3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

This Section covers a number of subjects which can be classified under the ‘design’ specification. It starts by a brief overview of the events which where used to define the range of loads imposed on structural components and equipment. The material properties of the metal and concrete used in construction of the structures and components of the plant are summarized. Criteria for choosing equipment characteristics are presented. The Section is concluded by describing the testing of equipment and the testing program completed during initial startup and initial power operation.

Only a brief description of some of the events which result in the imposition of loads (e.g. DB and SB-LOCA events) is given in this Section. A more complete description is provided in the Subsections that analyze the thermal-hydraulic response of the plant and the design and performance of the safety related equipment.

3.1 DESIGN LOADING

The structural components of the plant are designed in accordance with the specification set forth in "Design Safety Regulations of Nuclear Power Plants (OPB-88)" [16]. The generic requirement of this document is that safety-related systems and elements of nuclear power plants have to be able to fulfill their functions under all conditions. This implies that they have to accommodate stresses imposed by natural phenomena as well as mechanical, thermal, chemical and other impacts which may arise during design basis accidents.

3.1.1 External

The term "external events" (relative to a nuclear power plant) covers such natural phenomena as a earthquakes, flooding, strong winds, lightning, snow and ice, and such man-made events as aircraft crashes, industrial explosion, sabotage and terrorist action. On site fire and flooding are usually also considered as external events.

3.1.1.1 Air-Shock Wave

Buildings and other constructions unit 1 and unit 2 of the Ignalina NPP were designed and built without taking into account the influence of the air-shock wave, because corresponding requirements were put into operation only after the beginning of year 1987 [17].

Before construction started at the Ignalina NPP, the Research and Development Institute for Energy Technology, St. Petersburg (at that time Leningrad), Russia, performed a number of studies on the response of structures subjected to a step air shock wave of 10 kPa with the duration of up to 1 s, as required in the code [17].

The structural components under investigation were similar for all three units of the Ignalina NPP. The calculations show, that such a wave would destroy the following structures:

- the pre-fabricated ferro-concrete structures, the metallic framework of external and internal walls in the central hall of the equipment room of the main building (Fig.1.2, building A),
- the building structures of the turbine hall (Fig.1.2, building G) and the deaerator room (building D) of the main building,
- the building structures of the ECCS pressurized tanks (accumulators) above the zero elevation (Fig. 1.2, building B),
- the building structures of the service-water pump station (Fig. 1.1, buildings 1 and 2),
- the building structures of the redundant diesel power station (Fig. 1.1, building 40).

The studies showed that in this event the loading on equipment due to oscillations of the structures was below the load due to, for example, an earthquake of magnitude 6 on the scale MSK-64 (see Subsection 3.1.1.4).

3.1.1.2 Water

According to reference [18] an analysis of the "Rupture of the dam (weir) of the cooling water pond" was not necessary, because the source of cooling of the Ignalina NPP is the water from lake **Drūkšiai**. In order to justify this, changes of the water level in lake **Drūkšiai** must be considered. According to available records a maximum water level was 142.26 m in 1953, and the minimum -140.76 m (above see level) in 1964 [10].

In normal operation of the Ignalina NPP and for supporting calculation of water level in lake **Drūkšiai**, there is the addition of spring floods and rain-water, as well as the effects of the hydro-engineering complex on the river Prorva. In case of destruction of this structure and the dam of the hydroelectric power plant, the water level in lake **Drūkšiai** would drop to the level of river Prorva, which is 140.20 m.
In case of destruction of the earthen dam on the river **Drūkšė** and return of the flow to the old river-bed, the influx of water into lake **Drūkšiai** would decrease by about 20 %. Temporary elimination of the dam on the river **Drūkšė** would not lead to a sharp lowering of the water level in lake **Drūkšiai**.

3.1.1.3 Missiles

Safety analysis of nuclear power plants, according to reference [18], requires the consideration of an "Aircraft crash on the reactor hall". The consideration of this event is proposed in a list of hypothetical accidents defined in 1990 by the Kurchatov Atomic Energy Institute, Moscow, Russia. This requirement was imposed after completion of the Ignalina NPP.

The requirements for considering an aircraft crash therefore were not imposed on any RBMK plant. This was due mainly to three major considerations:

A) There were no such regulatory requirements at the time when the plants were developed and no such requirements were introduced by the regulatory bodies,
B) Such events are sufficiently unlikely, besides, the RBMK sites are situated reasonably far from airports,
C) Until recently, there were no reliable statistics on flight incidents and fatal accidents involving both civil and military aircraft, which could be used when considering such events regarding nuclear power plants.

Note, that the nearest air route Svir-Rokiškis is ten km to the West of the Ignalina NPP. In 1990 a total number of flights along the Lithuanian air routes was 65000. During the last 30-year period there were no commercial aircraft crash accidents in Lithuania

3.1.1.4 Seismic

Seismic stability is the ability of equipment and structures to maintain integrity during seismic loading. This implies the maintenance of strength, tightness, maintainability, nuclear and radiological safety and the absence of residual deformation, which encumber normal operation [19].

The standard of seismic stability in the former USSR was the MSK-64 scale, which was established by the Earth Physic Institute, Moscow, Russia. Fig 3.1 provides a comparison between the MSK-64 scale and two other scales in common use. According to the MSK-64 scale, forces correspond to the following acceleration ranges of soil for periods from 0.1 to 0.5 s [19]:

forces 5 - 0.12-0.25 m/s^2,
forces 6 - 0.25-0.50 m/s^2,
forces 7 - 0.50-1.0 m/s^2 and so on.

For Soviet-designed NPPs two levels of seismic impact were taken into account: (a) the design earthquake, and (b) the maximum possible calculated earthquake. The first is a maximum earthquake, which may happen during the life-time operation of the NPP. The second is the maximum possible earthquake in the area in question. As a rule, for design purposes the maximum calculated earthquake employs an MSK-64 scale one force higher.

During construction it is necessary to guarantee the seismic stability not only for the building being constructed, but also for equipment, reactor control and protection system, control and measuring devices and others.

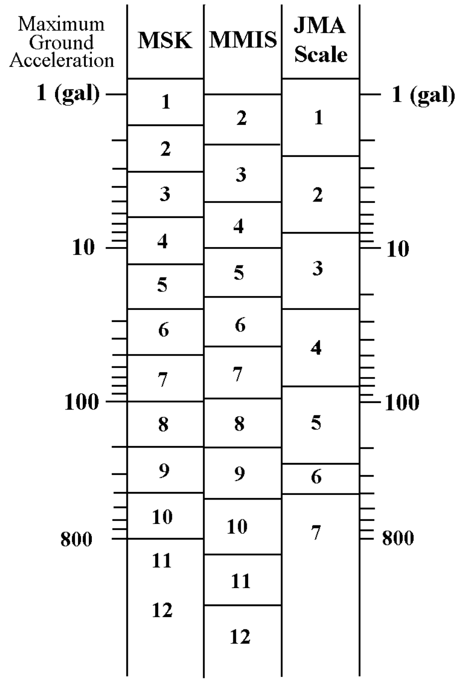

Fig. 3.1 The relation between seismic scales
MSK - Modvedev, Spanheuer, Karnik
MMIS - Modified Mercalli Intensity Scale
JMA - Japan Metrological Agency

Seismic Stability Categorization for the NPPs with RBMK - type Reactors

Depending on the need for functionability during and after the earthquake, all systems, equipment and structures of NPPs with RBMK-type reactors, are designed according the "Code for Designing of seismic-resistant nuclear power plants" [20]. The structures are divided into three seismic stability categories.

The first category is in turn divided into three groups. The first group encompasses systems or elements for which damage and loss of integrity could lead to releases of radioactive products in such amounts that they would result in radiation exposure to inhabitants above the established codes for a maximum design basis accident. The following systems of the NPP belong to this group:

- reactor,
- main circulation circuit,
- steam pipelines from the separator drums to the relief valves, and the valves themselves,
- feedwater supply system,
- MCP hydrostatic sealing feeding system,
- CPS cooling circuit,
- refueling system equipment, refueling machine, cranes of the central hall,
- equipment of spent fuel system, cranes in the service hall of the fuel cooling pond,
- radioactive safety control system,

- CPS equipment for combustion of explosive mixture.

The second group of the first seismic stability category includes the safety systems which protect the reactor core, the emergency heat removal systems from the reactor, and also the confinement system for radioactive products. The following safety systems belongs to this group:

- control and protection system,
- emergency core cooling system,
- MCC overpressure protection system, except steam discharge valves to condensers (SDV-C),
- reactor-space overpressure protection system (part of the ACS, which removes the steam from the reactor space to the fifth condensing pool),
- reinforced leaktight compartment system, including headings, hatches and doors,
- ACS towers,
- ACS heat exchanger and pumps,
- cutoff and sealing devices,
- system for reception of waste water from reinforced compartments,
- emergency power supply system,
- service water system,
- intermediate circuit system,
- ECCS makeup system,
- venting system of the safety system compartments and cable service,
- tanks of the CPS venting system,
- control safety systems, including redundant control systems.

Third group of the first seismic stability category includes those buildings, structures and equipment for which damage would lead to failure of reactor operations. This group consists of:

- control room structures (building A) and deaerators (building D),
- ECCS pressurized tanks (accumulator) building,
- redundant diesel-generators building,
- service water pump building,
- purified deminiralized water tank (V=1500 m^3),
- communication tunnels and support structures of the first seismic stability category.

The second seismic stability category encompasses those systems, equipment and structures (not included in the first category), for which failure can lead to radiation levels above the permissible annual level for normal operations. This category is divided into two groups.

The first group of the second seismic stability category includes systems, equipment and constructions, which are located inside of the accident confinement zone:

- heat exchanger and pumps of the purification and cooling system,
- bypass cleaning system,
- maintenance cool-down system of reactor,
- pipelines located inside of the ACS.

The second group of the second seismic stability category includes:

- ACS elements not mentioned above,
- control systems of devices, which separate structures of the second seismic stability category from structures of third category,
- turbine hall (building G),
- supporting structures of the second seismic stability category.

Systems, equipment and structures, which are not included in the first and second categories, belong to the third seismic stability category.

Seismic Stability of Structures, Equipment and Pipelines

Calculations of seismic stability criteria for the Ignalina NPP structures, equipment and pipelines were conducted by the Research and Development Institute for Energy Technology, St. Petersburg (at that time Leningrad), Russia. These calculations were performed using a linear spectral theory of seismic stability. Calculation results of the main and auxiliary facilities are shown in Table 3.1. The seismic stability of buildings is given according to the above mentioned MSK-64 scale.

For the Ignalina NPP area the design earthquake magnitude is 6 forces and the maximum possible calculated earthquake magnitude is force 7 according to the MSK-64 scale. This requirement implies, that some structures of the Ignalina NPP need to be strengthened. According to the above listed estimates, the following structures are subject to alterations:

A) Main building. The extent of reconstruction amounts to about 35-40 % of the entire volume of the unit. The deaerator building and reactor hall need to be replaced completely.
B) The ECCS pressurized tank (accumulator) building. Degree of reconstruction amounts to about 50 %.
C) Building of service-water pumps should be replaced completely.

Table 3.1 Seismic stability of the Ignalina NPP structures

Building	Seismic stability, force
Reactor building (A)	6
Deaerator building (D) and Turbine hall (G)	< 5
Building of the ECCS accumulators	5
Redundant diesel-generator building	7
Building of service-water pumps:	
- part below ground	7
- building frame	6
Primary system grade water tank	7
Trenches and tunnels for cables	7

A large part of the equipment within the first and second units of the Ignalina NPP do not comply with the seismic requirements. Calculated results of pipeline seismic stability show, that they correspond to standards [21].

Thus systems, equipment and structures of the Ignalina NPP do not fully comply with seismic stability standards. Measures aimed at reinforcing the existing building and equipment components are expensive, and are considered by plant experts as unfeasible. However, aftereffects of an earthquake will be diminished if the reactors are promptly shut down prior to the seismic wave approaching the plant. The implementation of the seismic monitoring system is under progress in accordance with Safety Improvement Program of Ignalina NPP [22].

3.1.2 Internal

The "internal" events which can impose loads on structures and equipment are events anticipated or postulated to occur as a result of plant failures, i. e. malfunction of the reactor's normal operating and control system.

3.1.2.1 Postulated Piping Ruptures

The safety analysis of the NPP with RBMK - type reactors considers the following ruptures of pipes and system components:

- fuel channel,
- water communication line,
- steam-water communication line,
- group distribution header,
- downcomer or the cofferdam of the separator drum,
- main steam line before the main steam gate valve,
- feedwater pipe,
- pipe or header of the MCP,
- service water pipeline,
- purification and cooling system pipeline,
- intermediate circuit pipeline.

This Subsection discusses the initiating event itself, symptoms of the accident, direct consequences and measures to eliminate or limit the anticipated consequences. A quantitative analysis of the most significant accidental transients are presented Section 11 of the Ignalina RBMK-1500 Source Book.

<u>Rupture of the Fuel Channel</u>

The following potential break locations of the fuel channel were investigated:

- rupture of the steam and water communications lines of the fuel channel
- full rupture of the fuel channel inside of the reactor block,
- partial rupture of a fuel channel.

The rupture of the fuel channel inside of the reactor block can be diagnosed by the following symptoms:

- pressure change in the reactor block,
- activation of "gas moisture increase" and "gas temperature increase" signals in the system for monitoring fuel channel integrity,
- activation of the "water flow increase" signal in the emergency channel cell,
- appearance of water in the drainage pipes from the reactor block,
- appearance of water in the drainage lines from the CPS channel siphon compensator,
- a considerable local decrease of graphite temperature.

When symptoms flagging a fuel channel failure are detected, the reactor emergency shutdown and cool-down is initiated. Reactor shutdown is initiated by either:

- a FASS signal generated automatically if the excess pressure in the reactor block increases to 7.5 kPa,
- or by the operator, using the AZ-1 button if emergency symptoms are indicated by control room instrumentation.

Simultaneous with the cool-down of the reactor, the damaged fuel assembly is removed from the fuel channel using the refueling machine. The emergency plug is utilized and the leakage is stopped by closing of the isolating and control valve of the damaged channel.

<u>Rupture of the Water-Communication Line</u>

The symptoms of a full or partial rupture of the water-communication lines are:

- activation of the "water flow increase" signal in the channel with the ruptured water-communication line,
- increase of moisture and air activity in the water communication compartment,
- increase of pressure in the water communication compartment,
- noise in the water communication compartment,
- level increase in the waste water tanks.

An increase of excess pressure to 2 kPa in the water communication compartments, activates the emergency protection (FASS, AZ-1) signal and the reactor power is decreased to zero. Subsequently the reactor cool-down process is initiated. If emergency symptoms are flagged, the reactor can be shut down by the operator using the button AZ-1.

After cool-down of the reactor, the fuel assembly is removed from the damaged fuel channel and an emergency plug is installed. Leakage is stopped by closing off the isolating and control valve.

Rupture of the Steam-Water Communication Line

A rupture of the 75 mm exiting line which carries the two-phase coolant from the core to the DS results in increased coolant flow rates within the damaged channel. Coolant flow is increased roughly in proportion to the break flow, consequently ability of coolant to maintain adequate temperatures within the channel is thus not impaired. During the pressurized phase of the transient critical flow conditions prevail for the break flow.

In case of the mentioned rupture, a leakage of the coolant from the MCC is insufficient to cause a strong change of the circuit parameters. However, because of loosing a large amount of active coolant, it is necessary to shut down and cool-down the reactor.

Accident symptoms:

- activation of the "water flow increase" signal in the channel with the ruptured steam-water line,
- increase of air moisture and activity,
- noise in the drum separator compartments,
- pressure increase in the steam-water compartments.

If any of the above symptoms appear, the operator must shut-down the reactor. The reactor emergency protection AZ-1 is activated by a pressure increase signal when the excess pressure reaches 2 kPa.

Water, which is discharged due to the rupture of the steam-water pipeline flows to the drainage lines of the 135m^3 tank. From this tank water could be taken to the ACS hot-condensate chamber, or after the elimination of accident consequences, to contaminated demineralized water tank, the capacity of which is 1500 m^3.

Rupture of a Group Distribution Header

Rupture of the 300 mm diameter group distribution header belongs to those accidents, which can cause considerable changes of the MCC parameters. The supply of coolant to the fuel channels connected to the ruptured group distribution header, depends strongly on the rupture location. The most severe rupture of the group distribution header is one which occurs beyond the check valve (taken in the flow direction). In this case all 43 fuel channels are cooled by backward flow of coolant from the drum separators. In this case the leakage rate is maximized, because the coolant flow, which is discharged through the restriction inserted in the MCP pressure header, is added to the leakage of the coolant from the emergency group distribution header through the 43 fuel channels from the separator drum and the ECCS headers.

The FASS reactor emergency protection signal is activated by an increase of excess pressure to 2 kPa in the water communication compartment. In order to simulate most unfavorable accident conditions, a complicating event is assumed. Simultaneous with the activation of FASS, a loss of off-site power is assumed to occur. In this case, the main circulation pumps and main feedwater pumps are not available. The ECCS is put into operation by the simultaneous excess pressure signals and the decrease of the pressure differential (up to 0.6 MPa) between the MCP pressure header and the separator drum. Water from the ECCS is taken to the affected damaged-half of the reactor. ECCS operation is discussed in more detail in Section 6.4.

Safeguards for confinement of this accident lead to the closure of gate valves in the head, suction and bypass lines of the MCP headers. All channels of the damaged-half of the reactor, which are not connected to the affected group distribution header, are cooled by direct ECCS flow under maximum possible flow conditions. After an increase of the water level in the separator drum of the damaged-half of the reactor the channels of the ruptured GDH are cooled by backward flow.

Water, which escapes through the rupture of the group distribution header to the water-communication compartment, flows by way of the drainage lines to the 350 m^3 tank. From this tank water could be taken to the ACS hot condensate chamber.

Rupture of the Downcomer from the Separator Drum

An accident rupturing one of the 325 mm, 16 mm wall thickness downcomers from the separator drum will have a more severe impact on cooling conditions of the reactor core. This is due to a lower discharge rate of saturated water from both sections of the ruptured piping. In this event, before the ECCS pumps are initiated, the reactor core is cooled by coast-down inertia of the MCPs.

To ensure most unfavorable accident conditions, simultaneous with the activation of FASS, a loss of off-site power is assumed. This leads to a shut-down of the turbine generators, the main circulation pumps and the main feedwater pumps. The reactor is cooled down by the ECCS pumps and the auxiliary feedwater pumps. Cladding failure does not occur, because coolant is provided to the reactor block without interruption.

Water, which escapes from the separator drum compartment, flows by way of the drainage lines to the 135 m^3 tank, which are designed for maximum pressure possible in case of emergency in the separator drum compartment. From this tank water could be taken to the ACS hot-condensate chamber.

An accident rupturing one of the water cofferdam of the separator drum which have an outside diameter of 325 mm and a wall thickness of 16 mm will have a more severe impact on cooling conditions of the reactor core. This is due to a lower discharge of saturated water from both sections of the ruptured piping. Before putting into operation the ECCS pumps, the reactor core is cooled while the MCP is costing down.

Rupture of the Main Steam Line Before the Main Steam Gate Valve

Rupture of the 600 mm diameter main steam pipeline is characterized by a significant pressure decrease in the main circulation circuit, as well as by the release of a large amount of steam through the break. System operating parameters, such as pressure in the MCC, pressure differential between the MCP pressure header and the separator drum, liquid levels in the separator drum, etc., in both halves of the reactor change in a similar fashion, because of the 400 mm diameter steam connection between the two MCC halves through 4 cofferdams and the 8 steam discharge valves of type SDV-C. Simultaneously a loss of off-site power is assumed.

The reactor emergency protection FASS or AZ-1 are activated by one of the following signals:

- pressure increase in the steam-water pipeline compartment,
- loss of off-site power supply,
- closure of the turbine emergency regulating valve,
- decrease of feedwater flow.

In addition it is anticipated that the emergency protection AZ-1 is activated by neutronic signals from the reactor: either by exceeding the power limit or by an increase of the reactor power excursion period.

Before the ECCS pumps are turned on, the reactor core is cooled by the inertial coast-down of the MCP's. Subsequently the ECCS provides cooling water to both halves of the reactor.

Rupture of the Feedwater Pipeline

The largest impact of a break in this component occurs for a full rupture of the 500 mm diameter feedwater pipeline extending between the separator drum and the feed control device. This regime is characterized by a significant dynamical variation of the main parameters of the MCC.

In the analysis a simultaneous loss of off-site power is assumed. In this case, turbine generators, main circulation pumps and main feedwater pumps are switched off.

The FASS and AZ-1 reactor emergency protections are activated by an increase of the gauge pressure 2 kPa in the separator drum compartment. The reactor is automatically shut down.

Water, from the ruptured of feedwater pipeline flows along drainage lines to the 135 m^3 tank. From this tank water could be taken to the ACS hot-condensate chamber or to the contaminated de-mineralized water tank.

Rupture of the MCP Header or Pipes Leading to the Header

The largest diameter component of the primary system is the 900 mm MCP pressure header. Loss of integrity of this component cuts of the regular coolant supply to one half of the reactor. Therefore, the rupture of this component is defined as the DB-LOCA for an RBMK plant.

Rupture of the MCP suction header starts a coast-down of the MCP pumps in the affected half .The pumps have a cavitation margin more than 1.5 MPa. Moreover, after stalling of the MCP's, their inertia provides a brief operating margin. This means, that the coolant supply to the affected half of the reactor is not terminated immediately.

The rupture of pressure or suction pipes to the MCP, which have an internal diameter of 750 mm, leads to a smaller loss rate of coolant and the initial failure of only one MCP pump.

Therefore, the instantaneous full cross-section rupture of the MCP pressure header, which has an internal diameter of 900 mm, with unimpeded discharge of coolant from both ends of the pipe while the unit is in full power operation is taken to be the maximum design basis accident.

The reactor emergency protection FASS is activated by a pressure increase signal in the reinforced leaktight compartments. A simultaneous loss of off-site power is assumed. The main circulation pumps and main feedwater pumps are deactivated. Diesel-generators are placed into operation.

The reactor emergency cooling system is switched on by simultaneous signals generated by the pressure increase in the reinforced leaktight compartments and the pressure differential between the MCP pressure header and the separator drum. The ECCS pumps and the auxiliary feedwater pumps are activated and provide cooling water at a flow rate of at least 275 kg/s in each half of the MCC. Water expelled through the break flows along drainage lines to a 150 m^3 capacity tank. From this tank water can be taken to the ACS hot-condensate chamber.

Steam produced by flashing of the hot pressurized water flows from the reinforced leaktight compartments through the steam supply corridors and passes to the ACS steam reception chamber. The response of the of the ACS is described in more detail in Section 6.2.

Rupture of the Service-Water Pipeline

At least three different emergency situations are possible:

A) Rupture of the pressure pipeline of the service-water pump before the check valve.

When the pressure decrease signal is received, the reserve pump is automatically switched on and the devices are switched into reverse. If during the 24 hours it is impossible to repair the malfunction, the unit must be shut down.

B) Rupture of the service-water pipeline in the segment extending from the check valve of the service-water pumps to the gate of the device-consumer pipeline.

For the case of a partial rupture of pipelines, which does not lead to a pressure decrease in the water line, the loss of coolant is eliminated without unit shutdown.

A large loss of water can lead to a cessation of the service-water supply required for normal operation. Interruption of cooling of engines and oil coolers of an MCP requires their shut-down, subsequent to this the reactor would be shutdown by the AZ-1 button. In this case a loss of off-site power is possible. It is the most serious sequence of the rupture of a service water pipeline. In this case to ensure a reliable supply of service-, this pressure water line is closed off, and their pumps switched to the non-emergency water line.

C) Rupture of the service-water pipeline at segment which connects this pipe to the system.

Isolation of the damaged section is performed automatically by a preventive signal from flow meters. It leads to the isolation of only one device-consumer from the group.

Rupture of the Purification and Cooling System Pipeline

The character of the accident resulting from the rupture of the purification and cooling system pipeline depends on the location of the rupture. For the Ignalina NPP design the following situations were examined:

A) Rupture of the PCS 150 mm diameter pipeline near the pressure header.
B) Rupture of the PCS 200 mm diameter pipelines at the section right after joining the 150 mm diameter pipelines before the limiting insertion.
C) Rupture of the PCS 300 mm diameter pipeline, which is used for water return to the mixers of the feed system between check valve and mixer.
D) Rupture of the 300 mm diameter water receipt pipeline to the PCS from the water cofferdam of the separator drum.
E) Rupture of a pipeline in the PCS compartments.

The maximum rate of coolant loss occurs in the first case. Depending on the rupture location, the reactor emergency protection FASS is activated by a pressure increase signal in the corresponding locations:

- reinforced leaktight compartments in case A,
- reinforced leaktight compartments in case B,
- feed system compartment in case C,
- separator drum or reinforced leaktight compartments in case D,
- purification and cooling system compartment in case E.

Fuel channels of the damaged-half of the reactor are normally cooled by coolant flow from the MCP. During the loss of off-site power or by a signal of water level decrease in the separator drum, water is taken to the damaged-half of the reactor by the ECCS pump.

Rupture of the Intermediate Circuit Pipeline

Rupture of the intermediate circuit pipeline is a very-low-probability event, because in this system pipelines are used, which are designed for pressures of 1.6 MPa. This is about three times higher than the maximum possible pressure in the system.

Rupture of this pipe would lead to the shutting off of the water supply to the MCP coolers and the heat exchanger. The reactor would be shut down by the operator and cooled down. Water level in the separator drums is supplied by the auxiliary feedwater pumps.

3.1.2.2 Assumed Missile Effects

According to the list of initial events [18], the following events, related to internal missile effects, are examined. Accidental dropping of:

- a fuel assembly during transportation to the spent-fuel pool by the refueling machine,
- a casket with spent fuel during its unloading from the hot cell to the spent-fuel pool or during transportation from the spent-fuel pool to the container,
- an irradiated fuel channel in transportation by crane during replacement,
- a transport container during transportation from the spent-fuel pool by car to the flushing chamber,
- a refueling machine, central hall crane or constructions onto the top of the reactor.

A fuel assembly being transported by means of the refueling machine is lifted and lowered with the grabber. This ensures a reliable grip of the fuel assembly and eliminates a possible release of the fuel assembly. To increase the reliability and safety, the refueling machine is inspected before each transporting cycle.

The fall of the casket with spent fuel, fall of spent fuel assembly, irradiated fuel channel and transport container during transporting from cranes are prevented by the following measures:

- using the grabber, which eliminates the spontaneous disconnection with transporting loads,

- precautionary lock is placed on the hook, which prevents the dropping of the grabber loop from the hook,
- cranes have a brake which is normally closed. This ensures reliable locking of loads during loss of power and from automatically disconnecting by a switch on the drive of the lifting mechanism, and the end switch, which ensures automatically stops of the hook-lifting mechanism, when it reaches a set limit,
- regular inspection of the load lifting mechanisms.

The possibility of fall of a transport casket from the second tier to the bottom of the spent-fuel pool is prevented, because the dimensions of the openings are smaller than the dimensions for the casket.

The consideration of the "Dropping of the refueling machine, the central hall crane or crane construction components on the top of the reactor" is proposed in a list of hypothetical accidents defined in 1990 by the Kurchatov Atomic Energy Institute, Moscow, Russia. This requirement was imposed after completion of the Ignalina NPP.

3.2 DESIGN OF STEEL STRUCTURES

3.2.1 Standard Practices

Requirements for materials inside the core, where they are subject to intensive neutron fluxes, and for materials of equipment and pipelines, which are not exposed in such a way, are quite different. All metals, which are used for equipment and pipelines of the nuclear power plant, must have excellent mechanical properties, such as high corrosion and erosion resistance, certain thermophysical and good technological properties. It means, they must have an ability to withstand deformation without cracking under cold and hot conditions as well as good weldability and machinability.

For materials inside the core, the cladding and pressure vessels present additional requirements such as low cross neutron capture cross sections. This is needed for supporting chain reactions, as well as for improved mechanical properties, especially for strength against cleavage fracture and resistance to plastic deformation upon intensive neutron exposure.

Neutron exposure can result in damage of metal and alloy crystal lattice. Each metal structure transformation involves a change of its mechanical and thermophysical properties. Metals subjected to neutron exposure, become tougher and more brittle, their thermal conduction is decreased, and their creep rates are increased.

Thus, main requirements for construction materials are:

- corrosion and erosion resistance of coolants for specified parameters, compatibility with fuel and fission products,
- satisfactory mechanical properties (strength, plasticity, creep) taking into account radiation, which can change these properties,
- high thermal conductivity,
- low cross-section for neutron capture,
- technological effectiveness, such as machinability and weldability.

Note that, compared to the VVER-type reactors, the requirements for materials of metal structures of RBMK-type reactors is less restrictive. It is possible to use simpler low-alloy steel because of the lower coolant pressure, neutron fluency and thickness of different construction elements.

3.2.2 Material Properties

The RBMK-1500 reactor is located in a cylindrical casing made from a sheet of heat-resistant low-alloy steel 10ChN1M, the thickness of which is 16 mm. Reactor casing, together with the top and bottom metal structures, the thickness of which is 40 mm and is manufactured from the same material, form the close reactor space. Chemical composition, physical-mechanical and thermophysical properties of the low-alloy steel are given in Tables 3.2 through 3.4.

Maximum permissible temperature for steel 10ChN1M is 400 ^{o}C [17]. This steel is not prone to thermal fracture by aging in the temperature range of 340-450 ^{o}C and duration time up to 10^4 hours. Steel 10ChN1M is sufficiently effective to all metallurgical conversions. The statistical data of mechanical tests of specimens show that the material is stable and has a high level of strength, plasticity and viscosity.

For welded structures, which are manufactured without a thermal treatment, the ability of steel to relieve stresses during operation is very important. At a temperature of 400 ^{o}C and the duration time of 10^3 hours an initial stress equal to 0.8 x s_y is reduced about 10 % and at temperature of 450 ^{o}C - by about 30 % [23].

Steel 10ChN1M with a content of nickel up to 1.4 % ensures full hardenability of sheets and, hence, uniform mechanical properties for thickness not more than 50 mm. The critical fracture temperature for steel 10ChN1M sheets for thickness of 30-40 mm is about 10 ^{o}C [23]. This steel has a high resistance for thermal fracture, and maximum rise of critical fracture temperature depends somewhat on aging temperature and the critical fracture temperature 10-20 ^{o}C [23]. Steel 10ChN1M has good radioactive resistance to fluency up to $6{\cdot}10^{13}$ n/cm^2 (E > 1 MeV) [21].

Main circulation circuit of the RBMK-1500 reactor is manufactured mainly from austenitic stainless steel 08Ch18N10T. Pressure and suction headers with corresponding pipelines, as well as the separator drum, are manufactured mainly from carbonic steel 22K, and the lining inside by austenitic stainless steel 08Ch18N10T. Chemical composition, physical-mechanical and thermo-physical properties of mentioned steels are given in Tables 3.2 through 3.4.

Table 3.2 Chemical composition of steels, used for the main equipment of RBMK-1500 reactors [23]

Steel type	Chemical composition, %									
	C	Si	Mn	Cr	Ni	Ti	Mo	S	P	Cu
10ChN1M	0.08-0.12	0.17-0.37	0.3-0.6	0.7-1	1.1-1.4	-	0.4-0.6	< 0.03	< 0.03	< 0.3
08Ch18N10T	< 0.08	< 0.8	< 2	17-19	9-11	< 0.7	-	< 0.02	< 0.035	-
22K	0.19-0.26	0.2-0.4	0.75-1	< 0.4	< 0.3	-	-	< 0.025	< 0.025	< 0.3

Table 3.3 Physical-mechanical properties of steels, used for the main equipment of RBMK-1500 reactors [21]

Steel type	Characteristic Specification		Temperature, °C											
			20	50	100	150	200	250	300	350	400	450	500	550
10ChN1M	Sheet	σ_u, MPa	540	530	520	500	491	491	461	451	441			
	thickness	σ_y, MPa	441	432	422	412	402	392	373	363	353			
	from 6 to 40 mm	ε_u, %	16	16	16	13	13	13	13	13	14			
		A, %	50	50	50	50	45	45	40	40	40			
	Weld-less tubing	σ_u, MPa	491	491	471	451	441	432	412	402	392			
	hot rolled with	σ_y, MPa	343	333	323	314	304	294	294	294	275			
	outside diameter	ε_u, %	20	20	20	15	15	15	15	15	15			
	60-168 mm and wall thickness from 6 to 32 mm	A, %	50	50	50	50	45	45	40	40	40			
08Ch18N10T	Rolled steel	σ_u, MPa	491	480	461	436	417	397	377	353	328	314	289	270
	and forging	σ_y, MPa	196	191	189	186	181	176	172	167	162	157	152	150
	with thickness or	ε_u, %	38	37	36	33	31	28	26	25	22	20	20	20
	diameter less than 200 mm	A, %	40	40	40	40	40	40	40	40	40	40	40	40
	Plates, forging	σ_u, MPa	491	477	456	426	417	382	358	333	303	289	260	235
	from ingot, sheet	σ_y, MPa	196	193	186	181	176	167	162	157	152	144	137	132
	bar and stamping	e_u, %	35	34	33	31	29	27	26	25	24	23	22	22
	with diameter from 40 to 200mm	A, %	40	40	40	40	40	40	40	40	40	40	40	40
	The same,	σ_u, MPa	491	475	446	421	392	368	343	314	289	260	235	206
	with diameter	σ_y, MPa	196	191	181	172	164	152	147	137	132	123	113	103
	more than	ε_u, %	35	34	33	31	29	27	26	25	24	23	22	22
	200 mm	A, %	40	40	40	40	40	40	40	40	40	40	40	40
	Piping	σ_u, MPa	510	471	461	441	421	421	412	412	402	382	353	333
		σ_y, MPa	216	206	206	196	187	187	177	177	167	157	147	147
		ε_u, %	35	32	30	28	27	26	26	26	25	25	25	25
		A, %	55	55	55	54	54	53	52	51	50	48	47	45
22K	Sheets with	σ_u, MPa	430	430	430	430	430	421	412	392				
	thickness from	σ_y, MPa	215	206	196	186	186	186	186	177				
	70 to 170 mm	ε_u, %	18	18	18	17	17	16	17	18				
		A, %	40	40	39	38	38	38	39	40				
	Forging with	σ_u, MPa	390	390	390	383	373	363	353	353				
	diameter from	σ_y, MPa	195	186	177	167	167	157	157	137				
	300 to 800 mm	ε_u, %	18	15	13	13	13	13	13	13				
		A, %	38	38	38	36	36	35	34	34				
	Forging with	σ_u, MPa	430	392	392	392	392	392	353	343				
	diameter from	σ_y, MPa	215	206	196	186	186	186	186	177				
	100 to 800 mm	ε_u, %	16	14	11	11	11	11	11	11				
		A, %	35	35	35	33	33	32	31	31				

σ_u - ultimate strength, MPa,
σ_y - yield strength, MPa,
ε_u - ultimate strain, %,
A - area reduction, %.

Table 3.4 Thermo-physical properties of low-alloy steels [23,24]

Temperature, ^oC	10ChN1M		08Ch18N10T			22K		
	α, 10^{-6}/K	E, GPa	α, 10^{-6}/K	E, GPa	λ, W/(mK)	a, 10^{-6}/K	E, GPa	λ, W/(mK)
20	-	210	-	205	-	-	200	-
50	11.5	207	16.4	202	-	11.5	197	-
100	11.9	205	16.6	200	16.3	11.9	195	49.5
150	12.2	202	16.8	195	-	12.2	192	-
200	12.5	200	17.0	190	17.5	12.5	190	47.7
250	12.8	197	17.2	185	-	12.8	185	-
300	13.1	195	17.4	180	18.8	13.1	180	45.5
350	13.4	190	17.6	175	-	13.4	175	-
400	13.6	185	17.5	170	21.4	13.6	170	43.5
450	13.8	180	18.0	167	-	13.8	165	-
500	14.0	175	18.2	165	23.0	14.0	160	41.5
550	14.2	170	18.4	162	-	14.2	-	-
600	14.4	165	18.5	160	24.6	14.4	-	39.3
700	-	-	-	-	26.8	-	-	-

α - coefficient of thermal expansion, 1/K,
λ - thermal conductivity, W/(mK),
E - Young's modules, GPa.

Table 3.5 Steel 08Ch18N10T properties at 20 ^oC after exposure to fast neutrons at different fluences [24]

Fluency, n/cm^2	σ_u, MPa	σ_y, MPa	ε_u, %
At exposure temperature of 100 ^oC			
-	675	340	53
10E17	730	380	48
5x10E17	720	530	45.5
4.3x10E18	875	710	37
9x10E18	780	680	34.5
10E20	880	780	23
At exposure temperature of 300 ^oC			
-	550	-	40
3x10E19	650	-	27
6x10E19	680	-	21
10E20	720	-	18
3x10E20	750	-	14
5x10E20	770	-	18
7x10E20	790	-	12

Maximum permissible temperatures for steels 08Ch18N10T and 22K are 600 ^oC and 350 ^oC, respectively [22]. Austenitic stainless steel 08Ch18N10T possesses high heat resistance right up to 550-600 ^oC [24]. This steel has high uniform corrosion resistance in the water of main circulation circuit up to 360 ^oC, and in water-steam mixture - up to 650 ^oC. Losses due to corrosion at temperatures 260-315 ^oC amount to 0.0003-0.0018 mm/year [25].

Cross-section of heat neutron absorption for steel 08Ch18N10t is $(2.7\text{-}2.9)\cdot 10^{-28}$ m^2. Neutron fluency influence to physical-mechanical properties of austenitic stainless steel is shown in Table 3.5.

With an increase of neutron fluency the elongation of steel 08Ch18N10T decreases. By exposure to temperatures higher than 350 °C partial annealing of the radioactive effects and, hence, reestablishment of physical-mechanical properties of steel 08Ch18N10T take place. The temperature of the annealing is about 0.5-0.55 of the absolute melting temperature. Exposure of austenitic stainless steel to temperatures higher than 600 °C leads to loss of fragility.

Neutronic exposure has practically no influence on the corrosion of steel 08Ch18N10T. It can be subject to corrosion cracking with simultaneous presence of chlorine and oxygen or other oxidizers and with the presence of stretching of the metal. If the stressed metal is in the surroundings, of only 1 % of the mentioned agent, cracking is not observed [24]. Corrosion cracking of the steel 08Ch18N10T is observed with the concentration of chlorides on the surface. It is observed that, if concentration of the chlor-ion on the surface of steel is in saturated steam which contain 0.3-0.4 g/kg oxygen, corrosion is 10^{-6} g/kg.

Steel 22K is sensitive to thermal fracture of the operating temperature of the NPP equipment. Maximum rise of the critical temperature of fracture depends very little on the temperature of aging. For heating under extended time of 10^4 hours it make up 10 °C at aging temperature of 340 °C and 20 °C at aging temperature of 400 °C [23].

As a construction material for manufacturing of cladding of fuel assemblies and fuel channels of the RBMK-type reactors zirconium alloys with admixture of niobium are used. Zirconium and its alloys have a small cross-section of absorption of heat neutrons equal to $(0.2\text{-}0.3)\cdot10^{-29}$ m^2. Zirconium keeps satisfactory mechanical properties at temperatures up to 390-400 °C. Zirconium practically does not interact with nuclear fuel, does not undergo plastic deformation either under hot or cold conditions, and has good weldability. Zirconium and its alloys have good corrosion resistance to reactor water at temperatures up to 370 °C.

If water possesses both oxygen and ammonia, corrosion resistance visibly decreases. This condition imposes definite limitations on the choice of the water regime.

At temperatures of 1200-1300 °C zirconium actively interacts with water. This process leads to the release of a large amount of heat. This condition is taken into account, when analysis of accident situations are performed.

Niobium not only improves mechanical properties of the alloy, but also neutralizes bad influences of admixtures to the corrosion resistance.

Rise of hydrogen content in zirconium and its alloys evoke a hydrogen-induced fracture, which is displayed mainly by a decrease of impact strength - at 20 °C it can decrease by 4-6 times. Hydrides have little influence to indexes of statistical strength and strain elongation.

Cladding of fuel assemblies use iodide-zirconium alloy with 1 % of niobium, and cladding of fuel channels use an alloy with 2.5 % of niobium. The physical-mechanical properties of the mentioned alloys as well as the Zyrcaloy-2 properties, are shown in Table 3.6. Maximum permissible temperature for using zirconium alloys with 1 and 2.5 % of niobium is 360 °C [21].

Influence of neutron exposure on zirconium and its alloys is similar to the influence on steel: ultimate resistance and especially yield strength increases, ductility decreases, creep accelerates. Influence of neutronic exposure to fracture of zirconium alloys with niobium is lower in comparison with that of Zyrcaloy-2. With rise of temperature up to 350-400 °C bad influence of exposure to creep decreases. Influence of exposure to physical-mechanical properties of alloy Zr+2.5 % Nb is shown in the Table 3.7.

Corrosion process are accelerated with increasing amounts of oxygen in the reactor water and the availability of exposure. Without exposure such a process was not observed. The influence of exposure to hydrogenation is less than that to oxidation. Hydrogen absorption by zirconium alloys with niobium is less visible than that by Zyrcaloy-2.

In the wall of the fuel channel of the zirconium alloy an accumulation of plastic deformation is observed during its life-time. Zirconium alloy with 2.5 % of niobium has good ductility at initial conditions. However, due to aging, exposure, creep and cyclic change of temperature, the ductility properties gets worse.

Fracture of fuel channels because of loss of long-term strength occurs at residual strain of 5-10 %. According to the studies of the exposure limit on long term strength increase [25]. Corrosion rate of alloy Zr+2.5 % Nb does not exceed 0.01 g/(m^2hour) during the 8000 hour testing, and under reactor exposure conditions, rise by 5-10 %.

Aluminum alloys are used for the CPS equipment. They have good corrosion resistance in the water at temperatures 100-250 °C. Maximum permissible temperature for these alloys is 190 °C [21]. Ultimate strength of these alloys at 20 °C is 150 MPa, and the yield strength is 40-50 MPa. At a temperature of 200 °C these indexes decrease to 90-100 and 30-40 MPa, respectively.

3.2.3 Failure Design Criteria

Similar to the design of other reactor types, the design of RBMK-1500 reactors employs a ‘safety factor’ in the determination of the strength of equipment and pipelines. Maximum stresses in structures must not exceed the permissible stresses. Permissible stresses are determined from the material characteristics at recommended temperatures and include the safety factor. In the code for strength calculations [21] the following recommended temperatures were given:

- 20 ^oC for aluminum and titanium alloys,
- 250 ^oC for zirconium alloys,
- 350 ^oC for carbon, alloy, silicon-manganese and high-chromium steels,
- 450 ^oC for corrosion-resistant austenitic and high-temperature chrome-molybdenum-vanadium steels and for ferrous-nickel alloys.

Table 3.6 Physical-mechanical zirconium alloy properties [24]

Alloy	Characteristics	Temperature, ^oC			
		20	**200**	**300**	**400**
Zr+1 % Nb	σ_u, MPa	350	260	200	180
	σ_y, MPa	200	160	120	90
	δ, %	30	31	33	38
Zr+2.5 % Nb	σ_u, MPa	450	320	300	270
	σ_y, MPa	280	220	200	180
	δ, %	25	24	23	22
Zyrcaloy-2	σ_u, MPa	480	250	200	170
	σ_y, MPa	310	150	100	70
	δ, %	22	34	35	36

δ - plasticity, %.

Table 3.7 Influence of fluency exposure on physical-mechanical properties of alloy Zr+2.5 % Nb [24]

Working	Fluency, n/cm^2	Test temperature, ^oC	σ_u, MPa	σ_y, MPa	ε_u, %	δ, %
Hardening	-	20	870	780	63	13
with 880 ^oC	10^{20}	20	1000	960	-	10
and aging	10^{21}	20	1110	1080	45	8
with 500 ^oC	-	300	580	530	75	14
during	10^{20}	300	720	680	-	13
24 hours	10^{21}	300	810	780	65	9
Hardening	-	300	580	480	70	13
with 960 ^oC	10^{20}	300	810	770	50	8
and aging	10^{21}	300	860	860	5	4
with 500 ^oC						
during 24 hours						

According to the same design code [21], nominal permissible stresses for equipment and pipeline components, which are pressure loaded, take the minimum of the following values:

$$\sigma = \min\{\sigma_u(T)/n_u,\ \sigma_y(T)/n_y,\ \sigma_{ut}(T)/n_{ut}\},$$

where σ_u - ultimate strength, σ_y - yield strength, σ_{ut} - minimum protracted strength during time period t.

For components of equipment and pipelines, which are loaded by internal pressure,

$n_u = 2.6$, $n_y = 1.5$, $n_{ut} = 1.5$.

For components of equipment and pipelines, which are loaded by external pressure, which is higher than the internal pressure

$n_u = 2.6$, $n_y = 2$, $n_{ut} = 2$.

This calculation procedure for permissible stresses is used for static loading conditions. Structural components subjected to variable, cyclic loading, as a rule, are damaged by lower stresses. Therefore, for the final design strength and stability conditions, final calculations are performed, which take into account all the expected loading and all the operation regimes. During these final calculations, the following characteristics are taken into account:

- static strength,
- stability,
- cyclic and long-term cyclic strength,
- resistance to cleavage fracture,
- prolonged statistical strength,
- progressing deformation,
- influence of seismic loading,
- resistance to vibration.

The final calculation procedure is described in detail in [21], pages 45-119.

3.2.4 Qualification Tests of Reactor Components

The materials of the equipment, including the materials used in the reactor core, must maintain during its operational life-time high strength with a sufficient level of ductility and high corrosion resistance. In order to avoid destructive failure the NPP, equipment is tested before installation for expected conditions during its life-time. The design, manufacturing, mounting and operation of the main metal components and their welds would be subjected to the requirements, which are regulated by the references [23, 27, 28]. These requirements are extended to vessels under pressure (including hydrostatic and vacuum), reactor vessels with their guard tanks and casings, pump vessels, pipelines and devices of first and second circuit of the NPP.

Hydraulic Tests [23]

Hydraulic tests are performed to control the strength and leakage of equipment, which is loaded by pressure. Hydraulic tests of this equipment are performed after the manufacturing and installation has been completed.

Pressure of the internal hydraulic tests (P_h) must not be lower than those calculated by the formula:

$P_h = K_h P\ \sigma(T_h)/\sigma(T)$ (lower bound),

and not higher than the pressure which would cause the tested equipment a nominal membrane stress to 1.35 s (T_h), and the sum of the nominal local membrane and the nominal flexural stresses to 1.7 s (T_h) (upper bound),

where, K = 1.25 for equipment and pipelines, and K = 1 for confinement and for guard tanks,
P is allowable pressure during tests at the manufacture plant or working pressure during tests after installation and operation, MPa,
$\sigma(T_h)$ is the nominal allowable stress at temperature of the hydraulic tests T_h, MPa,
$\sigma(T)$ is the nominal allowable stress at recommended temperature T, MPa.

For components, which are subjected to loading by external pressure, the following condition must be fulfilled $P_h < 1.25P$.

For a pressure P up to 0.49 MPa, the value of P_h should be more than 1.5P, but not less than 0.2 MPa. For a pressure P above 0.49 MPa, the value of P_h should not be less than (P+0.29) MPa.

Hydraulic tests of equipment would be performed at the temperature of the tested medium, where the metal temperature of the tested equipment is not lower than the minimum allowable, as calculated by the code [21]. In all cases the temperature of the test and the surrounding medium can not be lower than 5 ^oC. It is also possible to calculate the minimum permissible metal temperature T_h during a hydraulic tests from the correlation's:

$T_h > T_{k0} - 260 + 73 \cdot 10^{-6} S\sigma^2_y$, if $S\sigma^2_y < 3.5 \cdot 10^6$,

$T_h > T_{k0} - 17 + 3.1 \cdot 10^{-6} S\sigma^2_y$, if $3.5 \cdot 10^6 < S\sigma^2_y < 25 \cdot 10^6$,

$T_h < T_{k0} + 48 + 0.47 \cdot 10^{-6} S\sigma^2_y$, if $S\sigma^2_y > 25 \cdot 10^6$,

where, T is the critical temperature of metal fracture at initial conditions, ^oC,
S is the maximum nominal wall thickness of the equipment, mm,
σ_y is a limit of material yield stress at temperature 20^{o}C, MPa.

Pressures as well as temperatures of the hydraulic tests conducted after manufacturing, are indicated by the manufacturer in the equipment certificate.

The permissible metal temperature in the hydraulic tests during equipment operation is determined by the owner of the equipment on the basis of calculated strength, data from the equipment certificate, number of load cycles (which is known from the operational process) neutron fluency with energy E > 0.5 MeV and the surveillance-specimen test data.

The exposure time for the equipment subjected to pressure P_h during hydraulic tests must not be less than 10 minutes. After the exposure, the pressure is decreased to the value of $0.8 \cdot P_h$ and a visual inspection is performed wherever possible.

The measurement of pressure during the hydraulics tests has to be performed by at least two independent measuring channels. A measurement error can not exceed

5 % of the nominal value of the test pressure. Temperature must be controlled by devices with the sum error not higher than 3 % from the maximum value of the measuring temperature.

A test program is prepared before the hydraulic tests. After the test the data are prepared and documented.

The equipment is considered having passed the test, if during the test or visible inspection it does not reveal leaks or rupture of metal, pressure drop does not exceed a permissible limit, and if after the tests visible residual strains are not discovered.

Control of the State of Metallic Equipment During Operation [23]

Control of the state of the NPP metallic equipment is performed to uncover light structural defects, changes of physical-mechanical properties, as well as to estimate the metal. Estimation of the metal state during the equipment operation is performed using both indestructible and destructible methods. Metal testing by indestructible methods is performed using visual, capillary, supersonic and radiography methods. Destructible methods are used to test metal components and welds by means surveillance-specimens, namely:

- changes of mechanical properties (yield strength, temporary resistance, change in elongation, change in cross-section),
- characteristics of resistance of cleavage fracture (critical temperature of fracture, viscosity of destruction or critical crack opening),
- characteristics of solid and local corrosion (inter-granular stress corrosion),
- changes in characteristics of cyclic strength (fatigue strength).

Periodical inspection of metallic components by indestructible methods is performed on the following time scale:

a) first - not later than after 20000 hours of operation,
b) subsequent - not later than after 30000 hours from the previous test.

Tests of the surveillance-specimens, which are placed inside the reactor vessel, are performed not less than 6 times per expected life-time of the component. First unloading and tests of surveillance-specimen are performed after one year from the start of the operation, and subsequently - every three years during the first 10 years of operation on the condition, that during the first unloading the neutron fluency at the reactor vessel is not less than 10^{22} n/m^2, but not more than 10^{23} n/m^2 (E > 0.5 MeV). Surveillance-specimens for control of mechanical properties and its characteristics for redundancy to cleavage fracture are placed into the fuel channels. Detectors for measuring of neutron fluency and temperature are placed with the specimens. Surveillance-specimens are made by the manufacturer of the equipment.

Technological Control of RBMK-1500 Reactor Components

In order to avoid reactor component damage it is necessary to ensure an optimal thermal operation regime. Technological control of reactor components is provided by a system of temperature monitors of graphite cladding and reactor metal structures.

Temperature of cylindrical casing of the reactor is controlled along one generatrix at four elevation points. On top and bottom metal structures, which are under considerable thermal stresses, 30 points are monitored during stationary and transient regimes. Cable chromel-alumel thermocouple (type TChA-1449) placed inside protective case of corrosion-resistant steel with two modification - for putting into another special case and one without that case - is used for temperature measuring of metal structures. The use of special cases for positioning the TChA-1449 thermocouple does make it possible to change thermocouples.

The temperature of metal structures is automatically monitored by the information-computing system "TITAN" and in the case of excess of an assigned value, a deviation is depicted on the screen and in print. Information about temperatures of the metal structure is available upon operator's request.
The error of the temperature measurement does not exceed 2 %, the time constant of the cable thermocouple is less than 5 s, and the time constant of the thermocouples in the protective case - no more than 60 s. The magnitude of cumulative errors are estimated for internal heat-evolution in elements of thermometric devices and for heat exchange conditions using benchmarked objects. As a rule, methodical errors have a positive sign. This means that thermal converters give an excessive reading in comparison with the actual temperature value. Thus, there is some reserve in the control parameters.

Primary converters of type D-2373 are used for control of pressure differences between the cylindrical casing and the reactor space.

Quality Assurance of Welded Joints

A large number of welds are present in an RBMK type NPP. Welding is the main method for joining a considerable number of pipelines with elements of equipment, devices and metal structures, which were manufactured from carbon, austenitic and other special steals. Pipelines at Ignalina NPP are of different outside diameters ranging from less than 0.1 to more than 1.0 m. Total amount of welded joints reaches hundreds of thousands. Therefore, a continuous control of quality assurance of welds by appropriate measuring equipment is necessary.

For quality control of welded joints and surfacing the following methods are used [28]:

- visual inspection and measurement of main dimensions of welds,
- chromatic and luminescent detection (to reveal internal defects, which become apparent on the surface),
- magnetic-particle inspection (for ferro-magnetic steals, it is possible to find defects at depths up to 8-10 mm from the surface),
- radiographic inspection,
- ultrasonic detection,
- hydraulic tests of strength and leakage,
- control of welded joint tightness by means of helium and halogen leak detectors,
- laboratory testing methods (mechanical properties, metallography investigations, tests of inter-granular corrosion, etc.). This inspection should be carried out for the surveillance-specimens that are cut-out from the welded joints made according to the same welding technology using the same surfacing materials and heating facilities, welded using the same equipment and subjected to the same monitoring method as the joint under inspection.

The specimens are placed in a special container. A complete set should include the following specimens:

- specimens for determining the mechanical properties,
- specimens for determining the critical fracture temperature.

Six containers each of which is a single set, represent a suspension that is arranged in the reactor close to the bottom and top metal structures of the reactor. The suspension structure enables the removal of specimens without difficulties and to separate each section for handling and further testing of the specimens.

3.3 DESIGN OF CONCRETE STRUCTURES

3.3.1 Standard Construction Practices

The NPP concrete, which serves as a structural component, is also used as a material for radiological protection. For radiological protection of the NPP heavy concrete is used with volumetric mass density from 1.8 to 2.5 t/m^3 and particularly heavy concrete with density of more than 2.5 t/m^3.

Heavy concretes for radiological protection, which can be employed at temperatures up to 50 ^{o}C, are called ordinary heavy concretes. Concretes, which are used at temperatures from 51 to 350 ^{o}C, are called concretes for abnormally high temperatures. Concretes, which are used at temperatures higher than 350 ^{o}C are called heat resistance concretes.

Concrete is a composite material, which consists of a binder, fine and coarse aggregates. In most cases portland cement is used as binder, sand is used as fine aggregate. For preparation of coarse aggregate used in protection concretes, metamorphic rocks (serpentinite) are used, as well as metallic ores (magnetite, chromite, hematite),and artificial materials (chamotte, boron carbide).

For radiological protection of the Ignalina NPP ordinary heavy concrete with a density of 2.2 t/m^3 is used most widely, and to a smaller degree, particularly heavy concrete with a density of 4.0 t/m^3 [2]. Special protective concretes or solutions with complex chemical composition are used in very small amounts. At the Ignalina NPP serpentinite solution is used as a filling of top metallic structures. Density of the serpentinite solution is about 1.7 t/m^3 [2].

Use of concrete for radiological protection of nuclear reactors is possible provided the following conditions are met:

- the concrete has properties which are resistant to the effects of both neutron and γ-radiation and can maintain satisfactory mechanical and thermo-physical characteristics,
- temperature stresses, which arise in the concrete and reinforcement, deformations and cracks do not exceed permissible values,

3.3.2 Material Properties Used

All initial concrete properties are determined at normal temperature (20 ^{o}C) and depend on concrete consistency. Protective properties of concretes are determined by two main factors: chemical composition and nuclear density. Composition and protective properties of ordinary and particularly heavy as well as serpentinite concretes are shown in Tables 3.8 and 3.9.

Variation of structural and material properties caused by the influence of ionizing radiation, depends on two main factors, namely, composition and the radiation load on the material. After radiation all concretes (except chromite) increase in volumetric dimensions, and the density decreases. The influence of radiation on strength of concrete is shown in Table 3.10.

Compression strength of ordinary heavy concrete depends on the type of cement used and can reach 50 MPa. Heat conductivity and the coefficient of thermal expansion range up to 700 ^oC is 1.14 W/(mK) and $3.5{\cdot}10^{-6}\,^oC^{-1}$, respectively. Limiting permissible temperature of ordinary heavy concrete is 1200 ^oC [31]. Concrete exposure to neutron flux leads to a storage of radiation defects in the concrete, which can be a cause of variation of their mechanical and thermo-physical properties. The exposure of ordinary heavy concrete to a neutron fluence up to $5{\cdot}10^{19}$ n/cm^2 does not change properties significantly. Increasing the fluence to $1.45{\cdot}10^{20}$ n/cm^2 leads to visible variation of properties: decreasing of density to 7-11 %, thermal conductivity to 30-35 %, coefficient of linear thermal expansion by 5-10 times, compressive strength and elastic properties to 10-20 % [32].

Compressive strength of particularly heavy (chromite) concrete depends on the type of cement used and may reach 40 MPa. Heat conductivity and coefficient of thermal expansion at 700 ^oC is 1.57 W/(mK) and $3.4\mathrm{x}10^{-6}\,^oC^{-1}$, respectively. The permissible temperature operating limit of particularly heavy (chromite) concrete is 1700 ^oC [31].

Compressive strength of serpentinite concrete is 40 - 62.5 MPa Elastic modules of serpentinite concrete varies from 18200 to 6800M at temperature variation from 20 to 500 ^oC, respectively. Serpentinite concrete is a heat resistant material and has a coefficient of heat conductivity, which linearly decreases from 0.91 to 0.74 W/(mK) with temperature increase from 20 to 450 ^oC. Coefficient of linear expansion is constant in temperature interval 100-450 ^oC and equals $4.2{\cdot}10^{-6}\,^oC^{-1}$. Limit of permissible temperature for serpentinite concrete is 500 ^oC [31]. Maximum change of coefficient of linear expansion of serpentinite concrete is 1.3-1.7 % and is affected by exposure dose of $(1.3\text{-}1.7){\cdot}10^{21}$ n/cm^2. Strength of serpentinite concrete with increase of exposure dose $(1.3\text{-}1.7){\cdot}10^{21}$ n/cm^2 reduces to 40 % of the original strength. Variation of the modules of elasticity is nearly the same as the strength variation. Thermal conduction coefficient decreases by 13 % upon exposure of $1.7{\cdot}10^{21}$ n/cm^2. Coefficient of linear thermal expansion at exposed and nonexposed serpentinite concrete by repeated heating is the same $(6\text{-}7){\cdot}10^{-6}\,^oC^{-1}$ [33].

Table 3.8 Chemical compositions of different concretes [30]

Concrete	H	O	B	C	Na	Mg	Al	Si	Ca	Fe	S	F	Cr
	Chemical elements, kg/m^3												
Ordinary heavy	8	1275	-	-	-	-	110	774	137	46	-	-	-
Particularly heavy (chromite)	-	1200	-	2	29	194	175	145	119	263	2	21	1119
Particularly heavy (magnetite)	10	1150	-	-	-	-	30	661	164	1245	-	-	-
Serpentinite	25	1000	12	-	-	332	42	309	132	86	9	-	-

Table 3.9 Neutron and gamma quantum attenuation parameters of concrete [29,30]

Concrete	Cross-section of extraction	Storage coefficient for X = 1.0 - 1.5 m			Attenuation of gamma radiation with E = 3 MeV coefficient	
	Σ_r m^{-1}	K_i (W/m^2)/(n/cm^2)	K_h (W/m^2)/(n/cm^2)	K_g (W/m^2)/(n/cm^2)	μ, m^{-1}	μ/ρ, 10^{-3}m^2/kg
Ordinary heavy	8.0	10.2	105	130	8.52	3.63
Particularly heavy (chromite)	10.02	1000	100	900	11.84	3.63
Particularly heavy (magnetite)	9.5	18.0	8	100	11.90	3.63
Serpentinite	6.5	20.0	82	93	7.01	3.64

K_i, K_h - storage coefficients of intermediate and heat neutron, respectively, (W/m^2)/(n/cm^2),
K - storage coefficients of gamma radiation capture, (W/m^2)/(n/cm^2),
Σ_r - macroscopic extraction of protection material, m^{-1},
μ - linear coefficient of attenuation of gamma quantum flux density, m^{-1},
μ/ρ - mass coefficient of attenuation of gamma radiation, m^2/kg,
X - thickness of protection, m.

Table 3.10 Radiation influence to strength of concretes [29]

Concrete	Neutron flux, n/cm^2	Temperature, ^{o}C	Compressive strength, MPa
Ordinary heavy	0	20	12.5
	$(0.4\text{-}0.6)\cdot10^{20}$	100	12.5
	$(1.2\text{-}1.4)\cdot10^{20}$	150	10.0
	$(3\text{-}4)\cdot10^{20}$	180	4.5
	0	200	9.0
Particularly heavy (chromite)	0	20	12.0
	$(16\text{-}24)\cdot10^{20}$	550-650	4.8
	0	550	7.9
Serpentinite	0	20	9.0
	$(1.3\text{-}1.9)\cdot10^{20}$	100-150	9.5
	$(5\text{-}6)\cdot10^{20}$	200-250	7.0
	$(13\text{-}17)\cdot10^{20}$	250-300	4.0
	0	300	11.0

3.4 SELECTION OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.4.1 Compliance with Code Requirements

The main components of an NPP with the RBMK-type reactors is: the nuclear reactor itself, equipment of the MCC (including MCP, pipelines, devices), separator drums, turbines, generators and transformers.

Principal Components of the Main Circulation Circuit

The design power of the Ignalina NPP the RBMK-1500 is 1500 MW, though as noted previously after the Chernobyl event it was reduced to 1370 MW. The steam generation capacity of the nuclear reactor is determined by the nominal steam flow rate to the turbines, by a pre-set temperature in the turbine condenser, and all thermal loads for the internal requirements of the NPP, and requirements needs of the industrial site and the local residential settlement. The final calculated value of steam generation capacity is increased by a margin of at least 3 %.

The RBMK-1500 reactor has two independent main forced circulation circuits. Each MCC consists of four main circulation pumps, two separator drums and the necessary pipes and associated equipment

At the Ignalina NPP type CVN-8 main circulation pumps are used. The shaft of the pump has a flywheel to increase the MCP slow-down period to 120-130 s by loss of power of electrical motor. A moment of inertia of the CVN-8 is 9400 kgm^2, and the total moment of inertia of the pump complex is 13150 kgm^2.

The MCPs provide the main circulation circuits with the required flow and the pressure head. As a rule, parameters of the pump correspond to the characteristics of the off-site network and have a head margin of 2-4 %.

Internal pipe diameters of the MCC of Ignalina NPP are 0.02-0.9 m, and the flow rates are 2-12 m^3/s.

Selection of profile for the MCC pipelines and the strength calculations is performed in accordance with code rules for calculation of strength for the reactor equipment of the NPPs [21,23].

Principal Equipment of the Turbine Plants

An RBMK-1500 plant is equipped with two type K-750-65 turbines. Although the installation of two turbines in the NPP is somewhat expensive, it nevertheless enables to utilize the capacity of the unit more effectively, because in the RBMK-type reactors it is possible to do the core loading without reactor shutdown. Moreover, the availability of two turbines in one unit makes it easier to do general turbine repairs (during operation), which normally need a considerable amount of time.

Flow of the main feedwater pumps must correspond to the steam generation rate of the nuclear reactor with a margin of 10 %. In case of failure of one pump, the rest would provide the reactor operation by nominal capacity of all steam generating equipment. Three main feedwater pumps for each circulation loop are used at the Ignalina NPP based on the so-called 3 x 50 % principle.

Each unit of the Ignalina NPP is provided with six AFWPs. The capacity of each pump is sufficient to provide removal of residual heat from the core. Approximate flow of one AFWP is 2-4 % of the nominal feedwater requirement. Pipeline systems, as well as electrical cables to auxiliary feedwater pumps, are laid independently from the corresponding lines of the main feedwater pumps.

For reliable removal of thermal energy from the reactor during startup and shutdown of the unit special steam-relief valves are installed to the steam discharge to the turbine condensers (SDV-C), to the fifth pool of the accident confinement system (SDV-A) and to the deaerators (SDV-D). These steam discharge valves have a safety margin of 100 %. Their capacity is determined to provide an appropriate cool-down rate for the reactor. The removal of residual heat in the core as well as the heat, which is accumulated by equipment and coolant is taken into account.

Feedwater, which is directed to the reactor, should be well deaerated. To this end, the use of deaerators in thermal schematic was foreseen. All aerated water flows (drainage, overflows from drainage tanks, condensate overflows, etc.) are fed to the deaerators if they have a temperature exceeding 50 ^oC. In the opposite case, they are directed to the condensers of the turbines. Total capacity of the deaerators is selected based on the maximum feedwater flow with a margin not less than 10 %. Two deaerators are provided for each turbine plant. Feedwater margin in the tanks of main deaerators provides for one unit operation with full power during a time period not less than 180 s.

For water supply to the separator drum in case of emergency cool-down of the reactor a permanent margin of feedwater in special tanks is foreseen. Two such tanks are available, their capacity can provide sufficient coolant to sustain an NPP operation at full power of at least 300 s. On the other, hand an operation at power equal to 2 % from nominal (for removal of residual heat) should not be less than 10 hours. Loss of feedwater is supplanted by de-mineralized water.

Three condensate pumps are used for each turbine, each pump providing 50 % of the nominal flow. The head of the condensate pumps is designed to overcome the hydraulic resistance from the turbine condensers to the deaerators, including the water pressure in the deaerators.

3.4.2 Qualification Tests and Analyses

Operation of installed equipment and systems is permitted only after stringent control measures are taken and equipment acceptance tests are completed. Acceptance and pre-operational tests of installed equipment and systems usually begin 8-12 months before start-up of the NPP. They are performed by the constructors in cooperation with the customer. For the installing of the NPP equipment the following tests were performed:

- tests of individual equipment, tests of safety systems and of the automatic activation or deactivation of equipment,
- circulation of "cold" and "hot" coolant, flushing of NPP equipment and systems,
- test of reactor equipment and systems under 'hot' conditions with coolant parameters near to design values but without loading of nuclear fuel into the reactor core,
- startup of the reactor and testing of its nuclear-physical characteristics,
- start of energy generation and complex testing of all NPP equipment and systems at nominal power,
- tests at design power.

Individual Tests

Tests of Individual NPP components and equipment consist of the following:

- hydraulic or pneumatic tests for strength and leak-tightness of vessels, heat exchangers, shells of pumps, pipeline systems and other equipment,
- tests at low speeds of rotating mechanisms (electric motors, pumps, ventilators, electric drives),
- testing of equipment (engines, mechanisms, pumps) by operation at full load for the specified test time period.

Post-installation Flushing of the MCC and Turbine Hall Equipment

The equipment is flushed by "cold" and "hot" coolant to ensure a thorough cleaning of the internal cavities of the Ignalina NPP equipment and pipelines. Flushing by circulation of coolant in the MCC was performed in two steps: first, high velocity water was used to flush separate sections, the flushed water being directed to waste disposal. Second, the MCC was flushed with cold water at temperatures up to 100 ^{o}C and with hot water at temperatures 250-270 ^{o}C.

A large portion of the NPP equipment, systems and pipelines are subjected to high velocity water flushing:

- all fuel channels,
- top and bottom water communication lines,
- pipelines of the purification and cooling system,
- cooling circuit of the CPS channels,
- downcomers from the separator drum to discharge header,
- charging and discharging pipelines of the MCC.

During high velocity flushing of individual elements and pipelines it is necessary to ensure that the flushing velocity is 1.5 time higher than nominal design values, but not lower than 2.5 m^3/s.

A criterion that the "cold" flushing has been performed successfully is a water transparency of not less than 95 %. Hot flushing of the MCC is provided to ensure cleanliness of the internal cavities of equipment and the pipelines. Water is heated in the MCC by the operation of the MCP. During hot circulation flushing each half of the MCC is in turn connected to two MCP's. The water temperature is maintained at 250 ^{o}C at a flow rate of about 8 m^3/s. The water is cleaned continuous by mechanical filters and special grid filters, which are temporarily used in the group distribution headers.

Effectiveness of hot flushing is controlled not only by transparency, but also by monitoring the presence of corrosion products and chemical admixtures in the water. According to code, a transparency of the flushing water of at least 95% is required, corrosion products must decrease to o concentration of less than 10^{-3} g/kg, chloride - less than 10^{-4} g/kg, and the total hardness of water must be less than 10^{-6} g(equiv.)/kg. Oil and mechanical admixtures in that water are not permitted after hot circulation flushing.

A characteristic of the NPP with RBMK - type reactors is that the main quantity of metal corrosion products (up to 90%) in the MCC are generated by reaction occurring with feedwater. Therefore, the condensate-feed circuit and other auxiliary systems of the turbine room are subjected to careful cleaning from post-installation contamination as well as from corrosion products. Mechanical cleaning, high velocity flushing and chemical flushing of all the main and auxiliary systems of the turbine room are used for this purpose. For high velocity flushing of the main pipelines of the condensate-feed circuit, the water velocity has to beat least 3-4 m^3/s. Effectiveness of the cleaning procedures are controlled mainly by requiring water transparencies of at least 90 %.

Several equipment items and pipelines of the condensate-feed section require a more extensive cleaning procedure. Therefore, together with water flushing, a procedure using a chemically active (acidic) wash of selected pipes and equipment is performed. For this purpose a flushing solution at a flow rate of 0.2-0.3 m^3/s is circulated by special acid-resistant pumps. The procedure employs tanks for storage of solutions of chemical reagents and temporary pipelines. After the chemical flushing a special flush using ammonia is performed to neutralize acidic residues.

Acclimatization of Equipment at Hot Conditions

The RBMK equipment is operated under hot conditions only after completion of the complex testing program assuring the quality of the installed and manufactured components and systems. During the hot acclimatization of the reactor equipment and systems, the hydro-dynamical characteristics of the reactor and main circulation circuit, reliability of reactor control and protection systems, as well as other systems, which provide safe and reliable operation of the reactor are tested. Average duration of the warm-up of the NPP equipment is about 10-15 days. From the results of the warm-up period conclusions regarding the quality and reliability of installation of the equipment and pipelines are made. This includes the assessment of the readiness of the reactor and all its service systems including the loading of the nuclear fuel to the core of the reactor.

Startup of the Reactor

Startup procedures of the RBMK-1500 reactor include the determination of flux and power distribution of the initial core loading and the evaluation of the main neutronic and thermal characteristics of the reactor. In order to provide reliable control and to ensure nuclear safety during the loading of fuel assemblies into the core a highly sensitive CPSs is used in addition to the regular control and protection system. This CPS makes it possible to control the neutron flux and the reactor reactivity level, and provides an emergency shutdown if necessary.

Fuel assemblies are loaded in several stages. During the first stage criticality of the reactor is achieved. At room temperature this consists of 23-24 fuel assemblies, which have an initial enrichment of 2 %. In the second stage the number of fuel assemblies is increased to 916 and 154 additional absorbers and 56-60 CPS rods are inserted into the core. In the subsequent stages the initial core is loading is completed. During the determination of the neutron flux distribution of the initial core loading the power is on the order of 10^{-5}-10^{-4} % of the nominal value. At this low power the neutron-physical characteristics of the reactor, the effectiveness of the CPS rods and the reactivity margin are established. The composition of the initial core loading of the RBMK-1500 reactor consists of 1445-1455 fuel assemblies and 230-240 additional absorbers.

During the first stage single fuel assemblies are loaded into the reactor. In the subsequent stages it is feasible to load two assemblies. Total time for full scale core loading is about 20-25 days.

During startup of the reactor several measurement programs are carried out. This includes measurement of the following parameters:

- integral and differential effectiveness of the CPS,
- the reactivity variation generated by increase or decrease of water in fuel channels with fuel assemblies and in channels with CPS rods,
- the absolute measurement of neutron flux (neutron power) using activation methods and an initial survey of the measurement generated by emission detectors and ionization instruments. The relation of these measurement to the thermal power of the reactor.

The start-up phase of the RBMK-1500 reactor is completed by transferring control of low-power level to the CPS. This is a power level at which automatic controls can become effective and amounts to about 1 % of the nominal thermal power. At this power level the temperature reactivity coefficients are determined by temperature variation in the range of 100-150 ^{o}C. Heat-up of water in the MCC is accomplished by utilizing four MCPs (two by two at each half of the MCC).

Start of Power Generation and Testing of Reactor at Power

Testing of NPP equipment and systems for power operation is accomplished at power levels from 1 to 10 % of the nominal reactor power. Initiation of power generation for the RBMK-type reactors proceeds in several stages each of which encompasses a gradually widening range of tests.

Before completion of the first stage of energy startup, an adjustment and trial of the fuel channel failure detection system is performed. During the first energy startup stage, the main pipelines are cleaned by steam, which is generated in the reactor. To carry out the proper steam blow-through steam flows of up to 95-100 kg/s must be generated at a pressure in the separator drums of about 1.2-1.4 MPa. Therefore, the MCC water is heated to 80-100 ^{o}C by operation of the MCP, and subsequently fission power is increased to up to 8-10 % of the nominal power. The rate of temperature increase in the MCC must not exceed 10 ^{o}C/h.

Blow-through of steam through the main pipelines is performed in the following sequence: separator drum, main pipeline, main steam gate valve, pipeline from the main steam valve to the emergency regulating valve, temporary blowdown pipeline, discharge to atmosphere. The blow-through of the main pipelines is maintained foe 15-20 minutes, and the total time for blowdown of all the pipelines is 15-20 hours. To provide the necessary amount of steam, the deaerators, separator drums and tanks are filled with 3500-4000 m^3 of de-mineralized water.

The second stage of the startup involves adjustment of the main and auxiliary equipment of the reactor and turbine hall at powers of up to 10 % of the nominal as well as an adjustment of steam discharge valves, emergency steam reception systems, safety valves at the separator drum and bubble condensers. At this stage the reactor power is varied from 2 to 10 % from the nominal.

The third start-up stage involves a test startup of the reactor accompanied by testing of individual turbine generators under a relatively small load (75-100 MW).

A complete testing program of the NPP equipment and systems is carried out during 72 hours at specific power by means of nominal parameters in all circuits.

Approach to Design Power

Testing of the NPP at design power is performed in several stages, beginning from 10 % of the nominal power and subsequently in steps of 20-30, 40-45, 55-60, 75, 85-90 and 100 % of the nominal thermal power. At each stage, acceptance tests of the thermal-hydraulic components and electronic equipment are carried out.

The safety and reliability of reactor operation at a given power level is checked. This includes the following tests and procedures:

- optimization of power density distribution and determination of other thermal parameters,
- investigation of the degree of stress imposed on individual structural components and the NPP systems,
- analysis of transient characteristics of main equipment,
- estimation and evaluation of the heat balance in the MCC and in the turbine room,
- debugging of codes employed for the calculation of neutronic and thermal core properties, as well as for control of fuel channel parameters,
- adjustment of water purification units and provision of water-chemical regime in the MCC and other NPP circuits.