200/491) and one with mostly passive systems (VVER-1200/392M); estimated cost of construction $8 billion U.S. for two VVER-1200/491 units in Kaliningrad, $10 billion for two VVER-1200/491 units at Ostrovets in Belarus, $20 billion for four units at Akkuyu in Turkey

TABLE 3: GENERATION III AND III+ BOILING WATER REACTORS (BWRS) AVAILABLE FOR IMMEDIATE DEPLOYMENT

Design	Vendor	Type	Coolant	MWe	Notes
ABWR	GE-Hitachi	BWR	Light water	1350	Generation III; four units in operation in Japan; additional units under construction in Taiwan and Japan; offered in U.S.
ESBWR	GE-Hitachi	BWR	Light water	1333	Generation III+; offered in Finland, India, Poland & US; no units under construction or in operation.
EU-ABWR	Toshiba	BWR	Light water	1600	Generation III+; offered in Finland; protected against aircraft crash; double containment & core catcher provided; no units under construction or in operation; offered in Finland
Kerena	AREVA	BWR	Light water	1250	Generation III+; EUR certified in 2002; formerly known as SWR-1000; offered in Finland; no units under construction or in operation.
US ABWR	Toshiba	BWR	Light water	1400	Generation III; offered in the U.S.; no units under construction or in operation.

TABLE 4: GENERATION III AND III+ PRESSURIZED HEAVY WATER-MODERATED REACTORS (PHWRS) AVAILABLE FOR IMMEDIATE DEPLOYMENT

Design	Vendor	Type	Coolant	Net MWe	Notes
ACR-1000	CANDU Energy Inc. (SNC Lavalin)	PHWR	Light water	1082	Generation III+; refueled online; light water cooled and heavy water moderated; offered in Canada; no units in operation or under construction.
EC-6	CANDU Energy Inc. (SNC Lavalin)	PHWR	Heavy water	690	Generation III; refueled online; mid-life refurbishment required to re-tube the fuel and calandria tubes; single pre-stressed concrete containment; cost estimated between $3.8-5.9, including mid-life refurbishment & decommissioning; offered in Argentina, Canada, Jordan & Turkey; no units in operation or under construction.

TABLE 5: SMALL MODULAR REACTORS (SMRS) AND GENERATION IV REACTORS AVAILABLE FOR IMMEDIATE DEPLOYMENT

Design	Vendor	Type	Coolant	Net MWe	Notes
CAREM-25	CNEA & INVAP	Integral PWR	Light water	25	SMR; integral PWR; 16-17 month refueling cycle, replacing 50% of the core; passive residual heat removal system; passive gravity-driven borated water injection as a second shutdown system, and passive low pressure injection system (accumulators); pressure suppression containment constructed from reinforced concrete, 0.5 MPa design pressure; in-vessel retention (IVR) and PARs provided; under construction in Argentina adjacent to the Atucha site.
KLT-40S	OKBM Afriakntov	PWR	Light water	2×35	SMR; barge-mounted PWRs (2 reactors per vessel), with the or barge towed to the planned location and connected via a purpose-built dock to the land-based power grid; two units under construction for deployment at Vilychinsk on the Kamchatka peninsula near a Russian naval base and Petropavlovsk.
SMART	KAERI	Integral PWR	Light water	100	SMR; System-Integrated Modular Advanced Reactor; design certified in Republic of Korea; capacity when supplying 40,000 tonnes per day of desalinated water is 90 MWe; capacity when supplying district heating is 82 MWe; passive decay heat removal system; refueling every 36 months (conventional PWRs are 12-24 months)
HTR-PM 200	INET, Tsinghua University	Gas-cooled reactor	Helium	200	Generation IV gas-cooled reactor; under construction at Shidaowan in the People's Republic of China; no units in operation; two modules feed a single turbine.

TABLE 6: REMAINING GENERATION II REACTORS STILL UNDER CONSTRUCTION AND FOR WHICH ADDITIONAL CONSTRUCTION PLANS EXISTED IN APRIL 2013

Design	Vendor	Type	Coolant	Net MWe	Notes
CNP-300	CNNC	2-loop PWR	Light water	300	Generation II PWR; three units operating (one in the People's Republic of China), two under construction in Pakistan; core damage frequency estimated at 1.52×10^{-5}/a (scope unclear); reactor cavity flooding system for in-vessel retention (IVR) and PARs for combustible gas control; single containment with a steel liner
CNP-600	CNNC	3-loop PWR	Light water	610	Generation II PWR; four units operating at Qinshan, People's Republic of China; two more under construction at Changjiang in the People's Republic of China, and additional units are planned at Qinshan.
CPR-1000	CGNPC	3-loop PWR	Light water	1021	Generation II PWR; single pre-stressed concrete containment with a steel liner (the CPR-1000 containment is small in terms of free volume at 49,400 m^3 and a power level of 2895 MWt – or 17.1 m^3/MWt – compared with Temelin, similar power level of 3000 MWt, with free volume of 67,000 m^3 – or 22.3 m^3/MWt); 22 CPR-1000 units under construction in the People's Republic of China
PHWR-700	NPCIL	PHWR	Heavy water	630	Generation II PHWR; refueling at power; double containment (pre-stressed concrete inner containment with steel liner, reinforced concrete secondary containment); design includes a passive decay heat removal system (PDHRS); 29% net efficiency; 4 units under construction in India.
OPR-1000	KHNP	2-loop PWR	Light water	1000	Intended for Asian market (Vietnam, Indonesia); as of March 2013, nine units were operating in the Republic of Korea, with one more under construction; single pre-stressed concrete containment with a steel liner

TABLE 7: REACTOR DESIGNS POTENTIALLY AVAILABLE FOR NEAR-TERM DEPLOYMENT (2015-2020)

Generation III+ PWRs and BWRs
ACP300/ACP600 (CNNC) (Note 1)
ACPR1000+ (CGNPC) (Note 2)
APR+ (KHNP) (Note 3)
CAP1400/AP1400 (SNPTC) (Note 4)
VVER-1200A/501 (OKBM, Atomstroyexport) (Note 5)
VVER-1500/1800 (OKBM, Atomstroyexport) (Note 6)
Pressurized Heavy Water Reactors (PHWRs)
Advanced Heavy Water Reactor, AHWR (BARC) (Note 7)
Small Modular Reactors
HI-SMUR/SMR 160 (Holtec International) (Note 8)
IRIS (Consortium) (Note 9)
mPOWER (Babcock & Wilcox) (Note 10)
NuScale (NuScale Power LLC) (Note 11)
RITM-200 (OKBM Afrikantov) (Note 12)
VVER-300/478 (OKB Gidropress) (Note 13)
Westinghouse SMR (Westinghouse) (Note 14)
Generation IV Reactors
4S (Toshiba & CRIEPI) (LFR) (Note 15)
BN-1200 (IPPE) (Note 16)
SVBR-100 (AKME Engineering) (LFR, lead-bismuth) (Note 17)

Notes on Table 7

Note 1 ACP300 and ACP600 – The ACP300 and ACP600 designs are Generation III PWRs designed by CNNC in the People's Republic of China. The plants will have a double containment, an 18-24 month refueling cycle, digital I&C, and an intended 60-year plant life.

Note 2 ACPR1000+ – ACPR 1000+ is an 1150 MWe PWR design concept building on the CPR1000 design to develop a Generation III technology by CGNPC in the People's Republic of China. Compared with the CPR1000 design, the ACPR1000+ design features a 0.3g PGA seismic design (vs. 0.2g for CPR1000), a double containment (vs. single containment for CPR-1000), and an in-containment refueling water storage tank (vs. external refueling water storage tank for CPR1000). Unlike the CPR1000, China Guangdong Nuclear Power Company (CGNPC) claims independent intellectual property rights for the design, and plans to offer the ACPR1000+ for construction outside the People's Republic of China. The ACPR1000+ is stated in Chinese media to be compliant with both the EPRI URD and European Utility Requirements (EUR). According to the World Nuclear Association, CGNPC plans to have the design completed and available for export starting in 2014. Given that, the earliest conceivable "first concrete" date would be in 2015.

Note 3 APR+ - The APR+ is a Generation III+ version of the Generation III APR1400 PWR by KHNP in the Republic of Korea. KHNP stated on its web site that it completed FOAKE on the design, and is spending 2013-2015 optimizing the design (aiming for a 36 months construction duration, first concrete to fuel load) in preparation to enter foreign markets. The gross electrical output is increased to 1500 MWe by increasing the number of fuel assemblies from 241 in the APR1400 to 257 in the APR+. Most other APR1400 design features remain the same, except that four diesel generators are planned (instead of two in the APR-1400 design). The design has a goal of a core damage frequency of 1×10^{-6}/a or less (compared to the CDF for the APR-1400 design of 6.22×10^{-6}/a). Some details were available as of 28 March 2013 at http://cyber.kepco.co.kr/kepco_new/nuclear_es/sub6_2.html. According to the WNA, the APR+ is intended for the European market with a double containment and a core catcher (WNA, 2013f**)**.

Note 4 CAP1400/AP1400 – The CAP1400/AP1400 is a larger evolution of the AP1000 design by SNPTC and SNERDI in the People's Republic of China. The design is intended to be 4040 MWt/1520 MWe two-loop PWR with passive system technology similar to the AP1000 (WNA, 2013f). According to Sun Qin (Chairman of China National Nuclear Corporation), the first international sales agreement for a CP1400 nuclear power plant could be signed in 2013 (China Daily, 2013). The first of the CAP1400 reactors is slated to being construction at the Shidaowan site in Rongcheng in 2013 (Nuclear Street, 2013; SNPTC, 2012).

Note 5 VVER-1200A/501 – The VVER-1200A/501 is a concept proposal for a Generation III+ evolution of the VVER-1200 AES-2006 design involving two loops rather than four loops, and correspondingly larger steam generators. The ideas behind this concept are being able to transport the large components by rail and being able to design a more compact containment (smaller diameter; 40 meters vs. 44 meters in the VVER-1200/491). It is possible that this design will be bypassed by the VVER-TOI, an AES-2010, Generation III+ design with a planned power level of 3300 MWt/1300 MWe. The VVER-TOI is already planned for construction at a three sites in Russia (Kursk II Units 1-4; Kola II Units 1 & 2; Smolensk II Units 1 & 2), with construction planned to start on the first unit in 2014. The completed design is expected to receive Rostechnadzor approval in 2013 and then be submitted for EUR certification. There is some indication that the VVER-1200 units planned for Niznhy Novgorod and at Akkuyu in Turkey will also be VVER-TOI units (Artisuk, 2012; WNA, 2013a).

Note 6 VVER-1500/448 – The VVER-1500/448 PWR (4-loop) has a planned capacity of 4250 MWt/1560 MWe gross, and is equipped with a double containment. The design has an air-cooled passive heat removal system (PHRS) for the primary containment. IAEA lists the detailed design as being complete in 2011 (IAEA, 2011d).

Note 7 AHWR – The Advanced Heavy Water Reactor (AHWR) from the Bhabha Atomic Research Centre (BARC) is an advanced pressurized heavy water reactor with a design capacity of 920 MWt/304 MWe. The AWHR is designed to be fueled with Uranium-233/Thorium-232 oxide together with plutonium-thorium oxide, and is intended to be nearly self-sustaining in Uranium-233. The AHWR is planned to have 452 primary coolant channels in heavy water filled vertically-oriented calandria (all other PHWRs have a horizontally-oriented calandria), but the primary coolant is boiling light water. Reactor heat can be removed by a passive system consisting of isolation condensers submerged in a 6000 m^3 tank (Gravity Driven Water Pool, GDWP), which is adequate to cool the core for three days. Containment cooling is also done with a passive system. The design features a double containment, with a negative pressure maintained in the annulus between the containments (IAEA, 2009b). According to the IAEA ARIS data base, the AHWR is at the basic design stage. A version of the AHWR is planned for export since 2009. It is proposed to begin construction of the first AHWR in 2014 with initial operation planned for 2019, but this schedule appears to be optimistic considering that a site for the project had yet to be identified in March 2013 (WNA, 2013b), and that the design was identified by the IAEA in September 2012 as being at the basic design stage (IAEA, 2012f).

Note 8 HI-SMUR/SMR-160 – The Holtec Inherently-Safe Modular Underground Reactor (HI-SMUR/SMR-160) is being designed by SMR LLC, a Holtec International company, and has a design capacity of 160 MWe. The SMR 160 design has a small footprint of 5 acres. The design features a 4 year refueling cycle and a service life of 80 years. An agreement has been signed between Holtec and the U.S. Department of Energy in which DOE agreed to host the first SMR-160 at the Savannah River National Laboratory. A Design Certification application to the NRC is expected in 2013 (WNA, 2013e).

Note 9 IRIS – The development of the International Reactor Innovative & Secure (IRIS) was originally led by Westinghouse. Westinghouse withdrew from the project in 2010, and now the project is a 10-country multi-party arrangement involving Oak Ridge National Laboratory, the University of California at Berkeley, Ansaldo Nucleare, Politecnico di Milano, the University of Pisa, Politecnico di Torino, ENEA, Mangiaraotti Nuclear, Maire Tecnimont, ATB Riva Calzoni, SAIPEM (ENI), Rolls Royce, CNCN Research Center, NUCLEP Industries, ENSA Industries, Empresarios Agrupados, the University of Zagreb, Tokyo Institute of Technology, the Lithuanian Energy Institute (LEI), the ININ Research Center, and EESTI Energia. IRIS is a Generation III+, 1005 MWt/335 MWe net modular integral PWR. The reactor is housed in the 25-meter diameter spherical steel pressure suppression containment. External air cooling of the steel shell provides a passive heat sink to the atmosphere. In addition, there is an emergency heat removal system (EHRS) that cools the steam generators via heat rejection to the refueling water storage tank. Level 2 PSA estimates CDF = 2×10^{-8}/a, and LERF = 6×10^{-10}/a **(IAEA 2009b)**. Project proponents initially proposed submitting the design for Design Certification in the United States in 2008; this was later delayed to 2010. No Design Certification application had been made as of March 2013.

Note 10 mPOWER – The mPower reactor is 530 MWt/180 MWe modular integral PWR design. Generation mPower LLC (majority owned by Babcock & Wilcox Nuclear Energy, Inc.) signed a letter of intent with the Tennessee Valley Authority in 2011 which could lead to the construction of up to four mPower modules at the Clinch River site in Tennessee. A Design Certification application was projected by the vendor to occur in 2014, and the first module was projected by the vendor to be deployed by 2022. In late 2012, the U.S. Department of Energy selected the mPower design to receive federal funding to support technology development. As of March 2013, B&W planned to submit the mPower design for NRC design certification by mid-2014, and also planned to submit a construction permit application for NRC (for the Clinch River Site) by mid-2015. B&W expected to submit both Design Certification and a Construction Permit application in 2015, and expects to have both approved by 2018, with a goal of commercial operation of the first two mPower modules in 2022 (DOE, 2012a; WNA, 2013e). The design was identified by the IAEA as being at the conceptual design stage in September 2012 (IAEA, 2012f).

Note 11 NuScale – The NuScale design is a small modular reactor (integral PWR) with a 160 MWt/45 MWe net design capacity per module. The reactor is housed in a steel containment (deep vacuum design with a design pressure of 3.4 MPa) that is submerged in a steel lined underground concrete water-filled pool. From one to twelve modules can make up a power plant as currently designed. NuScale anticipated filing for Design Certification in the United States in 2012, but the application still had not been filed as of March 2013. The design was identified by the IAEA as being at the basic design stage in September 2012 (IAEA, 2012f).

Note 12 RITM-200 – The RITM-200 design (OKBM Afrikantov) is a small (175 MWt/55 MWe), integral pressurized water rector. The RITM-200 would use less than 20% enriched Uranium, and only need to be refueled every 7 years. The design life is planned at 40 years. The RITM-200 could be used in an icebreaker (two units per ship, and could also be used in marine oil drilling platforms or as a land-based reactor. OKBM Afrikantov has produced a brochure for the RITM-200 design (OKBM Afrikantov, Undated 3).

Note 13 VVER-300/478 – The VVER-300/478 is a small, two-loop version of the VVER design rated at 850 MWt/300 MWe gross. The intended design service life is 60 years. The design includes a passive decay heat removal system, a double containment (a steel-lined pre-stressed concrete primary containment & a reinforced concrete secondary containment), and a core catcher. The containment design pressure is 0.5 MPa, and the design leak rate is 0.2% volume per day. The construction duration is expected to be 54 months. As of 2013, the design is at the feability study stage (IAEA, 2011f).

Note 14 Westinghouse SMR – The Westinghouse SMR is a small modular integral PWR with a design capacity of 800 MWt/225 MWe net. As of March 2013, Westinghouse predicted it would file a Design Certification application in the United States in late 2013. Westinghouse filed for U.S. Department of Energy funding in a program that would have required a plant in service date in 2022 (however, Westinghouse was not selected for this program). Ameren Missouri was expected to file a Combined Operating License (COL) application with the NRC in 2013, and proposes to build five Westinghouse SMRs at the Callaway site (instead of an EPR) (WNA, 2013e).

Note 15 4S – The Toshiba 4S is a so-called "nuclear battery", a 30 MWt/10 MWe sodium-cooled fast reactor with infrequent refueling (every ten to thirty years). Emergency cooling is passively provided by a Reactor Vessel Auxiliary Cooling System (RVACS). The reactor is located underground in this design, and seismic isolators are used to reduce earthquake impacts. A guard vessel surrounds the reactor vessel. Toshiba worked on the design with the Central Research Institute of Electric Power Industry

(CRIEPI), Japan, and is working with Argonne National Laboratory and Westinghouse in the US (Westinghouse is majority owned by Toshiba). Toshiba originally planned to submit the 4S for Design Certification in the United States in 2009; this was later changed to mid-2013, but no submittal had been made as of March 2013. Licensing in the US is not expected to be completed before 2020, although this may be optimistic since IAEA in September 2012 stated that the design was at the conceptual design stage (IAEA, 2012f). A 50 MWe version of 4S is potentially available after 2020.

Note 16 BN-1200 – The BN-1200 (2800 MWt, 1220 MWe) is intended to be a Generation IV, large 4-loop sodium-cooled fast reactor. The basic design of BN-1200 was planned as of 2013 to be complete in 2015. The construction of the BN-1200 unit Beloyarsk was under consideration as of 2013. The design concept includes a 3-loop passive decay heat removal system, and a core catcher. The design target for CDF is less than $1{\times}10^{-6}$/a (Ashurko, 2013).

Note 17 SVBR-100 – The SVBR-100 is a small modular fast reactor cooled by a lead-bismuth eutectic. The reactor, steam generators, and main circulating pumps are all arranged in a monoblock vessel. The vendor estimated in 2009 that an initial production block would startup in 2017, and that serial production could commence in 2019. Since then, however, the 50% utility shareholder has changed from En+ (subsidiary of JSC EuroSibEnergo) to JSC Irkutskenergo, and the site was changed from Obninsk to Dimitrovgrad. The vendor still maintained as of 2012 that the 2017 startup date was on schedule, and in late 2012 the vendor indicated that construction would begin in 2013, and a projected duration for construction of 42 months from first concrete to startup (startup in 2017).

TABLE 8: REACTOR DESIGNS POTENTIALLY AVAILABLE FOR LONG-TERM DEPLOYMENT (AFTER 2020)

Small Modular Reactors (SMRs)
CAREM-300 (CNEA & INVAP) (Note 1)
FBNR (Fed. Univ. of Rio Grande do Sul, Brazil) (Note 2)
IMR (Mitsubishi & CRIEPI) (Note 3)
MARS (University of Rome La Sapienza) (Note 4)
UNITHERM (RDIPE, Russian Federation (Note 5)
VBER-300 (OKB Afrikantov) (Note 6)
Generation III+ Reactors
ABWR II (GE-Hitachi) (Note 7)
APR1000 (Korea Hydro & Nuclear Power) (Note 8)
Reduced Moderation BWR (JAEA & Hitachi) (Note 9)
SCOR (CEA (Note 10)
VVER-600/498 (OKB Gidropress) (Note 11)
VVER-640/407 (OKB Gidropress) (Note 12)
VVER-1800 (OKB Gidropress) (Note 13)
Generation IV Reactors
AHTR (Oak Ridge National Laboratory) (MSR) (Note 14)
ALLEGRO (European Consortium) (GFR) (Note 15)
ANTARES (AREVA) (VHTR) (Note 16)
AREVA SFR (AREVA) (SFR) (Note 17)
ASTRID (France) (SFR) (Note 18)
BREST-300 (RDIPE & NIKIET, Russia) (LFR) (Note 19)
CANDU-SCWR (CANDU Energy, Inc.) (SCWR) (Note 20)
ELFR (European Consortium) (LFR) (Note 21)
EM2 (General Atomics) (GFR) (Note 22)
European GCFR (AMEC) (GFR) (Note 23)
FlexBlue (DCNS) (PWR) (Note 24)
Fuji MSR (Fuji) (MSR) (Note 25)
GT-HTR (JAEA) (VHTR) (Note 26)
GT-MHR (General Atomics & OKBM Afrikantov) (VHTR) (Note 27)
HP-LWR (Karlsruhe Inst., of Technology & Others) (SCWR) (Note 28)
JSCWR (Tohshiba) (Note 29)
KALIMER-600 (KAERI) (SFR) (Note 30)
KAMADO (CRIEPI) (GFR) (Note 31)
LFTR (Flibe Energy) (MSR) (Note 32)
MOSART (Kurchatov Institute Consortium) (MSR) (Note 33)
MSFR (EURATOM) (MSR) (Note 34)
OKBM Afrikantov BN-1200 (SFR) (Note 35)
PB-GFCR (Argonne National Laboratory) (GFR) (Note 36)
PBMR (PBMR Pty.) (VHTR) (Note 37)
PFBR (SFR) (Note 38)
PRISM (GE-Hitachi) (SFR) (Note 39)
SSTAR (LLNL, LANL & ANL) (LFR) (Note 40)
Traveling Wave Reactor (TerraPower) (SFR) (Note 41)
VVER-SKD (OKBM Gidropress) (SCWR) (Note 42)
China Institute of Atomic Energy CFR-600 (SFR) (Note 43)

Notes on Table 8

Note 1 CAREM-300 – The CAREM-25 integral PWR is planned to be uprated to a 300 MWe reactor. This design was still at the feasibility study stage as of 2013.

Note 2 The Fluidized Bed Nuclear Reactor (FBNR) is a 218 MWt/70 MWe net pressurized light water reactor (an SMR) but with the CERMET fuel in spherical form. FBNR is intended to operate without the need for onsite refueling (new core every 25 months). The coolant is supercritical water. This reactor is at the concept description stage (the earliest stage of development).

Note 3 IMR – The Mitsubishi Integrated Modular Water Reactor (IMR) is an integral pressurized water reactor with a capacity of 1000 MWt/350 MWe. According to IAEA/ARIS, the conceptual design has been completed for the reactor as of mid-2011. Validation testing, research and development for components and design methods, and basic design are required for licensing. The target year to start licensing as of mid-2011 was 2020 at the soonest.

Note 4 MARS – The Multipurpose Advanced Reactor, Inherently Safe (MARS) is under design by the University of Rome La Sapienza and CEA (France). The design concept is for a 600 MWt /150 MWe modular PWR with a net thermal efficiency of 25% (much lower than conventional Generation II PWRs for which net thermal efficiencies typically are in the range of 30-33%). This design concept dates from the 1980s, but is still at the conceptual design stage.

Note 5 UNITHERM – The UNITHERM concept is being developed by NIKIET in Russia. The concept is for a 30 MWt/6 MWe autonomous co-generation reactor, with only annual maintenance visits.

Note 6 VBER-300 – The VBER-300 design (OKBM Afrikantov) is a small, modular, four-loop PWR with passive safety systems. A VBER-300 unit is under discussion with authorities in Kazakhstan for construction in Aktau (which used to house the now shut down BN-350 fast reactor used for electricity production and desalination). The VBER-300 has a double containment, with the outer reinforced concrete containment constructed from extra density reinforced concrete walls 1.5 meters thick. The reactor is planned to have a thermal capacity of 917 MWe and an gross electrical rating of 325 MWe. The designed service life is 60 years. The steam generators planned for the VBER-300 are a unique vertical, once-through design with titanium steam generator tubes (OKBM Afrikantov, Undated 4). The IAEA's Advanced Reactor Information System had a report on the VBER-300 design as of April 2013 at http://www.iaea.org/NuclearPower/Downloadable/aris/2013/30.VBER-300.pdf. As of April 2011, the conceptual design for VBER-300 had been completed.

Note 7 ABWR-II – The ABWR-II is planned to be an advanced, Generation III+ version of the Generation III ABWR rated at 4950 MWt/1638 MWe net (33.1% efficient). The steel-lined, reinforced concrete containment (0.31 MPa design pressure) will be nitrogen-inerted. PARs are to be deployed in the containment, but apparently not in the reactor building. Passive systems are included in the design for reactor cooling and containment cooling, both using isolation condensers in a common heat sink pool above the containment, with a one-day grace period. Level 1 PSA core damage frequency estimated at 4.52×10^{-8}/a (the PSA scope is not identified, but is probably limited to internal events at power). Emergency power sources are 2 diesel generators and 2 gas turbine generators. The ABWR-II design retains the reactor building concept of the Japanese ABWR (with light industrial grade construction above the containment). Construction duration identified as 29.5 months. The detailed design is not yet complete, but is expected to be complete by 2015 or soon after. Commercial introduction of ABWR-II is expected between 2012 and 2020. No indication has been identified suggesting ABWR-II has been bid for a nuclear power plant construction contract as of March 2013. Available design details can be found in the IAEA ARIS status report (IAEA, 2011c). In 2009, an executive from Hitachi-GE (the Japanese counterpart of GE-Hitachi, based in the U.S.), forecast construction of ABWR-II units beginning in about 2025 (Hanyu, 2009).

Note 8 APR1000 – The APR1000 is a design concept by KHNP to produce a smaller version of the APR1400 (2815 MWt/1000 MWe net). The design includes a single pre-stressed concrete containment with a design pressure of 0.494 MPa and a design leak rate of 0.1% volume per day. The design was at the conceptual design stage in 2011.

Note 9 Reduced Moderation BWR – The Reduced Moderation BWR is an advanced design concept in which the fuel rods are packed closer together, and the fuel assemblies are shorter than in the more conventional BWRs. The concept involves hexagonal fuel assemblies and Y-shaped control rods.

The fuel is expected to be 18% MOX surrounded by depleted uranium in the blanket region. The reduced moderation results in production of more plutonium, with an expected breeding ratio of about 1 (instead of 0.6 in more conventional BWRs). A conceptual design study was completed by JAERI and Japan Atomic Power Company in 1998, and according to IAEA the design remained as of April 2011 at the conceptual design stage. The planned capacity is 3926 MWt and 1356 MWe net. Given the current state of development, it is not clear whether the Reduced Moderation BWR will be developed in time to be deployed in 2020, or whether it will be overtaken by supercritical water-cooled reactor technology (SCWR) and/or sodium-cooled fast reactor (SFR) technology, both Generation IV concepts.

Note 10 SCOR – The Simple Compact Reactor (SCOR) is a conceptual design for a 2000 MWt/630 MWe net integral PWR from CEA in Cadarache. The design incorporates a passviee decay heat removal system (with a passive air-cooled heat exchanger) and a dedicated steam dump pool in the containment to which steam can be directed in case of a steam generator tube rupture (a unique feature of this design). The SCOR design relies on in-vessel retention (IVR) of core debris in case of a severe accident, which is achieved by reactor cavity flooding. The containment is inerted to prevent hydrogen combustion (IAEA, 2008b).

Note 11 VVER-600/498 – The VVER-600/498 is a medium power two-loop VVER with a design capacity of 1600 MWt/600 MWe gross. The design, although based on the VVER-1200, eliminates the core catcher in favor of the in-vessel retention/ex-vessel reactor cooling concept. The design includes a passive decay hea t removal system. The spent fuel pool is located within the containment. A double containment concept is followed in the design. The VVER-600/498 was at the conceptual design state as of mid-2011.

Note 12 VVER-640/407 – The VVER-640/407 is a medium power two-loop VVER with a design apacity of 1800 MWt/604 MWe net. The VEER-640/407 uses active safety systems, and has a double containment with a design pressure of 0.5 MPa and a design leak rate of 0.1% volume per day for the primary containment. The primary containment is a steel containment with a free volume of 50,000 m^3, and the secondary containment is a reinforced concete structure. A passive containment heat removal system and a passive steam generator heat removal system are included in the design (both are 4×50% systems). The basic design of the unit was complete as of mid-2011.

Note 13 VVER-1800 – The VVER-1800 would take the two loop VVER-1200A/501 concept and expand it to a three loop design producing 1800 MWe. This is apparently more of a design concept as of early 2013 (it is not, for example, listed in the IAEA ARIS data base of advanced reactor designs.

Note 14 AHTR – The Advanced High Temperature Reactor (AHTR) is a pebble bed fuel, molten salt cooled reactor concept coming from Oak Ridge National Laboratory (ORNL). The 2011 version of the design had a capacity of 3400 MWt/1500 MWe. As of 2012, the reactor was still at the preconceptual design stage. A recent UK NNL assessment characterized MSRs as a very immature design (UK NNL, 2012).

Note 15 ALLEGRO – A European consortium (CEA, ÚJV Rez, MTA Centre for Energy Research, VÚJE, and NCNJ Poland) is designing the ALLEGRO gas-cooled fast reactor (GFR) design is planned to be complete for construction in 2025. Potential sites for ALLEGRO include Bohunice in Slovakia, Dukovany in the Czech Republid, and Paks in Hungary. ALLEGO is planned as a 75 MWt reactor with a secondary water system and no power conversion system (i.e., heat is rejected to the environment) (VÚJE, 2012). A recent UK NNL assessment characterized GFRs as a very immature design (UK NNL, 2012). SNETP estimates that €400 million in development and another €700-800 million would be needed for ALLEGRO design and construction (a total investment of €1.1-1.2 billion) for a 70-100 MWe prototype reactor (SNETP, 2010).

Note 16 ANTARES - AREVA's ANTARES VHTR design features a 625 MWt/285 MWe reactor that can be used for electricity production or hydrogen production. The reactor is helium-cooled, and has a passive decay heat removal system (Reactor Cavity Cooling System, or RCCS). The ANTARES design was selected for the US Next Generation Nuclear Plant (NGNP) by the Department of Energy (DOE) in early 2012, and is at the Pre-Conceptual Design stage. The NGNP is an industrial prototype project; commissioning is not expected until 2021 at the earliest. The estimated cost for the ANTARES reactor as the Next Generation Nuclear Plant was $3.911 billion, estimated in 2007 (estimate does not include escalation) (DOE, 2010).

Note 17 AREVA SFR – The CEA/AREVA commercial sodium-cooled fast reactor is a planned commercial development from the ASTRID prototype reactor. Since ASTRID will probably not operate until 2023 or later, and since at least five years of successful operation (perhaps longer) would be required before a final design for the commercial SFR could be completed, it is not expected that a CEA/AREVA commercial fast reactor would be in operation before 2030 or later.

Note 18 ASTRID – The Commissiarat l'Energie Atomique (CEA) is designing the 1500MWt/600 MWe Generation IV ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) sodium-cooled fast reactor demonstration plant (integrated technology demonstrator). Detailed design and a construction decision are pending in about 2019 (the design was at the pre-conceptual stage in 2013), and operation in about 2023 or later (Alphonse, Perrin & Gama, 2013). A commercial sodium-cooled fast reactor based on ASTRID is expected by CEA to be available between 2040 and 2050 (Vasile, 2012). The design is expected to include a guard vessel surrounding the reactor vessel, a core catcher, and a natural air draft decay heat removal system. CEA is responsible for the core design; AREVA NP is responsible for the nuclear island and I&C; ALSTOM is responsible for the energy conversion system; Buoygues is responsible for civil engineering; and Jacobs Engineering is responsible for balance-of-plant engineering. Support is being provided by COMEX Nucleaire, EdF, Rolls-Royce & Toshiba (Béhar, 2013). The ASTRID prototype SFR is projected to require about €1 billion in development, and about €4 billion for the prototype design and construction, for a total investment of €5 billion to bring the ASTRID prototype to fuel load (SNETP, 2010).

Note 19 BREST-300 – The BREST-300 lead-cooled fast reactor (LFR, 700 MWt/300 MWe net) is still in planning stages, with an expected first unit operation in 2020 according to the latest plan. It was acknowledged in early 2013 that "*none of the technologies involved in the BREST system has been demonstrated*" (Rachkov, 2013). We consider it unlikely that the 2020 in-service date for the BREST-300 prototype unit will be met. (BREST is a Russian language acronym for Bystryi Reactor so Svintsovym Teplonositelem – Fast Reactor with Lead Coolant.) The BREST-300 design is eventually seen as giving way to a larger BREST-1200 plant design (2800 MWt/1200 MWe net) (Filin, et al., 2000). A recent UK NNL assessment characterized SCWRs as an immature design (UK NNL, 2012).

Note 20 CANDU-SCWR – As part of the Gen IV initiative, a supercritical water cooled reactor (SCWR) version of CANDU technology is being investigated. In 2009, it was predicted that a prototype plant could be designed, licensed, and built by about 2020. This appears to be an optimistic projection since there are materials problems to be solved and the design was in 2012 in the pre- conceptual design stage. A recent UK NNL assessment characterized SCWRs as a very immature design (UK NNL, 2012).

Note 21 ELFR – The ELFR (European Lead-Cooled Fast Reactor) would follow up on work completed in the European Commission's FP6 and 7 Framework research programmes (ALFRED and ELSY/LEADER). The European Lead-Cooled System (ELSY) is a 1500 MWt/600 MWe design concept. ALFRED is planned as a 100 MWe reactor, and it is estimated by SNETP that it will cost €1 billion to bring it to fuel load. Commissioning of ALFRED was forecst in 2012 to take place in 2025 (SNETP, 2012).

Note 22 EM2 – The General Atomics EM2 Gas-Cooled Fast Reactor design is an evolution of the GT-MHR design. The reactor is cooled by helium. The planned capacity of an EM2 unit is 500 MWt/240 MWe net. This design would require no refueling for thirty years. The reactor is sized so that it can be carried by a flat-bed truck. The reactor is planned to be installed in an underground caisson. A recent UK NNL assessment characterized GFRs as a very immature design (UK NNL, 2012).

Note 23 European GCFR – ALLEGRO (see Note 16, above) is meant to serve as a technology demonstrator for a later 2400 MWe GFR design. It is planned that the ultimate 2400 MWt GFR would use a Brayton cycle (direct use of helium in a gas turbine). The 2400 MWt concept involves three GFR modules in a steel guard vessel, housed in a containment building, driving three gas turbines and together driving a steam turbine (combined cycle) (Stainsby, 2012). A recent UK NNL assessment characterized GFRs as a very immature design (UK NNL, 2012).

Note 24 FlexBlue – The Direction des Constructions Navales Services (DCNS) is designing the 50-250 MWe FlexBlue reactor for undersea deployment. FlexBlue PWRs are meant to be deployed at depths of 60-100 meters offshore, with undersea cables connecting the electricity supply onshore. The concept calls for a plant housed in a cylindrical hull about 100 meters long and 12-15 meters in diameter, with a total mass of about 12,000 tonnes. The hull and power plant are intended to be transportable using a purpose-built ship. DCNS was in 2011 in phase 2 of development, working together with EdF, AREVA, and CEA. (DCNS is 74% owned by the French state.) (DCNS, 2011) It is reported by IAEA that a conceptual PWR SMR design called SHELF (a seabed based reactor) has been initiated in Russia (as of 2012) (Subki, 2012b).

Note 25 FUJI MSR – The FUJI Molten Salt Reactor (FUJI MSR) is being developed by the International Thorium Energy & Molten Salt Technology Company, Inc. (IThEMS) as a 450 MWt/200 MWe thorium molten salt reactor (MSR). (FUJI is apparently a Japanese language acronym.) The molten salt for the reactor would most likely be Flibe (LiF-BeF_2ThF_4-UF_4). The concept involves a design life of 30 years. A recent UK NNL assessment characterized MSRs as a very immature design (UK NNL, 2012).

Note 26 GT-HTR – The Gas Turbine High Temperature Reactor GT-HTR is a gas-turbine VHTR being developed by the Japan Atomic Energy Authority (JAEA) for nuclear cogeneration of electricity and hydrogen production. The reactor is a helium-cooled design rated at 600 MWt/275 MWe. The reactor and gas turbine are planned to be located underground. In July 2008, a prototype was forecast for about 2020, with deployment of commercial units beginning in about 2030. A 30 MWt high temperature test reactor (HTTR) began operating in Oarai, Ibaraki, Japan in 1999.

Note 27 GT-MHR – The GT-MHR design (600 MWt/287 MWe) is a Brayton cycle (helium gas turbine without steam generation) VHTR. The reactor and the power conversion system are located in an underground a reinforced concrete structure. Passive heat removal is accomplished by an air-cooled reactor cavity cooling system (RCCS). GT-MHR has a containment with a design leak rate of 1% volume per day **(IAEA 2009b)**. The design is being developed by General Atomics (United States) in partnership with OKBM Afrikantov (Russia) with support from Fuji Industries (Japan). A conceptual design was completed in 1997, and a preliminary design was completed in 2002 for a GT-MHR prototype. The design planned for deployment in Russia to consume excess plutonium is planned to operate on plutonium oxide fuel.

Note 28 HP-LWR – The High Performance Light Water Reactor (HP-LWR) is a development of the 42-month duration, €4.6 EU-supported Specific Targeted Research or Innovation Project (STREP) that began on 1 September 2006, involving a consortium of 10 partners (designated HPLWR Phase 2). The design involves supercritical water cooled reactor (SCWR) reactor technology, however housed in a cylindrical pressure suppression containment with a drywell end cap somewhat resembling the ABWR containment concept. The design status is at a conceptual design stage, and none of the key components have been tested. Transient accident analysis codes for supercritical water cooled reactors are still under development. A recent UK NNL assessment characterized SCWRs as a very immature design (UK NNL, 2012).

Note 29 JSCWR – The Japanese Supercritical Water-Cooled Reactoris a Generation IV supercritical water cooled reactor concept from Toshiba. The design was at the conceptual design stage as of April 2011. The design concept is for a 3681 MWt/1620 MWe net reactor with 44% net thermal efficiency. The containment in the JSCWR concept is a steel-lined pre-stressed concrete structure (IAEA, 2011g). A recent UK NNL assessment characterized SCWRs as a very immature design (UK NNL, 2012).

Note 30 KALIMER-600 – The 1523 MWt/600 MWe net, pool type, sodium-cooled fast reactor design was at the conceptual design stage in 2011. The KALIMER-150 design was accomplished through a collaborative effort with General Electric. The KALIMER-600 design was based on this earlier experience, and was selected as one of three Generation IV SFR concepts. The goal is to develop an advanced 600 MWe pool-type SFR for standard design approval by 2020 and completing construction by 2028 (JEONG, 2011). The KALIMER-600 desgin strongly emphasizes proliferation resistance, and for this reason has no breeding blanket. The design is intended as a break even reactor, producing as much fuel as it uses (i.e., breeding ratio of 1). A preliminary PSA (internal events at power only) estimated the CDF for KALIMER-600 at 1.2×10^{-6}/a (Kim et al., 2011).

Note 31 KAMADO – The KAMADO design involves a loop type, vertical pressure tube, fast breeder reactor with a design capacity of 3000 MWt/1000 MWe. It is further envisioned that the coolant will be supercritical carbon dioxide. As of 2011, the design was at the conceptual design stage. A recent UK NNL assessment characterized GFRs as a very immature design (UK NNL, 2012). KALIMER technology was at the conceptual design stage in 2011 (JEONG, 2011).

Note 32 LFTR – The Liquid Fluoride Thorium Reactor (LFTR) has been proposed by Flibe Energy in conceptual design form. No schedule basic engineering, first-of-a-kind engineering, and prototype deployment were identified. A recent UK NNL assessment characterized MSRs as a very immature design (UK NNL, 2012).

Note 33 MOSART – The Molten Salt Actinide Recycler & Transmuter (MOSART) is being pursued by the Kurchatov Institute, the Institute of High Temperature Electrochemistry, and the Institute of Technical Physics in Russia. The concept involves a 2400 MWt/1100 MWe , graphite-moderated reactor; the molten salt is Flibe. The proponents acknowledge that a "*substantial R&D effort would be required to commercialize MOSART*". As of 2007, the design was at the conceptual design stage (Ignatiev, et al., 2007). A recent UK NNL assessment characterized MSRs as a very immature design (UK NNL, 2012).

Note 34 MSFR – The Molten Salt Fast Reactor (MSFR) is a project of France and EURATOM. An assessment in the SNETP program suggests operation of an MSFR demonstration plant (20-50 MWt) starting in 2030, and operation of an MSFR prototype starting in 2040. The current status is described as being at the pre-design stage (concept studies) (SNETP, 2012). A recent UK NNL assessment characterized MSRs as a very immature design (UK NNL, 2012).

Note 35 OKBM Afrikantov BN-1200 – OKBM Afrikantov is working on the BN-1200 design (2800 MWt, 1220 MWe net). The design is due to be completed in 2014, and the first BN-1200 unit is scheduled (as of March 2013) to begin operation at Beloyarsk in 2020. OKBM Afrikantov plans to construct nine of the BN-1200 units by 2030 (WNA, 2013c). The BN-1200 design, which is expected to be complete by 2015, is planned to include a guard vessel (in case of reactor vessel leakage), a passive decay heat removal system, passive shutdown systems, and a core catcher (Ashurko, 2013).

Note 36 PB-GCFR – The Particle-Bed Gas-Cooled Fast Reactor (PB-GCFR) was the subject of a conceptual design study at Argonne National Laboratory in the early 2000s. The history of GCFR designs was described in 2009 (Van Rooijen, 2009). A recent UK NNL assessment characterized GFRs as a very immature design (UK NNL, 2012).

Note 37 PBMR – The Pebble Bed Modular Reactor (PBMR) is a 400 MWt/165 MWe helium-cooled, graphite moderated, high temperature reactor concept developed by PBMR (Pty) Ltd in South Africa. Up to 2010, it seemed likely that PBMRs would be built in South Africa. A final environmental impact report for the PBMR demonstration plant at Koeberg was issued in 2002. However, in 2010, the government cancelled the project. About €794 million was spent on the project, with 80% of the total coming from the South African government. American utility Exelon initially pushed for Design Certification review of PBMR, but stopped this in April 2002. In February 2004, PBMR Pty. Ltd. request pre-application review of the PBMR design, and NRC assigned a project number to the review. A letter of intent to see Design Certification was send to the NRC in March 2009, indicating a possible Design Certification application in 2013. Subsequently, Westinghouse withdrew from the project. There has been no further development of the PBMR since 2010.

Note 38 PFBR – The Prototype Fast Breeder Reactor (PFBR) is under construction at the Kalpakkam site in India. Design efforts for PFBR date to the 1980s. The reactor is a 1250 MWt/500 MWe net sodium-cooled fast reactor (SFR) planned to operate with MOX fuel. The design includes a core catcher and a rectangular containment building. An air-cooled heat exchanger system is included, but its operation is not passive as it requires operation of a blower to operate (Chellapandi, 2011). The IAEA has designated PFBR as available for immediate deployment (Subki, 2012b). The EHNUR project disagrees; it was April 2013 as this Chapter was written, and the prototype PFBR unit was still under construction, with the expectation that the prototype would be placed in operation by September 2014, with commercial operation following in September 2015 (The Hindu, 2013). EHNUR expects that at least five years of successful operation would be necessary in order to minimize risks arising from construction of an additional planned four FBR units based on PFBR. This would place the beginning of construction of the additional units beginning late in 2020. Construction of PFBR began in August 2004, and is continuing through (at least) September 2013 (nine years). This would place the earliest startup date for the next PFBR-based reactor in 2019. Considering the likelihood of further schedule extension in PFBR startup and commercial operation, the EHNUR project regards it as unlikely that the design could be commercially deployed before 2030.

Note 39 PRISM - The Power Reactor Innovative Small Modular (PRISM) is a modular sodium-cooled fast reactor design from GE-Hitachi. In one form or another, this design has been around since the late 1980s. GE-Hitachi is advocating the construction of a so-called Advanced Recycling Center (ARC) consisting of three 622 MWe power blocks of PRISM modules and an electrometallurgical separations plant. Each power block consists of two PRISM modules (840 MWt each) driving a single turbine, producing 622 MWe net. The reactor has a passive cooling system (Reactor Vessel Auxiliary Cooling System, RVACS). GE-Hitachi has a targeted deployment date of about 2020, but several previous deployment dates and already come and gone without result. Even if the design were available in 2020, it is likely that gaining Design Certification and constructing a prototype PRISM block would require 10 years (Boardman, 2001) (meaning the first block would go online in 2030). GE-Hitachi stated in 2009 that it was preparing a Design Control Document (DCD), a necessary step in order to apply for Design Certification (GE-Hitachi, 2009). In a letter to the NRC in March 2009, GE-Hitachi stated that it would submit PRISM for Design Certification in mid-2011. As of March 2013, however, GE-Hitachi had not still submitted the PRISM design for Design Certification.

Note 40 SSTAR – The Small Sealed, Transportable, Autonomous Reactor or SSTAR concept has been developed jointly by Lawrence Livermore National Laboratory, Los Alamos National Laboratory, and Argonne National Laboratory in the U.S. The planned power level is from 10-100 MWe, and the reactor can be transported on ship or by a heavy haul transport truck. The SSTAR is envisioned as either a lead-cooled or lead-bismuth-cooled fast reactor concept. A recent UK NNL assessment characterized SCWRs as an immature design (UK NNL, 2012).

Note 41 TWR – The Traveling Wave Reactor (TWR) is under development by TerraPower LLC in the United States. Conceptual designs for this sodium-cooled fast reactor have been created for TWR designs from 300-1000 MWe. A small amount of 10% enriched uranium is planned to be used to initiate the core, which then breeds Plutonium 239 within depleted uranium, and "burning" it in place without the need for chemical separation in a reprocessing plant. TerraPower has plans to complete the design and construct a 600 MWe TWR (designated TWR-P) by 2022 (TerraPower, 2013). This is considered to be an unrealistically ambitious schedule considering the conceptual design stage of the reactor in 2013, the need to complete the design, the need to have the U.S. NRC perform a design certification review, and then to site and construct the reactor. Even if things go extremely well (i.e., no significant issues are identified either in final design or design certification review), a more realistic schedule would have startup of the prototype in 2028 or thereafter.

Note 42 VVER-SKD – The VVER-SKD SCWR is being pursued by OKB Gidropress and TsNIITMASH (The Central Research Institute of Machine Engineering Technology). A recent UK NNL assessment characterized SCWRs as a very immature design (UK NNL, 2012).

Note 43 CFR-600 – The China Institute of Energy is planning a fast breeder reactor design called CFR-600 to be in operation in about 2023. The design is for 1500 MWt/600 MWe, fueled by MOX, with a breeding ratio of 1.2. The core damage frequency for CFR-600 is required to be below 1×10^{-6}/a, with a tentative large release frequency goal of less than 1×10^{-8}/a. The design includes an additional passive shutdown system, a passive decay heat removal system, and a core catcher. The detailed design is planned to be complete in 2017 (Zhang, 2013).

TABLE 9: NUCLEAR FUSION CONCEPTS (AFTER 2050)

EFDA Designs A, B, AB, C & D	(Note 1)
Field Reversed Configuration	(Note 2)
Fusion-Fission Hybrid	(Note 3)
Laser-Driven Inertial Confinement	(Note 4)
Reversed Field Pinch	(Note 5)
Spheromak	(Note 6)
Stellarator	(Note 7)
Z-Pinch Fusion	(Note 8)

Notes on Table 9

Note 1 In mid-2005, the European Fusion Development Agreement (EFDA) published the Power Plant Conceptual Study (PPCS), a conceptual design study of five fusion power plant designs of increasing complexity. The designs were designated PPCS A, B, AB, C, and D. All five of the designs are based on the tokamak concept as used in the JET and ITER experimental machines. PPCS A and B were based on limited extrapolations of the ITER design basis. In PPCS A, the blanket is a water-cooled lead-lithium design, whereas in PPCS B the blanket is a helium-cooled lithium silicate pebble bed concept. PPCS C and D are based on dual coolant blankets (helium and lead-lithium coolants with steel structures and silicon carbide insulators for PPCS C and a self-cooled lead-lithium coolant with a silicon carbide structure for PPCS D) (EFDA, 2005). PPCS AB was described in a separate paper, presented at ISFNT-7 in 2005 as a variation on PPCS B in which the tritium breeding material is lead-lithium rather than lithium silicate (Li_4SiO_4) (Maisonnier, 2005).

Note 2 A field reversed configuration fusion machine confines the plasma in a cylindrical chamber, instead of having a toroidal chamber as is done in tokamak or stellerator devices.

Note 3 In a fusion-fission hybrid device, the fusion machine, in additional to producing electrical power, is also used to convert Uranium-238 into Plutonium-239, which can then be used in a conventional fission power plant. This type of machine is referred to as a fission-suppressed hybrid fusion machine. Research institutes in the People’s Republic of China are pursuing this system in order to capitalize on the country’s more limited uranium resources. Research institutes in the United States are also pursuing hybrid machines for the same purpose, as well as one concept aimed at destroying long-lived radioactive materials contained in spent fuel from nuclear power plants. This type of system could also be used to produce Uranium-233 from Thorium-232.

Note 4 Laser-driven inertial confinement is a type of inertial confinement fusion in which powerful lasers are used to heat and compress a fuel target (normally a pellet that contains a mixture of deuterium and tritium). Examples of this type of fusion machine are the National Ignition Facility in the U.S., and the Laser Mégajoule facility in France.

Note 5 A reversed field pinch machine is a variety of toroidal machine in which the magnetic field pointing toroidally reverses its direction. Reversed field pinch machines are at RFX in Padua, Italy, the Madison Symmetric Torus (MST) at the University of Wisconsin-Madison in the U.S., the EXTRAP T2R in Sweden, and the TPE-RX in Japan.

Note 6 The spheromak is a type of compact toroid device that confines the plasma in a shape similar to a smoke ring or a coronal loop (such as seen on the sun). The advantage of these devices is a comparatively long confinement time for the plasma. The START machine at Culham Laboratories in the U.K. is a spheromak, as are the NSTX machine in the U.S., the Globus-M machine in Russia, and the newer MAST machine in the U.K. Researchers at the University of California at San Diego (UCSD) published a study of a spherical tokamak power plant design (ARIES-ST) in 2003 (for which the papers can be downloaded at http://aries.ucsd.edu/ARIES/DOCS/bib.shtml#ARIES-CS).

Note 7 A stellerator is a type of magnetic confinement system that uses a figure-8 confinement system. Stellerator machines are in operation at Wendelstein 7-X in Germany and the Large Helical Device in Japan. Researchers at the Max-Planck Institute for Plasma Physics in Garching are investigating possible stellerator fusion reactor concepts. Researchers at the University of California at San Diego (UCSD) published the results of a compact stellerator fusion power plant concept in the journal Fusion Science and Technology in 2008, for which the papers can be downloaded at (http://aries.ucsd.edu/ARIES/DOCS/bib.shtml#ARIES-CS).

Note 8 Z-Pinch is a type of plasma confinement system that uses an electrical current in the plasma to produce a magnetic field that compresses the plasma. Z-Pinch machines are located in the U.S. (especially at Sandia National Laboratories), France, Germany, the U.K., and Israel. The Z machine at Sandia is the largest X-ray generator in the world. In 2000, Sandia authors together with authors from universities and other laboratories published a paper outlining a Z-Pinch inertial fusion power plant concept (Derzon, et al., 2000). Another Sandia report (Spielman, 2000) provided more details as well as conference papers from Snowmass on inertial confinement fusion. A final report concerning Z-Pinch fusion energy was published by Sandia in 2006 (Cook, et al., 2006). NASA's Marshall Space Flight Center has examined the possible use of Z-Pinch fusion propulsion for interplanetary transportation.

TABLE 10: COMPARISON OF CANDU EC6 WITH PHWR-700

Parameter	EC6	PHWR 700
House Loads	50 MWe	70 MWe
Net Efficiency	33.1%	29%
Seismic Design	0.30g	0.214g
Core Damage Frequency	1×10^{-6}/a (target)	1×10^{-5}/a (target)
Large Early Release Frequency	1×10^{-7}/a (target)	1×10^{-6}/a (target)
Design Plant Availability	90%	90%
Average Discharge Burnup	7500 MWd/t	7000 MWD/t
Number of Coolant Channels	380	392
Number of Fuel Bundles per Channel	12	12
Steam Generator Tube Material	Incoloy-800	Incoloy-800
Containment Type	Double	Double
Primary Containment	Pre-Stressed Concrete with Steel Liner	Pre-Stressed Concrete with Steel Liner
Secondary Containment	Reinforced Concrete	Reinforced Concrete
Containment Design Leak Rate	0.2%/day	1.0%/day
Containment Design Pressire	0.5MPa	0.16 MPa
Decay Heat Removal	Active and Passive	Active and Passive
Duration of Passive DHR	168 hours	6 hours
Emergency Core Cooling – High Pressure	Passive (accumulators)	Passive (accumulators)
Emergency Core Cooling – Medium Pressure	Active	Not Present
Emergency Core Cooling – Low Pressure	Active	Active
Emergency Makeup to Calandria Vessel and Calandria Vault	Severe Accident Recovery and Heat Removal System (SARHRS)	Active (Diesel-Driven Fire Pumps)
Emergency Makeup to Steam Generator Secondary Side	Passive (Emergency Heat Removal System)	Active (Diesel-Driven Fire Pumps
Emergency Power	4 Diesels	4 Diesels
Fuel Bundle	37 fuel elements	37 fuel elements
Plant Design Life	60 years	40 years
Combustible Gas Management	PARs	Recombiners (Type Unknown)