Chapter 2
The Nuclear Power Plant with WWER440 Reactors

The acronym water–water energy reactor (WWER) refers to Soviet design water-cooled, water-moderated, and electricity-generating reactors. The designer of WWER440 reactors is Gidropress. The WWER1000 reactors are designed by Atomenergoproject.

The WWER440 reactor has a designed net electrical output of 440 MW(e) corresponding to thermal power output of 1,375 MW. The nuclear fuel is enriched uranium, and moderator as well as coolant is water with variable concentration of boric acid. Currently, the power uprate was performed in many WWER440 plants. The thermal power output is increased by about 7 % (1,375 MW $\times$ 1.07 $=$ 1,471.25 MW).

The WWER440 reactor belongs to the most prevailing type of light water reactors. Operation of WWER440 and the Western-type PWRs is based on the same principle. However, there are essential differences both in design and materials used.

The main distinguishing features of the WWER440 compared to other PWRs are as follows:

- horizontal steam generators with high feedwater capacity (3–4 times higher than in case of some Western-type PWRs),
- hexagonal fuel assemblies,
- no bottom penetrations in the pressure vessel and
- high-capacity pressurizer.

The first pressurized water reactor was commissioned in the former Soviet Union in 1963 at Novovoronezh. This unit, designated as WWER210, was followed by a second prototype, a 365 MW version that became operational in 1969. From these prototypes, a standardized 440 MW nuclear power plant (WWER440) was developed. The first WWER440 reactors have the standard plant design referred to as model V230.

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The design basis accident for the WWER440/V230 reactor is a pipe rupture with an effective 100 mm diameter. Special orifices reduce the flow to an amount equivalent to a diameter of 32 mm. The model V230 plants have limited capacity for emergency core cooling. The design of the emergency core cooling system (ECCS) differs among the plants. The majority of WWER440/V230 plants comprise a high-pressure safety injection system that provides coolant from a 800 m^3 borated water storage tank with provision for high-pressure recirculation of the coolant from the sump of the confinement. Normal make-up system is used to compensate losses during operation of the plant. In some older WWER440/V230 plants, a high-pressure injection system provided coolant from a tank, but had no provision for high-pressure recirculation. This system was also used for certain make-up functions performed during normal operation, including plant start-up and shutdown.

The first WWER440/V230 reactor was built in Novovoronezh Unit 3 and became operational in 1971. The WWER440/V230 reactors were in operation from the early 1970s in Armenia, Bulgaria, the former Czechoslovakia, the former German Democratic Republic and the former Soviet Union. At the present time, these reactors are in operation in Armenia (V270 model) and the Russian Federation.

A modernized version of WWER440, model V213, was a product of the first nuclear safety standards adopted by Soviet designers. This model has improved the emergency core cooling systems. Full-scope ECCS is installed with low-pressure safety injection pumps and hydro-accumulators. The accident localization system is upgraded using the bubble tower. This is the most significant addition to plant safety. The bubble tower is incorporating a large number of water trays serving as suppression pools in which extensive steam condensation occurs during loss of coolant accident (LOCA) conditions. For each unit, a set of pressure suppression trays is located inside a separate building adjacent to the reactor building. The design basis accident for the WWER440/V213 reactor is a double-ended guillotine break of the primary circuit pipe with an effective 500 mm diameter.

The WWER440/V213 reactor was commercially introduced in the former Soviet Union in 1980/1981 at Rovno (units 1 and 2). The WWER440/V213 reactor was also used at Loviisa nuclear power plant in Finland; however, the accident localization system was replaced by a Western-type containment structure of Westinghouse design. Currently, more than 15 units with a WWER440/V213 reactor are in operation in the Czech Republic, Finland, Hungary, the Russian Federation, the Slovak Republic and Ukraine. List of WWER440 plants is presented in Table 2.1.

In the next part of this chapter, overview of the WWER440/V230 and WWER440/V213 reactor design is provided [1–5].

Table 2.1 List of WWER440 plants

Power plant	Country	Reactors	Notes
Bohunice	Slovakia	2 × WWER440/V230 2 × WWER440/V213	Split in two plants: V1 with V230 units and V2 with V213 unit. Operation of the WWER440/V230 units was terminated in the end of 2006 (unit 1) and 2008 (unit 2). Power uprate of the V2 units was performed to 107 % of nominal power. Planned shutdown of the V2 units in 2025
Dukovany	Czech Republic	4 × WWER440/V213	Power uprate was performed on all units to 106 %
Greifswald	Germany	4 × WWER440/V230 1 × WWER440/V213 (3 × WWER440/V213)	The plant is decommissioned. Unit 6 finished but never operated, unit 7 and unit 8 construction was suspended
Kola	Russia	2 × WWER440/V230 2 × WWER440/V213	The V230-type units are modernized. Planned shutdown in 2018 and 2019
Kozloduy	Bulgaria	4 × WWER440/V230	Operation of the WWER440/V230 units was terminated in 2004–2007
Loviisa	Finland	2 × WWER440/V213	Western-type control systems and containment structures. Power uprate to 496 MW. The plant will be in operation until 2027 and 2030, conditional on safety reviews before 2015 and 2023
Metsamor	Armenia	2 × WWER440/V230	Both units were shutdown in 1988 due to earthquake in Armenia. The unit 2 was brought back to operation in 1995 as V270 model. Safety upgrading of the plant was performed
Mochovce	Slovakia	2 × WWER440/V213 (2 × WWER440/V213)	Units 3 and 4 are under construction; the units are planned to be operational in 2015 and 2016. Power uprate was performed on units 1 and 2 to 105 % of nominal power
Novovoronezh	Russia	2 × WWER440/V179	Both units (unit 3 and 4) are prototypes of WWER440. Safety upgrading of the units was performed. Planned shutdown in 2016 and 2017
Paks	Hungary	4 × WWER440/V213	Power uprate was performed on all units to 105 % of the nominal power Lifetime extension is considered. The plant may remain in operation for another 20 years beyond the 30-year design lifetime
Rovno	Ukraine	2 × WWER440/V213	Lifetime extension is considered beyond the 30-year design lifetime

2.1 Overview of the WWER440/V230 Reactor Design

2.1.1 The Original Design

The WWER440/V230 units were fundamentally designed as a twin unit plant with numerous interconnections between the units. These interconnections were explicitly considered in the original safety case. Additional safety improvements and modifications implemented over the last decades were also based on assumed operation of both units.

A sketch of the major buildings and components of WWER440/V230 plant is given in Fig. 2.1.

All WWER440 plants comprise six loops (see Figs. 2.2 and 2.3). Each loop has a horizontal steam generator, a main coolant pump and two main isolation valves. In addition, the pressurizer is connected to a loop. The reactor is a pressurized water reactor. The primary circuit is a closed loop cooling circuit and removes heat from the reactor core. The reactor coolant acts also as a moderator. The coolant is chemically treated water. The flow of coolant through the reactor (approx. 43,000 m^3/h) is driven by the main coolant pumps. Primary water with a temperature about 300 centigrade flows to the steam generators where its thermal energy is transferred to the secondary side.

The primary coolant system including steam generator heat transfer tubes is invariably in stainless steel. The secondary side including SG shells is in carbon steel.

The steam generated in the steam generators are supplied to the steam turbines. The secondary-side equipment of the plant, such as feedwater and condenser water

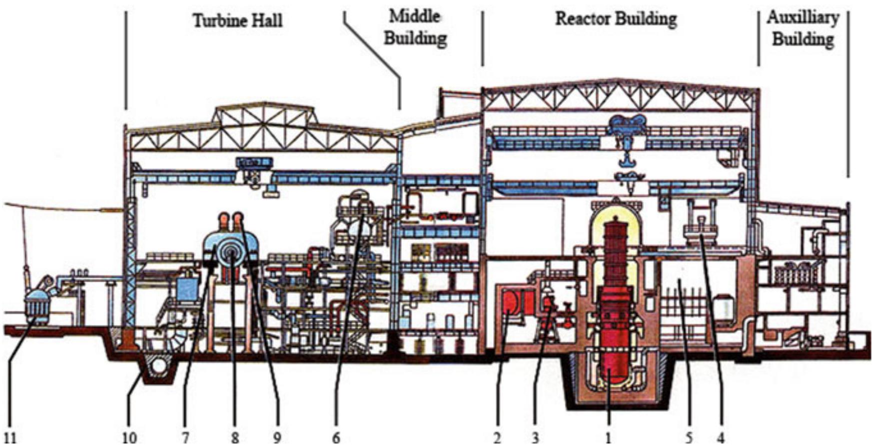


Fig. 2.1 Major buildings and components of the WWER440/V230 plant. *1* reactor, *2* steam generator, *3* main circulation pump, *4* refuelling machine, *5* spent fuel pool, *6* feedwater tank, *7* turbine, *8* generator, *9* steam piping, *10* cooling water piping and *11* transformer

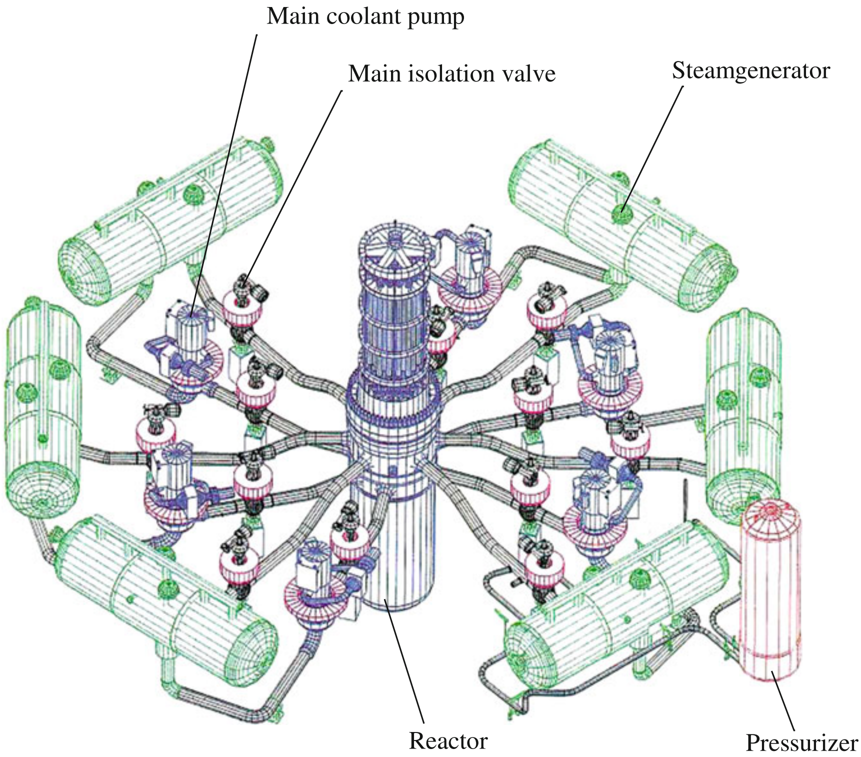


Fig. 2.2 Three-dimensional view of primary circuit of the WWER440 reactor

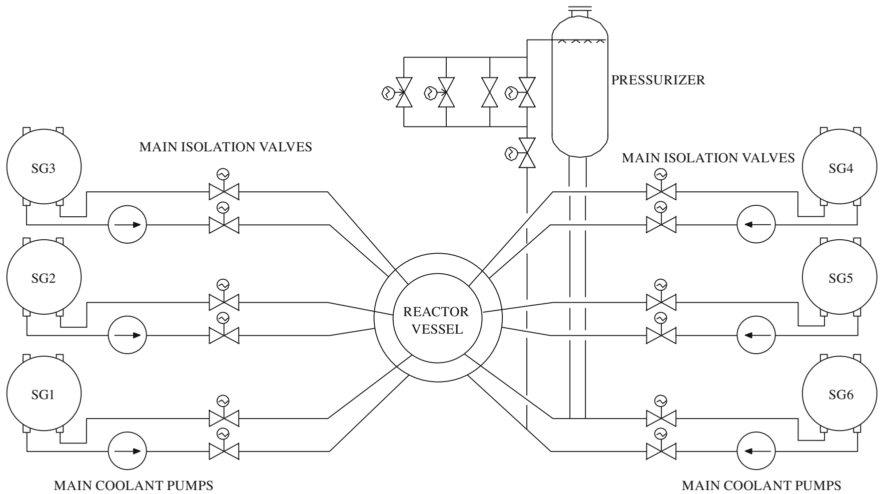


Fig. 2.3 The schematic of primary circuit of the WWER440 reactor

systems and related auxiliary systems, is located in the turbine building. Each nuclear power unit has two 220 MW(e) turbogenerators operating with saturated steam. The turbines are aligned on the same axis parallel to the reactor building.

Basic thermo-technical features of the reactor and primary circuit are as follows:

Thermal power output	1,375 MWt
Mean value of coolant media	280 °C
Inlet reactor coolant temperature	265 °C
Outlet reactor coolant temperature	292 °C
Peak cladding temperature	373 °C
Nominal operational pressure	12.26 MPa
Total volume of the primary circuit	223 m^3
Coolant media volume in the primary circuit at nominal level in pressurizer (4,700 mm)	207 m^3
Average fuel burn-up	23.6/26.7 GWd/TMU
Maximum fuel burn-up[a]	37/42 GWd/TMU
Average Fuel enrichment[a]	3.46/3.80 % U 235
Primary coolant additives	
H_3BO_3	0–12 g/l
NH_4^+	6–10 mg/l
KOH	2–16 mg K^+/l
Hydrazine	Only during start of cycle for oxygen removal (dosed around 50 l of 15 % substance)

[a] These values are changing during the plant operation. The first one is typical for the first half of operation, the second one is for the last two cycles and it is typical for 4-year fuel cycle

2.1.1.1 Primary Coolant System

The main components of the primary circuit (primary coolant system or reactor coolant system) are described below.

Reactor Vessel

The reactor pressure vessel is a vertical cylindrical vessel, 23.4 m in height and with a maximum diameter of 4.35 m. It houses internals which retain and support the core and provide for coolant distribution through the core. The vessel has a main joint and 6 inlet and 6 outlet nozzles above the core. Cylindrical forgings of a low alloy, high-strength steel are welded into a cylinder with hemispherical bottom from the vessel. The vessel head is bolted on to complete the pressure boundary and to locate the penetrations for the control rod assembly drives and for the

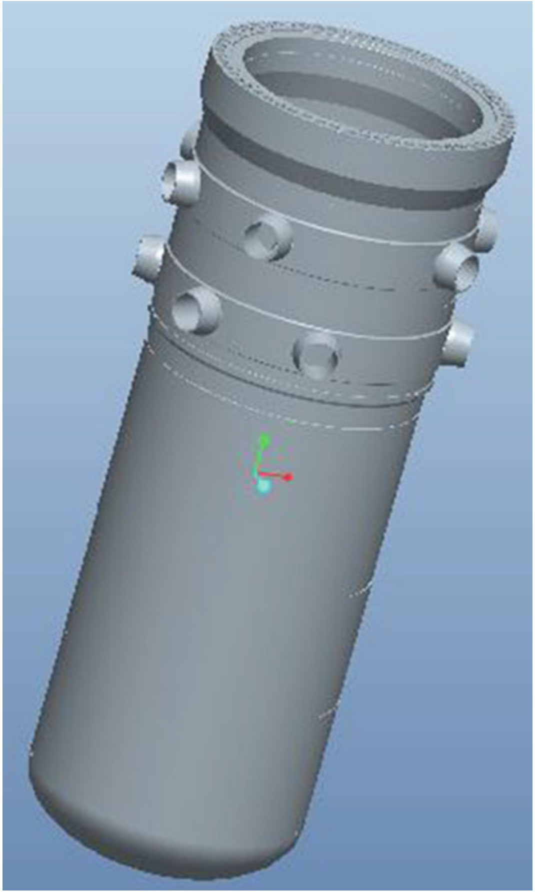


Fig. 2.4 General view of the reactor vessel

thermocouples. Materials used, in particular the metal of the weld located at the height of the core, are susceptible to embrittlement by fast neutrons. Neutron fluency is greater than in other PWR pressure vessels owing to the small annual space between vessel and core. The vessel geometry is a consequence of design requirements to allow transport by rail. The vessel walls are lined with a stainless steel cladding of about 10 mm thickness.

The reactor vessel is installed in a concrete pit and its support is shaped as an annular tank filled with water which also serves as a biological shield.

General view of the WWER440 reactor vessel is provided in Fig. 2.4.

The reactor core is composed of hexagonal fuel assemblies with 126 fuel rod positions each. Control rod assemblies are combination of fuel assembly and an absorbing extension. The WWER440 reactor uses a rack-and-pinion-drive mechanism to move the control rods. The reactor core consists of 276 fuel assemblies, 37 control rods and 36 shielding assemblies, so-called dummy elements which in total gives 349 of all assemblies in the reactor core.

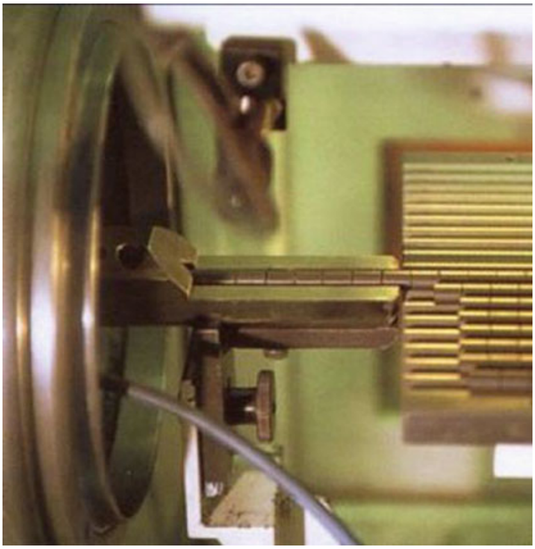


Fig. 2.5 The WWER440 fuel pellets

The fuel rod is made about 7.5 mm-diameter UO_2 pellets with a density of 1.04×10^4 kg/m^3, stacked in a Zr–Nb (1 %) tube of 9.1 mm outside diameter. The fuel pellets are designed with a central hole about 1.5 mm in diameter to reduce the fuel temperature and to lower the probability of melting in transient conditions (see Fig. 2.5). The fuel rods are filled with helium and sealed welded at both ends. Space at the top of the pellet stack allows for fuel expansion during operation, and a spring is placed there to hold the pellet stack. The initial loading composition for fixed fuel assemblies consists of about one-third each of assemblies of 1.6, 2.4 % ^{235}U-enriched fuel (see Fig. 2.6). The fuel follower of the control rod assembly has 2.4 % enriched fuel. Owing to the smaller moderator to fuel ratio, the neutron spectrum of the WWER440 reactors is harder than in other PWRs.

The control rod fuel follower design is similar that of the fixed fuel assembly, except for the grab head which is designed to connect to the absorber extension and the end closure which is provided with a liquid damper. The absorber consists of a hexagonal stainless steel shroud shaped like the fuel assembly. The wall of the shroud and the borated steel inserts mainly absorb thermal neutrons. The absorption is intensified by the water inside the control rod (neutron trap type). To move the control rod assembly, a rack-and-pinion drive is used. When the control rod assembly is moved up by the drive system, the absorber part comes out of the core and is replaced with the fuel follower. The 37 control rod assemblies are subdivided into 6 groups (banks) with 6 control rod assemblies each symmetrically distributed over the core. In normal operation, all banks are in the upper position except working group 6 with 7 control rod assemblies, which is dipped into the core so as to be more effective (see Fig. 2.7).

The control rod assemblies serve for reactor scram and planned shutdowns by pushing the fuel part away from the core and inserting the absorber part. This

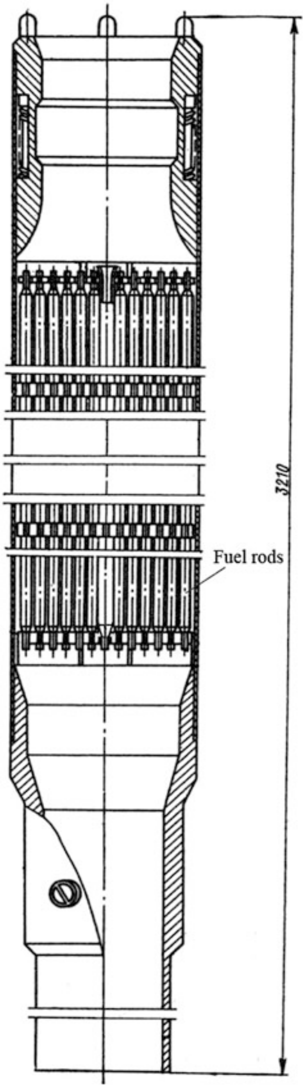


Fig. 2.6 The fuel assembly with fuel rods

mode of the control rod assemblies drops under the influence of gravity only. The power control rod assemblies are also used for control purposes to maintain the reactor at the specified level, to transfer the reactor from one power level to another and for compensation of rapid changes in reactivity due to temperature, power effects, poisoning, etc. A second means for reactivity control is boron dissolved in the reactor coolant. This is needed to compensate for fuel burn-up and to ensure deep subcriticality in cold shutdown.

In normal operating conditions and in anticipated operational occurrences, no cladding failures of fuel melting should occur. To meet these requirements, the

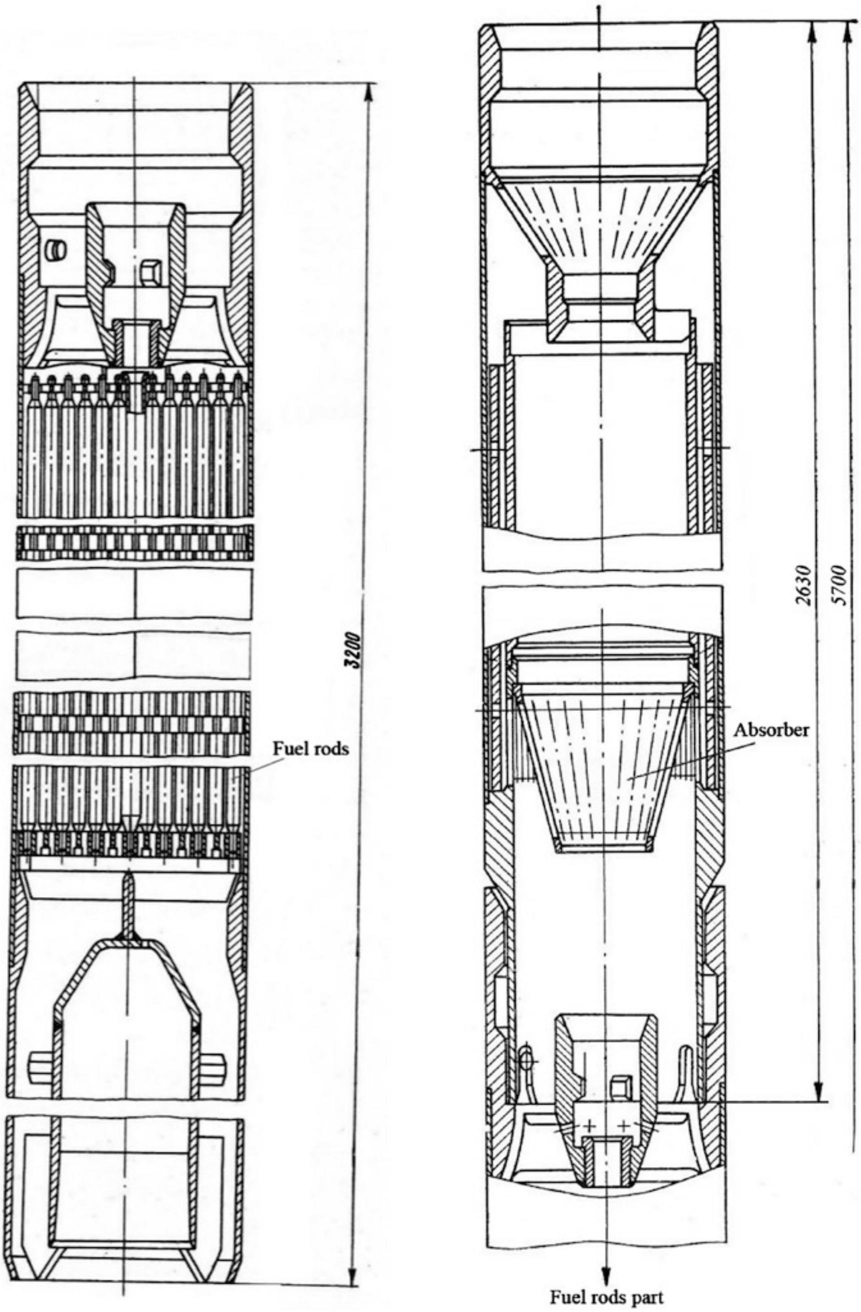


Fig. 2.7 The control rod assembly with fuel rods and absorber

main design limits of core operation are as follows: average linear power of the fuel rod in the core is 12,900 W/m, and the peak linear power is 32,500 W/m which is about 40 % lower than in present PWRs. Furthermore, the peak heat flux from the fuel rod surface is 1.20 MW/m^2, and the average core power density is 83 MW/m^3. The design average burn-up of uranium in the discharged fuel assemblies in steady-state operation conditions with three partial refuellings per full cycle is about 30 MW d/kg U.

There is a system for measurement of power distribution consisting of 210 assembly coolant outlet thermocouples located in the assembly coolant channels about 0.2 m above the top of the fuel pins. The original core design was equipped with an in-core flux monitoring system based on wire activation which has since been deactivated at all sites.

Steam generator

Six steam generators are located around the reactor (Fig. 2.8). Steam generator is designed for generation of dry saturated steam in required quantity and quality by way of primary circuit coolant heat removal. Steam generators constituted the boundary of the primary circuit coolant.

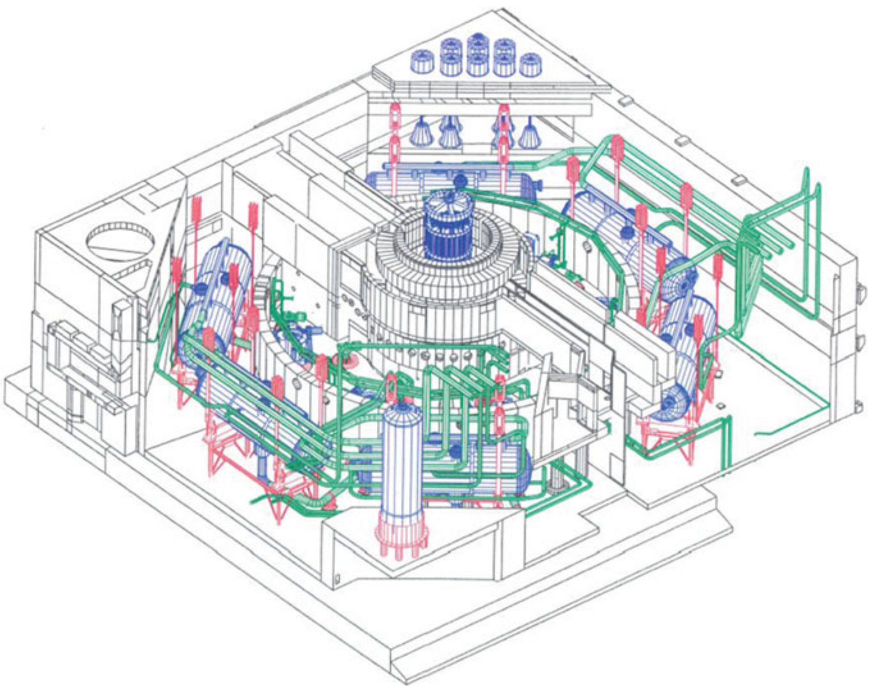


Fig. 2.8 Steam generator layout at the plant

Steam generator is a single-shell recuperative horizontal-type heat-exchanging unit with submerged heat-exchanging surface and consists of the following main components:

- shell with nozzles of various purposes;
- heat-exchanging surface with support assemblies;
- primary circuit coolant headers;
- main feedwater supply and distribution devices;
- emergency feedwater supply and distribution devices;
- air vent lines and lines for flanged connections and enclosures leak-tightness monitoring;
- louvre separator;
- perforated steam receiving tube sheet.

The steam generator design is presented in Fig. 2.9.

SG shell consists of shell sections, stamped elliptic bottoms and forged nozzles connected by welding. The shell design provides access for inspection of the internals from the secondary circuit side. A Ø 500 manhole with split-flanged

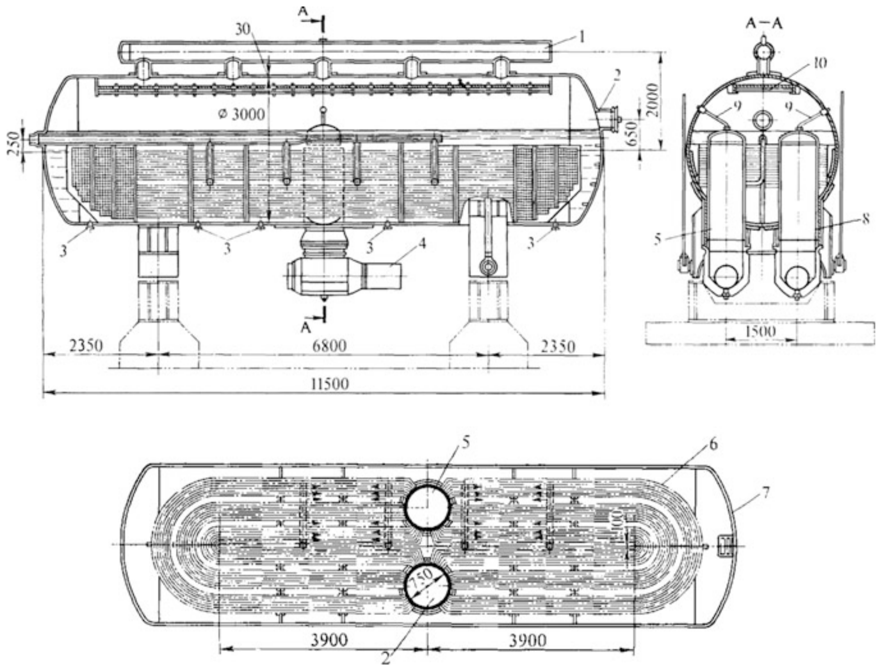


Fig. 2.9 Steam generator design. *1* steam header, *2* manhole, *3* blow-down and drainage fittings, *4* MCP nozzle, *5* inlet header, *6* heating surface, *7* steam generator shell, *8* outlet header, *9* primary air vent and *10* louvre separator

connection is provided in the elliptical bottom for that purpose. Two Ø 700 manholes with split-flanged connections are provided for inspection of primary circuit headers.

The shell has:

- two manholes for primary circuit header maintenance-Ø 700 mm;
- manhole for access to secondary-side cavity-Ø 500 mm;
- five steam extraction nozzles-Ø 250 mm;
- one feedwater supply nozzle-Ø 250 mm;
- fitting Ø 80 mm and two fittings Ø 50 mm for blow-down;
- nozzle Ø 80 mm for water drainage from SG and additional emergency feedwater supply;
- two nozzles for welding the primary circuit headers-Ø 1,100 mm;
- six fittings on the shell and bottom for connection of three single-chamber levelling vessels-Ø 25 mm;
- six fittings for connection of double-chamber levelling vessels-Ø 25 mm;
- two fittings for primary circuit air vents-Ø 10 mm;
- four fittings for monitoring flanged connections leak-tightness-Ø 10 mm;
- four fittings for blow-down of primary circuit header pockets Ø 25 mm.

Heat-exchanging surface consists of 5,536 tubes of 16×1.4 mm diameter made of 08H18N10T steel, bent in U-shape coils. The tubes are horizontal, with corridor-type arrangement in bundles with height spacing 30 mm, and horizontal spacing 24 mm.

The coils are connected to the primary circuit headers. Their ends are welded to the headers on their internal surface. The guaranteed depth of fusion is not less than 1.4 mm. The heat-exchanging tubes are expanded by the header wall thickness.

The tubes are spaced using bent and flat plates (spacing straps) providing for uniform tubes arrangement in the heat-exchanging bundle.

The design of the spacing unit and its material rules out any tube damage, including corrosion damage. At the same time, the possibility is provided for tubes' thermal expansion movements along their axes.

Primary circuit headers are designed for coolant distribution to heat-exchanging tubes, coolant collection and removal. The header is welded in its lower part to the shell nozzle through a reducing ring, while the upper part is free inside the shell nozzle with a 4 mm circular clearance.

The header lids are provided with rings (displacers) for coolant leaks restriction from primary to secondary circuit to nominal diameter DN 32 in case of the lid break-off due to possible corrosion damage of M48 studs used to seal the header.

The central part of the primary circuit header has holes with heat-exchanging tubes secured in them.

Installed in the upper part of steam generator shell is the louvre separator made of louvre packages and a perforated sheet designed for steam velocity equalization in the steam generator separation devices.

Main feedwater distribution device consists of a header and distribution pipes, having feedwater outlet holes along it. The feedwater is supplied to the "hot" side of the tube bundle.

Steam header is located above the steam generator. It consists of a DN 400 mm pipe, bottom, five header nozzles.

Main Coolant Pump

The main coolant pumps, providing coolant circulation in the loops, are low-inertia canned motor pumps with a capacity of 7,000 m^3/h. The two major parts of the MCPs, the hydraulic part and the electric motor, are arranged in one unit which is very tight. The coastdown time in the event of loss of electric power is about 3 s. To compensate for this, two of these MCPs are powered by two auxiliary generators directly coupled with the turbine sets which have a longer coast down time. In the event of a turbine trip, these generators provide electrical supply for about 100 s. The MCPs are cooled by an intermediate cooling water system which is in turn cooled by the service water system. The MCP housing is made from austenitic steel.

The reactor coolant pump used at most WWER440/V230 units is the reactor coolant electric pump GTsEN-310, manufactured by JSC Kirov Factory in St. Petersburg, Russia. The pump type is vertical, centrifugal, single-stage canned (leak-tight) pump with integrated 3-phase direct current induction motor.

The reactor coolant pump consists of the following main units and components (see Fig. 2.10):

- main electric pump (1);
- cooler (2), designed to cool the coolant circulating in the independent circuit (6);
- fan (3), forcing air circulation through the stator-end winding cavities and air cooler system (5);
- frames (4) for installation of the fan (3) and air cooler;
- GTsEN-310 cooling system, consisting of the independent circuit pipeline (6), with a check valve (10), component cooling circuit water cooling system, stator air cooling system (5);
- casing (7);
- air cooler (8);
- auxiliary pump (11), creating coolant circulation in the independent circuit given the MCP shutdown;
- support frame (12) with ball bearings (9), foundation plates and striking wedges (13);
- biological shield (15); and
- cooler (2).

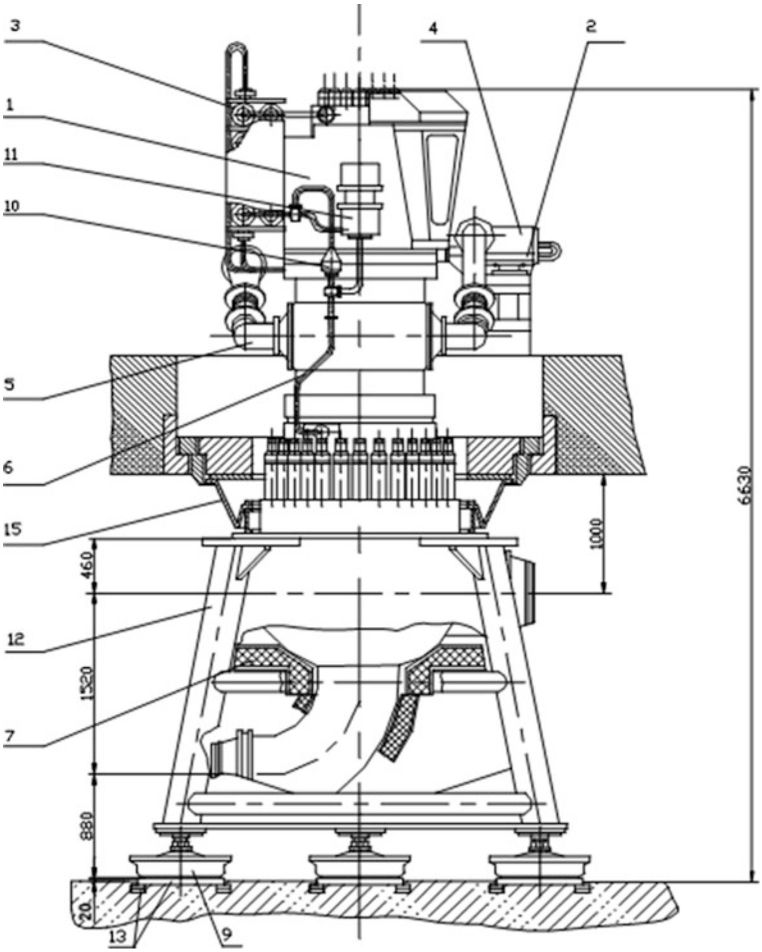


Fig. 2.10 Main coolant pump

Intermediate Cooling System of MCPs

The intermediate cooling system of MCPs is designed to cool the following primary circuit equipment: MCPs including autonomous circuit heat exchangers and air coolers, primary circuit water clean-up cooler, safety injection pump bearings, pressurizer relieve tank, controlled releases from primary circuit coolers and sampling coolers.

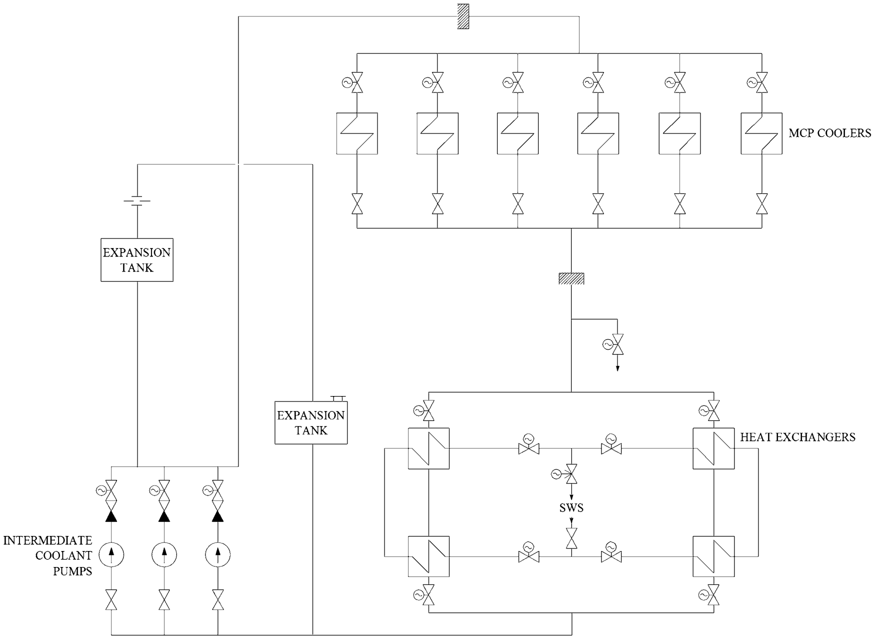


Fig. 2.11 Intermediate cooling system of MCPs

The cooling system cools by heat transfer from the component being cooled into the cooling system water.

One pump is operable, and two are in the standby mode. The water is circulated around each of the coolers in parallel.

The water is then transferred into the heat exchangers where it is cooled by transfer of heat into the service water. The water is then passed back into the suction line of the pumps. Cooling of CP coolers is presented on Fig. 2.11.

Pressurizer

The pressurizer and connecting pipelines are components of the primary circuit pressurizing system and are designed to create and maintain pressure in the primary circuit.

The pressurizer is connected to the non-isolable part of the hot leg of one circulation loop by two lines of nominal diameter 200 mm (see Figs. 2.12 and 2.13). In the event of an excessive rise of pressure in the pressurizer, the coolant is discharged into the bubbler tank (pressurizer relieve tank) through two pilot operated safety valves. The pressure can be reduced by spraying into the pressurizer and can be increased by using electric heaters. The vapour volume of the pressurizer is connected by an additional line, also a nominal diameter 200 mm, to

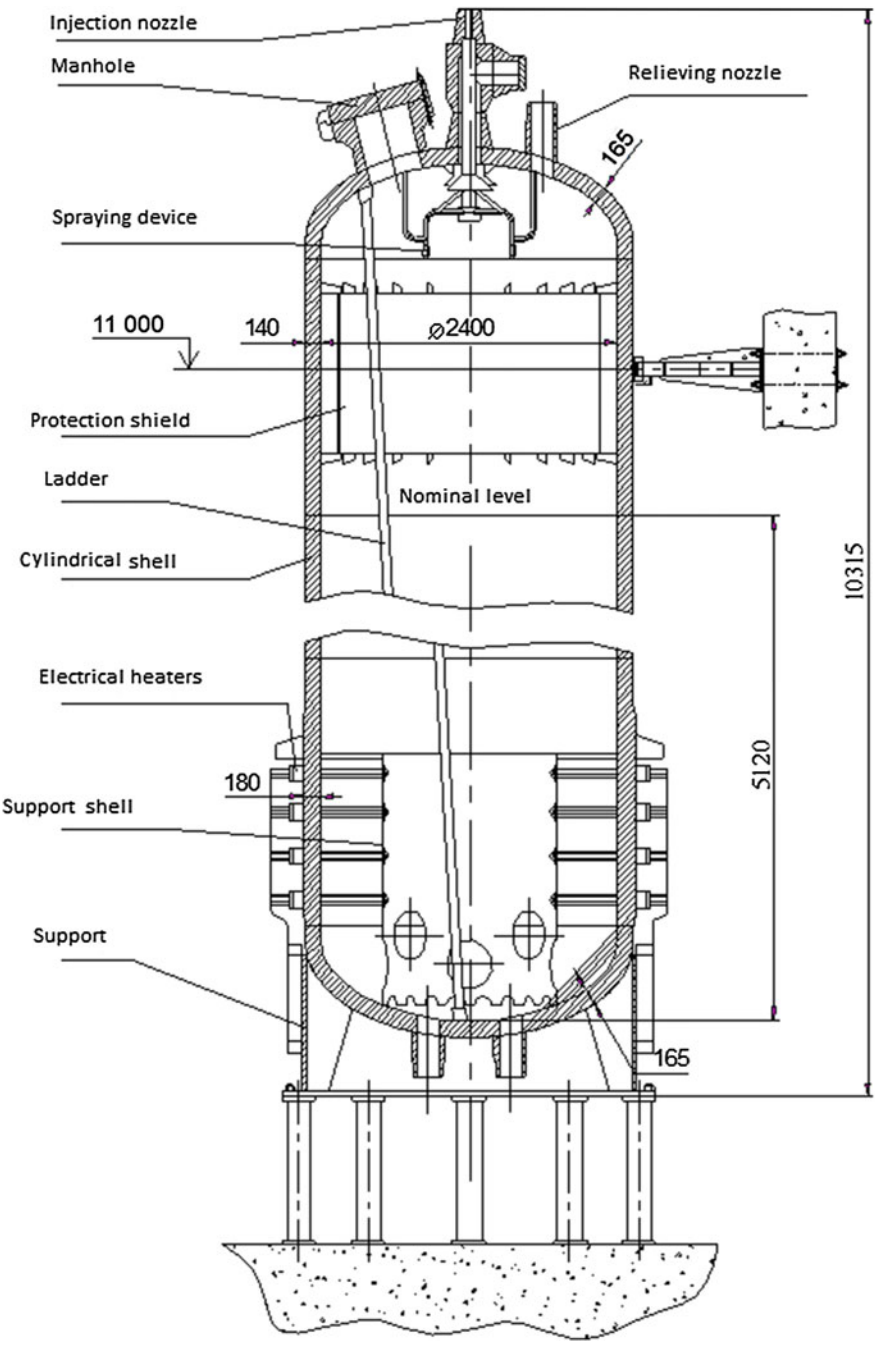


Fig. 2.12 Pressurizer

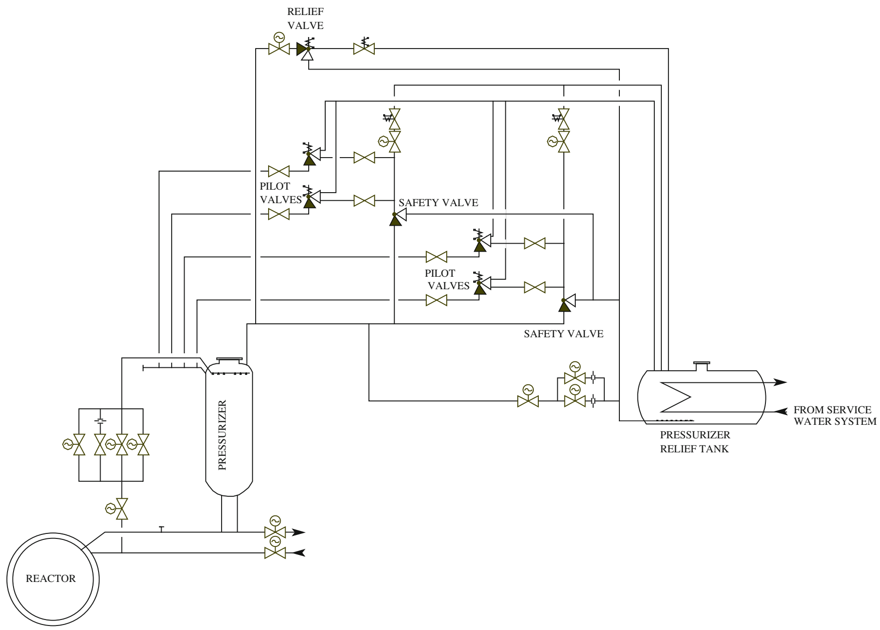


Fig. 2.13 Pressurizer and its safety and relief valves

the cold leg in order to achieve pressure equalization in the event of strong pressure transients.

The pressurizer is a vertical vessel, installed on a cylindrical support, its vessel consists of a body frame and internals (injection manifold, thermal protection shield, ladder, support shell and heating elements).

The pressurizer body frame consists of a cylindrical shell, an area of holes for heating elements and two elliptical end plates. The support shell fixes 108 heating elements in their operating (horizontal) position and provides a natural circulation circuit during operation of heating elements. In the middle part of the support shell, there are holes with holders for heating element unit installation, whereas in the lower part, there are holes for medium passage during natural circulation and for internal inspection of the body frame.

On the cylindrical part of the pressurizer, there are nozzles for levelling vessel connecting pipelines as well as sampling and pressure nozzles and cases for temperature monitoring indicators.

In the elliptical top end plate, there is a manhole for inspection of pressurizer internal surface, an injection nozzle and a nozzle for steam dump through pilot operated relief valves of the pressurizer. The manhole connector is sealed by two nickel rod-shaped gasket rings with a leakage control nozzle between them.

In the upper steam part of the pressurizer, there is an injection manifold designed for equal water spraying in the steam space. Cold water is sprayed through four nose-pieces, located at an angle of 90° to each other.

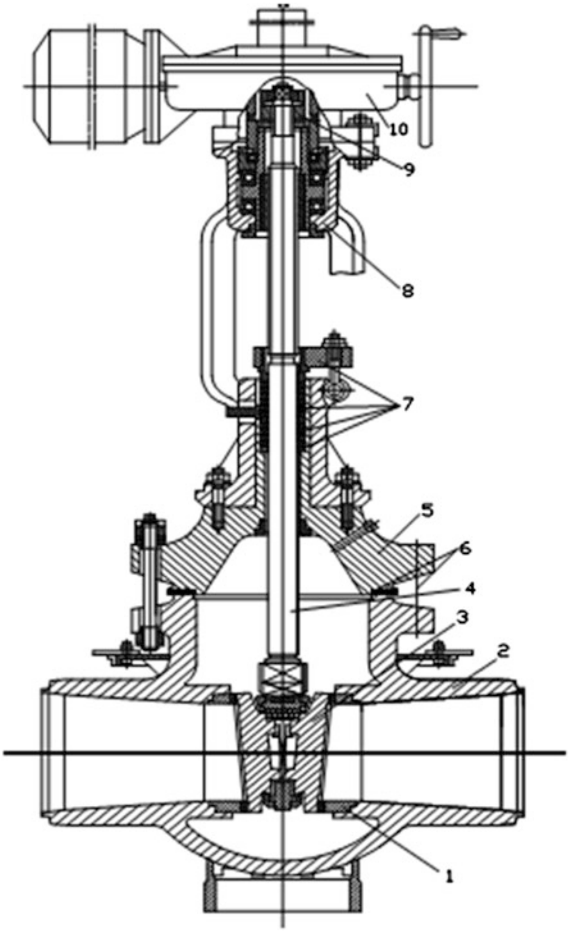


Fig. 2.14 The main isolation valve. *1* welded seats, *2* body, *3* gate unit, *4* spindle, *5* cover assembly, consisting of the cover itself and a yoke, *6* the body-to-cover sealing unit, *7* the cover-to-spindle sealing unit, *8* guiding unit, *9* spindle stopping unit and *10* electric drive

The thermal protection shield protects the pressurizer body frame from impact of cold water coming from the injection manifold and represents a cylindrical shell.

Main Isolation Valve

The main loop isolation valves are designed to isolate a circulation pool for its removal from operation for maintenance during the plant outage and loop removal to "hot" standby under normal operating conditions. General view of MIV is provided in Fig. 2.14.

The valve body is a ball-shaped casting with two welded nozzles and a flange. The valve seats are welded to the nozzles inside the body. Ribs are available inside the valve body that provide for required gate position relative to the seats during its opening and closing actions.

The flange of the body and the side part has the holes of Ø 10. The body flange has 26 stud holes. The valve body is cast-welded, and the cover and yoke are made of castings. The power parts of the gate are made of forgings. The valve gate is of wedge type with mechanical disc hold-down against the seats with a possibility of sealing water supply to the middle cavity.

The MIVs are intended to isolate each of the six reactor circulation loops. Pressurized water of the make-up system is used for sealing the MIVs to prevent the ingress of the coolant into the disconnected loop. The valves are controlled by an electric motor with a closure time of about 80 s. The housing is fabricated from cast austenitic steel.

Main Coolant Piping

The main coolant pipe is made from tough austenitic steel with an outer diameter of 560 mm and a wall thickness of 32 mm. All pipes are connected to the equipment and valves by means or argon welding. Layout and equipment support provide temperature expansion self-compensation within the specified strength limits for austenitic steel.

Primary Make-up System

The primary circuit make-up system is designed to perform the following functions: compensation for minor leakages from the primary circuit, measurement and maintenance of boric acid concentrations in the primary circuit, maintaining the water chemical conditions in the primary circuit, pressurizer cooldown, pressure test of the primary circuit. For the purpose of safety analysis within PSAs, only compensation of leakages is considered in.

The make-up system is constructed from the following items (Fig. 2.15):

- injection tank—deaerator (atmospheric type)
- injection water heaters
- three injection pumps
- measuring instruments
- water-level control in the injection tank

The make-up tank contains the water inventory for the primary circuit make-up system; the tank has an operational volume of 18 m^3. The tank is used to deaerate the make-up water, and the regeneration heater in the primary clean-up system normally used to heat up the make-up water prior to injection into the primary circuit.

For compensation against minor leakages from the primary circuit, when the pressurizer level drops, a make-up pump is started, and at a high-level drop in the pressurizer, all pumps in standby mode are started. The boron solution is drawn

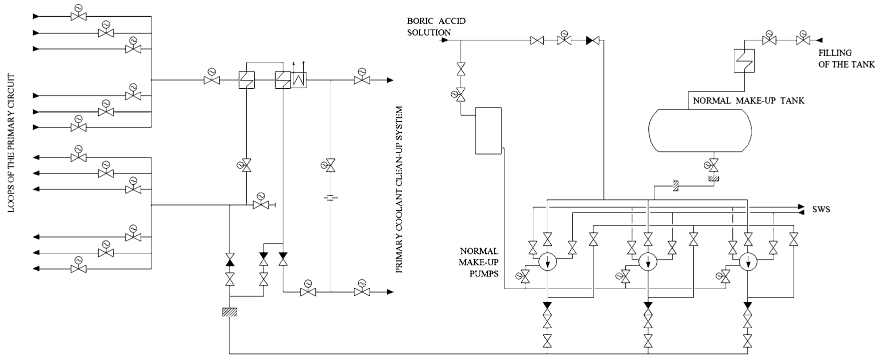


Fig. 2.15 Primary make-up system

from the make-up tank and pumped through the primary circuit water clean-up heaters and into the primary circuit loops. The pumps are stopped after the pressurizer reaches normal level.

Primary Coolant Clean-Up System

In the case of a primary circuit LOCA, the first operator action required which changes the status of the plant is the disconnection of the water clean-up system from the primary circuit. In this way, any leaks within the clean-up system will be isolated. If the water level in the pressurizer continues to fall, the operator then starts primary circuit coolant loop isolation.

In the safety analysis, only functional fault trees of the clean-up system are modelled as failing to be isolated from the primary circuit. Therefore, the only system components considered in this analysis are the valves used for isolation.

Water from the primary circuit flows out through the valves into a common header (see Fig. 2.15). The primary circuit water then passes through a series of heat exchangers and then to the primary coolant clean-up system. After cleaning, the water is injected back to the primary circuit using the primary make-up system.

2.1.1.2 Confinement

The confinement is a closed compartment system which confines the main equipment of the primary circuit, such as the RPV, the SGs the MCPs the MIVs and the pressurizer. In addition, the high-pressure safety injection system, the spray systems and the borated water storage tank are located here. The total net volume of the confinement, about 14,000 m^3, is relatively small and designed to withstand an excess pressure of 0.1 MPa. A lower pressure is kept in the inaccessible compartment than in the accessible compartments and the atmosphere by means of a ventilation system. An air recirculation system with a cooler is installed to remove the heat from the confinement system. Within the compartment containing the SGs and the MCPs, there are three lines of nozzles of the spray system to reduce pressure and to bond iodine.

The pressure-resistant confinement is vented to the atmosphere via nine flaps (eight dump valves of 1 m diameter each and one dump valve of about 0.5 m diameter). They do not open in the event of a design basis LOCA (nominal diameter of 32 mm) if the spray system remains operational. If it fails, then one flap should open at 0.08 MPa. The opening of all flaps at 0.10 MPa protects the pressure-resistant compartment system in the event of a break with a nominal diameter of 200 mm, which could happen if the largest connecting line to the main primary circuit breaks. The flaps should reclose when the pressure drops. The WWER440/V230 plants are not designed to contain the consequences of postulated major accidents, such as the full break of a main loop line.

2.1.1.3 Safety Systems

High-pressure Safety Injection System

The high-pressure safety injection system is designed to compensate losses from the primary circuit which exceed the capability of the normal primary make-up system. It also acts to decrease reactivity and to maintain the reactor subcriticality during accident. During normal plant operation, the system is in a standby state. The system schematic is in Fig. 2.16.

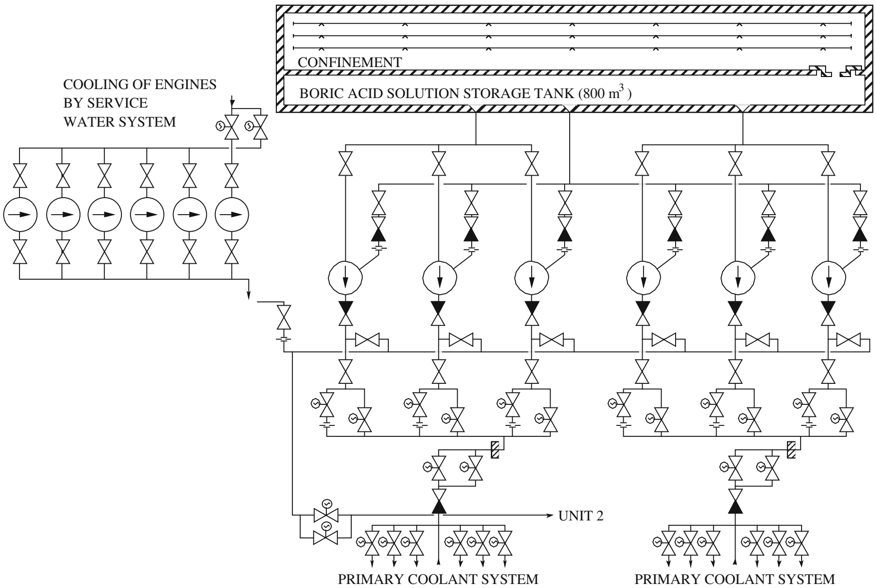


Fig. 2.16 High-pressure safety injection system

The six safety injection pumps are divided into two groups. One group of pumps supplies boric acid solution into the header of the primary circuit feed to clean-up. Second group of pumps supplies boric acid solution into the header of primary circuit water return from the clean-up.

Each pump via manual valve is connected to the common recirculation header. This is the test line for the pumps. This collector is interconnected with the second unit. In case of failure of unit 1 pump, the pumps of unit 2 can be used (if enough time is available to prepare the lines).

The delivery of each pump is to the main and auxiliary feed line. The main feed lines have valves to ensure the operation of the pumps at a primary circuit pressure of 7.85–12.26 MPa. The auxiliary feed lines have valves and associated throttle diaphragms to ensure the operation of the pumps at a primary circuit pressure of 0–12.26 MPa.

The high-pressure safety injection system can be started automatically when the pressurizer level drops significantly from large primary leak (pressure drops in the primary circuit to below 11.77 MPa and pressurizer-level drops to 700 mm) or pressure drops in the primary circuit to below 9 MPa. Given the starting signal, two pumps are started in each group with the switch set to "Operation 1" and "Operation 2" mode. If any of the pumps fail to start or fail to run given a start, the pump in the same group with the switch set to "Reserve" is started automatically.

From the start of any pump, the MOVs are opened to supply service water to cool the heat exchangers of the pump motors. At the same time, the MOVs in the main and auxiliary feed lines and MOVs in the group header are opened and MOVs in the drainage lines are closed.

Confinement Spray System

The confinement spray system is designed to decrease pressure and temperature in the confinement during LOCA by pumping boric acid solution through spray nozzles into the confinement atmosphere. During normal plant operation, the system is in the standby state.

A supply of $KOH + N_2H_4 + H_2O$ solution to the spray pump suction is used to bind radioactive aerosols of the isotope I-131. The scheme of the spray system is shown in Fig. 2.17.

The confinement spray system comprises the following items: spray pumps, heat exchangers and boric acid storage tank.

The system pumps are started if the confinement pressure is >15 kPa or in load-sequencing system after loss of offsite power given the confinement overpressure. From the start of the pump, the valves in the lines to nozzles are open.

The valves on the service water supply lines to heat exchangers are opened automatically when the temperature in the pump suction pipe work reaches 50 °C given the start of the pump.

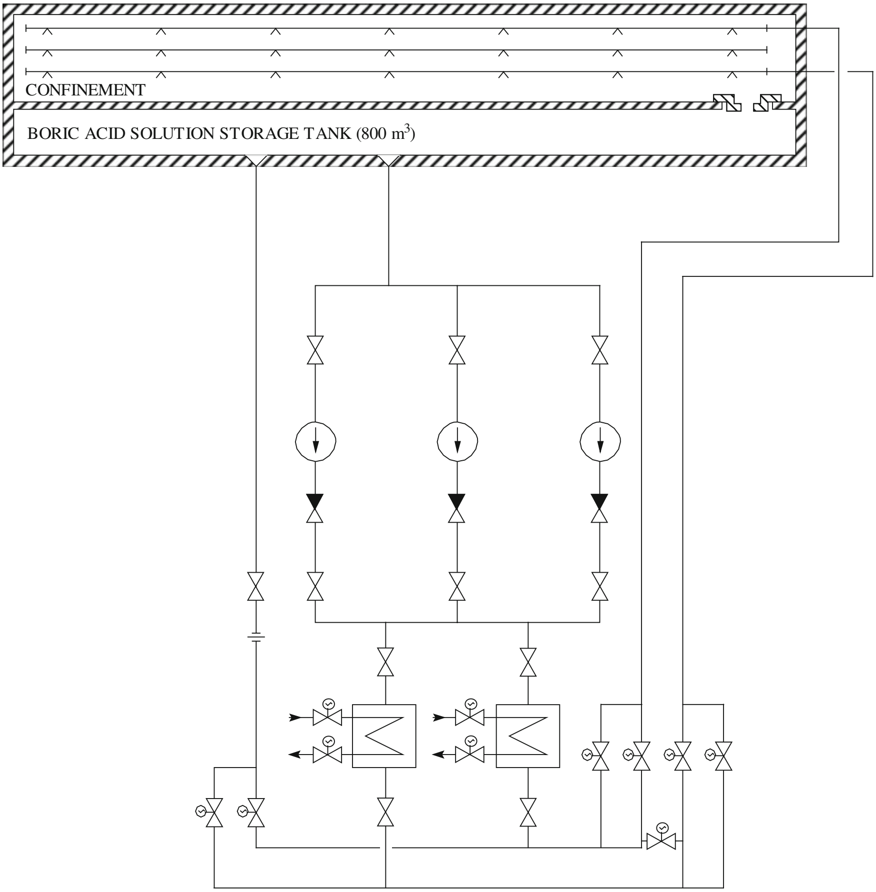


Fig. 2.17 Confinement spray system

The valves in the recirculation line will open with a time delay of 30 s, provided that the valves are not open or the confinement pressure is less than 10 kPa.

Reactor Protection System

The reactor protection system is designed to cause automatic interruption or slowdown of fission reaction on the detection of a number of accident situations. The system provides four levels of safety interventions: HO1, HO2, HO3 and HO4. HO1 results in a rapid simultaneous insertion of all control road groups, into their lower-end positions with velocity of 20–30 cm/s. This is achieved by switching off supplies to all the control rod drive mechanisms (CRDMs).

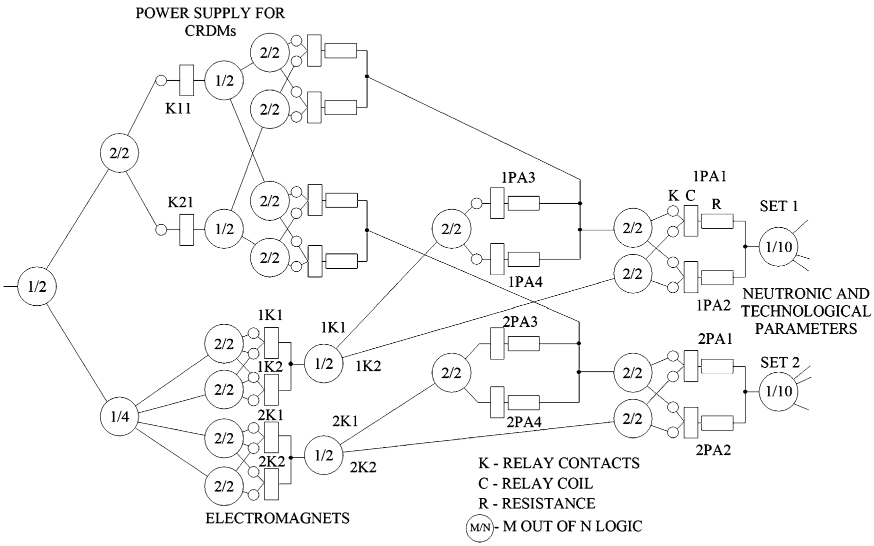


Fig. 2.18 Logic of the reactor protection system

HO2 results in a sequential rapid insertion of control rod groups with a velocity of 20–30 cm/s. This is achieved by switching off supplies to the CRDMs sequentially (in the reverse order to that in which the rod groups were withdrawn). The drop of each group starts when the previous group reaches the second zone of the ten zones associated with the rods position within the core (numbered top to bottom).

HO3 results in the insertion of the controlling rod group with the velocity of 2 cm/s. This is followed by sequential driving of control rod groups downward.

HO4 prevents withdrawal of the control rod groups.

The initiation of each protection level actuates the protection level below it. That is, HO1 actuates HO2, HO2 actuates HO3, and HO3 actuates HO4.

The effects of HO2, HO3 and HO4 last only for the duration of the appropriate trip signals. However, for HO2, if three groups have reached their end position, the action will continue. The effect of HO1 is not terminated with the termination of its trip signal.

At 100 % power, the rod groups are all raised. Group 6 is used as the controlling group. The CRDM clutches are held onto the threads of the control rods by the continued energization of electromagnets (each control rod has its own electromagnet). Logics of the reactor protection system (HO1) are presented in Figs. 2.18, 2.19 and 2.20. Logic to start the HPSI pumps is presented in Fig. 2.21.

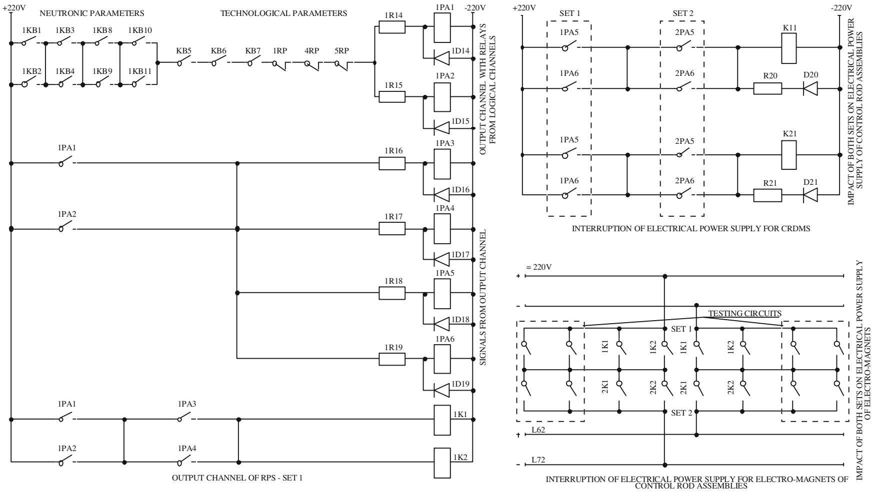


Fig. 2.19 HO1 signal generation to trip the reactor

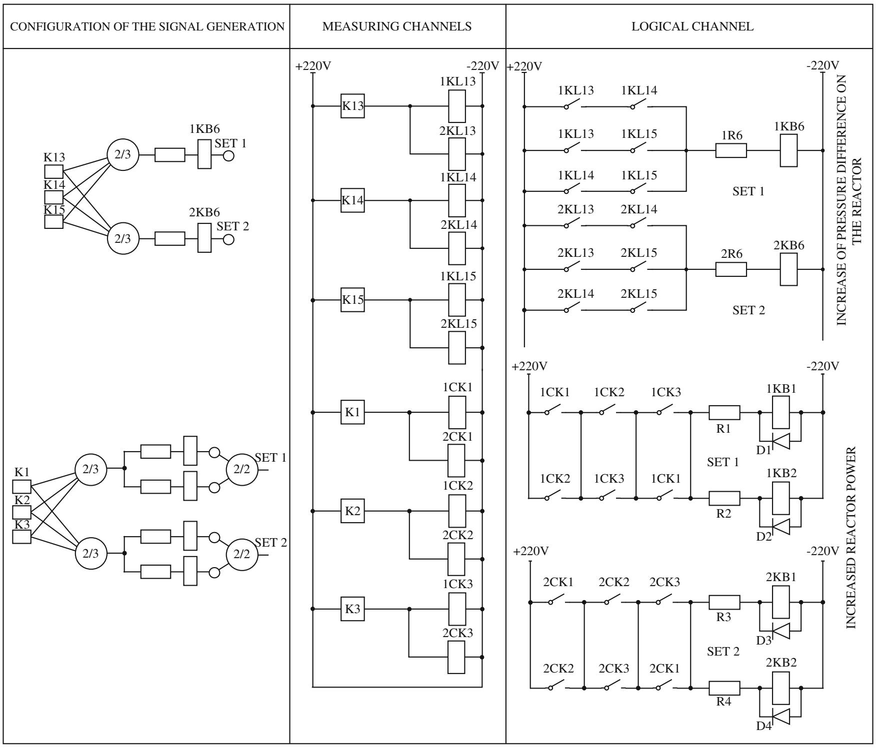


Fig. 2.20 HO1 signal generation in the logical channels

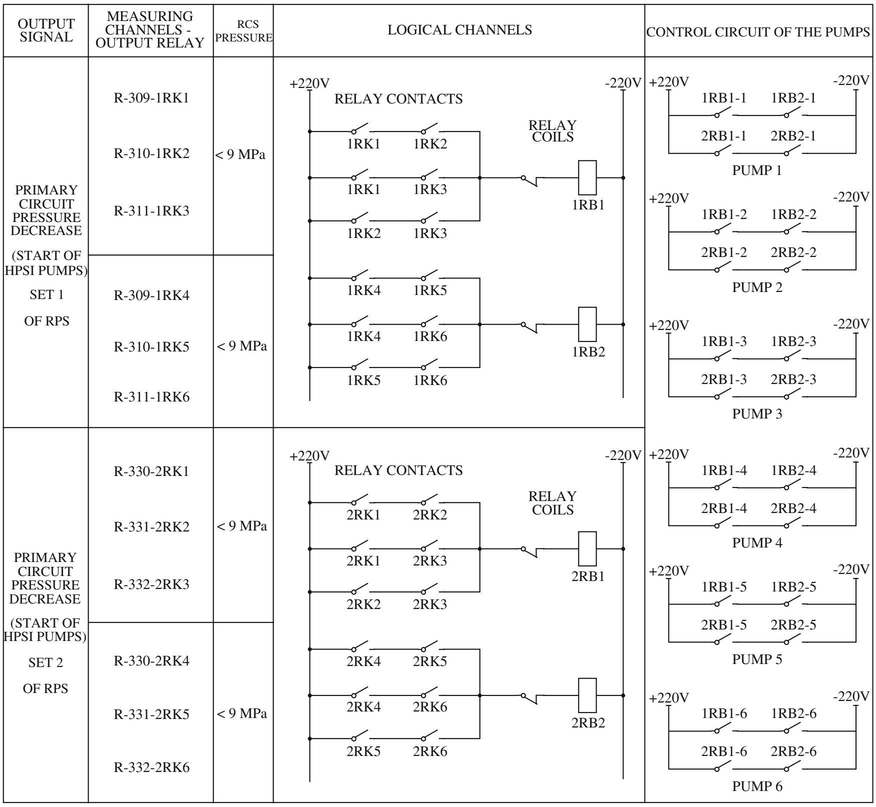


Fig. 2.21 Signal generation to start the HPSI pumps

2.1.1.4 Secondary Circuit

Main Steam System

Steam is produced in six SGs by heat exchange between the primary and secondary circuits. Six steam lines lead the steam from SGs to the steam header. The steam is then supplied to two turbines via four steam lines (two turbines per unit, each turbine has high-pressure and low-pressure parts). The steam header can be separated into two halves by two MOVs and one quick closing valve (located between the MOVs). The valves are normally open (see Fig. 2.22).

The steam lines of SG 1, 2 and 3 are connected to the first half of the header, and the steam lines of SG 4, 5 and 6 are connected to the second half of the header.

On each steam line, there is a quick closing valve and an MOV which have to isolate the SG following a steam line break or header break.

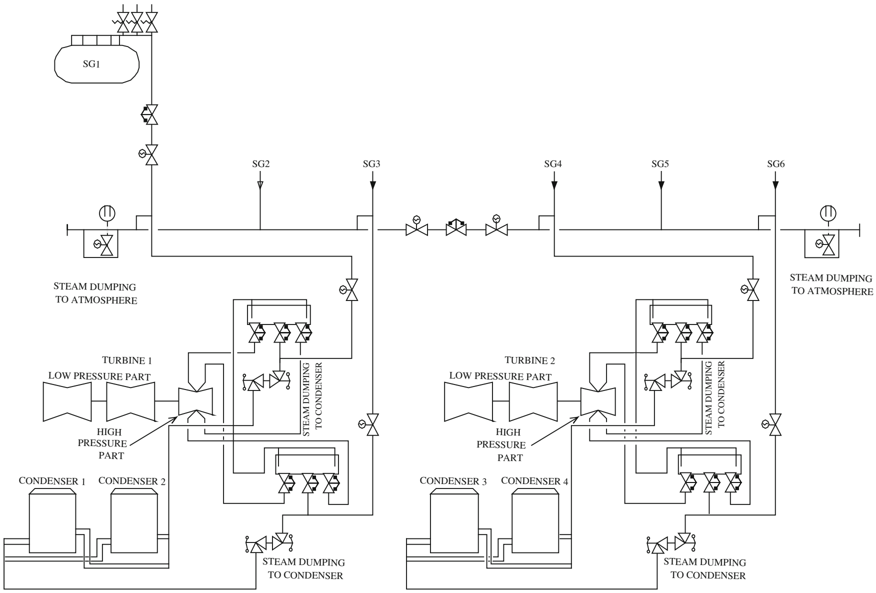


Fig. 2.22 The main steam system

Each SG has three safety relief valves to prevent secondary-side overpressure. These relief valves are operated directly by the pressure of the steam against an air supply and spring.

Two steam dump stations to the atmosphere are located on the steam header. They also have a safety function to protect against secondary-side overpressure. There are motor-operated relief valves which are automatically opened when steam header pressure exceeds 4.9 MPa.

The bypass of the turbines is allowed using steam dump stations to the condensers. Quick-acting hydraulically operated valves are used. On closure of the turbine quick-acting valves (due to a low-main steam header pressure or a reactor trip), these bypass valves will automatically open. The steam is then passed directly to the main condensers.

The steam is condensed in the main condensers. The condensed water is pumped to a series of low-pressure pre-heaters and demineralizers before entering the feedwater tank. After leaving the tank, the condensed water is pumped through a series of high-pressure feedwater heaters before entering the SGs.

If the main feedwater system supplying water to the SGs is not available, the auxiliary feedwater system is used to supply SGs.

Main Feedwater System

The main feedwater system forms part of the secondary circuit which also includes the steam system, the electrical generation system, e.g. the high-pressure and low-pressure turbines and the condensate system.

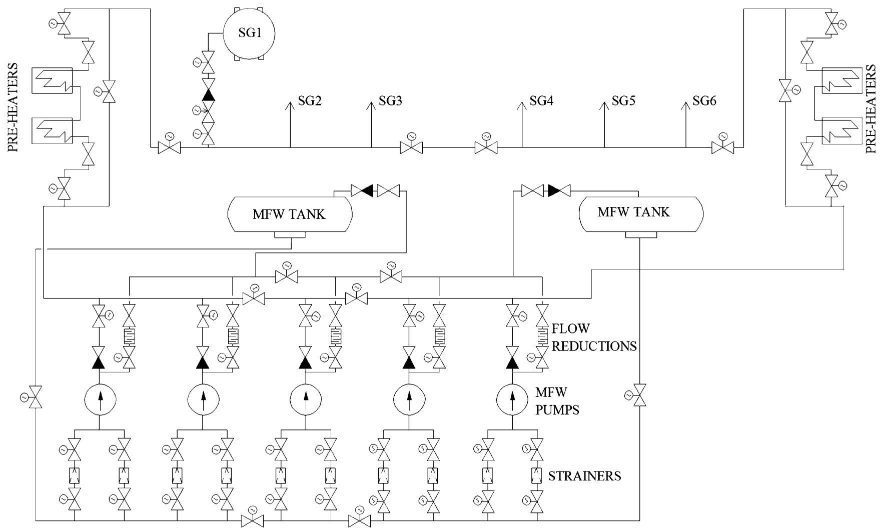


Fig. 2.23 The main feedwater pumps

The condensate system supplies water to the main feedwater tanks and the main feedwater system transfers this water to the steam generators raising its pressure and temperatures with pumps and pre-heaters. In the steam generator, the water is heated to give steam by the primary circuit which in turn cools the reactor core.

There are five pumps, (four for full output plus an automatic standby, see Fig. 2.23). The pumps are required as follows:

Up to 5 % rated output	1 Main feed pump
Between 5 and 30 % rated output	2 Main feed pumps
Between 30 and 50 % rated output	3 Main feed pumps
Over 60 % rated output	4 Main feed pumps

In the full power level 1 PSA, 100 % power only is considered, requiring 4 pumps, together with the post-trip requirements, of any one pump being required.

At 100 % full power, four pumps are running discharging to a common feed header, supplying all six SGs for full power operation. The reserve pump is designed to start in following loss of one of the running pumps. The segregation valves for half unit operation are all open. Both main feed tanks are in use each supplied by its own half unit condensate system.

Both pre-heaters are in use raising feedwater temperature from 164 °C nominal maintained tank temperatures to steam generator supply temperature. Post-trip, the pre-heaters are not required to function and the bypasses may be used.

During half unit operation, e.g. following a TG trip, the system may be segregated into two halves, one of which continues to cool the core while generating electricity for safety systems, self-consumption and grid use. Segregation may also be used during maintenance on one half unit.

Following a breach of the secondary circuit or a primary to secondary tube leak, the steam generator is isolated including the main feed line.

Following a reactor trip, main feedwater may be available provided power supplies to pumps are maintained.

One common suction header is installed between both the feedwater tanks and the feedwater pumps. Each feedwater pump is joined to the common header by two branch pipes fitted with pertinent filter screens (strainers) and isolation valves. Both branch pipes are joined together before the pump to one pipe that is joined to the pump suction branch. The redundant suction system arrangement makes screen cleaning at full operation possible.

The main feedwater pump discharge lines are through non-return valves and motor-driven isolation valves joined to a common discharge header that deliver feedwater to the high-pressure regeneration system of both unit turbines.

The standby pump is situated in the middle of the area, and its discharge line can be reconfigured towards any unit turbine by means of valves installed in the feedwater header.

Each main feedwater pump has its own lubricating oil supply system that provides motor and pump bearings pressure lubrication.

All feedwater pumps are fitted with the minimum bypass lines to the feedwater tank. The bypass lines enable the pumps to operate safely under conditions of reduced flow or in case of closed discharge lines. The main feedwater pump motors, the oil coolers and the pumps themselves are cooled by circulation cooling water.

Auxiliary Feedwater System

The auxiliary feedwater system delivers water to the steam generators in the event of a trip of the main feedwater system. Water is derived from the main feedwater tanks which are common to the main feedwater supply. The main feedwater suction header, is from two unions cross-connected pipe work provides a redundant path to the auxiliary feedwater pumps. Manual valves that are normally in the open position isolate the pumps during maintenance. The pump system is organized on a one out of two redundant bases.

A minimum bypass line back to the main feedwater tanks allows safe operation of the pumps if the discharge header output flowrate is reduced. Each discharge line has an orifice plate, check valve and motor-operated valve.

A common discharge header carries the water from the pumps and then to a line independent of the main feedwater system to the steam generators. Cooling of the auxiliary feedwater pumps is by means of the service water system.

The system is in standby mode at 100 % of power. The first pump automatically starts upon receipt of a safety signal of low SG level in any of the six SGs. Failure

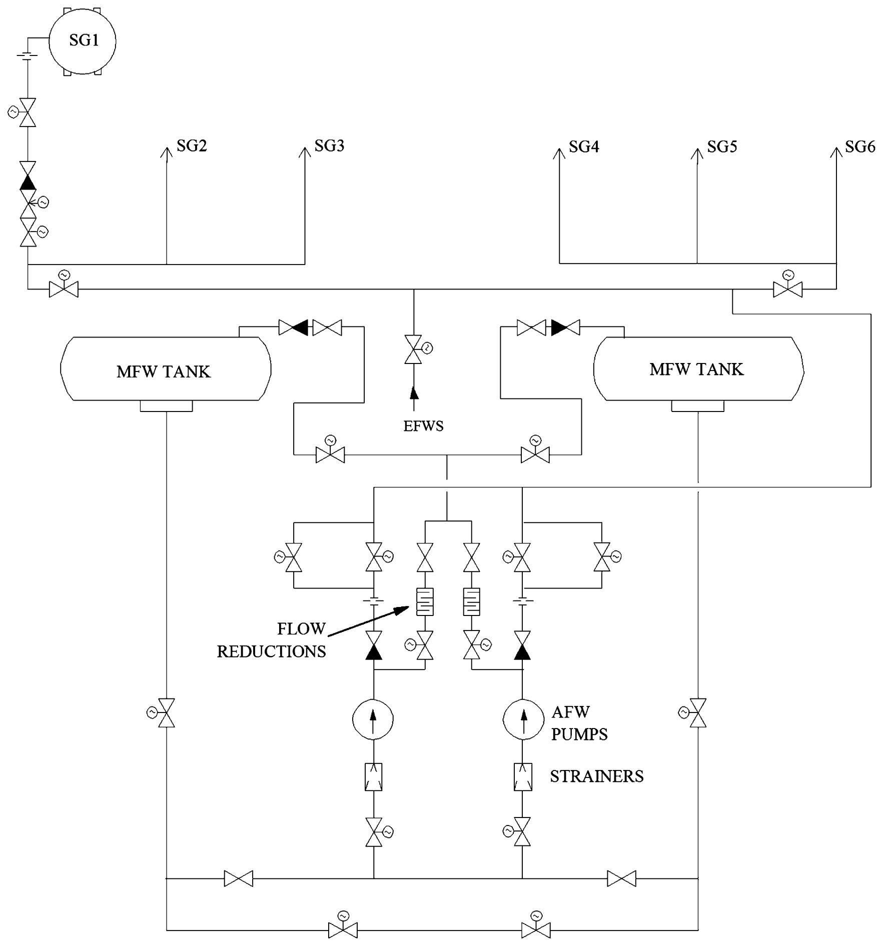


Fig. 2.24 The auxiliary feedwater system

to start this pump after 10 s initiates the start of the second pump. If a failure of the first pump occurs during its operation, the second pump in this case receives signal to start. During the concurrent loss of self-consumption and reserve electrical power supplies, the pump start signals are derived from the load-sequencing system (Fig. 2.24).

Emergency Feedwater System

There is another feedwater system which may be used if the main and auxiliary feedwater systems cannot be used and if certain conditions are met. The

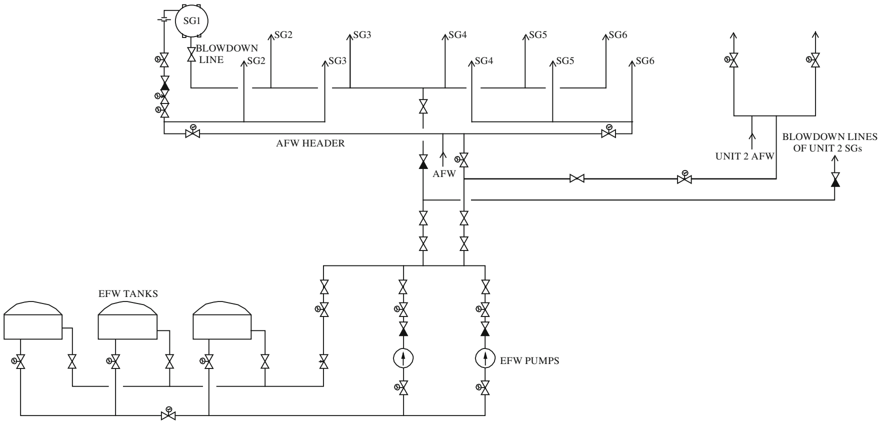


Fig. 2.25 The emergency feedwater system

emergency feedwater system supplies unheated water from the three main demineralized water storage tanks (1000 m^3 volume capacity of each) via one of two pumps. This is a common system for both units. The water is supplied to AFW header and the blowdown lines of SGs (see Fig. 2.25). The pumps are started manually by the operator.

The tanks are heat-insulated, and during extreme frosts, they are heated. Each pump has a minimum flow line with a three-way check valve and a manual isolation valve back to the tanks.

Demineralized Water 1 MPa System

The system is used to supply water to the main feedwater tanks if water from the condensate system is not available. During normal plant operation, the system is in the standby state.

The pump suctions are connected to the three demineralized water tanks, which are also used to supply the emergency feedwater system, Fig. 2.26. Water delivery from the demineralized water 1 MPa system can be used for both units. During normal plant operation, all valves in the lines of tank interconnections are open.

Condensate System

The steam condensate system is used to close the loop in the secondary circuit. Steam raised by the heat exchange from the nuclear reaction is used to drive turbines to produce electrical energy. After the steam has passed through the turbines, it enters the steam condensate system (main condensers). Here, the steam is condensed and returned to the main feedwater tanks.

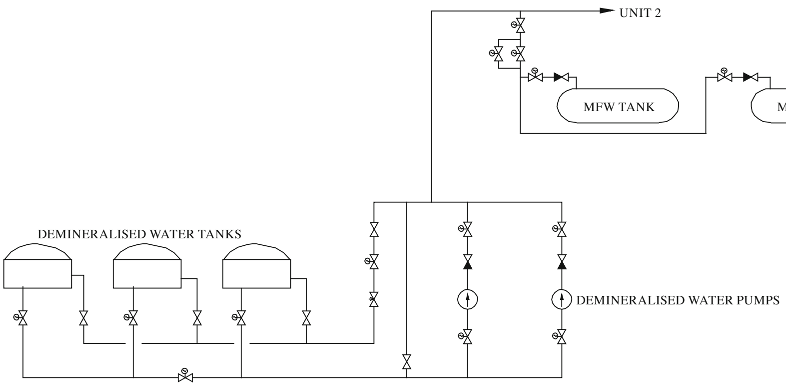


Fig. 2.26 The demineralized water 1 MPa system

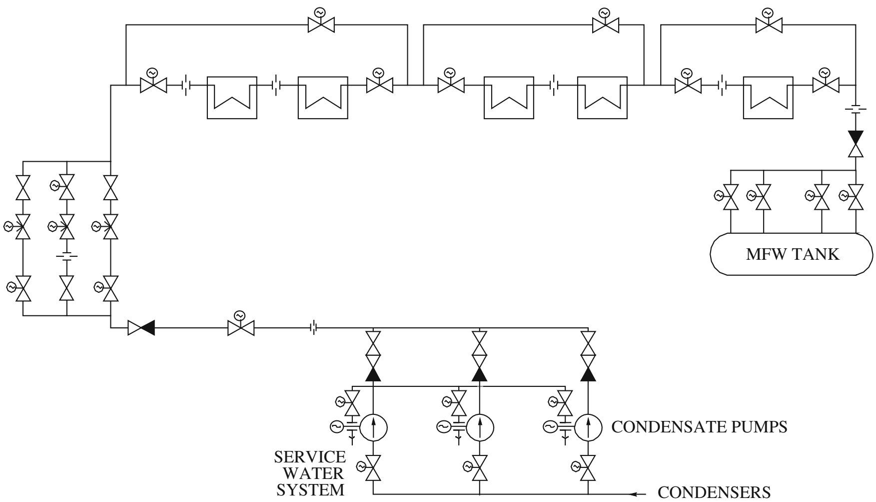


Fig. 2.27 Condensate system and low-pressure pre-heaters of TG1

The system is arranged in two separate and identical halves. One half services the steam that has passed through TG1 and returns its condensate to the first main feedwater tank. The other services the steam that has passed through TG2 and returns its condensate to the second main feedwater tank. The first half of the condensate system with low-pressure pre-heaters is shown in Fig. 2.27.

Each half of the system comprises two condenser units (K1 and K2). A vacuum should be maintained in condenser units. Steam should enter the condensers from either the turbine or steam lines via the steam dump stations to the condenser. Cold main circulation water is passed through the internal tubes of condensers (see Fig. 2.28).

There are three condensate pumps (normally two in operation) in each half of the system which take condensate from condenser units. These pump the condensate through the supply route to the main feedwater tank. As the condensate moves along the supply route, it passes through five low-pressure pre-heaters. These are heat exchangers which use steam/condensate taken from various stages of steam passage through the turbines, to heat the condensate as it enters the main feedwater tank. Two pumps are normally in operation with one pump in reserve. This reserve starts automatically on tripping of an operational pump. Start-up of the condensate pump generates a signal to open valve in the service water supply line in order to cool it. The valve is closed by a signal generated after the trip of the corresponding pump.

At 100 % power, both halves of the steam condensation system should be in operation receiving steam from the turbines of TG1 and TG2, respectively. In each half, two condensate pumps should be in operation, the third in standby.

The supply route is a section incorporating three parallel paths containing regulating valves. The central path is the start-up path, and its control valve should be shut. One of the other two paths should be open with control valve regulating condensate flow. The third path should be shut and available for standby use.

2.1.1.5 Support Systems

Circulating Cooling Water System

This support system supplies the cooling water for the turbine condensers and other components (Fig. 2.28). The system draws water from two circulating water tanks and uses four pumps to deliver water to the consumers via two supply lines. The two circulating water return lines from the condensers distribute the warm water to four cooling towers. At the cooling tower, some of the water is vaporized and carried away to atmosphere by the updraft created by the warm vapour and warm air. The cooled water rains down to the sump and flows through screens to the two circulating water tanks. The water loss, as result of evaporation in the

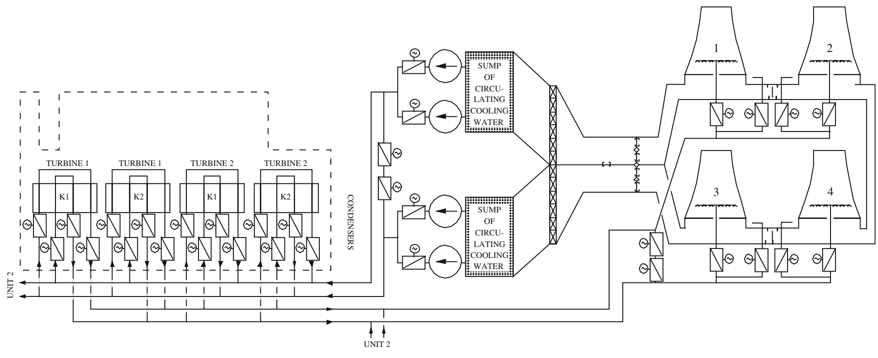


Fig. 2.28 Circulating cooling water system

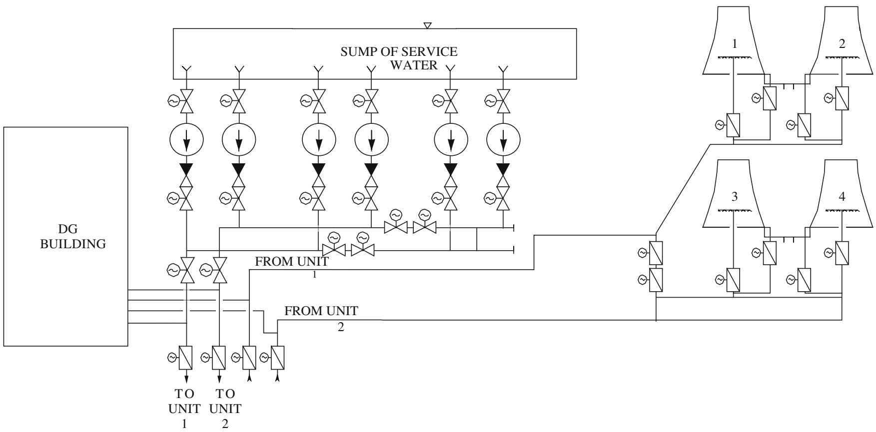


Fig. 2.29 Service water system

cooling towers, must be replaced. This is achieved by the water from external water sources. Each condenser is connected to each supply line. The lines are interconnected to allow variable cooling process. The water temperature in the condenser is increased by 10.3 °C, what was the basis for cooling tower dimensions.

Service Water System

This support system has to cool its consumers during both normal plant operation and accident situations. The system draws water from a single-service water tank and uses six service water pumps (two pumps are continuously in operation) to distribute it to the consumers. This is a common system for both units. There are two supply lines each serving the consumers in their own unit and a number of common consumers. The return water joins the circulating water system return line and passes to the cooling towers.

Each pump has a check valve and MOV in its discharge line. The discharge of the pumps is connected to two delivery lines which are interconnected (Fig. 2.29).

For each unit, there is a service water supply backed by an emergency service water tank which is used after station blackout to supply a subset of user systems.

Given the loss of offsite power, the working pumps are restarted by the load-sequencing system. Given the working pump trips, the reserve pump will start automatically.

The following consumers are cooled by the service water system:

- diesel generators,
- high-pressure safety injection pumps,
- heat exchangers of confinement spray system,
- heat exchangers of intermediate cooling system of MCPs,

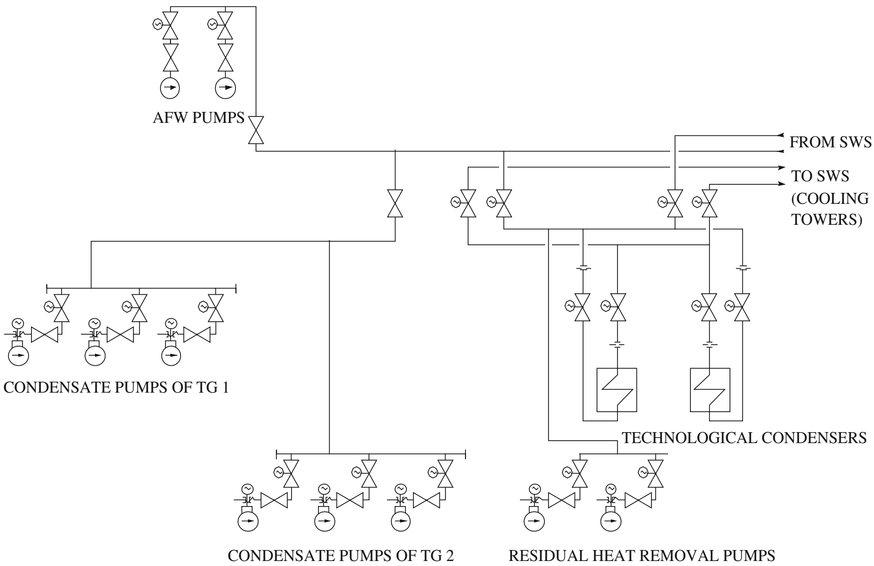


Fig. 2.30 Service water system in the TG hall

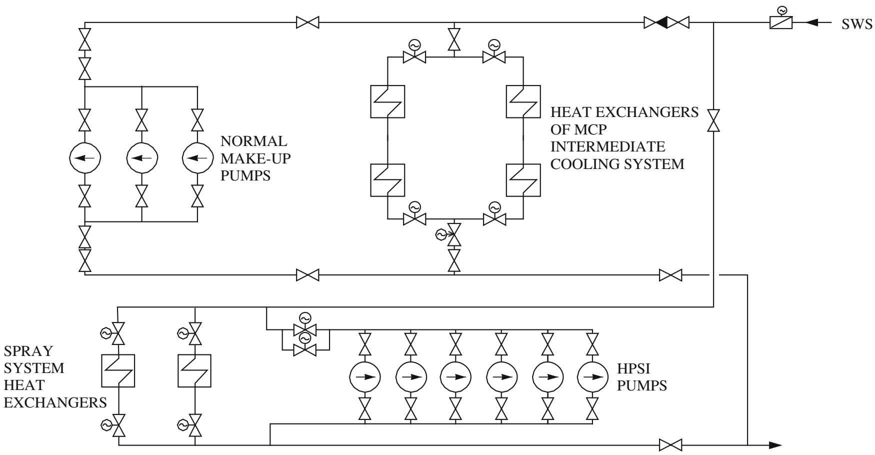


Fig. 2.31 Service water system in the reactor building

- primary circuit make-up pumps,
- auxiliary feedwater pumps,
- condensate pumps,
- residual heat-removal pumps,
- technological condensers, etc.

The service water system in the reactor building is presented in Fig. 2.30 and in the TG hall in 2.31.

Residual Heat-removal System

The cooldown of the primary circuit below 140 °C is provided by the residual heat-removal system (RHR). It is designed to cool down the primary circuit to approximately 50 °C. Heat from the primary circuit is removed through the steam generators.

The primary cooling circuit comprises a technological condenser, a reducing section and two RHR pumps for each unit.

The equipment is common and can be used for cooling both units. The equipment consists of two systems, both systems being identical. Each unit has one pressure-reducing station, a technological condenser and two residual heat-removal pumps.

The system follows the following route:

- residual heat-removal pump,
- residual heat-removal pump discharge pipeline,
- main feedwater discharge collectors,
- high-pressure heaters or their bypass,
- steam generator feedwater pipelines,
- steam generators,
- steam lines and steam header,
- pressure reduction station,
- technological condenser (feedwater tank),
- suction of residual heat-removal pump.

The primary circuit cooldown process is divided into two stages (steam-water and water–water stages). In the first stage, steam from the steam generators passes through the steam header and pressure-reducing station to the technological condenser where it condenses. The technological condensers are cooled by service water. The condensate is fed into the feedwater tank.

The pressure of the steam before reaching the reduction station is approximately 4.7 MPa. After passing through, the pressure is reduced to 0.4 MPa. The steam pressure in the technological condenser depends on the feedwater tank pressure (0.1–0.7 MPa).

If the pressure in the feedwater tank is higher than in the technological condenser, the condensate cannot be supplied to the tank and the condenser level increases. In such cases, the condensate is supplied to the condensate storage tank.

In the second stage, the condensate is circulated using residual heat-removal pumps from the technological condensers to the SGs. Water, from the feedwater tank, is fed to the suction of the pumps to compensate losses (Fig. 2.32).

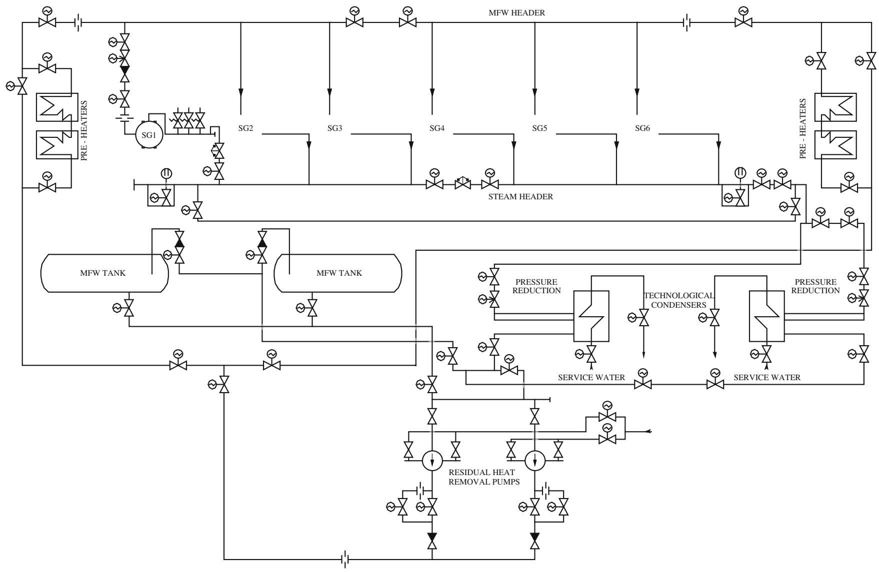


Fig. 2.32 Residual heat-removal system

Normal Power Supply System

The normal electrical power supply of the unit is based on two main and two auxiliary turbogenerators. The main turbogenerators feed the grid via the line used to export to the grid. The main turbogenerators also feed the unit's main self-consumption transformers. Output from the main TGs is at 15.75 kV, and it is transformed down to 6 kV for the supply of the non-essential loads.

The auxiliary turbogenerators supply the selected main coolant pumps among other consumers. Supply of the 6 kV buses is given via self-consumption transformers T11 and T12. The main distribution 0.4 kV buses are fed from the 6 kV buses via transformers 6/0.4 kV. The lower distribution buses are fed from the 0.4 kV main distribution buses.

A simplified line diagram of the normal power supply system is provided in Fig. 2.33.

Failure of the normal internal supply is mitigated by provision of a reserve supply system which is described below. After the reactor trip, the transfer of the unit self-consumption to the reserve power supply is needed.

Reserve Power Supply System

The reserve power supply system supplies the unit self-consumption loads during a loss of normal power supply after reactor trip.

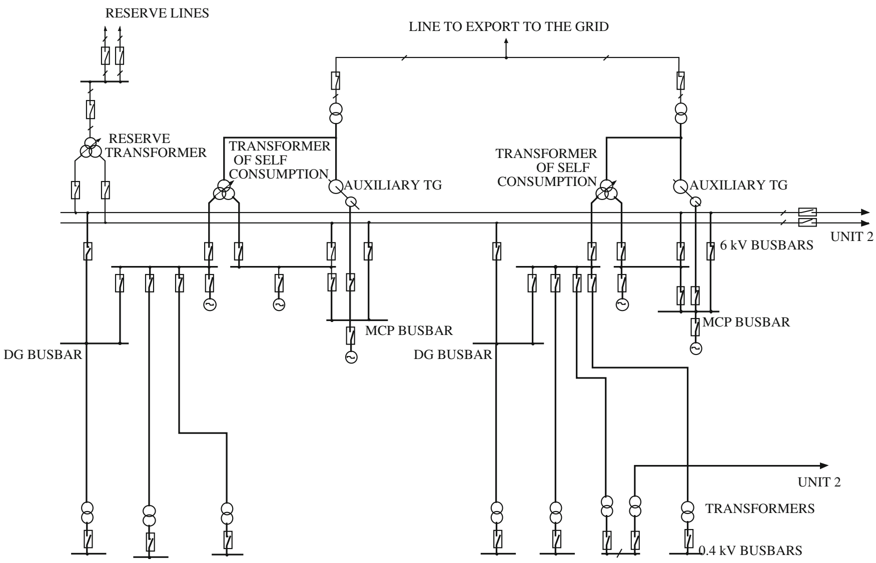


Fig. 2.33 Normal, reserve and essential emergency power supply

A simplified diagram of the system is provided in Fig. 2.33.

Given both TG's trip and following the reactor trip, the unit self-consumption is fed from the reserve transformer. This line is independent from the export line.

The secondary side of the reserve transformer is connected to the 6 kV reserve busbars. Loss of power on the reserve busbars causes automatic transfer to the reserve busbars of unit 2.

Essential Emergency Power Supply System

The essential emergency power supply system supplies the essential loads in the event of loss of internal and reserve power sources.

The essential emergency power supply system consists of two independent subsystems. Each subsystem consists of the following equipment:

- diesel generators,
- 6 kV buses,
- 6/0.4 kV transformers,
- 0.4 kV distribution buses,
- 0.4 kV lower distribution buses.

The simplified schematic of the essential emergency power supply system is shown in Fig. 2.33.

Given loss of power from normal and reserve power supply system, the diesel generators are started automatically and they are connected to the 6 kV buses.

The starting signal is generated. At the same time, signals are generated for disconnection of sectional breakers, breakers of reserve power supply system and all great consumers supplied from the DG-backed buses. After successful DG start and disconnection at least one out of two sectional breakers and one out of two breakers of reserve power supply system, the signal is generated for closing of DG breakers. Then, the consumers are restarted using the load-sequencing system.

Uninterruptible Power Supply System

The system supplies critical loads in the event of the loss of all onsite and offsite power. The equipment supplied by this category of supply includes instrumentation, control, and actuation devices for a range of safety, front line defence and shutdown emergency systems.

The uninterruptible power supply system is created from two identical independent subsystems. Each subsystem consists of following equipment:

- rectifier,
- invertor,
- main distribution bus 0.4 kV (AC),
- motor generator set,
- main distribution bus 220 V (DC),
- lead-acid accumulator battery.

The simplified schematic of the uninterruptible power supply system is shown in Fig. 2.34.

2.1.2 Safety Upgrading of the WWER440/V230 Reactors

The WWER440/230 reactors were designed in the early 1960s. They conformed with industrial standards available at that time but they do not comply with currently acceptable safety requirements. The WWER440/V230 nuclear power plants operated in different countries have been reviewed by several international organizations. In particular, the IAEA conducted review missions to the sites with the purpose of defining urgently required safety improvements. Action plans were defined for upgrading of these plants. On the other side, many plants with this type of reactors were shut down and operation terminated.

Safety upgrading of other plants were performed (Bohunice V1 NPP, Kola 1,2 NPP, Novovorenezh 3,4 NPP) and high level of safety, comparable of Western-type PWR, was achieved.

Safety assessments show that the early WWER440/V230 reactors have many inherent safety features that are absent on Western-type reactors and after upgrading can more than compete on safety terms with reactors of their vintage, or younger, in the West. An example is the water volume in the SGs. Given total loss

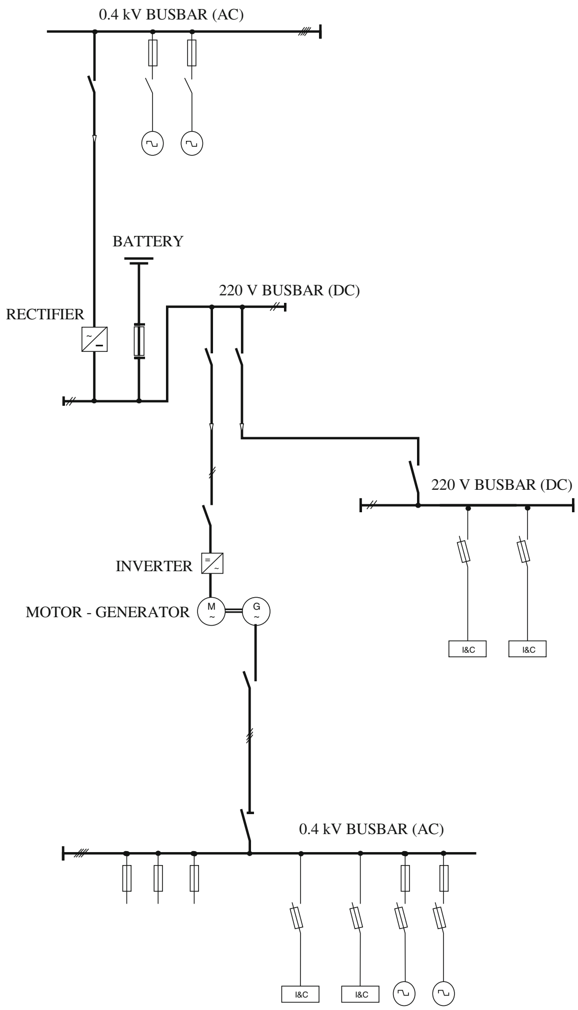


Fig. 2.34 The uninterruptible power supply system

of feedwater supply, the WWER440/V230 reactors have enough feedwater for at least three hours. The Three Mile Island accident has shown us that in case of the Western-type PWRs within half an hour the SGs dried.

Despite of high safety level of the Bohunice V1 plant after modernization, its operation was premature-terminated (the unit 1 operation in 2006 and the unit 2 operation in 2008). The Kola and Novovorenezh plant are still in operation. The safety upgrading of the Bohunice V1 reactors is described below. It was a pilot project for reactors of this type and after the completion of the final reconstruction the Bohunice V1 plant became the safest power plant with WWER440/V230-type reactors which met the current safety requirements [6, 7].

The operational safety of the plant has been questioned by international groups and also by governments of neighbouring countries. The former Czechoslovak Safety Authority initiated the small reconstruction of the plant. 81 safety measures were defined, and continued operation of the plant after 1992 was made conditional on their implementation to an agreed schedule. Within the small reconstruction, the safety measures were implemented in both units until 1995. The most important safety measures implemented within the "small reconstruction" were the following [2, 6, 7]:

- annealing of the reactor's pressure vessel of both units,
- significant improvement of the confinement hermetic area tightness,
- increase in the reliability of the heat transport in the secondary circuit,
- increase in performance of emergency power supply system of safety sections,
- seismic reinforcement of the structures, systems and components,
- safety measures to improve fire safety,
- installation of the new diagnostic systems.

The operation of the plant after 1995 was permitted only under condition that the safety level will be further enhanced by gradual reconstruction. So the gradual reconstruction was started in 1996, and it was finished in 1999 for unit 2 and in 2000 for unit 1.

Within the gradual reconstruction, the following main changes were performed in the plant configuration:

- A new computerized reactor protection and ESFAS system was installed.
- The ECCS system was modified. Two LPSI pumps were installed and the number of HPSI pumps was reduced from 6 to 4. This modification together with the confinement improvements allowed the plant to cope with double-ended guillotine break of RCS piping with 500 mm diameter.
- Modification of the confinement spray system.
- Primary RHR line was installed into the borated water storage tank which is utilized for emergency residual heat removal using HPSI pumps and the confinement spray system heat exchangers.
- The second redundancy of the EFW system was added.
- New essential service water system was built up.
- Steam dump stations into the atmosphere were installed on each steam line. They allow emergency residual heat removal in the form of secondary bleed and feed.
- New confinement pressure suppression system was installed. The post-gradual modification confinement status meets the requirement of the new design basis accident. It ensures the confinement integrity even in the event of a guillotine break of the primary piping with 500 mm diameter.
- The primary bleed and feed hardware and procedures were improved.
- The safety system components, buildings and structures were seismically qualified, including interactions.
- Complete physical separation of redundant systems was done.

In 2003 and 2004, symptom-based emergency-operating procedures (EOPs) prepared by Westinghouse were introduced in the Bohunice V1 plant for all operating modes to increase operational safety and reliability of the operators. These new procedures replaced the previous event-oriented procedures for the liquidation of failure states. The accident of Three Mile Island plant (1979) confirmed that the operational staff had not sufficient support for the accident liquidation in the valid event-oriented procedures. The analyses of the accident captivated the attention to the human reliability. They unambiguously confirmed the low reliability of operator in the extreme stress situations, if his activity is not supported by the convenient procedures. The need for the increasing of the human reliability leads to the development of the symptom-based procedures to minimize the possibility of human commission and omission in the stress situation. US NRC charged the Westinghouse company by the development of these procedures. Their using on the individual NPP began from the half of 80 years. At the present time, symptom-based emergency procedures are standard requirement of nuclear authority in the all mature countries with operational NPPs.

Level 1 and 2 full power, low power and shutdown PSAs were performed to quantify the benefit of the small and gradual reconstruction of the Bohunice V1 plant from the risk reduction point of view. Direct objectives of the analysis were to estimate the core damage frequency and large early release frequency, to identify dominant initiating events and accident sequences with highest contribution to the risk and to show that the safety requirements are met.

In the next part, the reconstruction of the Bohunice V1 plant is described in more detail to illustrate safety improvement of the plant [6].

2.1.2.1 Computerized RPS and ESFAS

RPS

The RPS is designed to cause automatic interruption or slow-down of fission reaction given accident situations. The system provides two levels of safety intervention: HO1 and HO3 (the levels HO2 and HO4 of actual system are excluded).

- HO1 results in a rapid simultaneous insertion of all control rod groups, into their lower-end positions, with a speed of 20–30 cm s^{-1}. This is achieved by switching off supplies to all the CRDMs.
- HO3 results in the insertion of the controlling rod group into the core with the speed of 2 cm s^{-1}. This is followed by sequential driving of control rod groups downwards. The effects of HO3 last only for the duration of the appropriate trip signals.

The computerized RPS consists of two identical full separated trains—redundancies. Each redundancy is able to evoke the reactor trip. The redundancy comprises the following:

- Initiation level—independent measuring channels (MU)
- Sensor signal processing level (GA)
- Data acquisition level (ER)
- Data processing level (VR)
- Drive control level—relay breakers of CRDM's electromagnets power supply.

The protected parameters are usually measured by three independent measuring channels for each redundancy. Measuring channel consists of sensor and I/U transmitter which transforms the current signal to voltage signal and provides sensor power supply.

The analogue signals from measuring channels enter to the system TELEPERM XS (Fig. 2.35). The three data acquisition computers (ER) receive the analogue signals from three measuring channels, convert them from analogue-to-digital signals and form the limit signals. The logic functions are implemented, and the actuation signals are formed in the three processing computers. Logical processing of the signals is carried out redundantly in all processing computers. At the drive control level, the components are actuated using a coupling relay in the switchgear via a 2 out of 3 logic gate of the 6-contact type from the processing computers (VRs) of trains 1, 2 and 3 of both redundancies 1 and 2.

ESFAS

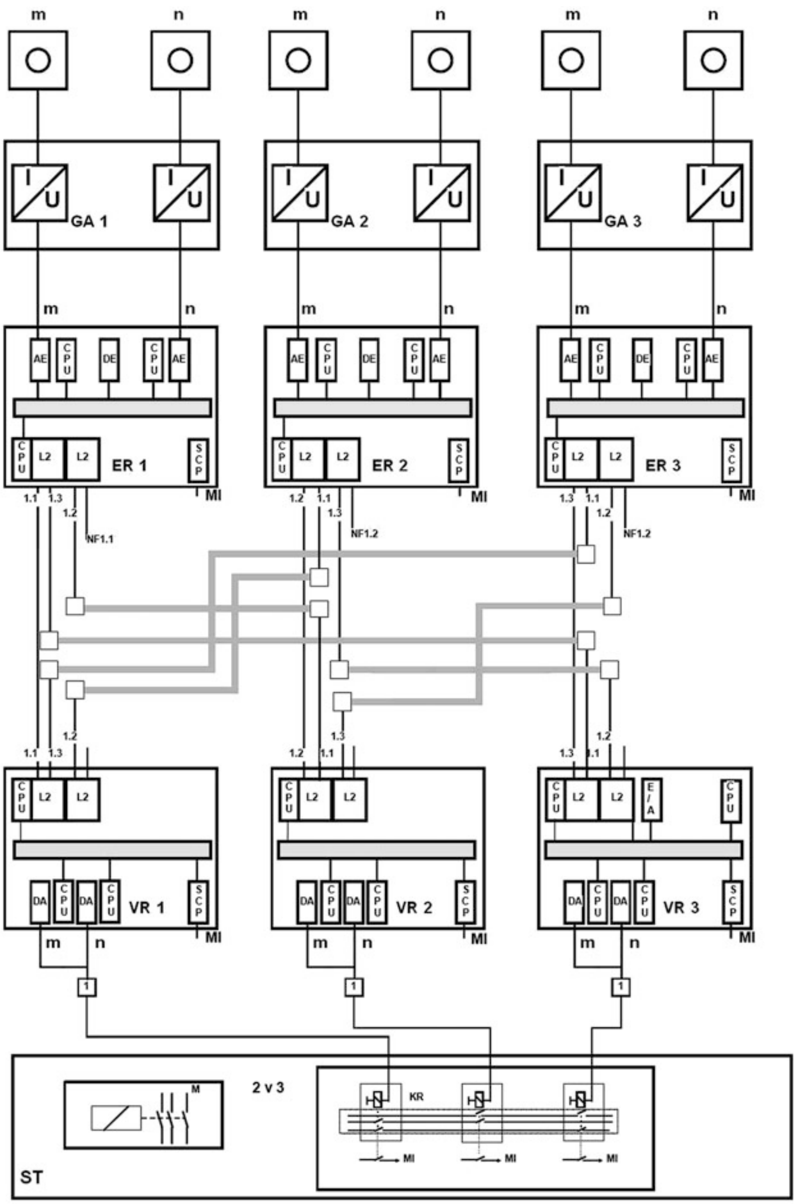
The ESFAS is designed to cause automatic activation of different safety systems to cool down the reactor, to inject water into the primary and/or secondary circuit in emergency conditions and to prevent the radioactive release outside confinement during LOCA.

The computerized ESFAS consists of two identical full separated trains—redundancies. Each redundancy is able to actuate safety feature. The redundancy comprises the following:

- Initiation level—independent measuring channels (MU)
- Sensor signal processing level (GA)
- Data acquisition level (ER)
- Data processing level (VR)
- Actuation level.

The protected parameters are usually measured by three independent measuring channels for each redundancy. Measuring channel consists of sensor and I/U transmitter which transforms the current signal to voltage signal and provides sensor power supply.

The analogue signals from measuring channels enter to the system TELEPERM XS. The three data acquisition computers (ER) receive the analogue signals from 3 measuring channels, convert them from analogue-to-digital signals and form the limit signals. The logic functions are implemented and the actuation signals are formed in the three processing computers. Logical processing of the signals is



GA Power supply sensor and signal processing
ER Computer of data acqisition
VR Computer of data processing
ST Control part (relay)
MI Announcement interface
MSI Announcement and service interface

Fig. 2.35 Computer structures for one RPS redundancy carried out redundantly in all processing computers. At the actuation level, the components are actuated using a coupling relay in the switchgear via a 2 out of 3 logic gate of the 6-contact type from the processing computers (VRs) of trains 1, 2 and 3.

ESFAS ensures the following safety functions:

- start of the confinement spray pumps and the transition to the recirculation,
- start of ECCS pumps (HPSI and LPSI pumps),
- the isolation of the confinement,
- the isolation of SGs,
- feedwater supply for SGs,
- limitation of the steam pressure through the steam dump stations to the atmosphere,
- turbine trip by quick closing valves,
- the accident localization,
- DG start and load-sequencing programme.

2.1.2.2 High-pressure Safety Injection System

The high-pressure safety injection system is designed to compensate primary coolant losses in case of LOCA and to add negative reactivity to the primary circuit. The system is in a standby state during normal reactor operation.

The system consists of two independent functionally identical subsystems with a common emergency borated water storage tank. Two high-pressure pumps of each subsystem have the common suction collector with the low-pressure pump of the corresponding subsystem. This line is designed to ensure that no pump cavitation will occur even all pumps in given subsystem operating in parallel (flow 1,040 $m^3 h^{-1}$) to the depressurized primary circuit and at the water temperature of 100 °C in the tank.

One high-pressure subsystem (pumps 1 and 2) delivers borated water to the unisolable cold part of the loop 2 and the second subsystem (pumps 3 and 4) into the unisolable cold part of the loop 4. The emergency cooling is not connected to the loop with the pressurizer. This secures the efficiency of the cooling also after rupture of the connecting pipe between the loop and the pressurizer (see Fig. 2.36).

Operation of HPSI system in case of LOCA:

1. Large LOCA: This category includes leakages of equivalent leak diameter of 200–500 mm. In case of large LOCA (e.g. double-ended guillotine break in a reactor coolant pipe), the pressure in the primary system drops rapidly. The leakages are compensated by the operation of HP pumps in cooperation with LP pumps.
2. Medium LOCA (100–200 mm): This category includes leakages of equivalent leak diameter of 100–200 mm. Medium LOCA is compensated by the parallel operation of the HP and LP pumps.

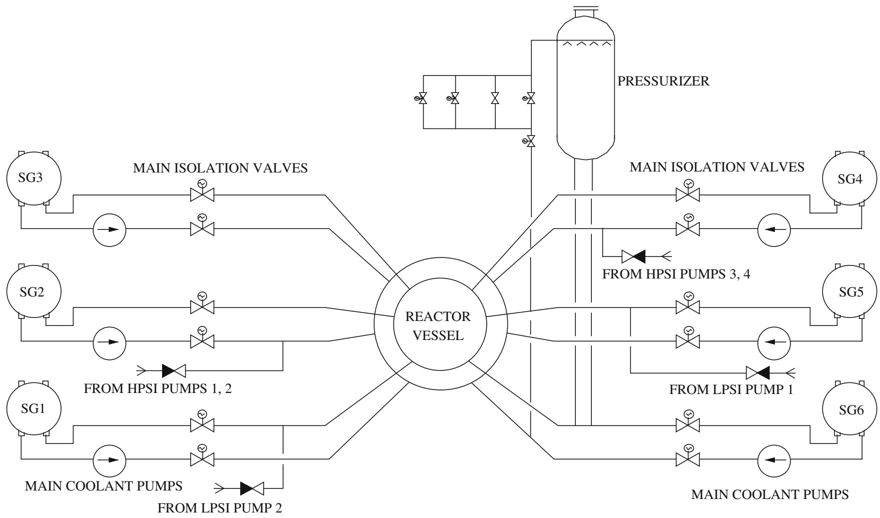


Fig. 2.36 Connection of HPSI and LPSI pumps to the primary circuit

3. Medium LOCA (32–100 mm): This category includes leakages of equivalent leak diameter of 32–100 mm. Medium LOCA is compensated by the operation of the HP pumps or LP pumps given manual depressurization of primary side after the failure of HP pumps.
4. Small LOCA: This category includes leakages of equivalent leak diameter of 0–32 mm. Due to the high primary system pressure, the leakages are compensated by HP pumps. The injection rate of the HP pumps can be greater than the escaping coolant blowdown rate, so that heat removal via steam generators and the secondary side is required. Given failure of all HP pumps, the LP pump can be used to compensate losses after manual depressurization of the primary circuit.

Operation of HP system in case of earthquake: the pressurizer water level is maintained and the boric acid concentration in the primary system is ensured by the HPSI pumps. In addition, the system can be used to cool down the plant after an earthquake via the primary side to the cold subcritical condition (50 °C), using the bleed line to the emergency borated water tank (see the text below). Reactor trip after an earthquake causes a drop in the primary coolant temperature and subsequently a drop in the pressurizer water level which has to be compensated by HPSI pumps. After the drop in the pressurizer water level, the HPSI pumps will be started automatically.

2.1.2.3 Low-pressure Safety Injection System

The low-pressure safety injection system is designed to compensate the primary coolant losses in case of a LOCA. The system is in a standby state during normal reactor operation.

The system consists of two independent, functionally identical subsystems with a common emergency borated water tank. The first subsystem (pump 1) delivers borated water into the unisolable part of the hot leg of the loop 1, and the second subsystem delivers borated water into the unisolable part of the hot leg of the loop 5 (see Fig. 2.36). Delivery of the boric acid solution into the hot leg of the loops is more efficient in case of a double-ended guillotine break than delivery into the cold leg.

The LPSI pumps have a common intake line from the emergency borated water tank with the HPSI pumps. This line is designed to ensure that no pump cavitation will occur even all pumps in given subsystem is operating in parallel to the depressurized primary circuit and at the water temperature of 100 °C in the tank.

Start-up of LP pumps is initiated by ESFAS given LOCA, and the primary circuit pressure is less than 3.3 MPa. Then, the LP pumps operate in parallel with the HP pumps.

2.1.2.4 Confinement Spray System

The confinement spray system is designed to decrease the confinement pressure and to cool the boric acid solution storage tank in case of a LOCA. The system is in the standby state during normal plant operation.

The system consists of two independent functionally identical subsystems with common boric acid solution storage tank. The tank is common with HPSI and LPSI systems, Fig. 2.37.

Each subsystem has one spray pump and heat exchanger cooled by the essential service water system. The first heat exchanger is cooled by first subsystem, and second heat exchanger is cooled by second subsystem of the essential service water system.

The spray pumps are started by ESFAS if the confinement pressure reaches a threshold. The motor-operated valves are normally open in standby state in both pump suctions.

Manual valves in the suction of the spray pumps are open during normal reactor operation. The motor-operated valves in the supply lines to nozzles are normally closed. Two motor-operated valves are installed in parallel in both supply lines to the spray nozzles. These valves open or close by ESFAS signals. Successful operation of given subsystem requires opening of 1 out of 2 valves or closing both valves.

The motor-operated valves are installed in supply lines penetrating the confinement wall. They allow the isolation of the supply lines. The valves are in open position.

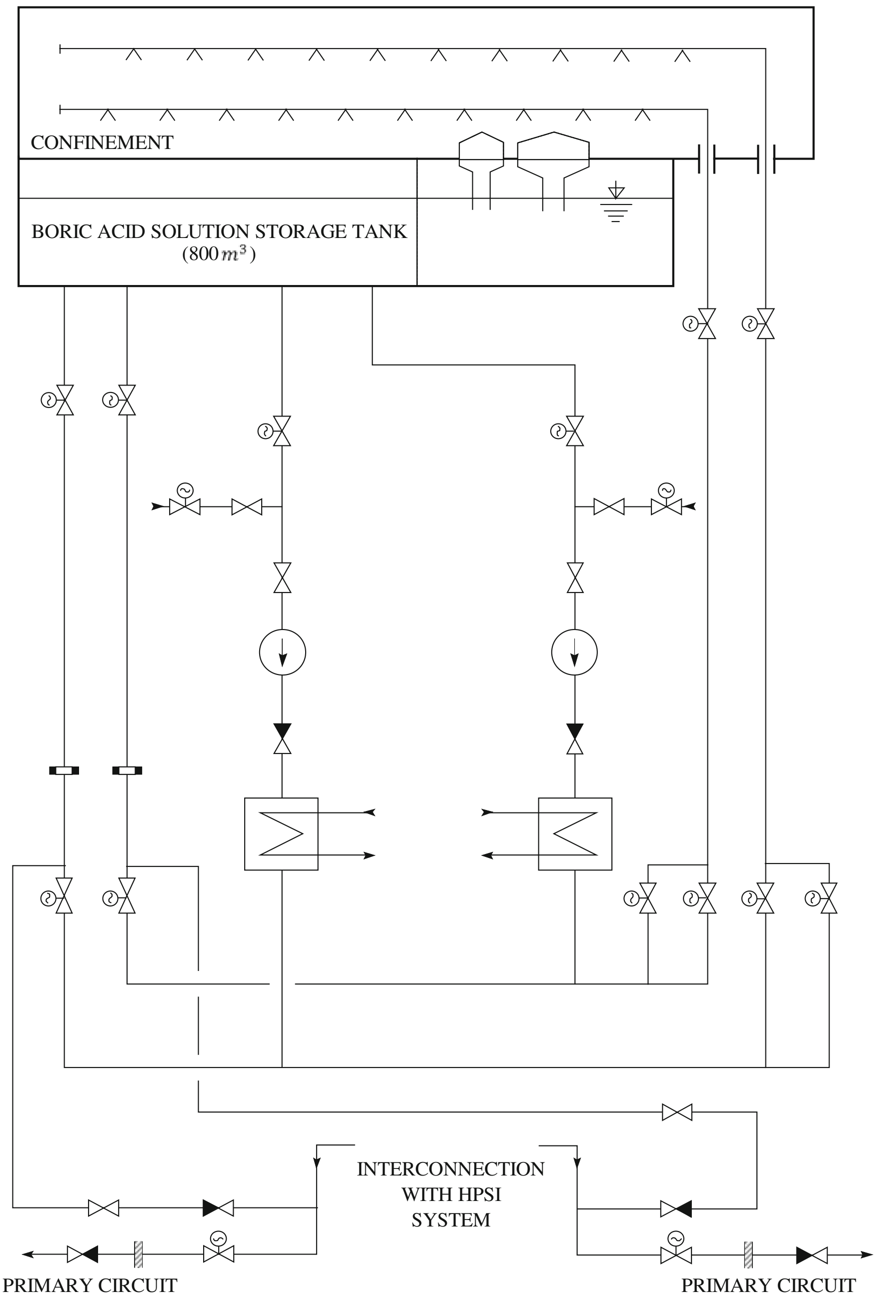


Fig. 2.37 Confinement spray system

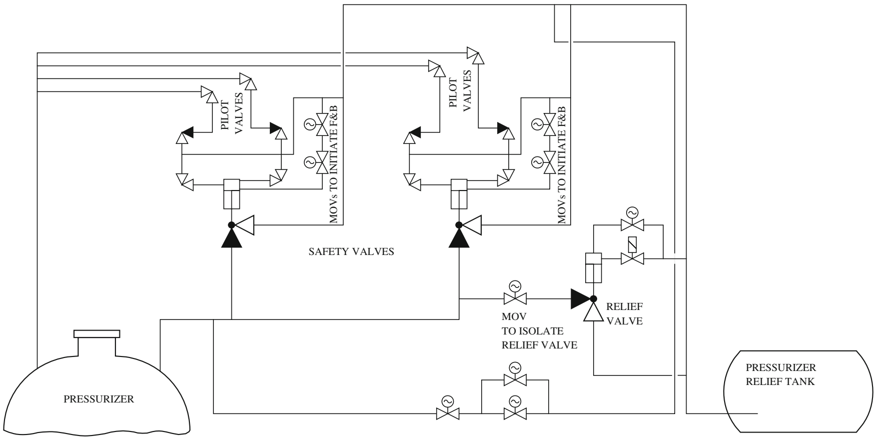


Fig. 2.38 Pressurizer safety and relief valves

Each subsystem has recirculation line to the emergency borated water tank. The line is used for the pump testing. The confinement spray system decreases confinement pressure in case of the confinement overpressure. If the confinement pressure is less than 0.095 MPa, the spraying is stopped and the spray pumps are used for recirculation cooling of the emergency borated water tank (in case of LOCA). This signal causes the automatic closure of the valves in supply lines to the nozzles and opening of the valves in the recirculation lines.

Using the manual valves, the confinement spray system can be connected to the high-pressure safety injection system and can be used for reactor cooling down using the bleed and feed to the borated water storage tank. This way of the operation is used only during unavailability of the high-pressure pumps.

2.1.2.5 Primary Bleed and Feed

The primary bleed and feed operation can be used for cooling down of the reactor through the pressurizer safety valves given the loss of the primary to secondary-side heat removal.

The pressurizer is equipped with two safety valves and one relief valve to protect the reactor coolant system against the inadmissible overpressure. The basic schematic is shown in Fig. 2.38.

Both safety valves have two spring-loaded pilot valves with electrical bias. The spring-loaded valves open the safety valves during plant transients when the primary circuit overpressure arises.

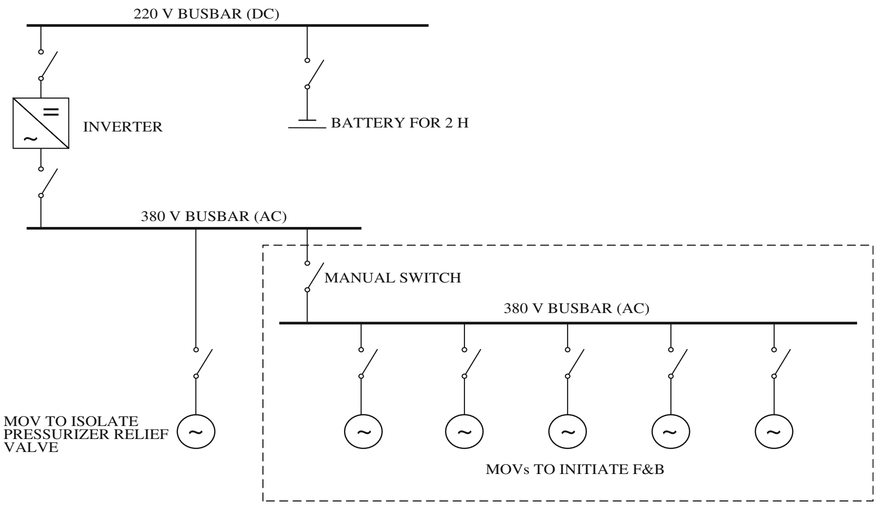


Fig. 2.39 Electrical power supply for MOVs to initiate primary feed and bleed

Each safety valve has a test line with the motor-operated valves. These valves are used during the test of the safety valves or for the initiating the primary bleed and feed operation by the operator. During normal reactor operation, the electrical power supply to these valves is disconnected. This is the way how to prevent spurious operation of safety valves in case of a fire.

Triggered by the drop of water level in all six SGs below 1 m, the alarm signal “Prepare for bleed” will be initiated. To make the bleed function ready for the operation, first the bleed busbar must be connected to the power supply locally in the switchgear cabinet (see Fig. 2.39). Secondary the key switch in the earthquake proof control desk in the main control room must be turned on. Now, the bleed function is ready for the operation. Ready for the operation is indicated in the control room by the alarm “Prepared for bleed”. The bleed operation is started if the level of SGs is less than 300 mm. Before opening a safety valve, a HPSI pump is started.

The relief valve has on its input a motor-operated valve which is normally open. The output of the relief valve is a solenoid valve. In parallel to this valve, the motor-operated valve is added, which is open by the operator only in case of bleed. Its position is signalled to the control room and as electrical power supply is disconnected.

The relief lines from the two safety valves and the relief valve lead to the relief tank. This tank is a horizontal cylindrical vessel with an integral heat exchanger. Its overall volume is 15 m^3.

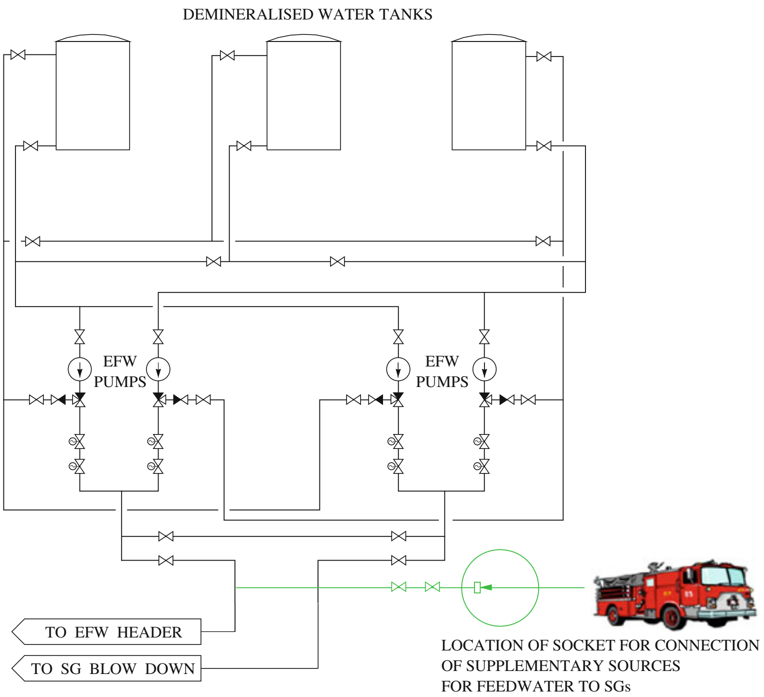


Fig. 2.40 Emergency feedwater supply system

2.1.2.6 Emergency Feedwater System

The emergency feedwater system supplies demineralized water to the steam generators from three storage tanks in the event of unavailability of the main and auxiliary feedwater systems. It is a common system for both units. During normal reactor operation, the system is in standby state.

The emergency feedwater system consists of two diverse and independent subsystems with a sufficient degree of physical, electrical and fire separation and resistance to flooding and earthquake. One subsystem injects water into the EFW header and then to the main feedwater lines in both units, the other subsystem into the blowdown lines of the SGs in both units. Each subsystem contains two pumps, one is electrically supplied from unit 1 and the other from unit 2. Within safety upgrading of the plant, another two pumps are added to the system (Fig. 2.40). In addition, mobile supplementary water sources can be used to supply SGs.

Three demineralized water storage tanks are the water source of the system. Both pumps assigned to unit 1 are connected to the first tank, and both pumps assigned to unit 2 are connected to the third tank. The second tank can be connected to unit 1 as well as to unit 2 by opening the manual valves. Each tank can be shut off separately by the manual valves. Each tank has a volume of 1,000 m^3. The tanks are heat-insulated, and during extreme frosts, they are heated.

Each pump has a minimum flow line with a three-way check valve and a manual isolation valve. The minimum flow lines of the pumps of unit 1 and unit 2 join and return to the respective demineralized water tank. The emergency feed-water system has possibility to be supplied by water from external sources. The connections are possible in each demineralized water tank. The tanks are used as water source also for the mobile source used to supply SGs given loss of the main, auxiliary and emergency feedwater supply.

The discharge line of each pump is equipped with a three-way check valve, a regulating valve which protects the pump from overloading and motor-operated valve which is normally closed.

2.1.2.7 Essential Service Water System

The essential service water system is common for both units. The system is constructed to cool the important consumers during normal plant operation and accident situations. The system is designed for the following accidents:

- LOCA with loss of offsite power in unit 1 and unit 2,
- Earthquake in both units simultaneously,
- Loss of offsite power in unit 1 and unit 2,
- Fire with loss of offsite power in unit 1 and unit 2.

Important consumers of the essential service water system are the following:

- confinement spray system heat exchangers,
- confinement spray pumps,
- high-pressure safety injection pumps,
- low-pressure safety injection pumps,
- diesel generators,
- auxiliary feedwater pumps.

The service water system consists of two subsystems which are common for both units. Each subsystem has four pumps divided into two groups serving to unit 1 and unit 2, respectively. The pumps of the one subsystem supply water to common header and then to particular consumers, Fig. 2.41. Finally, the water returns back to the storage tank passing through the cooling towers.

The system operates as a closed circuit during normal power operation; however, service water leakages are compensated by supplying the water from the river which is water source for the plant. The leakages are supposed to be 5 %.

Each subsystem has an emergency storage tank of volume of 80 m^3 which is used to ensure water inventory given all pumps tripped due to loss of offsite power. It prevents the possible failure of the pumps during restart in load-sequencing system.

One pump of each subsystem is in operation during normal power operation. The other pumps are in standby state. The associated motor-operated valves in the cooling towers are opened with start of the fan coolers in low-speed operation.

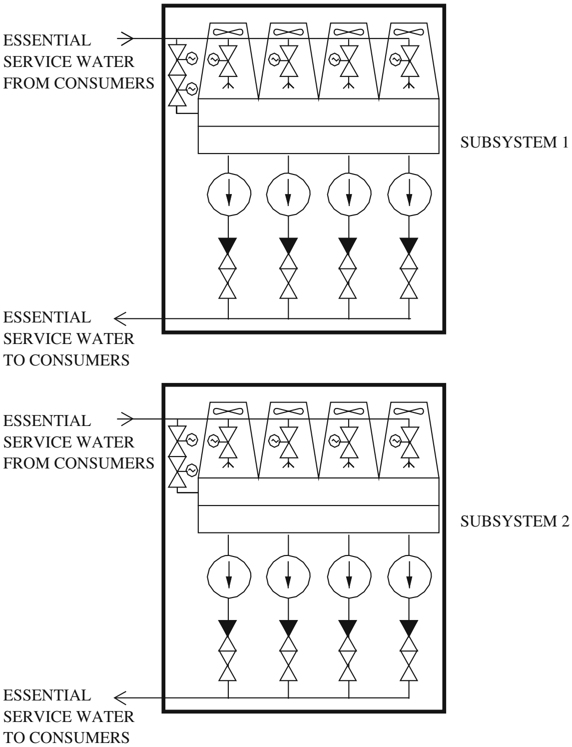


Fig. 2.41 Essential service water system

2.1.2.8 Steam Dump Stations to the Atmosphere

The steam dump stations to the atmosphere installed in steam lines ensure controlled steam removal to the atmosphere in cases when the steam dump stations to the condenser, technological condensers and steam dump stations to the atmosphere (on the main steam header) are unavailable.

The stations are as follows:

- seismically resistant and electrically supplied from the seismically resistant buses,
- protected against the consequences of steam line break,
- periodically tested to achieve high system reliability,
- designed for removal of steam-air mixture,
- designed to cool the reactor to the temperature, which correspond to 0.2 MPa pressure of the secondary side,
- the pressure value to open is higher as for SG relief valve to avoid frequent opening of them.

At each main steam line, one main steam dump station is installed. Each station consists of an isolation and a regulating valve. The schematic of the steam dump

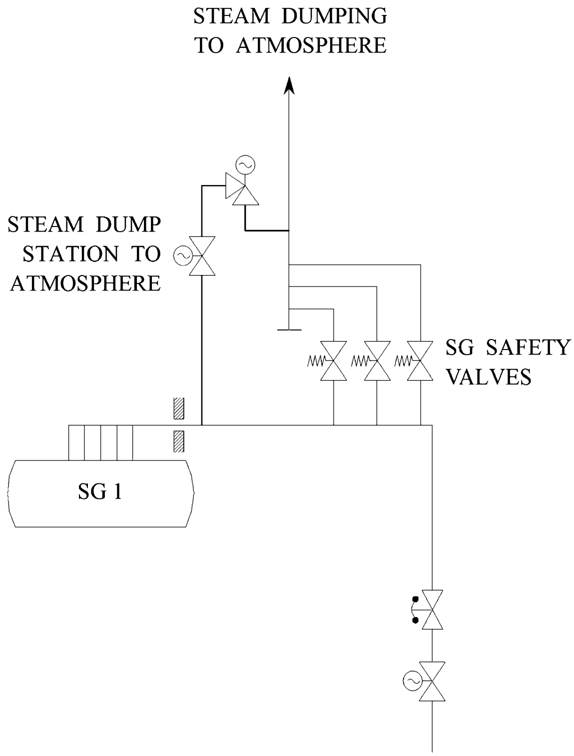


Fig. 2.42 Steam dump station to the atmosphere of SG1

stations to the atmosphere installed on SG1 is shown in Fig. 2.42. Each SG has installed the same steam dump station.

2.1.2.9 Emergency Residual Heat-removal System

The emergency residual heat-removal system is used for the cooldown of the plant to the cold subcritical conditions using the bleed line to the borated water storage tank. The system can be used after a seismic event when the normal secondary-side cooldown system (technological condenser) may be not available. The steam lines are isolated. The primary circuit has been depressurized to 1 MPa and cooled down to 130 °C by the secondary side using the steam dump stations to the atmosphere installed on the steam lines. Then, the cooldown is continued by the heat removal from the primary circuit through the bleed line to the borated water tank (Fig. 2.43). The spray system is started up in recirculation mode to cool the tank.

The heat removal is realized by the spray system in recirculation train through the spray system heat exchangers which are cooled by the essential service water system. The primary circuit is supplied by the HPSI pumps or LPSI pumps. Operation of 1 out of 4 HP pumps is necessary to recirculate the primary coolant.

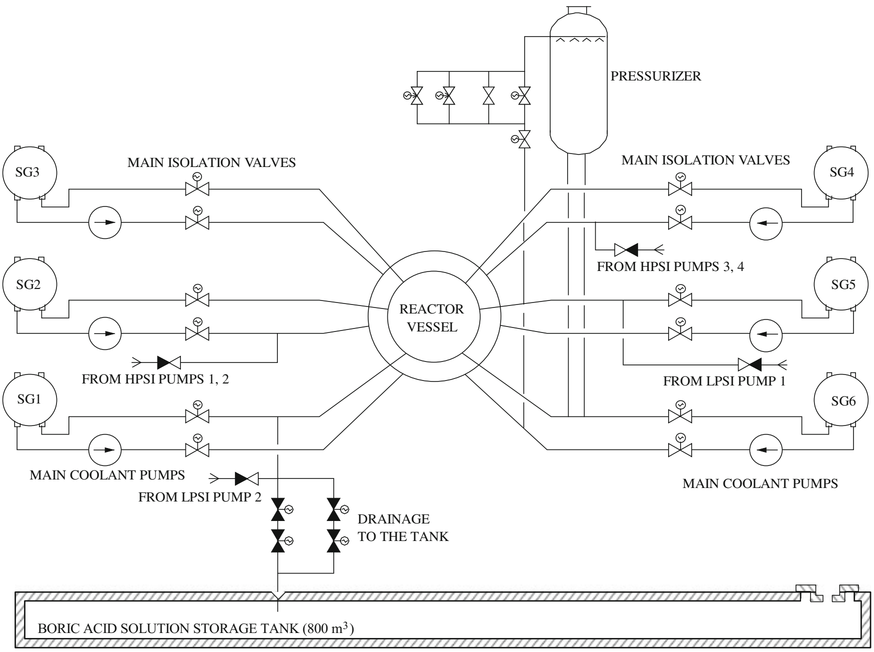


Fig. 2.43 Emergency residual heat removal *bleed line* (to the tank)

The primary circuit is cooled down in such a way to the temperature of 50 °C in the reactor outlet. In addition, the confinement spray system pumps can be used to inject water into the primary side.

2.1.2.10 Confinement Overpressure Protection

For confinement overpressure protections, the following components are available:

1. twelve confinement blow-off valves (DN1200, opening/closing pressure 50 kPa),
2. six jet condensers into the 800 m^3 borated water storage tank with blowing piping (DN1200) and two venting valves (DN150, opening pressure 1 kPa) protecting the tank against overpressure,
3. four rupture membranes (DN800, opening pressure 30 kPa)
4. spray system with heat exchangers.

The first three systems are passive systems working with high reliability. The blow-off valves have to prevent the confinement overpressure given a LOCA. If the opening pressure is achieved, the valves will open, and after pressure decreases, the valves reclose. All valves are set to the same opening and closing pressure.

The jet condensers are used to mix the steam with borated water of the storage tank and improve the efficiency of condensation process. In addition, the condensers prevent the tank from pressure peaks.

The blowing piping is led into the reactor hall. It has to prevent overpressure in the borated water storage tank. Motor-operated valve is installed in the piping which is closed automatically within 10 min from the accident beginning. Given that the valve fails to close, it does not effect the confinement overpressure protection and core cooling process.

The venting valves are used to vent the tank to the confinement atmosphere. The venting lines are equipped with check valves to prevent backflow.

The four rupture membranes are installed due to effective using of confinement rooms for overpressure protection.

After a LOCA, the passive confinement overpressure protection is initiated and the confinement spray system is started. Simultaneously, the confinement ventilation system is automatically isolated. Failure of confinement spray system leads to loss of confinement integrity and consequently to the loss of primary coolant. No isolation of ventilation systems can lead to release of radioactivity into the environment (Fig. 2.44).

2.2 Overview of the WWER440/V213 Reactor Design

2.2.1 Original Design

The WWER440/V213 reactor differs from the older model V230; in that, the model V213 has additional accident localization features and a full-scope ECCS. The most significant addition to the accident localization system (bubble tower) is a pressure suppression system incorporating a large number of water trays serving as suppression pools in which extensive steam condensation occurs during LOCA conditions. For each unit, a set of pressure suppression trays is located inside a separate building adjacent to the reactor building and part of the confinement [4].

A sketch of the major buildings and components of WWER440/V213 plant is given in Fig. 2.45.

The model V213 was designed to mitigate the effects of a double-ended guillotine break of the primary circuit piping with 500 mm diameter, i.e. the largest piping diameter in the reactor cooling system. The WWER440/V213 plant incorporates redundant, independent emergency core cooling systems including high-pressure safety injection pumps, low-pressure safety injection pumps and hydro-accumulators. In the next part of the text, the ECCS, emergency residual heat-removal system and the accident localization systems are described for

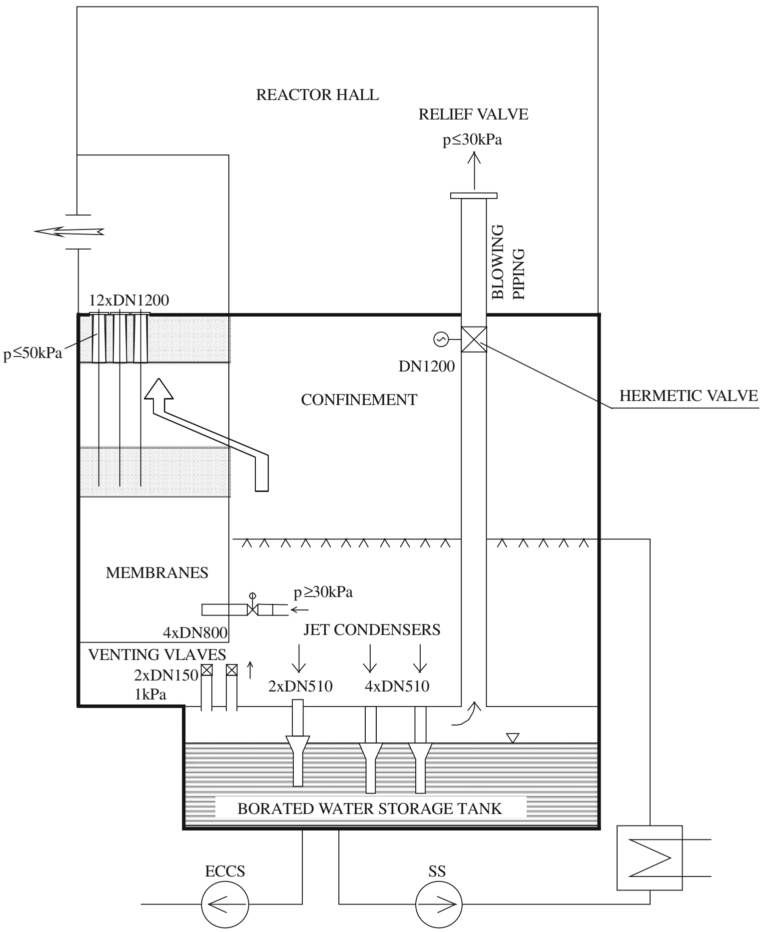


Fig. 2.44 Confinement overpressure protection

the WWER440/V213 reactor. The other systems are the same or similar to the WWER440/V230 reactor.

2.2.1.1 High-pressure Safety Injection System

The HPSI system is designed for:

- compensation of losses from the primary circuit,
- increasing boric acid concentration,

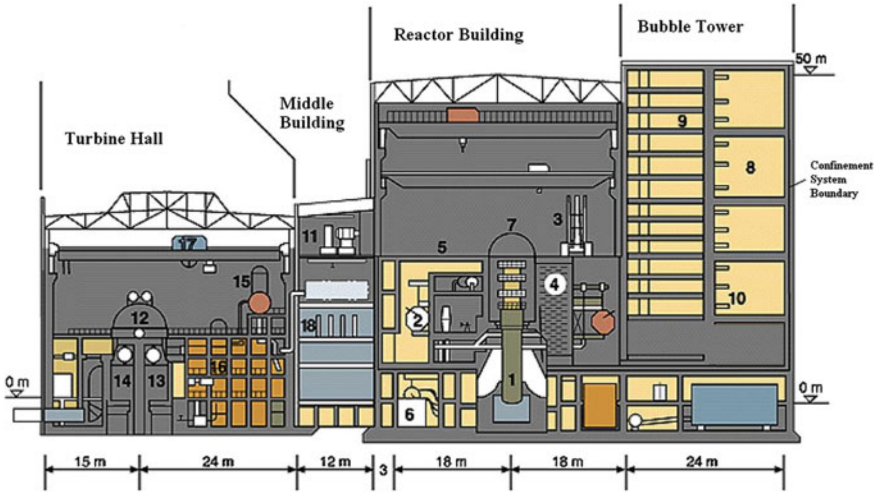


Fig. 2.45 Major buildings and components of WWER440/V213 plant. *1* RPV, *2* SG, *3* refuelling machine, *4* spent fuel pool, *5* reactor hall, *6* ECCS, *7* protective cover, *8* air receiver, *9* bubble tower trays, *10* went lines, *11* machine room of HVAC, *12* turbine, *13* condenser, *14* turbine block, *15* technological condenser, *16* pre-heaters, *17* turbine hall crane and *18* I&C compartments

- compensation of the positive reactivity effects caused by violation of the parameters of the secondary circuit,
- ensuring the subcriticality of the core in emergency situations due to leakage in RCS and its auxiliary systems.

During normal plant operation, the system is in standby state. The system consists of three independent and identical subsystems. They are mechanically, electrically and structurally separated. So, a subsystem does not affect functionality of the neighbouring subsystems in case of an accident. 1 out of 3 subsystems is required for accident mitigation, i.e. the system is designed with 200 % of redundancy. The schematics of the subsystems 1, 2 and 3 are shown in Figs. 2.46, 2.47 and 2.48.

The HPSI system is actuated by ESFAS signal upon LOCA. The air-operated valves in the pump discharge are open given the ESFAS signal.

The main discharge lines are connected to the unisolable parts of the cold loops of RCS.

After the level decrease in the HPSI tanks, the suction of the high-pressure safety injection pumps transfers to the LPSI tanks. Following the level decrease in the LPSI tanks, the injection phase is terminated and the recirculation phase is started. The valve in the suction from the confinement sump is being open and the water is pumped back to the primary circuit through the spray system heat exchangers cooled by service water.

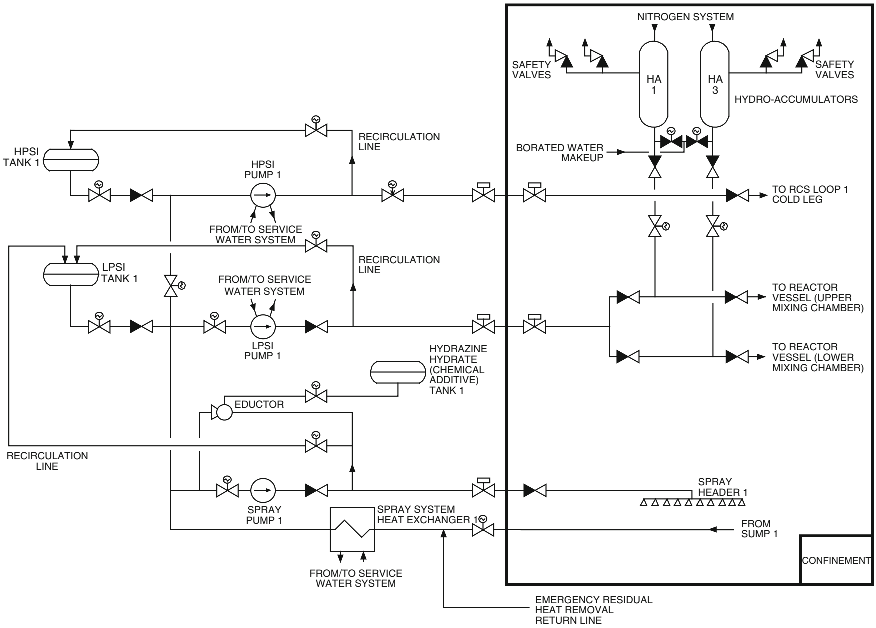


Fig. 2.46 Simplified schematic of emergency core cooling and spray system—redundancy 1

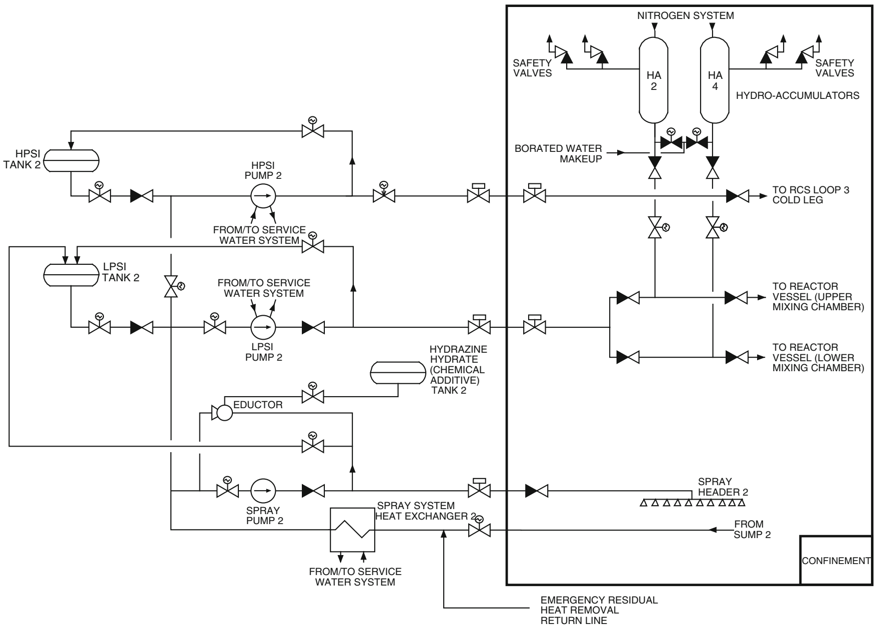


Fig. 2.47 Simplified schematic of emergency core cooling and spray system—redundancy 2

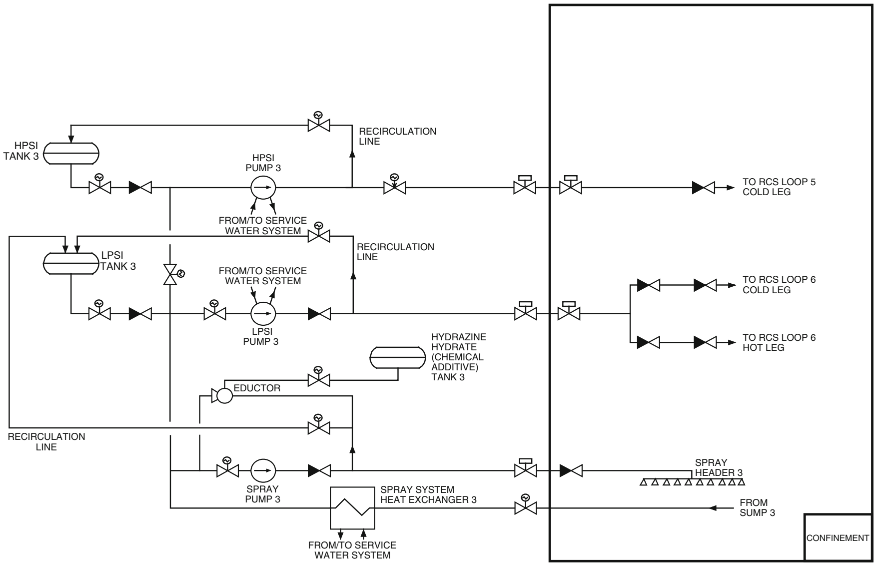


Fig. 2.48 Simplified schematic of emergency core cooling and spray system—redundancy 3

2.2.1.2 Low-pressure Safety Injection System

The LPSI system is used to compensate losses from the primary circuit and to increase boric acid concentration in case of accident. The system is also used for compensation of the positive reactivity effects caused by violation of the parameters of the secondary circuit.

During normal plant operation, the system is in standby state.

The system consists of three independent and identical subsystems. The schematics of the subsystems 1, 2 and 3 are shown in Figs. 2.45, 2.46 and 2.47. For accident mitigation, 1 out of 3 subsystems is required, i.e. the system is designed with 200 % of redundancy.

The system is used to compensate large LOCAs of the primary circuit, at the unsuccessful attempt to maintain the pressure by the high-pressure safety injection pumps and to cool down the core after large LOCA.

The discharge lines are connected to RCS through the air-operated valves and check valves. The subsystem 1 and subsystem 2 are connected to the pipes of hydro-accumulators. The subsystem 3 is connected to the non-isolable part of loop 6.

The system starts automatically from ESFAS signals upon LOCA. The pumps are started; the suction and the discharge lines are prepared for operation; and the air-operated valves, the regulation valves in the discharge lines and motor-operated valves in the suction lines from the LPSI tanks are open. The valves in the recirculation of the pumps back to the tanks are closed (testing lines).

The LPSI system supplies boric acid solution to RCS from the LPSI tanks. Following the level decrease in the tanks, the injection phase of operation is finished and the recirculation phase is started. The valves in the recirculation lines are open, and the valves in the suction lines from the tanks are closed. The LPSI pumps inject borated water to RCS from the confinement sump via the net construction and spray system heat exchangers which are cooled by service water.

2.2.1.3 Hydro-accumulators

The system is used for the emergency core cooling in case of LOCA. This is a passive system. Given the reactor pressure decrease below 6 MPa, the system automatically floods the core due to expansion of nitrogen. Given minimal level in HA, the float valve is closed to prevent escape of nitrogen into RCS. The system is actuated without initiation signal and electrical power supply. After depletion of own energy (given by pressure difference between hydro-accumulators and RCS), the system must be replaced by LPSI system. The system is in standby state during normal operation of the plant.

The system consists of two independent subsystems, each with two HA connected with the reactor vessel by piping lines (see Figs. 2.46 and 2.47). One subsystem supplies the boric acid solution above the core and second subsystem below the core. The boric acid solution is under the nitrogen pressure in the hydro-accumulators. Two safety valves are installed on each hydro-accumulators.

The pressure difference between RCS and HA is maintained by two check valves. One check valve is located at the reactor vessel to prevent leakage of coolant from RCS in case of piping break between the vessel and HA. The second check valve is located near to hydro-accumulator.

Between the check valves, the motor-operated valve is located. It prevents the draining of hydro-accumulator during planned plant outage connected with decreased RCS pressure under initiating pressure of hydro-accumulators. These valves are normally open during reactor operation but are closed manually by operator following the reactor shut down.

2.2.1.4 Emergency Residual Heat-removal System

The Bohunice V2 plant and the Mochovce plant (unit 1 and unit 2) are equipped with emergency residual heat-removal system. The system is used to:

- cool down the reactor core after seismic event,
- restore natural circulation in the RCS after loss of natural circulation,
- remove residual heat from the reactor core during shutdown state (refuelling outage).

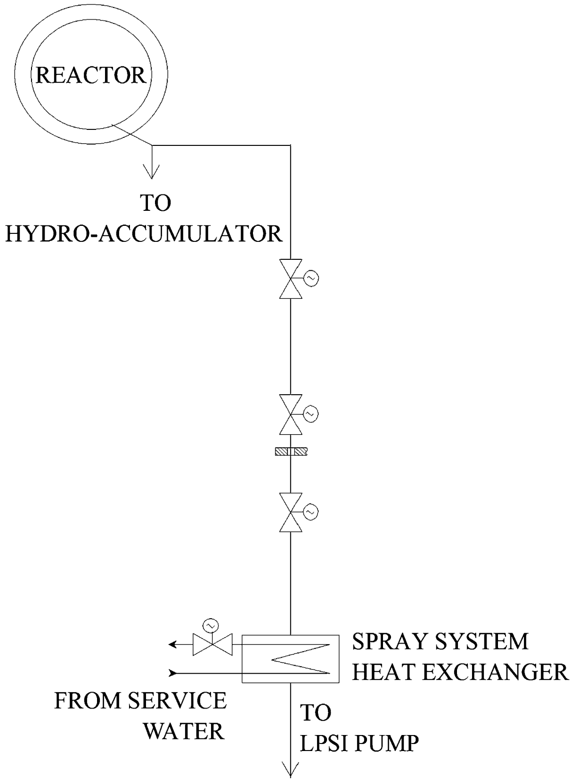


Fig. 2.49 Simplified schematic of emergency residual heat-removal system—redundancy 1

The system consists of three independent and identical subsystems. For accident mitigation, 1 out of 2 subsystems is required, i.e. the system is designed with 200 % of redundancy.

Before the system is activated, the closed circuit must be created using manual manipulations (without automatics). The closed circuit consists of LPSI pump, discharge line, reactor vessel, pipeline train, heat exchanger of the confinement spray system and suction train.

The circulation of H_3BO_3 is ensured by the LPSI pump. The flow can be regulated using the control valve. Given the start of LP pump, the circulation is started automatically through the permanent recirculation train. The measurement device of boric acid is installed in this train. This device is used for measuring of H_3BO_3 concentration behind the heat exchanger, and thus, the heat exchanger tightness check is performed. The permanent recirculation train is in operation always with the operation of LP pump.

The system can be activated if the RCS pressure is less than 0.5 MPa, and the water temperature at the output of the reactor is less than $\leq$130 °C. The schematic of subsystem 1 is shown in Fig. 2.49.

2.2.1.5 Confinement

The most important safety system, protecting confinement integrity and safety of NPP operation, is the bubbler tower. It is located in the accident localization cavity. In cooperation with the spray system, it is used to depressurize the confinement in case of LOCA. This is a passive system independent on power sources (Fig. 2.50).

The bubbler tower is connected by a corridor with other compartments of the confinement. The steam-air mixture is transferred under twelve levels of the bubbler tower.

The bubbler condenser trays are filled up with borated water with concentration of 12 g/kg. In addition, borated water in the bubbler tower contains hydrazine hydrate at concentration of 100 mg/l for retaining of iodine during accident.

Space behind the bubbler tower tanks is connected with gas traps (air receiver) via double-check valves (DN 500 mm). There are four gas traps (air trap). Each gas trap is connected to three levels (trays) of the bubbler tower.

The total water capacity is approximately 1,400 m^3. The water temperature is between 40 and 60 °C. If a high-energy pipe break occurs, the confinement pressure and temperature increase. If a LOCA occurs, the overheated coolant from the primary circuit rapidly evaporates. The steam release mixes with air, and it heats the confinement atmosphere and the pressure increases. Overpressure pushes the air from SG box into adjacent compartments of confinement. The steam-air mixture proceeds via corridor into the accident localization cavity.

The steam-air mixture enters under each level of bubbler tower. The steam is condensing and the non-condensable gases proceed through the check valves DN 500 into the gas traps.

In case of large LOCA, the pressure in the confinement exceeds 200 kPa and the check valves are disabled. The borated water is pushed out from the bubbler tower. The water flows down the structure forming the ceiling of lower tank and collect in containers on the front wall of each level of the bubbler condenser tanks. Then, the water is sprayed via front-perforated surface into accident localization cavity. It forms passive sprinklers which decrease pressure inside the confinement.

The confinement underpressure arises during severe accident if the confinement spray system is in operation for long time. The designed basis internal underpressure is the absolute value of 78 kPa. The underpressure increasing over the design basis value can lead to failure of confinement structure. Vacuum breaker was installed (as system for severe accident management) to prevent this failure.

Vacuum breaker of the confinement should be initiated to achieve the design criteria for confinement underpressure. If one out of three redundant systems of confinement spray system fails to trip and continues in operation (the system continues in confinement spraying), the vacuum breaker shall prevent the failure of confinement liners. The vacuum breaker of confinement is part of the bubbler tower. The vacuum breaker releases the gasses from collecting gas traps to the confinement using the piping line DN 200 equipped with MOVs and check valves, and the confinement pressure is increased.

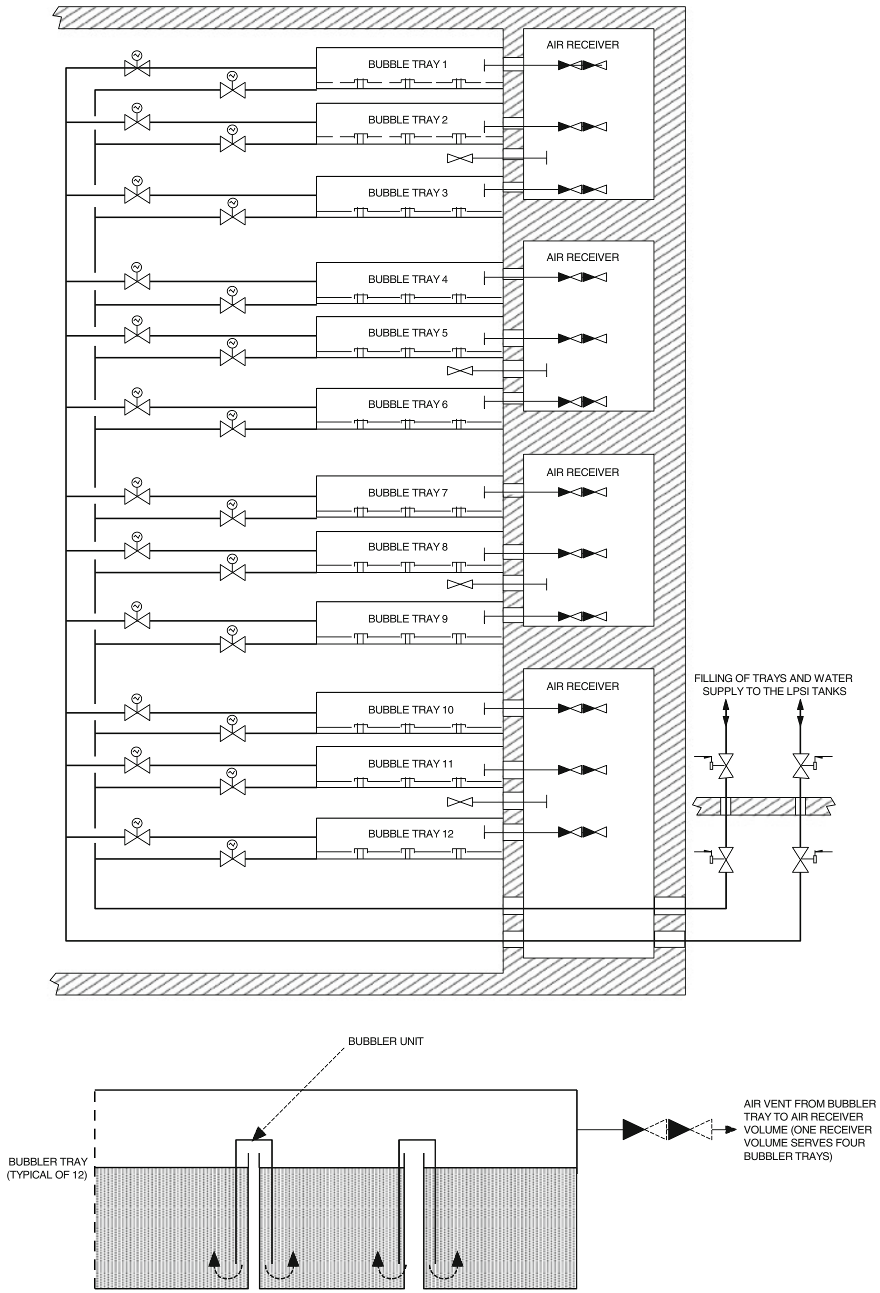


Fig. 2.50 Bubble tower

In addition, the water of the bubble tower can be used in case of not isolated interfacing LOCA or SG tube rupture outside confinement to compensate losses. The water can be supplied to the LPSI tanks or to the confinement floor. Then, the HPSI or LPSI pumps compensate losses of the primary circuit. The water train to the LPSI tank must be prepared and a pump used to supply water. The drainage to the confinement floor is accomplished by the opening of the filling lines for several levels of water trays at the same time, which leads to cascading water drainage and overflow of water in the lowest floor and then the overflowing of the confinement.

2.2.2 Safety Upgrading of the WWER440/V213 Reactors

At the present time, many WWER440/V213 plants are being refurbished with the reactor units fully operated. The refurbishment is aimed at increasing the plant's nuclear safety. Computerized reactor protection systems and ESFAS are installed, seismic upgrading of SSCs are being performed, fire safety is being enhanced, quality and safety culture are increased. In addition, the activities are focused on improvement of operational effectiveness in the form of power uprate and plant lifetime extension.

Symptom-based EOPs are implemented in the plants for all operating modes.

Currently, severe accident management systems and guidelines are being implemented in the WWER440/V213 plants [8–11].

2.2.2.1 Severe Accident Management Systems

Severe accidents occur in WWER440/V213 reactors if sufficient water is lost from the primary circuit for the core to be uncovered and overheated such that rapid fuel pin cladding oxidation occurs. Severe accident management is the composite of actions which would be taken to recover from the accident state and to prevent or mitigate the release of fission products to the environment. In the context of severe accidents, the recovery actions mean to quench the overheated core material and establish a safe stable state with a heat transport path to remove the heat generated by the debris. It should be noted that actions to accomplish the recovery process will also, in general, act to mitigate the release of fission products from the fuel, the primary circuit or the confinement if the confinement pressure boundary is breached or bypassed. Water addition to cool the core material and to ensure a heat sink is the primary means of recovering from a severe accident state.

The following systems are being implemented in the WWER440/V213 plants for the purpose of severe accident management:

- Primary circuit depressurization,
- Emergency water source for water injection into RCS, spent fuel storage pool and spraying of confinement,
- Emergency power supply,

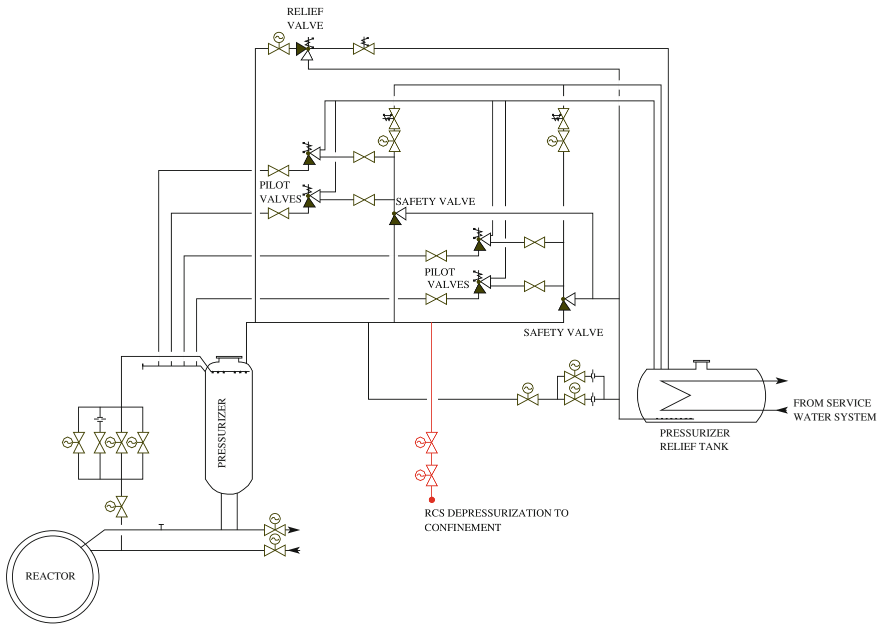


Fig. 2.51 Additional valves installed for depressurization of primary circuit

- Flooding of reactor cavity for external cooling of RPV,
- Vacuum breaker of confinement,
- Hydrogen management in confinement,
- Technical support centre.

Primary Circuit Depressurization

The system is used to depressurize the primary circuit in severe accident conditions. The purpose is to prevent discharge of debris from RPV at high-pressure and to allow water injection from the low-pressure safety injection systems. The RPV is depressurized to the pressure less than 2 MPa [12]. A new piping path is added to the node of pressurizer with MOVs (Fig. 2.51).

The depressurization can have positive and negative impacts on accident progression. The important positive impacts are presented below:

- Depressurization increases opportunity for injecting water into the primary circuit from low-pressure systems and accumulators where appropriate. Depending on the plant-specific configuration and extent of the depressurization, this could initiate injection from low-pressure ECCS, external water sources, fire water systems, etc.
- Depressurization increases steam flow through the overheated core region. At high depressurization rates, the increased steam flow through the core would likely cool the core and decrease the fuel rod temperatures. It is noted that

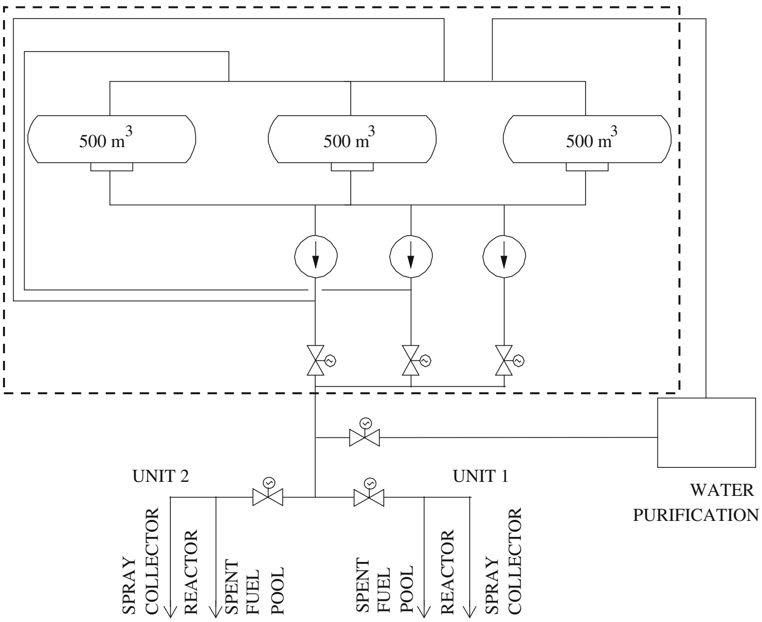


Fig. 2.52 External water sources

depressurization without injection would only result in a temporary cooling of the reactor core.

- Depressurization reduces the stress on the steam generator tubes which would help in protecting this important containment boundary against failure. This is also true for other parts of the reactor coolant system, such as the reactor vessel.

Negative impact of depressurization is that it increases the rate at which hydrogen could be discharged into confinement since it reduces the extent to which hydrogen could be retained in the primary circuit. Depressurization would increase the flow of fission products into confinement. This includes confinement regions where the fission products could be scrubbed.

Emergency Water Source

The emergency water source is used to ensure: (1) the subcriticality of molten core debris, (2) the integrity of confinement through sprays, (3) the cooling of molten core by embedding in phase of arresting core in RPV and (4) cooling of spent fuel pool by injection of coolant [12]. The coolant inventory is placed in external tanks. The tanks are connected with suction pump collector. Deliveries of 3 pumps are connected into a common path which is supplying coolant for both units (see Fig. 2.52).

Another phenomenological uncertainty is the potential for the core to return to a critical state. Since the control material could be relocated as a result of core damage, the addition of inadequately borated water into the core would increase the neutron moderation and could potentially result in a critical configuration. If borated water, typical of that considered in design basis analyses, is added to the core, there would be no potential for returning to critical state. If the core were to become critical, the power could be limited by the configuration itself or by the rate of water addition. In the former case, the fuel pin configuration is the most reactive state for the fuel. If the configuration is altered by the collapse or shattering of fuel pellets, the coolant fraction would decrease from about 0.6–0.5 or less. This reduces the moderator fraction and increases the flow resistance through the core debris, likely resulting in coolant boiling within the debris bed and a further reduction in the moderator fraction. Hence, compaction of the fuel makes the fuel less reactive. For the latter case, as long as the rate of reactivity addition is slow (the response times for the major feedback mechanisms are much faster than the rate of power rise in the fuel), the power level would be bounded by that value which could completely vaporize all the injection flow rates.

A third phenomenological uncertainty is the rate of recovery of a badly damaged core when water is added. Since the fuel configuration would have been greatly changed and compacted, the rate of cooling is much less than that which would be anticipated for fuel pin configurations. In fact, the experience in the TMI-2 event demonstrated that completely covering the core at about 200 min into the accident did not prevent a significant fraction of the fuel from remaining molten and draining into the lower plenum approximately 30 min later. Hence, there is a significant uncertainty with respect to how much water could be used by the core to quench the material, i.e. the core may be submerged but the coolability of the core is dependent on the specific configuration (porosity) of the debris. Therefore, high flow rates may fill the reactor vessel faster without increasing the core cooling. Once condition BD has occurred and has been reflooded, the extent and rate of hydrogen generation can only be estimated. In particular, the porosity of the relocated material would not be known; hence, the cooling rate and the accessibility of steam to the debris would also be unknown. However, it can be concluded that these two are related in the following way. An open debris bed would quench rapidly and would likely result in some additional hydrogen created at a relatively large rate. Conversely, a debris bed with very low porosity may cool very slowly, or not at all, with a significant amount of hydrogen created at a slow rate. Clearly, if the exothermic oxidation reaction proceeds too rapidly, the debris will melt, close the porosity and likely stop both the cooling and the hydrogen production. If local cooling ceases, the debris would heat up, melt and likely relocate. This change in geometry could both increase the cooling of the debris and increase the potential for hydrogen generation during the cooling-quenching process. The experience at TMI-2 provides some guidance with respect to this uncertainty. More detailed information is provided in [13].

Fig. 2.53 Mobile DG set

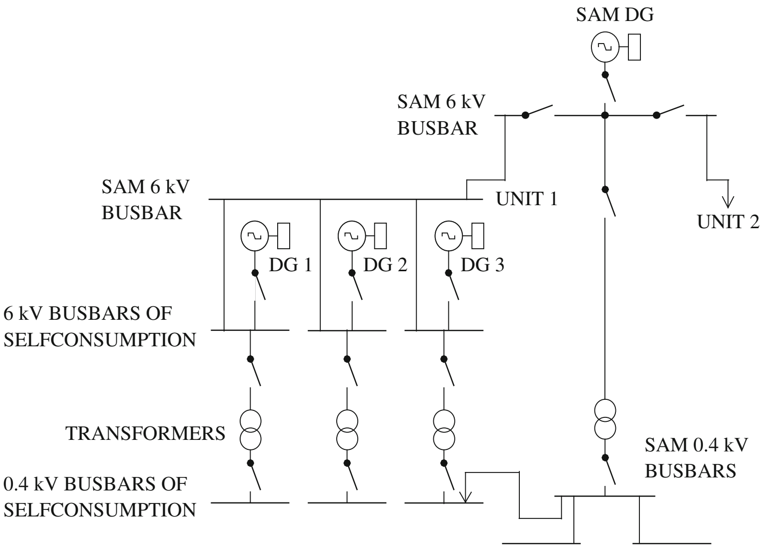


Fig. 2.54 Connection of SAM mobile DG set to the schematic of self-consumption

The Emergency Power Supply

The emergency power supply for severe accident conditions is a new DG SAM (diesel generator 0QG) with nominal voltage of 6 kV [12]. It is a container-type DG which can be connected with self-consumption of the plant (see Figs. 2.53 and 2.54).

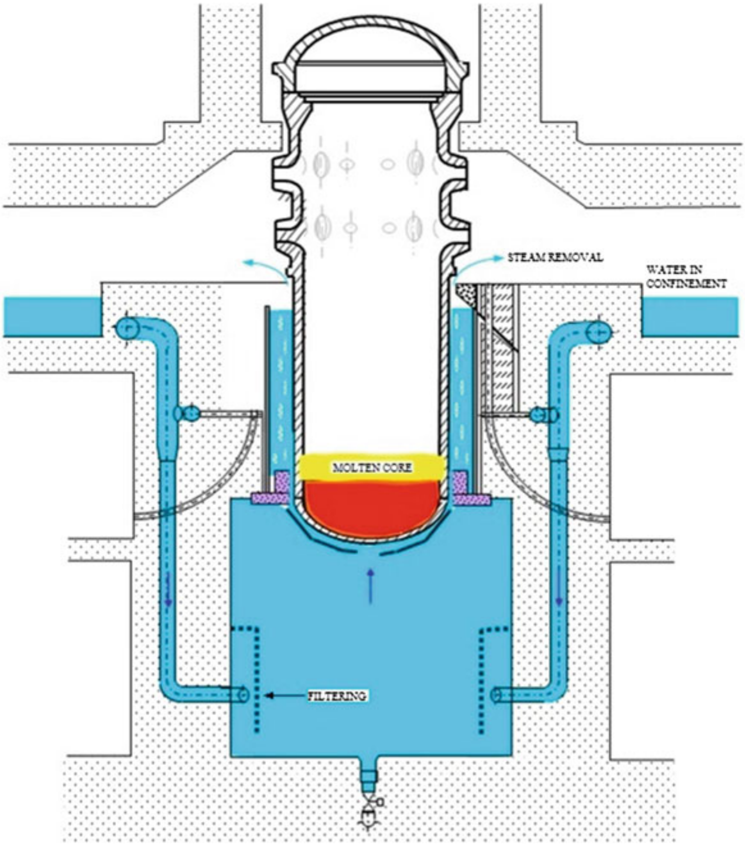


Fig. 2.55 External cooling of the reactor vessel

External Cooling of the Reactor Vessel

Flooding of the reactor cavity is used as external heat removal of RPV (Fig. 2.55). It is a system for stabilization and localization of corium in RPV, and it solves two basic tasks: (1) preventing the non-returnable loss of coolant from the confinement to the ventilation centre (removing deficiency of the original design of WWER440/V213 for LOCA events over reactors postament) and (2) reliable and long-term flooding of reactor cavity providing successful strategy for implementation of corium stabilization inside RPV [14]. Given flooding of the reactor cavity, recirculation of coolant and heat removal from RPV is possible.

If the accident progressed to the state with slumping of debris into the RPV lower plenum, then external cooling could have three major functions:

- Nucleate boiling on the entire outer surface of the RPV lower head would mitigate the temperatures developed in the steel. This would assure that the vessel would not fail as a result of creep rupture conditions.
- The heat removal through the lower head, should the RPV lower plenum subsequently dry out, would be a significant fraction of the decay heat generated in the debris, i.e. 25 % of the energy developed in the debris.
- Thirdly, if the cylindrical part of the vessel was submerged, all the decay heat could be removed if coolant injection to the primary circuit is not available. Safety analyses show that this could prevent RPV failure [13]. Hence, a safe stable state would be established.

Vacuum Breaker

Vacuum breaker ensures removing of deep underpressure from the confinement during operation of spray system. The liner of containment wall can be potentially damaged and can lose the tightness capability under accident conditions. This situation can happen when the containment spray pumps fail to stop under containment low-pressure conditions. The containment breaker is controlled manually by operator or automatically. The vacuum-breaking process is provided by releasing the gases from the bubbler tower into the confinement. This process causes pressurizing the confinement [12]. Four release piping lines are installed to air receiver of the bubble tower (Fig. 2.56).

Hydrogen Management

The hydrogen management prevents hydrogen explosion and subsequent failure of confinement. The system for hydrogen management is based on determination of hydrogen source term. This system ensures controlled removal of hydrogen during the representative accident sequences and prevents possibility of unacceptable rapid combustion forms of hydrogen in confinement (acceleration of flame, transit from deflagration to detonation and detonation of hydrogen). The system of hydrogen management is based on passive hydrogen recombiners. Given installed sufficient capacity in the plant, the control of hydrogen source term is possible. The advantage of this approach is continuous control of hydrogen removal without unwanted associated phenomena (e.g. high temperature as consequence of using the hydrogen igniters during combustion), which can endanger, for example, some of sensitive equipment in the confinement (e.g. measures necessary to use the SAM guidelines). Disadvantage of the proposed approach is the need of transient inerting confinement atmosphere using steam at some sequences with fast generation of hydrogen (shutdown of the spray system) (Fig. 2.57) [12].

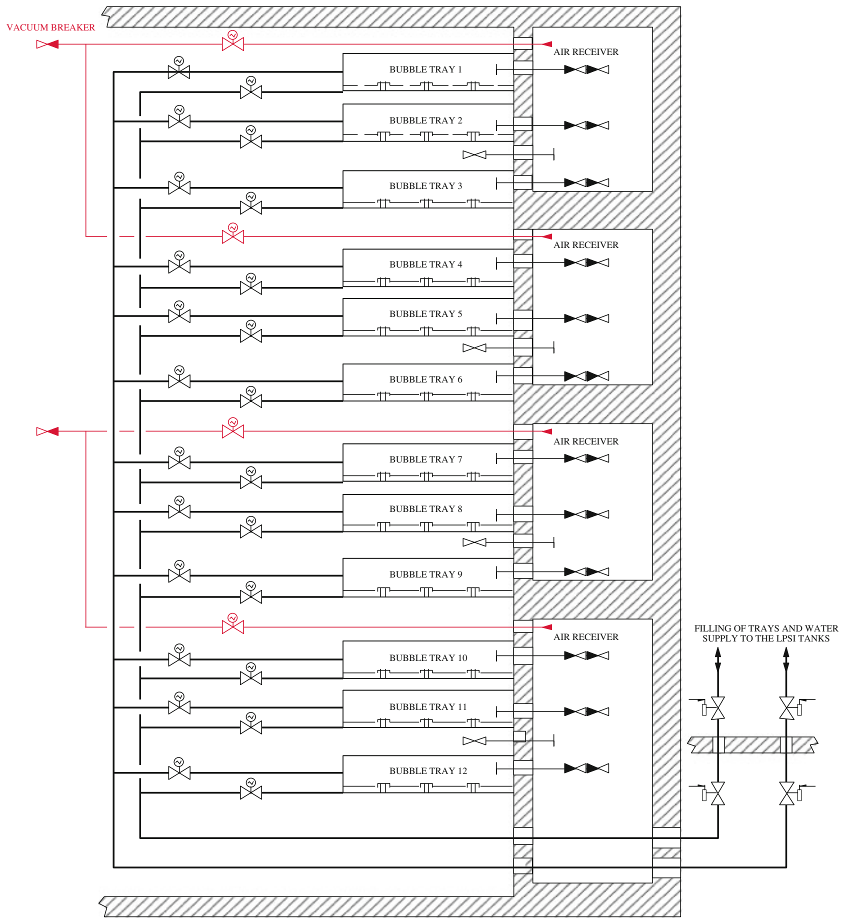


Fig. 2.56 Vacuum breaker installed on the bubble tower

Technical Support Centre

Technical support centre is installed in the plant with required information about the parameters needed for severe accident management.

Loviisa SAM Systems

The Loviisa plant is equipped with Western-type containment. The safety measures implemented within the severe accident management project of the plant are shown in Fig. 2.58 [15].

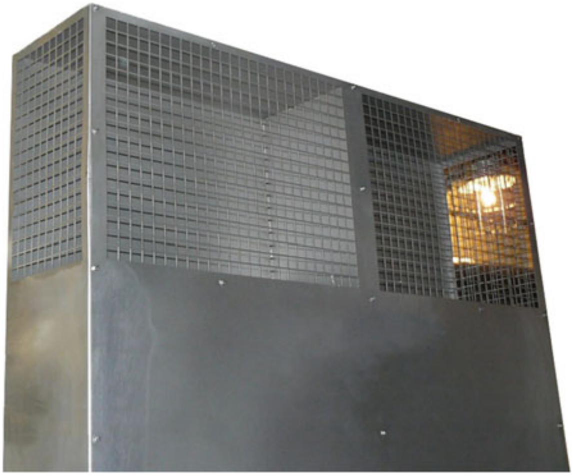


Fig. 2.57 Recombiner in the confinement

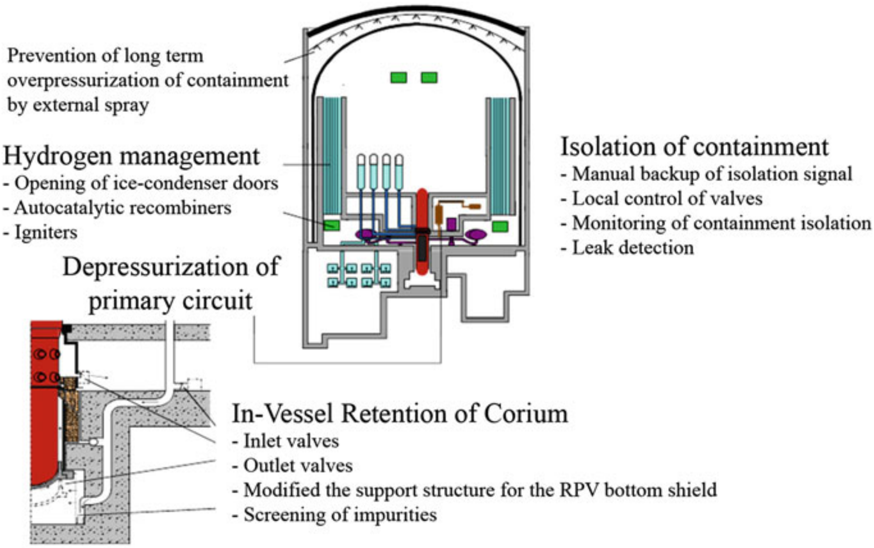


Fig. 2.58 Severe accident management in the Loviisa plant

Sever Accident Management Guidelines

There are currently three levels of guidance for the operating staff of a WWER440/V213 plant. The first level, termed operating procedures, focuses on plant operation during the time that plant parameters are within an acceptable range. The second level, termed abnormal operating procedures, focuses on restoring the function of systems which could impact overall plant operating margins. The third

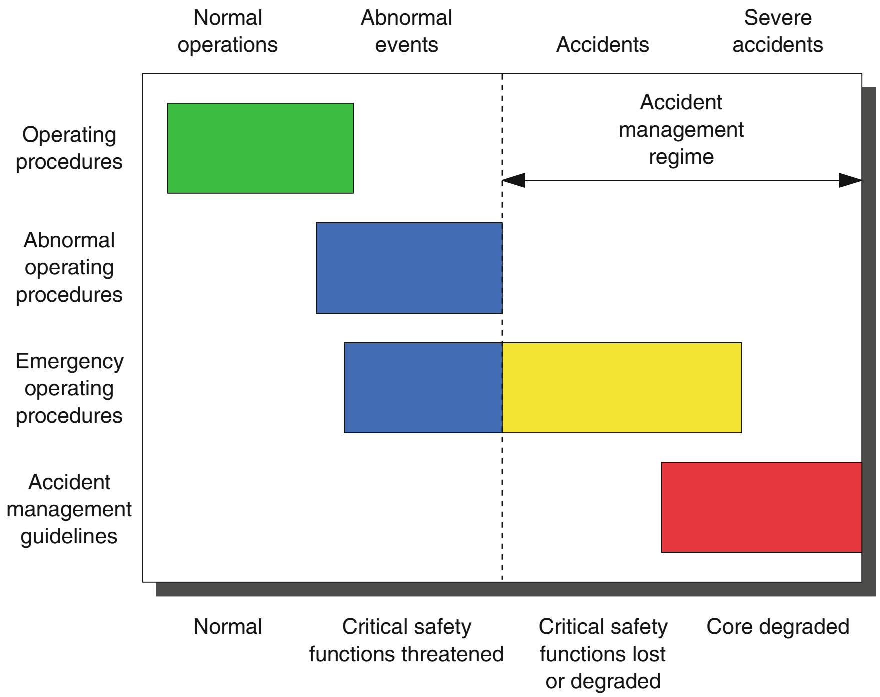


Fig. 2.59 Overview of procedures and SAMG of WWER440/V213 reactors

level, termed EOPs, is aimed at bringing the plant to a safe, stable state following a reactor trip signal. These procedures represent the initial phase of utility accident management and have been formulated around the essential safety functions such as reactivity control, adequate core cooling and removal of the heat generated in the reactor core [12].

For conditions leading to a severe accident state, most, or all, of the systems considered in the EOPs would be lost for a sufficient time to uncover the core and to result in the overheating of the fuel and cladding to temperatures causing extensive cladding oxidation. However, the functions to be accomplished are the same, but perhaps more focused due to the accident state.

Severe Accident Management Guidelines (SAMG) provide support for plant personnel to manage accident after core damage independently on the type of initiating event, which causes core damage.

As well as EOPs, also SAMGs are based on symptom-oriented approach. All actions taken by personnel are initiated according to the symptoms (directly measurable parameters which are representing the state of the plant), e.g. pressure, level, temperature, flow. Previous identification of the plant state is not required, when decision is taken about a specific step. Only knowledge of symptoms is required.

Overview of the use of operational and emergency procedures and SAMG depending on the state of the plant is shown in Fig. 2.59.

SAMGs have the following characteristics:

- Confinement protection and minimizing the release of radioactive substances into the environment.
- Interactions of operators are performed in accordance with the key parameters of the plant, respectively, with symptoms of ongoing accident. The required recovery actions are identified to maintain the key parameters within the limits ensuring stable conditions of the plant.
- During interactions in accordance with SAMGs, all applicable indications are taken into account regardless of uncertainty in measurement.
- When state of the plant cannot be identified from measurement (e.g. flammability of confinement atmosphere and potential possibility of secondary criticality), supporting calculations shall be performed.
- Identification of initiating event is not necessary for selection of severe accident suppression or mitigation strategy. Interactions of operator (unless special actions are defined) are appropriate regardless of initiating event or accident sequence.
- Specific actions are based on existing configuration of the plant systems. After changes of systems for severe accident management, the guidelines have to be updated in a way to reflect the actual state of the plant.
- All systems applicable in severe accident management, regardless of safety qualification, have to be used for mitigation of impacts on the inhabitant health and safety also in case of low probability of success. Integration of systems into SAMGs does not require changes over the original plant design.
- Limitations and interactions of operators are based on realistic best estimate technical calculations and not on traditional licensing or design basis analyses and assumptions. Uncertainties are also evaluated in the best estimate mode.
- Uncertainties in the knowledge of accident progression are large. Some aspects of severe accidents are still unknown, the possible modelling ways are limited, and for certain phenomena, the deterministic analysis results are even questioned.
- Interactions according to the guidelines have positive but also potential negative consequences. There are time periods for some actions when potential negative consequences of interactions are expected.
- Due to phenomenological uncertainties of severe accidents, it is not possible to describe human interactions exactly as it is done in EOPs. Interactions are based on specific symptoms in certain time, and it is necessary to take into account the possible recovery actions in the decision-making with objective to protect the plant personnel and inhabitants in vicinity of the plant.
- Interactions are coordinated from technical support centre (TSC). Team of specialists decides about the interactions to be performed. This process is based on common regulations and strategies mentioned in guidelines, and it is based on the analysis of situation (performing of interactions is knowledge-based).
- Interactions are performed mainly by the personnel of control room according to instructions from TSC.

- Possibility of interactions is limited by specific radiological situation.
- Decisions about serious actions with possible release into the environment have to be coordinated and accepted by accident response organization.

The structure of SAMGs is described below:

Severe accident control room guidelines (SACRG):

SACRG-1: Initial response
SACRG-2: Transients after TSC is functional
SACRG-3: Open confinement, initial response
SACRG-4: Transients after TSC is functional, open confinement

Severe accident guidelines (SAG):

SAG-1: Depressurize RCS
SAG-2: Inject into containment
SAG-3: Inject into RCS
SAG-4: Reduce fission product releases
SAG-5: Inject into the SGs
SAG-6: Control containment conditions
SAG-7: Reduce containment hydrogen
SAG-8: Spent fuel pool guideline (independent guideline)
Diagnostic flow diagram

Severe challenge guidelines (SCG):

SCG-1: Mitigate fission product releases
SCG-2: Reduce containment pressure
SCG-3: Control hydrogen flammability
SCG-4: Control containment vacuum
Severe challenge status tree

Severe accident exit guideline (SAEG):

SAEG-1: TSC long-term monitoring activities
SAEG-2: SAMG termination

References

1. Bohunice V1 nuclear power plant PSA, Electrowatt Engineering Services (UK), Main report and appendices, Supported by Commission of the European Community, Horsham,West Sussex, Feb 1994
2. ECCS and safety related Systems PSA of the Bohunice V1 plant. Basic engineering of the plant reconstruction. PHARE project NS02/91, prepared by AEA Technology plc (UK), Warrington, April 1997
3. Reactor operation in the Novovoronezh NPP, Czechoslovakian Nuclear Regulatory Authority, UISJP, Prague (1974)

4. Definition of baseline WWER440/V213 systems and evaluation of plant response to postulated accident. SAIC report, San Diego, California, Dec 1987
5. Design basis and design features of WWER-440 model 213 nuclear power plants. IAEA-TECDOC-742, Vienna (1994)
6. Kovacs, Z. et al.: Probabilistic safety assessment of the Jaslovske Bohunice V1 nuclear power plant following the gradual reconstruction of the plant. Journal article in safety of nuclear power—Bezpecnost jadrovej energie, Prague, 7/8, 2001 (in Slovak)
7. Kovacs, Z. et al.: PSA based upgrading of older European plants with WWER440 type reactors. In; International Conference: Probabilistic Safety Assessment in the Nuclear Industry, Café Royal, London, 26–27 Nov 2001
8. Kovacs, Z. et al.: Results of stress tests of European nuclear power plants after the Fukushima-Daiichi accident. Journal article in safety of nuclear power—Bezpecnost jadrovej energie, Prague, 9/10, 2012 (in Slovak)
9. Kovacs, Z. et al.: Reliability centred maintenance of nuclear power plant facilities. Journal article in safety of nuclear power—Bezpecnost jadrovej energie, Prague, 1/2, 2011 (in Slovak)
10. Kovacs, Z. et al.: Towards new nuclear safety. Journal article in safety of nuclear power—Bezpecnost jadrovej energie, Prague, 3/4, 2013 (in Slovak)
11. Kovacs, Z. et al.: Safety of nuclear power plants in the 21st century. Journal article in safety of nuclear power—Bezpecnost jadrovej energie, Prague, 1/2, 2012 (in Slovak)
12. Bugar, I., Nano, J.: Severe accident management projects of Slovenske elektrarne. WANO MC Seminar SAM, Kiev 26–28 Mar 2013
13. Severe accident management guidance technical basis report, vol. 1, 2. EPRI report no. 1025295 (2012)
14. Bognar, B.: Development of EOP, SDEOP, SAMG and consideration of multi unit accident aspects in Paks NPP. WANO MC Seminar SAM, Kiev 26–28 Mar 2013
15. Raunio, T., Paajanen, A.: Severe accident management guidelines during outage at Loviisa NPP. WANO MC Seminar SAM, Kiev, 26–28 Mar 2013

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