

## CHAPTER 20

# *Nuclear Power Reactors*

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We concentrate in this chapter on existing and some future power reactors and discuss the features common to each generic type as well as some peculiarities of special designs. Waste generation and handling at the power plants is also described generically.

Electric power producing reactors have been built to 1500 MW<sub>e</sub> but all sizes in the range 400 – 1400 MW<sub>e</sub> are common. With increasing size the cost of the electric power usually decreases. Power reactors of ~ 200 MW<sub>e</sub> are competitive in some areas of high energy costs, such as in many developing countries. These smaller sized plants have some advantages: they are more reliable due to less stress of the components, they can be built more easily in remote areas because the smaller components are easier to transport, the electric grid may be more capable of accepting the moderate power addition, and they can more easily be designed with inherent safety features. Table 20.1 summarizes the characteristics of some operating nuclear power reactors.

### 20.1. Thermal reactors

The large majority of nuclear power reactors is based on fission by thermal neutrons and the discussion will begin with this class of reactors.

TABLE 20.1. Characteristics of some operative nuclear power reactors

Type	Example	Power (MWe)	Fuel (% <sup>235</sup> U)	Can	Mode-rator	Cool-ant	Coolant temp. (°C)	Coolant press. (MPa)	Net eff. (%)	Burn-up (MWd/kg)	Temp. coeff.	Core size (h × d m <sup>2</sup> )
PWR	Diablo Canyon 1, USA	1124	93 t UO <sub>2</sub> (4.4)	Zr	H <sub>2</sub> O	H <sub>2</sub> O	320	16	32.1	45	Neg.	3.66 × 3.37
VVER <sup>†</sup>	Novovoronezh 5, Russia	1000	86 t UO <sub>2</sub> (4.4)	Zr	H <sub>2</sub> O	H <sub>2</sub> O	324	16	32.5	28.6	Neg.	3.53 × 3.16
BWR	Oskarshamn 3, Sweden	1205	143 t UO <sub>2</sub> (3.1)	Zr	H <sub>2</sub> O	H <sub>2</sub> O	286	7	36.4	32	Neg.	3.68 × 4.60
PHWR	Bruce 5, Canada	915	135 t UO <sub>2</sub> (nat.)	Zr	D <sub>2</sub> O	D <sub>2</sub> O	305	9.5	30.4	7.7	Pos.	5.94 × 5.67
RBMK	Sosnovy Bor 1, Russia	1000	218 t UO <sub>2</sub> (2.4)	ZN	C	H <sub>2</sub> O	284	7.0	32.0	22.2	Pos.	7.0 × 11.8
GCR	Wylfa A1, UK	570	593 t U (nat.)	Mgn	C	CO <sub>2</sub>	370	2.8	27.4	5.4	Pos.	9.1 × 17.4
AGR	Hinkley Point B3, UK	660	119 t UO <sub>2</sub> (3.4)	St	C	CO <sub>2</sub>	655	4.2	38.5	30	Pos.	8.31 × 9.11
LMFBR	Phenix, France	250	4.9 t (Pu, U)O <sub>2</sub> + UO <sub>2</sub> <sup>‡</sup>	St	-	Na	560	0.1	44.3	100	Neg.	0.85 × 1.39

Zr = Zircaloy, ZN = Zr + 1% Nb alloy, Mgn = Magnox, St = Stainless steel. <sup>†</sup> PWR-like russian design. <sup>‡</sup> as 19% Pu MOX.

## 20.1.1. Pressurized water reactors

As mentioned in Chapter 19, the name implies that a pressurized water reactor is cooled by hot high pressure water, either H<sub>2</sub>O (PWR, VVER) or D<sub>2</sub>O (PHWR). In the PWR and VVER types the coolant is also used as moderator whereas a separate D<sub>2</sub>O containing moderator tank is normally used in the PHWR type. These power reactor types have several things in common: primary – secondary coolant circuits separated by heat exchangers (*steam generators*), a *pressurizer* to adjust primary system pressure and often *chemical shim control* for adjustment of the excess reactivity with fresh fuel.

The pressurized water reactor is generally preferred for propulsion purposes (military surface vessels and submarines), partly because it can react faster on changes in power demand than many other types of thermal reactors.

(a) *PWR*. The main components of a pressurized light water moderated and cooled reactor (PWR) station have been described in §19.1 and Figure 19.2.

The important features of the PWR core are shown in Figure 19.12 and described in §19.13.1. The design with a pressure vessel without facilities for fuel handling under pressure necessitates a yearly shutdown for fuel replacement. However, this is not considered a great inconvenience as a PWR anyhow has to be shutdown on a regular basis for safety inspection of pipes, welds, etc, and for routine maintenance. The need to shut down the reactor and open the pressure vessel in order to replace fuel also makes the PWR (as well as the VVER) resistant to concealed nuclear proliferation.

Leaking tubes in the steam generators have plagued many PWRs. The leaks are usually caused by corrosion. Such leaks are normally fixed by plugging the affected tube. However, too many plugged tubes leads to a reduction in permitted power output; finally requiring either permanent shut down or replacement of the steam generators. In the beginning of 2000 many PWRs have already had their steam generators replaced (52 reactors) and replacements for another 21 reactors are planned.

Another generic problem has been the development of cracks in the control rod guide tubes of inconel which penetrate a PWR pressure vessel head. Such cracks are difficult to locate and repair because of the confined space between the tubes. After some time leaks usually develop when the cracks grow larger, ultimately requiring replacement of the vessel head. Crack formation can probably be prevented by a proper selection of material, its pre-treatment and meticulous adjustment of the water chemistry.

(b) *VVER*. In the former USSR and CMEA a reactor type similar to the PWR is used, referred to as the VVER, cf. Tables 19.1 and 20.1. The older VVER reactors have no containment; except two in Finland (Loovisa 1 & 2) where a Westinghouse type containment was added. The fuel elements have a hexagonal cross section and the steam generators are horizontal rather than vertical. The stainless steel control rods contain boron carbide as neutron absorber. A VVER reactor contains more water in the vessel than a typical PWR. This gives a higher heat capacity allowing a VVER to withstand interrupted coolant flow for a considerably longer time than a PWR. The pressure vessel in the VVER is exposed to rather high neutron fluxes and after many years of operation the wall has to be annealed (heated) in place to remove radiation induced brittleness. The VVER, like the PWR, has to be shutdown for refuelling.

(c) *PHWR*. The advantage of using heavy water for neutron moderation is that its low thermal neutron capture cross section ( $\Sigma_a$  (th), Table 19.3) improves the neutron economy

(i.e.  $1 - p$  becomes much less than in LWRs) so that even natural uranium can be used as fuel. The good neutron economy and harder energy spectrum (cf. Table 19.3) leads to about twice as high plutonium yield in heavy water reactors as compared to LWRs. The first large heavy-water-moderated reactors were built in the United States around 1950 for production of materials for fission weapons. Presently the only commercial PHWR type is the CANDU (CANadian Deuterium Uranium) design. The reactor vessel is a horizontal cylindrical tank filled with  $D_2O$ , through which several hundred horizontal aluminum tubes pass. Within each of these *pressure tubes* are bundles of zircaloy clad natural  $UO_2$  pins. The bundles are cooled by pressurized heavy water ( $\sim 8$  MPa), which in the steam generator produces light water steam for turbines. This pressure-tube-tank design is known as a *calandria*. In some designs the heavy water in the moderator tank can be dumped down into an empty tank below the reactor to allow an immediate shutdown, at the sacrifice of shielding in such an event. In other designs (Bruce, Table 19.7) the calandria is surrounded by a vessel full of ordinary water, which provides shutdown shielding; shutdown is then achieved by absorber rods and injection of gadolinium nitrate into the moderator. The calandria design permits on-line refueling.

Early CANDU-PHWR were built in the 200  $MW_e$  size, but the design can be upgraded to 750  $MW_e$  (Table 20.1). Using natural uranium the burnup is comparatively low, about 9000  $MW_{th}d t^{-1}$ .

The neutron economy makes it feasible to run PHWRs on the thorium cycle with a conversion ratio of  $\sim 0.9$ . Such a cycle has to be started and operated either with a mixture of  $^{233}UO_2$ , enriched  $^{235}UO_2$  and  $ThO_2$ , or – if isotope enrichment is not used – through the following three step fuel schedule: (i) running on natural uranium and extracting Pu; (ii) running on Th + Pu and extracting  $^{233}U$ ; (iii) running on Th +  $^{233}U$ .

The investment cost for CANDU reactors is higher than for LWRs because of a more complicated design and the large amount of  $D_2O$  required. The running costs are claimed to be lower; the standard loss of  $D_2O$  is  $\sim 1$  kg  $d^{-1}$ . The neutron capture by D forms tritium in considerable amounts, so the  $D_2O$  becomes contaminated by TDO. Also TD and TDO can leak into the surroundings, which may cause difficulties in biological safety. Two methods have been used to limit the amount of T present: i) In an expanding PHWR system part of the  $D_2O$  (contaminated by T) in older reactors can be transferred to new reactors and replaced by fresh heavy water thereby providing a temporary relief, ii) when a troublesome inventory of T has been built-up, the  $D_2O$  is transferred to a special isotope separation plant where tritium is removed.

### 20.1.2. Boiling water reactors

In boiling water reactors (BWR, ATR and RBMK types) the coolant is brought to boiling while passing along the fuel pins. As mentioned in §19.13.2, no separate steam generator and secondary loop exist and the design has less complexity. In addition to negative temperature coefficients both for fuel and moderator, boiling water reactors have a negative *void coefficient*. This means that bubble formation (*void*) along the fuel pins reduces the reactivity. It is therefore common practice to control the reactor power by the main circulation pumps; increased circulation initially reduces the voids (and fuel temperature) thereby increasing power production until the previous void fraction has been almost

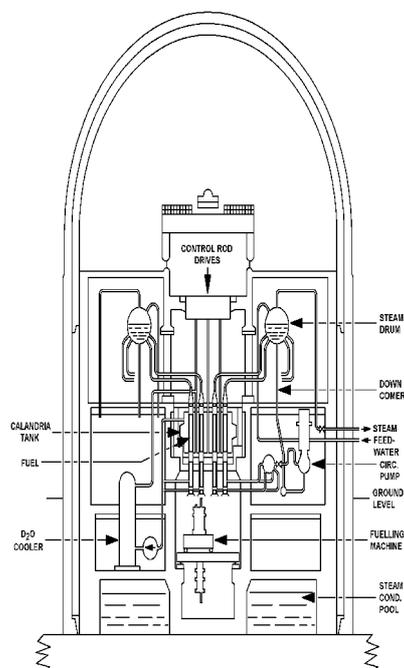


FIG. 20.1. Schematic drawing of the Japanese Advanced Thermal Reactor (ATR).

restored at a slightly higher average fuel temperature. However, a very large negative void coefficient should also be avoided as it can cause an undesirable amplification of pressure fluctuations in the reactor vessel, generating rapidly increasing power oscillations.

In reactors with *pressure suppression (PS) containments* the steam outlet from the reactor vessel is located in the so-called *upper dry well* (Fig. 19.16). In the case of excess pressure in the tank, safety valves at the steam outlet open, releasing steam into the water in the *wet well* through *blow-down pipes* (the wet well condensation pool typically contains  $\sim 2000 \text{ m}^3$  water) thereby condensing the steam. Steam from an inadvertent pipe break inside the containment will also be forced down into the condensation pool. Figure 19.16 shows some emergency systems for the ABB-Atom (Sweden) BWR reactor. In case of multiple failures leading to a too high pressure in the containment, the rupture disk will break and release steam and gas from the wet well through a special filter, thus preventing a failure of the containment.

(a) *BWR*. Most of the typical features of the BWR type is described in §19.13.2. Like the PWR and VVER types, a BWR has to be shut down yearly in order to replace spent fuel. Published data suggests that many BWR:s have a marginally higher net efficiency than the typical PWR. The response of a BWR to an increase in power demand is rather slow because of the void effect and it is therefore not used as power plant for propulsion purposes.

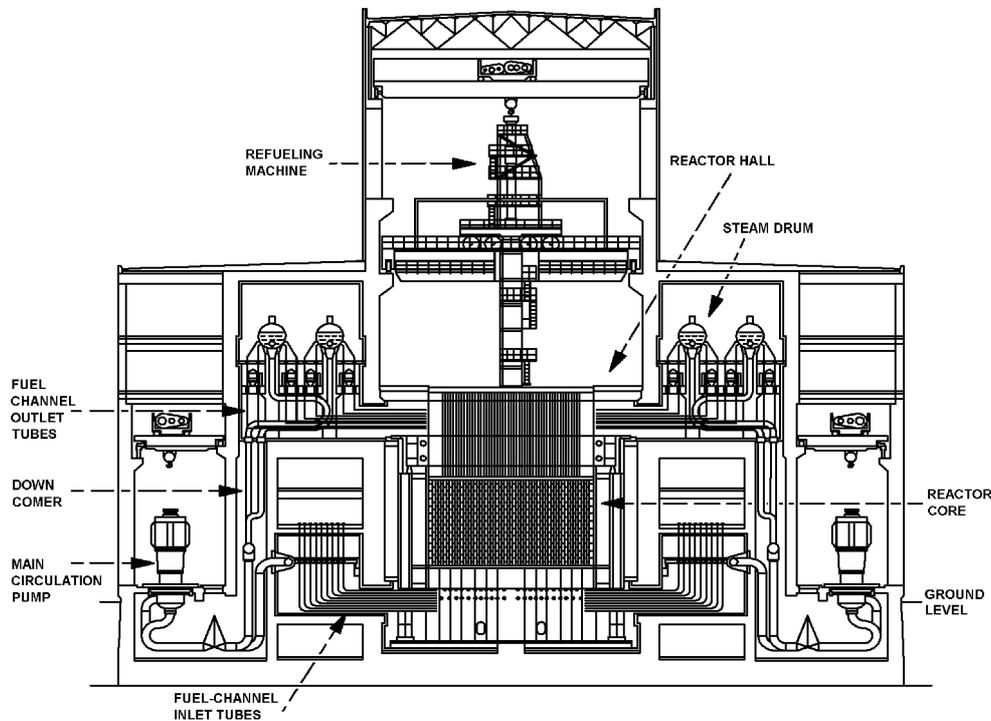


FIG. 20.2. Russian graphite-moderated boiling light water cooled power reactor, RBMK.

(b) *ATR*. The Advanced Thermal Reactor (ATR) is a Japanese  $D_2O$ -moderated boiling  $H_2O$  cooled MOX-fuelled Pu-burner existing as a prototype reactor, see Fig. 20.1. The design has a calandria with vertical pressure tubes, cooled with light water which is allowed to boil in the tubes. This design permits on-line refuelling. ATR reactors are seen as a complement to breeder reactors. A large  $600 MW_e$  ATR reactor will require a fuel supply of  $\sim 460$  kg Pu per year.

(c) *RBMK*. The RBMK is a graphite moderated, boiling  $H_2O$  cooled, pressure tube reactor of Russian design, see Figure 20.2. The use of many pressure tubes (Zr-Nb alloy), each normally containing two fuel bundles, in a large graphite moderator makes it possible to design a boiling water reactor of much larger size and power than possible with a single pressure vessel; up to  $> 2000 MW_e$ . The typical RBMK has  $1000 MW_e$  output, but two units with  $1500 MW_e$  were built in Lithuania. Furthermore, like in the CANDU design, the use of pressure tubes makes on-line refuelling possible; removing one of the reasons for a yearly shut-down period. Monitoring of individual pressure tubes makes it possible to detect and locate failure rather easily leading to a high plant reliability and availability. Subdivision into several coolant loops also increases the safety of the reactor system. The effective moderator makes it possible to operate the RBMK reactors with fuel only slightly enriched in  $^{235}U$ . Boron carbide containing control rods in special channels through the moderator are used to control neutron flux and power distribution.

The many (1452 in a standard RBMK-1000) pressure tubes used makes a pressure tube failure less disastrous than a large pipe break or vessel failure in many other water cooled reactor types. This led to an initial design without any effective containment (6 RBMK-1000 units). However, the 2nd generation RBMK-1000 (8 units) and the RBMK-1500 (2 units) have a pressure suppression pool for steam condensation in case of pressure tube rupture. Operating statistics shows that the design has been very good with regard to availability and annual load factor.

The disastrous Chernobyl reactor accident highlighted some of the undesirable properties of this reactor type in a dramatic way. The use of a large, very effective, graphite moderator in combination with a light water coolant gives the reactor a large positive void coefficient when operated below  $\sim 20\%$  of full power with natural uranium or low enriched fuel; the reactor has a better neutron economy with empty or steam-filled pressure tubes than with water-filled pressure tubes. This feature allowed one of the Chernobyl reactors to increase its power from below 20% to 100 times full power in about 4 seconds, causing a catastrophe. Furthermore, the large core makes the neutron fluxes in different parts of the core only weakly coupled, e.g. one or more regions of the large core may still be critical when one section is made subcritical by control rods. After the Chernobyl disaster, safety margins in the RBMK reactors have been increased by using higher enrichment in the fuel, special absorbers, an improved control system and better operating procedures. The consequences of this accident are discussed in §§5.10.2 and Chapter 22.

It may be noted that the RBMK reactor design was chosen by the USSR (despite warnings from the Soviet Academy of Sciences) because it was better suited to available production facilities than the VVER types (which required the manufacture of large pressure vessels). The rapid introduction of RBMK reactors in the USSR made previously used energy resources (oil and gas) available for export to the West, giving a very needed hard currency income.

### 20.1.3. *Special reactor systems*

So far we have described some reactors designed for conventional power production. In §19.12, reactors for research or isotope production were discussed. We have purposely not included reactor designs which presently are considered obsolete (e.g. the sodium cooled graphite moderated reactor such as Hallam USA, 75 MW<sub>e</sub>), designs which presently seems to have little promise for the future (e.g. gas cooled graphite moderated reactors, GCR), or are unsuited for large scale power production (e.g. the organic moderated and cooled thermal heterogeneous reactor, Piqua, USA, 11 MW<sub>e</sub>). Most reactors of these old types have been, or are planned to be, shut down.

Nuclear power reactors may be used for other purposes than large scale production of electricity (or heat) for community consumption. A new type of helium cooled and graphite moderated prototype reactor, using a special thorium based high-temperature fuel, has been built at Orai (Japan). The design goal was to heat helium to  $\geq 1000^\circ\text{C}$  by the 30 MW<sub>th</sub> reactor. Presently operation is permitted with a gas outlet temperature of about  $900^\circ\text{C}$ . The hot helium produced is intended to drive a new, continuous, chemical two-cycle process for production of hydrogen from water, thus converting nuclear energy and water directly into a possible fuel for combustion engines. For safety reasons the reactor is located under ground.

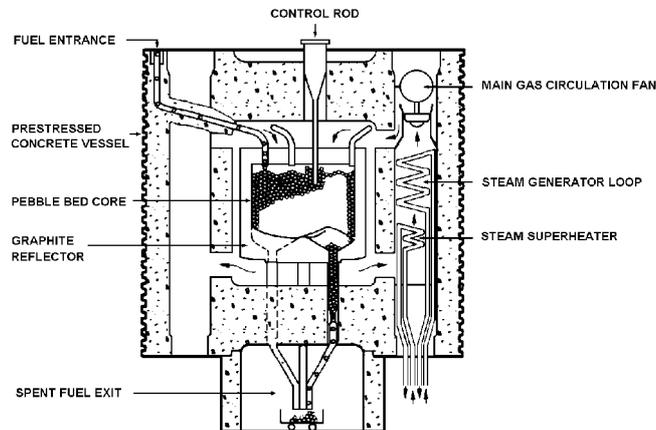


FIG. 20.3. Schematic drawing of the German pebble bed reactor (THTR).

A rather unique design is the pebble bed reactor, see Figure 20.3. It is a helium cooled graphite moderated reactor with a core consisting of a bed of spheres; each about 6 cm in diameter. The fuel is initially based on a mixture of  $^{235}\text{U}/^{232}\text{Th}$  carbides as microspheres in a graphite matrix, see §21.7. The fuel spheres are covered with a layer of silica or silicon carbide. This type of reactor can operate as a thermal breeder or converter with its own  $^{233}\text{U}$  replacing the  $^{235}\text{U}$  in the initial fuel. However, in order to recover new  $^{233}\text{U}$ , the spent fuel has to be reprocessed. Spent fuel is withdrawn at the bottom and new fuel added at the top of the core. The possibility to use slightly different sizes of fuel spheres would permit a mechanical sorting of the spent fuel according to initial type before reprocessing. Also this reactor can produce helium at a rather high temperature. A 15  $\text{MW}_e$  prototype was operated at Jülich (Germany) from 1967 to 1988 and a larger 300  $\text{MW}_e$  reactor was built and operated for about one year (THTR 300). Discussions are under way in South Africa to build a version of the pebble bed reactor that is claimed to be inherently safe.

Both the Japanese and German designs described above have a fuel that can withstand extreme temperatures without melting. Hence a core melt accident in such reactors is practically impossible. Even at high temperatures the fuel integrity is likely to be uncompromized, thus strongly reducing the possibility of radioactive releases to the environment in case of a severe accident. However, after air ingress into an overheated core, burning graphite might cause a release of radioactive material.

The large amount of energy stored in the small volume of fuel makes nuclear reactors suitable as energy sources when repeated refueling is a disadvantage. For example, a nuclear reactor was used for a long time as a power source in the Arctic. Several nuclear powered merchant ships have been built in order to determine their feasibility, but none is presently in use. The now scrapped USSR icebreaker *Lenin* (44 000 shaft horse power, launched in 1959) had three reactors of the pressurized water type, of which one was a spare. Nuclear power is used in many naval vessels in the USA, Russia, UK, China and France. The first nuclear powered vessel was the US submarine *Nautilus*, launched in 1955; in 1958 it made a famous northern voyage from the Pacific to the Atlantic, passing under

the ice at the North Pole. It could travel 250 000 km (~ 6 times around the earth) without refueling. At the end of the year 2000, about 250 nuclear powered vessels (8 icebreakers and the rest naval vessels - mostly submarines) were in operation. These vessels correspond to a total of about 400 reactors.

The present nuclear fleets are dominated by advanced submarines which now number several hundred. Their reactors (small PWR:s) are fuelled by enriched uranium metal (20-50%  $^{235}\text{U}$ ) alloyed with zirconium. The high initial enrichment in combination with burnable poisons and the use of extra neutron shielding to protect the pressure vessel gives a long service life of both the fuel and the reactor; from ~ 4 years between refuellings for the Russian icebreakers to ~ 30 years for the 48 MW reactors used in the French Rubis-class submarines. Hence, many submarines today have reactors that can operate without fuel replacement or pressure vessel annealing during the expected life of the submarine itself. Because of their long endurance submerged, high underwater speed, and the difficulty of detecting and destroying these submarines they are considered to act as deterrents against a nuclear war.

The world's first large nuclear powered naval vessel was the US aircraft carrier *Enterprise*, which has 8 PWR reactors, producing 300 000 shp. With a displacement of ~ 80 000 t (and ~ 100 airplanes) it can cruise at 35 knots for 60 days without refueling.

Finally, 35 small nuclear reactors have been employed by the former USSR and 1 by USA as high power energy sources in satellites. Two naval-type reactors on barges are also being built in Russia to be used for heat and electricity supply in remote locations in the arctic. Reactors have also been developed in the US for use as aircraft and rocket engines.

## 20.2. Chemistry of water cooled reactors

The conditions in water cooled reactors are approximately: temperatures  $\leq 350^\circ\text{C}$ , pressures  $\leq 16$  MPa, and intense  $\gamma$ - and n-irradiation. This causes potentially severe corrosive conditions.

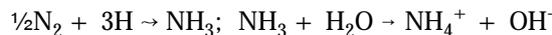
The radiation decomposition in water cooled and/or moderated reactors is considerable. About 2% of the total energy of the  $\gamma$ - and n-radiation is deposited in the water. We have seen in §7.6 that this produces  $\text{H}_2$ ,  $\text{O}_2$ , and reactive radicals. In a 1  $\text{GW}_{\text{th}}$  BWR the oxygen production is about  $1 \text{ l min}^{-1}$ , but it is considerably less in a PWR. Because of the explosion risk from  $\text{H}_2 + \text{O}_2$ , the two are recombined catalytically to  $\text{H}_2\text{O}$  in all water reactors.

Both the radicals and the  $\text{O}_2$  formed increase the corrosion rate of reactor materials. In a BWR the corrosion rate of stainless steel (18% Cr, 8% Ni) is about  $10^{-4} \text{ mm y}^{-1}$ , leading to the release of small amounts of Fe, Cr, Co, and Ni into the cooling circuit. Many of the activation products of these are highly radioactive, e.g.  $^{58}\text{Co}$  and  $^{60}\text{Co}$ . The corrosion products form insoluble voluminous colloid-like products referred to as *crud* (said to be the acronym for Chalk River Unidentified Deposits). In addition to polymeric metal hydroxides, the crud contains small amounts of other materials in contact with the wet circuit: zirconium from the canning material, copper from the condenser system, silicon and organic material from the water purification systems, boron from the boric acid control system, together with fuel and fission products from leaking fuel, etc. Deposition of crud on the fuel element surfaces may block cooling channels, and, because of its poor thermal

conductivity, may also cause a local overheating leading to a penetration of the can and a subsequent release of fuel and fission products into the reactor water. The radioactive corrosion products carried through the cooling loop create a serious radiation problem for the reactor personnel. Considerable effort is put into the development and selection of corrosion-resistant materials. Equally important in this respect is the selection of water conditions which minimize corrosion and the deposition of such products within the reactor core, and effective water cleaning systems. In a PWR the crud is transported through the whole primary cooling circuit, but removed in the purification circuit. In a BWR the crud accumulates in the reactor vessel; therefore water boiling reactors have a special cleaning circuit attached to the reactor vessel. The cleaning circuits are discussed in §§19.13.2 and 20.4.

The amount of crud deposited in the reactor core can be partly removed during shutdown by mechanical cleaning or by washing with chemical decontamination solutions. These solutions contain mildly oxidizing agents such as alkaline permanganate and/or organic complexing agents like ammonium citrate (APAC treatment), oxalic acid or EDTA.

There are a number of ways to reduce corrosion. One is to increase the pH of the water to  $\sim 8$  by adding alkali, e.g. LiOH or  $\text{NH}_3$  ( $\leq 10$  ppm). While reactors of US type use  $^7\text{LiOH}$  in order to reduce the formation of tritium from  $n, \gamma$ -capture in  $^6\text{Li}$ , the VVER reactors normally use KOH. When ammonia is used, the radiolysis yields  $\text{HNO}_2$  and  $\text{HNO}_3$ ; it is necessary to add  $\text{H}_2$  gas to shift the equilibrium from the acidic products. At an  $\text{H}_2$  concentration of  $\sim 2$  ppm, the concentration of dissolved  $\text{O}_2$  is greatly reduced. Instead of  $\text{NH}_3$ , hydrazine or  $\text{N}_2$  may be added to the water to increase the pH via the reactions



The hydrogen content of the water must be kept fairly low to avoid brittleness of zircaloy by hydride formation. Hydride formation may also lead to mechanical deformation of zircaloy parts. Surface pre-treatment is important in order to reduce these effects.

The purity of the reactor water is checked by measurement of pH, conductivity, turbidity, oxygen concentration, radioactivity, etc., either by sampling or – more usual – by on-line analyses.

### 20.3. Breeder reactors

The possibility to breed excess fissile nuclei by consumption of fissile + fertile nuclei increases with increasing fission factor  $\eta$  according to §19.9. Since  $\eta$  depends on the fission cross section, which is highly energy dependent (Fig. 19.3), the neutron spectra in reactors must be considered. In a reactor with large amounts of highly moderating atoms, the original unmoderated spectrum is strongly shifted towards lower energies. However, even under optimal moderating conditions, a thermal Maxwell neutron distribution is never achieved in a reactor, but can be closely approached in thermal columns of research reactors. Because breeding occurs through neutron capture in fertile  $^{232}\text{Th}$  and  $^{238}\text{U}$  (the *only* fertile atoms in nature), with maximum probability between 600 and 700 eV, it is desirable for breeding that the neutron flux not be too low in this region. Just as breeding

cannot be achieved with a thermal Maxwell neutron distribution it cannot be achieved with a fission neutron energy spectrum. Even if attempts are made to produce very "hard" neutron spectra (i.e. high energy), some moderation always occurs. Thus in a fast reactor, in which as few as possible light atoms are used, the neutron spectrum is still shifted strongly towards a lower energy region (Fig. 19.4) so that typically the neutron flux at 0.25 MeV is higher than that at 1 MeV. The curves shown for thermal and fast reactors in Figure 19.4 are only indicative, depend strongly on the particular reactor and fuel design, change with time, and vary over the core.

Because breeding is only possible for  $\eta > 2$ ,  $^{233}\text{U}$  is the favored fissile material for breeding in thermal reactors, and  $^{239}\text{Pu}$  for breeding in fast reactors. Considering the shift towards lower neutron energies in fast reactors it is seen that  $\eta$  for  $^{239}\text{Pu}$  is always greater than for  $^{233}\text{U}$ . Thus  $^{239}\text{Pu}$  can be more rapidly obtained by breeding than  $^{233}\text{U}$ . Consequently much more effort has been put into the development of fast plutonium breeders than into thermal  $^{233}\text{U}$  breeders.

### 20.3.1. The thermal breeder

In the thermal breeder reactor,  $^{233}\text{U}$  is produced from  $^{232}\text{Th}$ . The reactor may be designed either with a core containing a mixture of  $^{232}\text{Th}$  and  $^{233}\text{U}$ , or with a central zone (*core*) of  $^{233}\text{U}$  surrounded by an outer layer (*blanket*) of  $^{232}\text{Th}$ . However, it is necessary to minimize parasitic neutron capture in structural materials including monitoring systems, control rods, etc. Calculations have shown that a conversion ratio of 1.06 should be possible.

As an alternative to using a solid fuel matrix, both core and blanket can be liquid. In the HRE-2 (homogeneous reactor experiment), USA, the core consisted of a  $^{233}\text{U}$  sulfate solution in  $\text{D}_2\text{O}$ , while the blanket was a suspension of  $\text{ThO}_2$  in  $\text{D}_2\text{O}$ . The project was terminated because of various technical difficulties.

The molten salt reactor (MSR) was a somewhat similar design tested at Oak Ridge, USA, for some years. Here  $\text{UF}_4$  and  $\text{ThF}_4$  were dissolved in an eutectic melt of  $\text{BeF}_2$ ,  $\text{ZrF}_4$  and  $\text{LiF}$  (or  $\text{NaF}$ ). The reactor may be homogeneous with respect to  $^{233}\text{U}$  and Th, or of the core and blanket type in which case graphite may be used as moderator. The molten salt system has several advantages: (i) radiation effects are less in molten salts than in solid fuels, (ii) the expense of fabricating complicated fuel elements is avoided, (iii) the fission products can be removed continuously, (iv) fuel recharging can be made continuous, (v) the core contains little absorbing structural materials, and (vi) the system has a high negative temperature coefficient and is therefore easy to control. Points (iii) and (v) makes it possible to obtain a conversion ratio of 1.07, i.e. a 7% breeding gain; the fissile doubling time has been estimated to be 20 y. In the MSR high salt temperatures ( $\leq 700^\circ\text{C}$ ) were used. Nevertheless, the system could be operated near atmospheric pressure because of the low vapor pressure of the melt. The salt is also inert to air and water. By using a dump tank for the core, the risk of criticality accidents was minimized. The on-line fission product removal required novel chemistry, and these problems were never fully solved.

Figure 20.4 shows the production of heavy atoms in the  $^{232}\text{Th}$  system. Because  $^{233}\text{Th}$  is produced until the fuel is removed from the core (or blanket), the long half-life of  $^{233}\text{Pa}$  (27 d) requires long cooling times before reprocessing, if a maximum yield of  $^{233}\text{U}$  is to be obtained. Furthermore, when continuous on-line reprocessing with removal of  $^{233}\text{Pa}$  is not

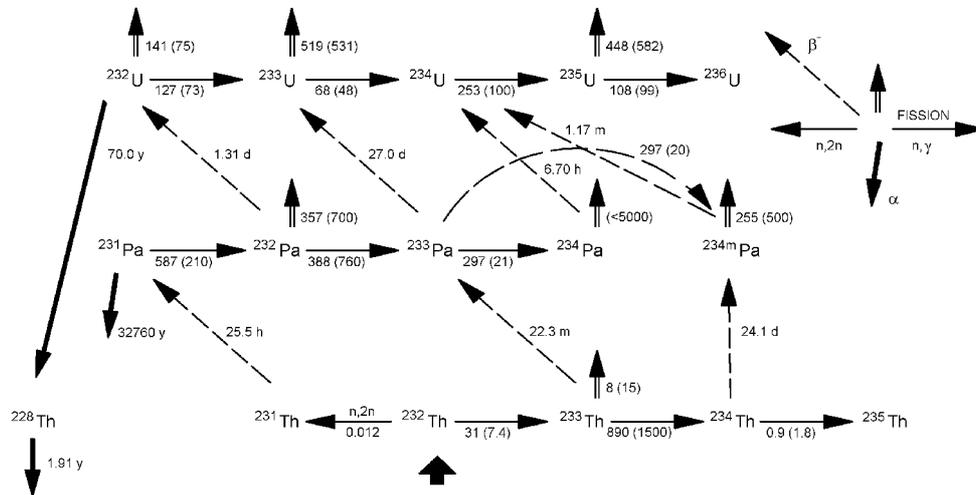


FIG. 20.4. Nuclear reactions in irradiated thorium. Numbers are half-lives or effective reaction cross-sections for a power LWR; 0.025 eV data are given within parentheses.

used, the decay of  $^{233}\text{Pa}$  will lead to a slow increase in reactivity with time after shutdown due to a slightly increasing amount of  $^{233}\text{U}$  from decay of  $^{233}\text{Pa}$ . The thorium would become contaminated by the isotope  $^{234}\text{Th}$  and 240 d would be required to reduce this activity to 0.1%. The  $^{233}\text{U}$  would be contaminated by  $^{234}\text{U}$  and  $^{232}\text{U}$ ; the latter constitutes a problem because of its relatively rapid decay into the  $^{228}\text{Th} \rightarrow ^{224}\text{Ra} \rightarrow$  chain. These isotopic contaminations of the fertile and fissile products in the  $^{232}\text{Th} \text{--} ^{233}\text{U}$  fuel cycle would cause handling difficulties, which are more severe than for the  $^{238}\text{U} \text{--} ^{239}\text{Pu}$  fuel cycle. At present no  $^{233}\text{U}$  breeder exists.

Several research groups have argued that a subcritical thorium breeder, driven by an accelerator based high intensity neutron source, might be a good and safe future energy producer. Using a molten salt/thorium fuel and coolant combination in continuous loops through a  $\text{D}_2\text{O}$  moderator tank or a graphite moderator it could generate a relatively high power from a small inventory of fissile material. It is also possible to design a fast, lead/bismuth cooled, Th-based subcritical breeder by using an intense external n-source. However, no such devices have yet been built and tested.

### 20.3.2. The fast breeder

The principal design of a fast breeder reactor consists of a central core of plutonium in which fission occurs, surrounded by an outer blanket of  $^{238}\text{U}$  in which neutrons are captured to form new  $^{239}\text{Pu}$ . This blanket is surrounded by a reflector, usually of iron. The fission yield curve of  $^{239}\text{Pu}$  is similar to that of  $^{235}\text{U}$  (see Fig. 14.9) but its heavy mass peak is shifted up by a few atomic mass units. Some neutron capture occurs in  $^{239}\text{Pu}$  yielding fertile  $^{240}\text{Pu}$ , which through another neutron capture produces fissile  $^{241}\text{Pu}$ ; similarly, successively higher elements are formed in time. In a strong neutron flux their

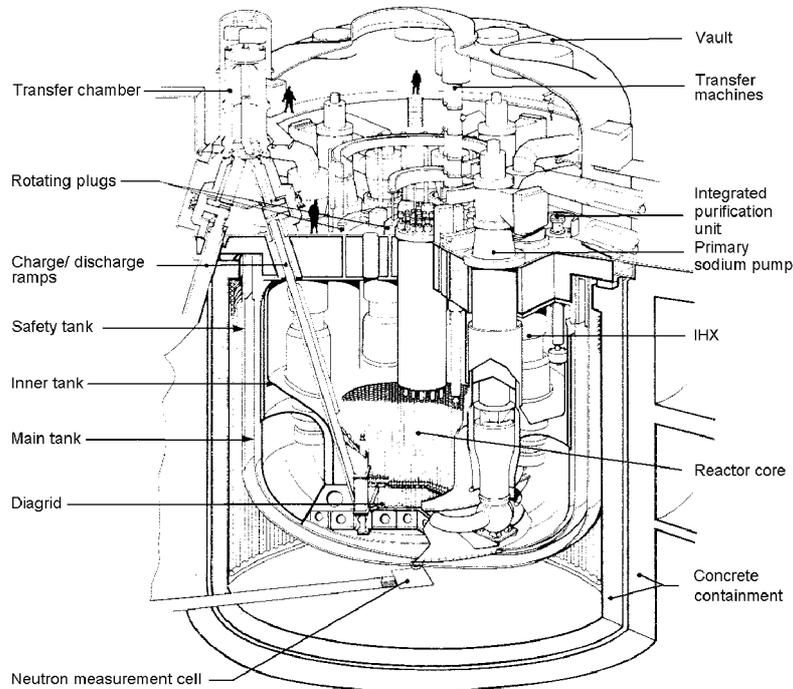


FIG. 20.5. Vertical section of European prototype fast power reactor Creys-Malville. The vessel was filled with 3 300 t of liquid sodium.

fate is destruction by fission. Thus, in the long run, by conversion of  $^{238}\text{U}$  into  $^{239}\text{Pu}$ , etc., fission energy is always released and, in principle, all the  $^{238}\text{U}$  can be used for producing fission energy. In practice a value of  $\sim 70\%$  is considered more realistic. Still this would mean  $\sim 100$  times more energy than is available from fission of  $^{235}\text{U}$  only. Since the fast breeder reactor concept extends the fission energy resources by about a factor of 100, using a proven technology, it makes uranium the largest fossile energy resource presently available on earth.

The conversion efficiency of a reactor of this design can be estimated from

$$C = \eta + \epsilon' - p' - \Lambda' - 1 \quad (20.1)$$

where  $\epsilon'$  is the contribution to the neutron production from fertile nuclides,  $p'$  is the neutron loss due to parasitic absorption, and  $\Lambda'$  due to leakage. Some representative values are

	$\eta$	$\epsilon'$	$p'$	$\Lambda'$	$C$
Core	2.56	0.27	0.049	1.15	0.63
Blanket	-	0.10	0.020	0.032	-
Total	2.56	0.37	0.069	0.032	1.83

It should be observed that the neutrons lost from the core are caught by the blanket.  $C$  is higher than in any other known chain-reacting system.

The longest experience with fast research reactors has been provided by the BN-350 reactor in the former USSR. It has been operated at a power level up to 1000 MW<sub>th</sub> since 1972. Larger (100 – 1200 MW<sub>e</sub>) prototype fast breeder power reactors have been designed in the former USSR, France, the UK, and Japan. The largest prototype FBR ever built and operated is the Creys-Malville plant (located in France), which could produce 1 200 MW<sub>e</sub>.

Figure 20.5 shows the Creys-Malville reactor; in operation from 1986 to 1997. The thermal rating was 3000 MW<sub>th</sub> (gross 1240 MW<sub>e</sub>); the neutron flux was very intense,  $\leq 6 \times 10^{19} \text{ n m}^{-2} \text{ s}^{-1}$ , which puts a severe strain on construction materials. The core consisted of an inner and an outer fuel zone surrounded by a blanket, a steel reflector and a neutron shield, all immersed in liquid sodium (3 300 t) contained in a 25 cm thick austenitic steel vessel at atmospheric pressure. The reactor vessel was enclosed in a safety vessel and surrounded by a cooled concrete (biological) shield. The fuel was made of pins placed in hexagonal shrouds. The fuel zones contained 15 vol.% PuO<sub>2</sub> and 85 vol.% depleted UO<sub>2</sub>, while the breeding blanket only contained depleted UO<sub>2</sub> (0.2% <sup>235</sup>U). The fuel pins were clad in 0.5 mm stainless steel. The total plutonium amount was 4 800 kg. The neutron shield consisted of hollow steel tubes. Because the sodium becomes extremely radioactive due to <sup>24</sup>Na formation, the whole primary cooling system was contained in the reactor vessel. By means of a sodium/sodium intermediate heat exchanger within the reactor vessel (but outside the neutron flux) heat from the primary circuit (542°C) was transferred to a secondary sodium stream (525°C) and transported to a sodium/water steam generator which produced 487°C steam at 17.7 MPa. Heat exchanger and piping were made of steel.

The prompt neutron lifetime in a fast reactor is about 1000 times smaller than in a thermal reactor. A reactor which can go critical on the prompt neutrons only would be exceedingly difficult to control. Therefore, fast reactors are designed to depend on the delayed neutrons (like thermal reactors). The time period ( $t_{\text{per}}$ ) is large enough to allow reactor control through the use of neutron absorbing rods. Since the neutron spectrum is such that several percent of the flux is in the resonance region, control rods with boron can be used; in practice boron carbide (the carbon atom reduces the neutron energy further) and/or tantalum (which has large absorption peaks  $\geq 10^4$  b at 3–100 eV) are used.

Although the temperature coefficient of the Creys-Malville reactor was negative, this is not so for fast Pu-fuelled reactors with harder neutron spectra, e.g. the Russian BN-350. If the fast reactor becomes overheated, the core could be deformed, making it prompt critical. The power would then increase rapidly with a doubling time in the microsecond range, and a severe accident would be unavoidable. To prevent this the core is designed to achieve negative reactivity upon sudden power transients by using <sup>238</sup>UO<sub>2</sub> in the core. When the temperature rises, doppler broadening occurs in the <sup>238</sup>U resonance capture region, and consequently more neutrons are consumed by <sup>238</sup>U, which limits the power increase. The shorter the neutron lifetime, the less would be the power excursion. In addition the thermal expansion of the fuel reduces the reactivity, which makes the temperature coefficient smaller or more negative.

The total breeding ratio in the Creys-Malville reactor was 1.24. In BN-350 a breeding ratio of 1.5 can be obtained because it has less neutron absorbing materials. Still higher breeding gain can be achieved in fast reactors at some sacrifice of safety control.

Considerable radioactivity is induced in the sodium in the primary cooling circuit of a liquid metal cooled fast breeder reactor (LMFBR):  $^{23}\text{Na}(n,\gamma)^{24}\text{Na}$ ,  $^{23}\text{Na}(n,p)^{23}\text{Ne}$ , etc.  $^{24}\text{Na}$  has a 15 h  $t_{1/2}$  and emits energetic  $\gamma$ 's. The primary cooling loop must therefore be well shielded. Its activity is a nuisance only in case of repair work in the primary system, requiring considerable waiting time before the loop can be approached. The sodium dissolves many of the corrosion and fission products eventually released. To remove these the primary loop is provided with cold trap purification systems.

#### 20.4. Reactor waste

In the "ideal" nuclear reactor all fission products and actinides produced are contained in the fuel elements. As mentioned in §19.17, there are four processes in a water cooled reactor through which radioactivity leaves the reactor vessel; in all cases the carrier of activity is the coolant:

- (i) induced radioactivity in the cooling medium (and moderator if separate);
- (ii) corrosion products containing induced activities (mainly from materials in the feed water);
- (iii) leaked fission products and actinides from faulty fuel elements;
- (iv) fission products of actinides deposited on surfaces in the core ("tramp uranium").

We order the discussion of reactor wastes according to the phase which contains the radioactivity (gas, liquid or solid) and end by a brief discussion of waste from decommissioning of nuclear reactors.

##### 20.4.1. Gaseous wastes

The intense flux of fast and thermal neutrons induces several radioactivities in  $\text{H}_2\text{O}$ :  $^2\text{H}(n,\gamma)^3\text{H}$ ,  $^{16}\text{O}(n,p)^{16}\text{N}$ ,  $^{18}\text{O}(n,\gamma)^{19}\text{O}$ , and  $^{18}\text{O}(n,p)^{18}\text{F}$ . Considerable activities of  $^{13}\text{N}$  are produced through the reactions  $^{13}\text{C}(n,p)^{13}\text{N}$  and  $^{14}\text{N}(n,2n)^{13}\text{N}$  with carbon in the steel, and  $\text{CO}_2$  and nitrogen in the water.  $^{14}\text{C}$  is produced through reactions with  $^{13}\text{C}$ ,  $^{14}\text{N}$  (mainly dissolved to 10 – 60 ppm in the fuel), and  $^{16}\text{O}$ ; 400 – 800 GBq  $^{14}\text{C}$  is produced per  $\text{GW}_e$ -year, mainly contained in the fuel. Activation products from B, Li and Ar occur when the cooling water contains these atoms due to the reactions  $^{10}\text{B}(n,2\alpha)^3\text{H}$ ,  $^6\text{Li}(n,\alpha)^3\text{H}$ , and  $^{40}\text{Ar}(n,\gamma)^{41}\text{Ar}$ . In a PWR the tritium content may be 40 – 400  $\text{GBq m}^{-3}$  because of the boric acid and lithium content, while 0.04  $\text{GBq m}^{-3}$  is more typical for a BWR. Though most of these activities are short lived many emit energetic  $\gamma$ -rays in their decay. In a BWR the active gases are transported with the steam to the turbines, where a considerable fraction of the decays occur, thus raising the background in this part of the power station to such a level that personnel cannot be allowed permanently in the turbine hall during operation. Also some radioactivity may be released through leaking gland seals; however, this can be very much reduced by adding a little inactive steam from a small electrically heated boiler to the high pressure side of the seals.

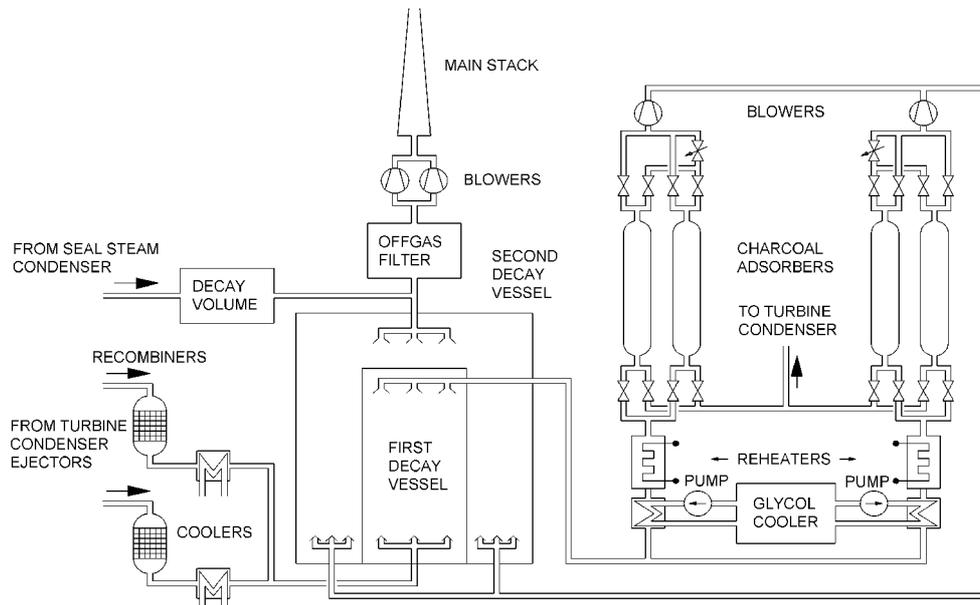


FIG. 20.6. BWR ventilation and off gas treatment system (ABB Atom).

Other activities are also found in the steam:  $^{24}\text{Na}$ , activated corrosion products like  $^{60}\text{Co}$ , and fission products like  $^{91}\text{Sr}$ ,  $^{99\text{m}}\text{Tc}$ , and  $^{137}\text{Cs}$ . These activities are many powers of ten lower than for the gases.

To take care of gaseous products transported with the steam, efficient drainage and suction (air ejector) systems are installed at the condenser side of the turbines. Here most of the activity is swept out and passes through a hydrogen-oxygen recombiner and a delay system (delay times of  $\sim 30$  min are used in older and  $\sim 30$  h in modern BWRs) before the rest of the gases are caught in filters, consisting of absorbing material (e.g. charcoal). Because some of the activity may leak out into the reactor and turbine buildings, the air flow is from areas with low risk to areas with higher risk for contamination. The building ventilation system also contains filters before the air is vented through the stack. Figure 20.6 shows the principle of the off gas system of a modern BWR. By operating the system at reduced pressure leakage to the environment can be avoided. When the charcoal absorbers become saturated they can be heated and the adsorbed gas driven off back into the turbine condenser, thus causing a recirculation which multiplies the available decay time.

The primary release of gaseous activities in a PWR is at the de-aerator after the heat exchanger in the primary loop. The hydrogen-oxygen recombiner is also located here. The gases are compressed and stored for up to 2 weeks before they (via filters) are vented through the stack; these volumes are much smaller than for a BWR, whose off-gas also contains air leaked into the turbine on the condenser side.

If fission products are released by faulty fuel elements, the gaseous ones appear most rapidly in the cooling water:  $^{133}\text{Xe}$ ,  $^{135}\text{Xe}$ ,  $^{85}\text{Kr}$ ,  $^{133}\text{I}$ , etc. The noble gases can be delayed in sand-filled hold-up tanks (permitting shortlived activity to decay) and the remaining

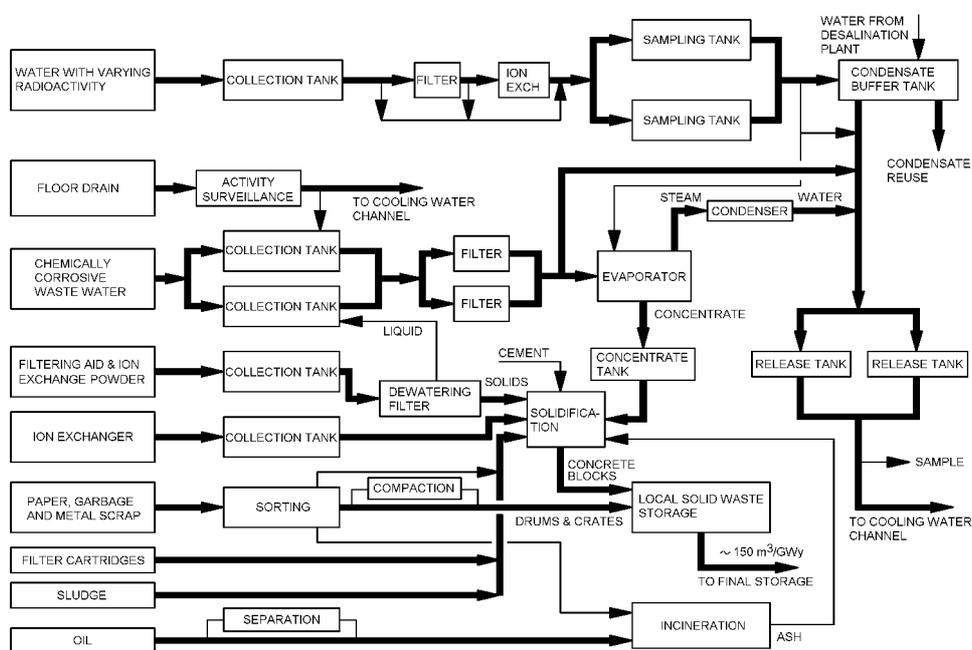


FIG. 20.7. Typical waste management system for a nuclear power station, but many variations exist.

activity retained in charcoal filters. Iodine can be removed by various special techniques. As mentioned above and indicated in Figure 20.6, many BWR plants cyclically regenerate the charcoal filters feeding the released activity back into the coolant system. Some actual release values are given in Table 19.7, see also Ch. 22.

The presence of radioactive noble gases in the off-gas is monitored continuously. Abrupt increases occur when fuel pins are penetrated. By considering the different half-lives and activities it is often possible to make a rough differentiation between tramp uranium, old leaks, pinholes and large new leaks in the fuel.

#### 20.4.2. Liquid wastes

Corrosion and fission products appear in dissolved ionic form and "precipitated" in the crud, depending on the chemistry and water conditions. Most of the corrosion products giving rise to induced activity enter with the feed water. The dominating activated corrosion products are  $^{51}\text{Cr}$ ,  $^{54}\text{Mn}$ ,  $^{59}\text{Fe}$ ,  $^{58}\text{Co}$ ,  $^{60}\text{Co}$ ,  $^{65}\text{Zn}$ , and  $^{124}\text{Sb}$ , and the dominating fission products are  $^3\text{H}$ ,  $^{131}\text{I}$ ,  $^{134}\text{Cs}$ , and  $^{137}\text{Cs}$ . Other fission products and actinides are released in minor amounts depending on the kind and size of fuel element leak. These products are continually removed by cleaning circuits. The aqueous chemistry, and related corrosion and waste problems, of operating reactors is one of the main areas for (nuclear) chemists working in the nuclear power industry.

In a PWR a continuous liquid stream is withdrawn from the coolant in the primary circuit, on the back side of a main heat exchanger. After cooling in another heat exchanger, the water is allowed to pass a filter (e.g. of the precoat type, 25  $\mu\text{m}$  particle size) and a demineralizer which removes ionic species as well as all particulates. The liquid waste streams are detailed in Figure 20.7, which is typical for both PWRs and BWRs. The system is designed to permit segregation of waste streams according to characteristics and type of treatment. The largest liquid waste source from a PWR is the steam generator blow-down, while the largest activities originate at the clean-up systems (condensate demineralizer and filter back-washes); the "demineralizer" may be an inorganic substance (e.g. kieselguhr) or an organic ion exchanger (among the most efficient ones is the Powdex system, which consists of very finely ground,  $\sim 400$  mesh, organic cation and anion exchangers). The boric acid adjustment system of some PWRs may be considered as part of the cleaning circuit; it is discussed in §19.13.1.

A BWR has three liquid purification circuits: one after the condenser for the feed water, one with withdrawal of a small stream from the reactor vessel, and one for the water in the fuel storage pool. The condensate purification system contains similar demineralizers as for the PWR. The reactor water purification system consists of a cooler followed by a mixed bed ion exchanger; also other substances have been used (magnetite, kieselguhr, etc.). A typical capacity figure for the purification system is  $\sim 100 \text{ m}^3 \text{ h}^{-1}$ .

The radioactivity becomes very high in the purification systems. The organic ion-exchange resins used are limited to absorbed doses of  $\sim 10^6$  Gy. The filters are back-washed and resins changed through remotely controlled systems, when measurements show that too much activity is leaking through, the pressure drop becomes too large, or water quality drops. At present resins are not regenerated.

As an additional safety, when filters and demineralizers show insufficient purification of circulating water and of water for release to the environment, an evaporator system may be incorporated. Such a system is normally a part of the PWR boric acid control circuit.

A nuclear power plant contains numerous liquid streams, not all indicated in Figure 20.7, and small controlled amounts of radioactivity are released from the plant. Typical releases are given in Table 19.7 and Chapter 22; these are actual measured values, as pointed out in §19.17 normally constituting a small fraction of the legally permitted releases.

#### 20.4.3. Solid wastes

Solid radioactive wastes are produced from many systems and purification circuits of the reactor station, see Figure 20.7. Many contaminated items can be decontaminated by proper treatment, saving both money and waste storage space. Low level waste can also sometimes be "declassified" and disposed of as normal, inactive, waste or scrap. Combustible solids may be reduced in volume through incineration. Several techniques are used for fixation of wastes in such a way that they can be safely stored with a minimum of surveillance. These techniques all have in common an enclosure of the activity in an insoluble material (normally concrete or bitumen) and in blocks of sizes and activities small enough to be handled with a fork lift.

The waste amounts are typically  $\sim 500 \text{ m}^3$  ILW and  $\sim 5000 \text{ m}^3$  LLW annually from a 1000  $\text{MW}_e$  power station, but vary considerably between different stations depending on

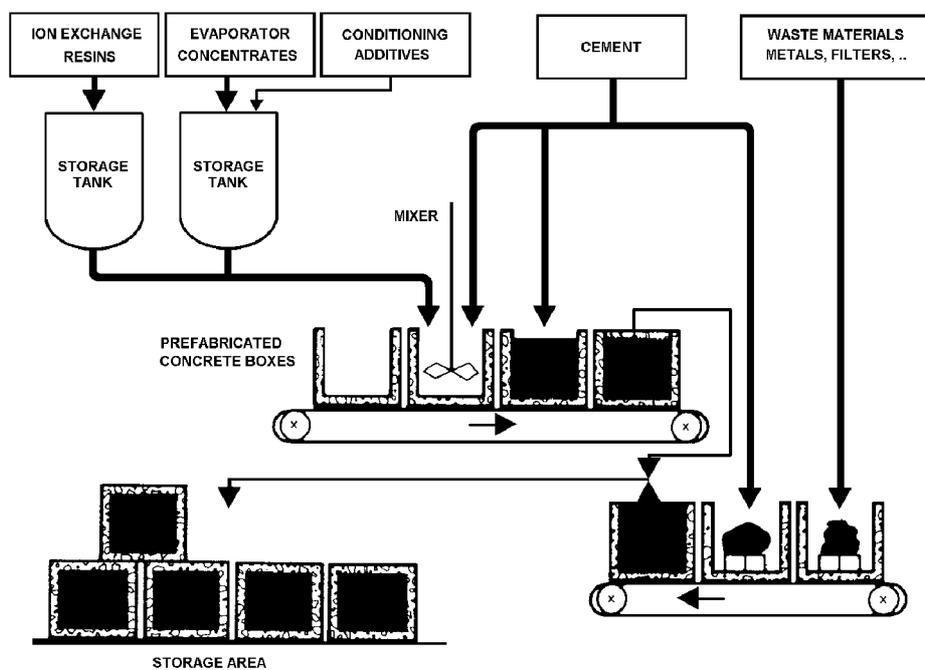


FIG. 20.8. Waste solidification in concrete blocks.

purification and concentration techniques used. The following annual averages are thought to be fairly representative for most LWRs: spent ion-exchange resins  $\sim 30 \text{ m}^3$  ( $\sim 50\%$  water), evaporator residues  $0\text{--}20 \text{ m}^3$  ( $\sim 80\%$  water), filter (condenser) concentrates  $\sim 10 \text{ m}^3$ , other wastes from primary reactor loop  $\sim 5 \text{ t}$ . The activity in the resins is typically  $2\text{--}7 \text{ TBq } ^{60}\text{Co}$ ,  $\sim 0.5 \text{ TBq } ^{137}\text{Cs}$ , and  $\sim 0.05 \text{ TBq } ^{90}\text{Sr}$ , and in evaporator and filter concentrates it is  $< 1/10$  of the resin amount.

Figure 20.8 shows waste solidification and fixation with cement, using prefabricated concrete boxes. This technique is useful both for contaminated or activated equipment and for resins, sludges, etc. The technique, however, increases the waste volume considerably, by a factor 4 to 40 times the untreated waste.

Concrete is a cheap fire and corrosion resistant material. However, active species, especially easily soluble ions such as cesium, can be leached from it by water. Addition of plastic binders to the concrete in order to improve its properties have been suggested. In the continuous bitumen extruder process for semi-liquid wastes all water is directly eliminated, considerably reducing the waste volume. The bitumen mixture is placed in steel drums of standard size (150–200 l). When additional shielding is desired, the filled drums are placed into disposable or reusable sleeves of concrete, iron, or lead. Such a sleeve of 12 cm lead weighs 7 t, and reduces the surface dose rate by a factor of  $\sim 10^3$ . A typical unshielded bitumen drum may have a surface dose rate of  $1 \text{ Sv h}^{-1}$ , necessitating remote handling.

Comparing cement and bitumen, bitumen produces smaller waste volumes (often 5 times less than concrete) with lower initial leach rates. The disadvantage of bitumen is its inflammability and lower radiation resistance; a fire could release large activities of fairly long-lived nuclides. Further, bitumen in wet areas has poor ageing properties, slowly deteriorating. However, for low and medium active solid wastes, bitumen is presently the preferred fixation material at many power stations. About 300 bitumen drums (200 l) with concentrated waste are produced annually per 1 300 MW<sub>e</sub> reactor in Germany; about 50 of these contain medium active waste and have outer concrete shielding (outer volume ~ 400 l).

If the waste does not contain any fission products, a storage time of 50 y would be sufficient to allow all activity (mainly <sup>60</sup>Co) to decay to negligible amounts; storage at plant site would be feasible. If the amount of <sup>90</sup>Sr, <sup>137</sup>Cs, or actinides is high, much longer storage times are required, and a special storage site will be required. If possible, it is desirable to separate these products from other waste; as an alternative the isolated <sup>90</sup>Sr, <sup>137</sup>Cs, and actinides could then be added to the high level waste of reprocessing plants.

The cementation and bitumenization techniques are also used for intermediate level wastes arising in other parts of the nuclear fuel cycle, e.g. reprocessing. The long-term storage of the solidified waste is described in Chapter 21 and some of its possible future impact on the environment in Chapter 22.

#### 20.4.4. Decommissioning nuclear reactors

Most of the waste from the dismantling of decommissioned nuclear reactors is non-radioactive. Also, large volumes can be decontaminated and "declassified" which permits them to be treated as normal industrial waste. Of the remaining waste, most contains low levels of relatively short-lived nuclides similar to wastes from normal reactor operation. Hence, it can be treated and disposed of in the same manner, e.g. put in bitumen or concrete, and stored in repositories for LLW and ILW (§21.9). The volumes may be rather large, however. Some internal parts of the reactor have relatively high activity. The induced activity is limited to part of the pressure vessel (closest to the core) and some core components.

The inner surface of the pressure vessel is normally covered by highly radioactive deposits. These deposits can be dissolved and removed by a repeated treatment with suitable chemicals. As an example, after more than 20 years of operation the pressure vessel of the BWR reactor Oskarshamn 1, Sweden, has recently been decontaminated to such an extent that workers could enter the empty vessel to perform necessary modifications and tests with only a cylindrical radiation shield placed inside the central part of the vessel wall to protect against induced radioactivity in the wall. Upper and lower parts of the pressure vessel are normally fairly well shielded from the neutron flux and only contain a low level of induced radioactivity. For chemical decontamination, i.e. dissolution of deposits, weak acids will dissolve hydroxides and complex formers (such as chelating agents) bind metal ions in soluble complexes. However, in some countries (e.g. Sweden) authorities forbid the inclusion of strong complex formers in the reactor waste. Therefore, in the case of Oskarshamn 1, oxalic acid was used; after dissolution of the deposits and fixing the metal ions on ion-exchange resins oxalic acid was destroyed by ultraviolet light yielding CO<sub>2</sub> and

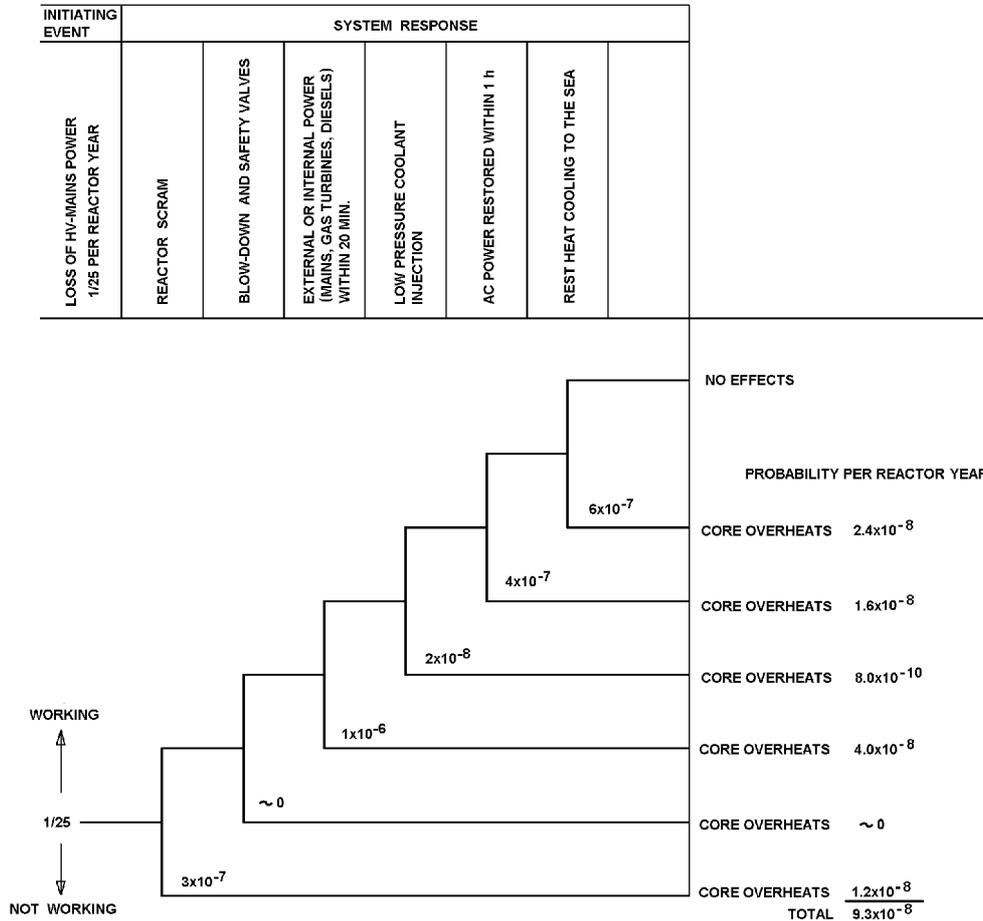


FIG. 20.9. Example of an event tree for a hypothetical BWR.

H<sub>2</sub>O. Thus, a pressure vessel of a decommissioned reactor could, after chemical decontamination, be cut into smaller sections, some of which would only be slightly radioactive. The resulting scrap can either be encapsulated in concrete and put in a repository for normal reactor waste (Swedish plans), or melted to be used as raw material for new reactor parts where the remaining level of activity is acceptable (France).

### 20.5. Safe operation of nuclear reactors

The major risks from reactor operation stem from operator mistakes (< 10%) or failure of critical hardware (> 90%). The risk from human errors is reduced by recurrent training of personnel, especially persons working in the control room, using mockups and

simulators. Thereby personnel in critical functions train their behavior in normal situations, but also in situations which seldom or never occur during routine operation of the reactor. Also exchange of experience and the results from analysis of actions violating or threatening to violate safety instructions help to reduce the risk for human errors. Risks from hardware failures or malfunctions are minimized by a continuous work aiming to identify risk dominating sequences and critical hardware. Once identified, such sequences should be modified, procedures changed, equipment replaced or duplicated in order to reduce the overall risk until it reaches an acceptable level.

Figure 20.9 shows an example of an *event tree* starting with an *initiating event* to the left and branching each time an action takes place. The upward branch corresponds to a success of the action specified at the top, whereas the downward branch represents a failure. Each branch is normally the compounded effect of actions in a sub-tree describing the function and interaction of affected hardware. The figures at each downward branch are estimated probabilities for a failure of the action at the top. The line representing the outcome of an event is then followed to the right; either to a final state or to another branch. To the extreme right is the final state with its estimated probability. For example, the most important risk in Fig. 20.9 arises from failure to restore AC power within 20 minutes; probability  $(1/25) \times 10^{-6} = 4 \times 10^{-8}$  per reactor year. For independent outcomes of events in a branch (e.g. probability of failure of too many of the available safety valves together with failing operation of the blow-down pipes), the probability is the product of the probabilities at the branches taken in the sub-tree. In case of common mode failures, the compounded probability is the same as the probability of a single event of that type. The probabilistic event tree is an example of the type of tools used to identify the current risk dominating event chain. When carried out meticulously, operator training, experience feedback and risk analysis help to continually improve the safety of reactor operation.

## 20.6. Exercises

**20.1.** The gas space (volume  $6000 \text{ m}^3$ ) in the PS system of a large BWR is filled with  $\text{N}_2$  at 97 kPa and  $25^\circ\text{C}$ . Assume the same temperature in the dry and wet wells and that the heat of evaporation of water and its heat capacity are  $\sim 2.26 \text{ MJ/kg}$  and  $\sim 4.18 \text{ kJ kg}^{-1} \text{ }^\circ\text{C}^{-1}$ , respectively, up to  $\sim 100^\circ\text{C}$ . The  $2000 \text{ m}^3$  water in the condensation pool is at  $25^\circ\text{C}$ . The primary system contains steam at  $287^\circ\text{C}$  and 7 MPa (energy content  $\sim 2.8 \text{ MJ/kg}$  steam). How many kg of such steam could be dumped into the condensation pool before the pressure in the PS system exceeds 0.1 MPa?

**20.2.** The amount of steam found in example 20.1 is released into an air-filled large dry containment where the pressure may not exceed 0.7 MPa. Assume steam and air in the containment can be treated as ideal gases. Estimate the necessary containment volume.

**20.3.** In the Chernobyl accident, reactor output increased from  $\sim 20\%$  to 100 times full power in about 4 seconds. Assume a constant reactivity excess during this time. a) What was the average reactor period? b) How much energy was released during this time?

**20.4.** KOH is normally used for pH control in VVER reactors. Which radioisotopes will be formed from potassium in the coolant?

**20.5.**  $^{232}\text{U}$  is formed in a thermal  $^{232}\text{Th}$  based breeder. a) By which chain of reactions is it formed? b) Will the choice between a continuous or discontinuous (once a year) reprocessing of the Th blanket affect the isotopic composition of the isolated uranium (assume that all other actinides remain in the blanket)?

**20.6.** Use eqn. (19.33) to estimate the maximum thermal efficiency for the Creys-Malville plant.

**20.7.** Assume that the probability of not restoring AC power within 20 minutes is reduced by a factor of 5 in Fig. 20.9. a) Which is now the risk dominating event chain? b) Is it better to reduce the total risk in Fig. 20.9 by this change or by improving the stability of the HV mains with a factor of 2 by e.g. building more power stations?

**20.8.** What is the fate of the gas released into the turbine condensers during regeneration of the charcoal absorbers in Fig. 20.6?

### 20.7. Literature

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