2. Design Basis Accidents

Design basis accidents are postulated accidents to which a nuclear plant, its systems, structures and components must be designed and built to withstand loads during accident conditions without releasing the harmful amounts of radioactive materials to the outside environment. Any DBA is controlled by the reactor safety systems with insignificant off-site consequences, but may require long shutdown for correction or repair.

The concept of DBA is very useful both for normal and abnormal operation. Design basis accidents are used in the design of a nuclear power plant to establish the performance requirements for reactor structures, systems and components.

More serious accidents that may involve significant core degradation and/or pose the real danger of a significant release of radiation to the environment are classified as beyond DBA or severe accidents. These accidents have an extremely low probability of occurrence, so low that full control of them is not considered in the design.

A set of design basis accidents is postulated for each type of reactor, covering the consequences of all combination of failures. According to the "Typical Contents of Technical Justification of Nuclear Power Plant Safety" [5] the following groups of DBA are considered for the RBMK-type reactors:

I. Reactivity initiated accidents
II. Loss-of-flow transients
III. LOCA
IV. Accidents induced by equipment failures
V. Fuel handling accidents
VI. Other accidents.

Design basis accident*s* are classified by type of initiating events. Reactivity initiated accidents cover such initial events as single rod withdrawal or voiding of control rod channel cooling system. Single or multiple main circulation pump trip are examples of loss-of-flow transients. Loss-of-coolant accidents include full or partial breaks of group distribution header, breaks of main steam line in different locations, etc. Loss of preferred power or turbine trip can be referred to accidents initiated by equipment failure. Fuel handing accidents cover failures during refueling machine operation. Flooding, fire, earthquakes are examples of the last group of accidents referred as other accidents (external events).

The design limits prescribe that for any DBA:

- fuel cladding temperature must not exceed 1200°C;
- the local fuel cladding oxidation must not exceed 18% of the initial wall thickness;
- the mass of Zr converted into ZrO_2 must not exceed 1% of the total mass of cladding;
- the whole body dose to a member of the staff must not exceed 50 mSv;
- critical organ (i.e., thyroid) dose to a member of the staff must not exceed 300 mSv.

According to the latest safety regulations issued in 1988, even in the case of double-ended break of a pipe with the maximum possible diameter, certain limiting conditions must be met without excessive consequences to the environment. For the Ignalina NPP, an instantaneous guillotine break of the MCP pressure header which internal diameter is 900 mm with unimpeded discharge of coolant from both ends of the pipe during the unit operation at full power is taken as the ultimate design basis accident. In accordance with the single failure requirement, this event is assumed to coincide with a failure of a check valve in one group distribution header to close. The analysis performed for ultimate design basis accident has shown that design limits will not be exceeded.

3. Main Safety Functions and Systems

According to the IAEA “Code on the Safety of Nuclear Power Plants: Design” [6] safety systems are “systems important to safety to assure the safe shutdown of reactor or residual heat removal from the core, or to limit the consequences of anticipated operational occurrences or accident conditions". Anticipated operational occurrences are events that are expected during the operating life of nuclear reactor, as opposed to the accident conditions which are events that occur unexpectedly and used as design basis for engineered safety futures to limit the release of radiation. For the RBMK-1500 reactor, the above mentioned main safety functions are assured by the following safety systems: Control and Protection System, Emergency Core Cooling System, Emergency Feedwater System, Accident Confinement System, Pressure Relief System and Reactor Cavity Overpressure Protection System.

3.1 Control and Protection System

The reactor protection system consists of several subsystems, which are designed to run back the reactor power or to scram the reactor in response to a variety of trip signals. A power runback to 60% of full power occurs in response to AZ-4 signal. A power runback to 50% of full power occurs in

The power runback is accomplished by inserting control rods until the desired power level is achieved. A reactor scram can be either a fast-acting scram (FAS signal) or the normal scram (AZ-l signal), which is slower. A fast-acting scram is intended to quickly insert control rods for conditions which could lead to the rapid generation of positive reactivity. The slower AZ-l rod insertion is effected on response to transients which would not produce a rapid positive reactivity insertion.

3.2 Emergency Core Cooling System

The primary purpose of the ECCS is to provide coolant makeup to the main circulation circuit. This is required to mitigate the effects of loss-of-coolant accidents or for transients resulting in a loss of main feedwater. The ECCS**,** Figure 3.1, consists of two subsystems: short-term ECCS and long-term ECCS. The short-term ECCS consists of high pressure accumulators containing emergency coolant and pumped injection from the main feedwater pumps.

This subsystem is activated on signals indicating a break in the main circulation circuit. The purpose of this subsystem is to provide immediate cooling to the failed

ECCS coolant is therefore

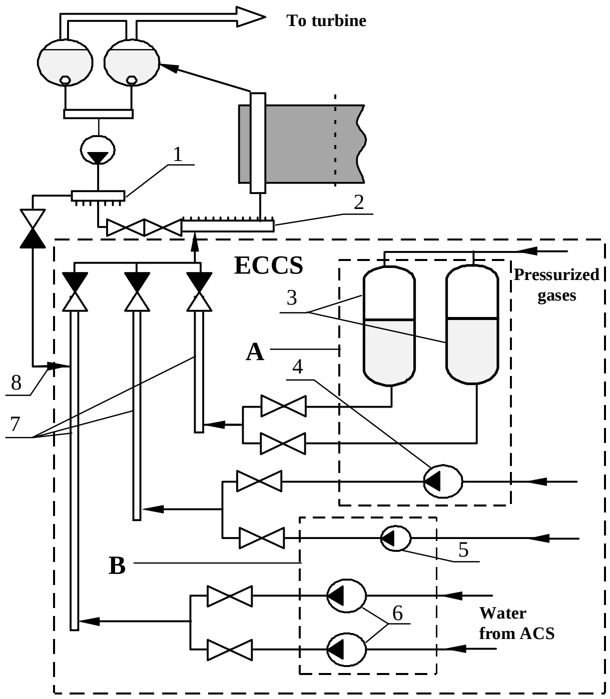

Fig. 3.1 Emergency core cooling system flow diagram [4]:
A - short-term cooling subsystem; B - long-term cooling subsystem; 1 - MCP pressure header; 2 - group distribution header; 3 - ECCS accumulator; 4 - main feedwater pump; 5 - auxiliary feedwater pump; 6 - ECCS pump for pumping water from ACS hot condensate chamber; 7 - ECCS header; 8 - ECCS bypass

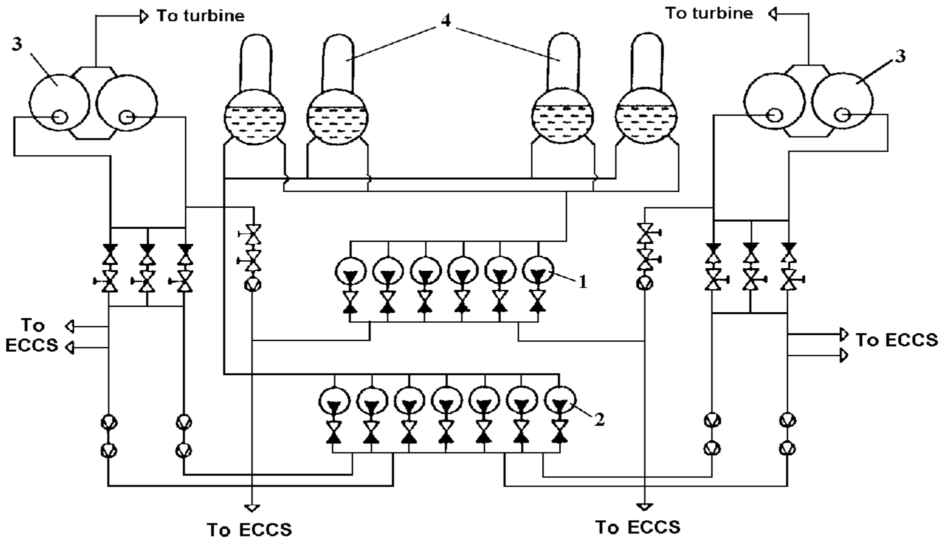

Fig. 3.2 Emergency feedwater system flow diagram:
1 - emergency feedwater pump; 2 - feedwater pump; 3 - steam separator; 4 - deaerator

injected into the GDH on the affected side of the reactor. The long-term ECCS consists of six motor-driven ECCS pumps (plus one standby pump) and/or five auxiliary feedwater pumps (plus one standby pump). The primary purpose is to provide the long term reactor coolant inventory control. This coolant is injected into both halves of the reactor.

3.3 Emergency Feedwater System

The flow diagram of EFWS system is shown in Figure 3.2. When activated, the EFWS provides makeup water to the main circulation circuit. This system can act as part of the ECCS long-term subsystem, as described in Section 3.2., or it may act independent of the ECCS. In the former mode of operation, the emergency coolant is injected into the GDH. For some upset conditions, the ECCS is not activated, but the makeup is still required, e.g. low feedwater flow. For such conditions, the EFWS provides additional coolant to the steam separators.

3.4 Accident Confinement System

The largest and most important system protecting the plant personnel and the environment from radiation in the event of an accident, is the accident confinement system. The ACS structure is shown schematically in Figure 3.3. The system used at the Ignalina NPP belongs to the pressure suppression category of containment's. The ACS must ensure the protection of personnel and environment from radioactive contamination above authorized limits following any DBA, including the ultimate DBA. This is achieved in the following way:

- the steam discharged during the accident is condensated reducing the pressure in the accident areas, and correspondingly, the release of radioactive materials from these areas;
- radioactive materials released during the accident are held in an enclosed compartment until they can be decontaminated;
- in the first stage of an accident, clean air is released from the ACS;
- a sprinkler system is used to condensate the steam in the remaining ACS compartments.

The release of clean air and the use of sprinklers allows the ACS pressure to be reduced to the relative atmospheric pressure, which decreases the possibility of release of radioactive materials to the environment. The ACS also serves as a water reservoir. This water is used for emergency core cooling, as well as to condensate any steam discharged by the safety/relief valves.

The defence-in-depth concept is not completely applied to Ignalina NPP with respect to confining all parts of the reactor coolant circuit, e.g. upper sections of pressure tubes, steam-water communication lines between steam separators. If a break occurs in these lines, steam is discharged directly to the atmosphere.

The free volume of the ACS is relatively small (less than 2800 m^3) with a high degree of compartmentalization. It is obvious that high H_2 concentration of 1% cladding oxidation can locally occur in some of the compart-ments. Analyses have shown that flammable concentrations are avoided for DBA, but no evidence was provided that the hydrogen removal capacity is sufficient to cope with the maximum credible hydrogen production during beyond DBA.

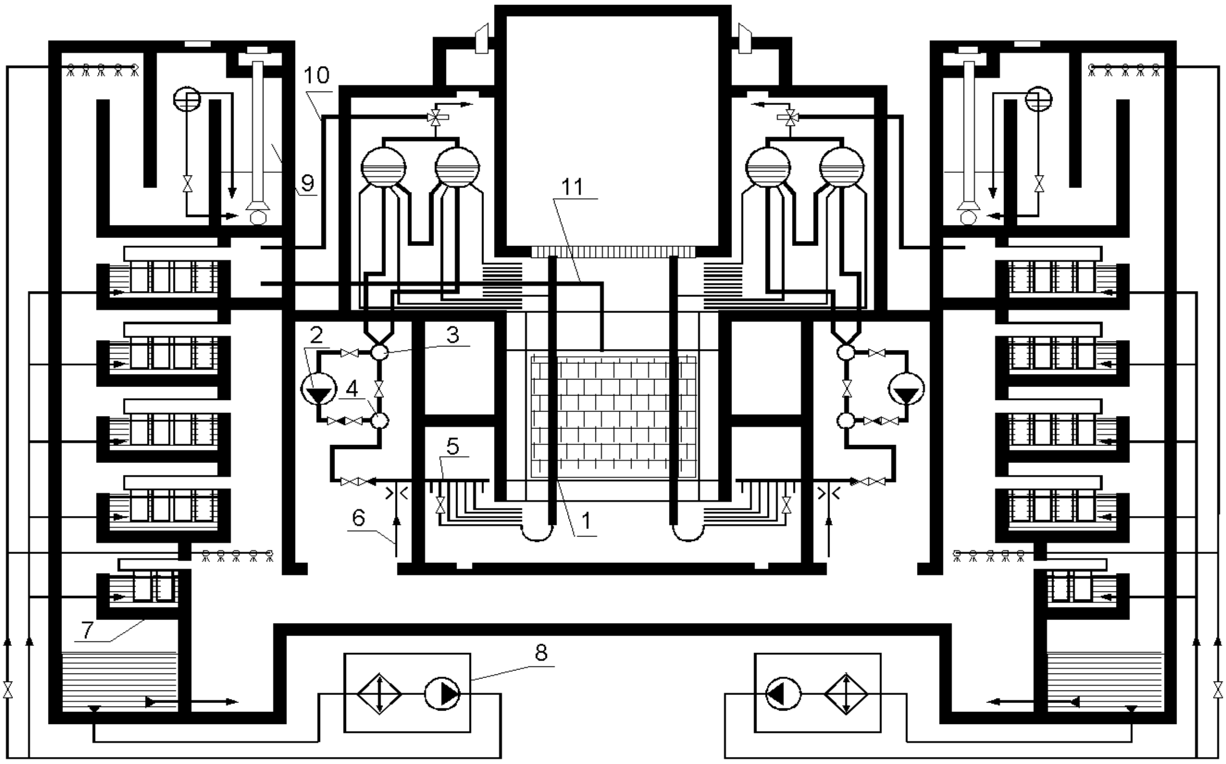

Fig. 3.3 Simplified schematic of the accident confinement system [4]:
1 - fuel channel; 2 - main circulation pump; 3 - suction header; 4 - pressure header; 5 - group distribution header; 6 - ECSS header; 7 - suppression pools; 8 - ACS heat exchanger; 9 - air discharge pipe section; 10 - steam pipe from MSVs and SDV-A; 11 - air-gas mixture from the reactor cavity

3.5 Pressure Relief System

Normally, the steam flows through the turbine control valves to the high pressure turbines. These valves are isolated on a number of plant upset conditions. Steam, however, continues to be generated in the core. To control pressure in the MCC, this steam is relieved either through the steam discharge valves, or through the main safety valves. The SDVs are power-operated and operate through an active control system to maintain pressure. Eight SDV-C valves provide a steam exhaust to path directly to turbine condensers. These valves are used for pressure control during a normal plant shutdown, as well as operational upset conditions. Two additional SDV-A valves relieve steam to the suppression pool of ACS system. There are twelve MSVs which are set to operate in three banks. When the MCC pressure reaches the specified setpoints for the first bank, valves of the first bank open and discharge steam to the suppression pool. If pressure continues to rise, the second and third bank valves may also open. Table 3.1 gives a list of setpoints, reset parameters and other characteristics of SDVs and MSVs.

Table 3.1 Setpoints and parameters of steam discharge and main safety valves

Valve	**Number of valves**	**Capacity per valve, kg/s**	**Opening pressure, MPa**	**Reset pressure, MPa**	**Delay for opening/ closing, s**	**Opening/ closing time, s**
SDV-C	8	152.78	7.06	6.77	-	10/10
SDV-A	2	97.22	7.16	6.87	-	10/10
MSV, Bank 1	2	97.22	7.46	7.16	4/10	0.5/0.5
MSV, Bank 2	4	97.22	7.55	7.26	4/10	0.5/0.5
MSV, Bank 3	6	97.22	7.65	7.36	4/10	0.5/0.5

3.6 Reactor Cavity Overpressure Protection System

Protection of the reactor cavity against overpressurization is an important part of the safety system of RBMK-1500. A simplified schematic of this system is shown in Figure 3.4. Cavity pressure exceeding 0.31 MPa has been described as having the possibility to lift the upper-head shield assembly breaking the reactor seal, the pressure tubes, and affecting the operation of other safety functions. The requirement for overpressure protection system is to have the capacity for coping with DBA involving one pressure tube rupture. The new overpressure protection system that has the capacity to withstand multiple rupture of up to 9 pressure tubes under conservative assumption of simultaneous ruptures was installed at INPP in l996. Pressure relief is provided by pressure relief tubes which connect the reactor cavity to the ACS. The reactor cavity venting system vents the steam-gas mixture from the cavity to the pressure suppression pools where steam is condensated while the gas is retained in the leaktight spaces.

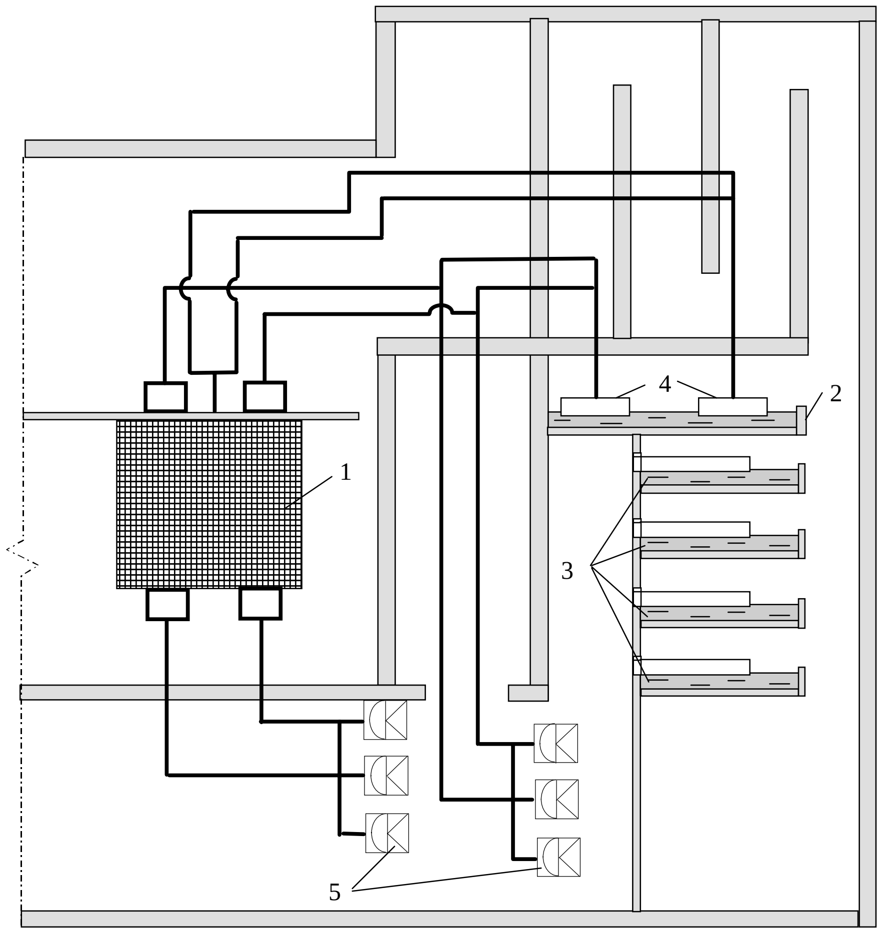

Fig. 3.4 Simplified schematic of the reactor cavity overpressure protection system:
1 - reactor; 2 - the fifth ACS suppression pool; 3 - suppression pools 1-4; 4 - steam distribution devices; 5 - membrane safety devices