STATE ENTERPRISE
IGNALINA NUCLEAR POWER PLANT

REPORT ON THE PERFORMANCE OF "STRESS-TESTS" AT THE IGNALINA NPP

English version

Ignalina NPP
2011

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1 OBJECTIVE

The objective of this report is to present to VATESI the results of additional analysis of the Ignalina NPP safety limits in accordance with the requirements of Annex 1 of the ENSREG Declaration dated 13 May 2011 "EU "stress tests" specifications".

2 SCOPE

The report is intended to prepare the national report of Lithuania on the results of "stress tests" at the Ignalina NPP in accordance with the requirements of Declaration of the ENSREG dated 13 May 2011.

3 RESPONSIBILITY

The Decommissioning Director of the Ignalina NPP is responsible for the issue of this report. The experts, involved in the development of the report, are responsible for the preparation and submission of the report materials.

4 LIST OF DEFINITIONS AND ABBREVIATIONS

32M basket	32M type Transport Basket
AHS	Additional Hold-Down System
ALS	Accident Localization System
ALT	Accident Localization Tower
Ass.	Assembly
ATS	Automatic Transfer System
AZ-BSM	Russian abbreviation for system of emergency protection and fast reduction of power
BCS	Blowdown and Cooling System
Bld.	Building
BRU-B	Fast-Acting Pressure-Reducing Valve with Steam Dumped into ALS
BSRCh	Bottom Steam Reception Chamber
BWR	Boiling Water Reactor
CBTI	Central Boiler-and-Turbine Institute (Russia)
CC	Cooling Circuit
CCR	Central Control Room
CH	Central Hall
ChWPS	Chemical Water Purification System
CP	Condensate Plate
CPS	Control & Protection System
CPW	Chemically Purified Water
DBE	Design-Basis Earthquake, causing a concussion of maximum intensity at the construction site over a period of 100 years
DD	Decommissioning Directorate
DDS	Dismantling and Decontamination Service

DEM	Deaerator Emergency Makeup
DF	Diagnostic Flowchart
DG	Diesel Generator
DGH	Distributing Group Header
DMD	Documents Management Department
DPMS	Decommissioning Projects Management Service
DPSS	Deputy Plant Shift Supervisor
DPW	Domestic Potable Water
DS	Drum Separator
DSFSF	Dry Storage Facility for Spent Nuclear Fuel
ECCS	Emergency Core Cooling System
ECCSP	Emergency Core Cooling System Pump
ECR	Emergency Control Room
EDPP	Emergency Diesel Power Plant
EFA	Erbium FA with Uranium-Erbium Fuel
EFWP	Emergency Feed Water Pump
EIA	Environmental Impact Assessment
ENSREG	The European Nuclear Safety Regulators Group
EWP	Embedded Water Pipelines
FA	Fuel Assembly
FASS	Fast Acting Scram System
FCh	Fission Chamber
FC	Fuel Channel
FE	Fuel Element (rod)
FIHC	ISFSF Fuel Inspection Hot Cell
GDPS	Gas Discharge Purification System
GMCU	Gaseous Mixture Combustion Unit
GSZP-87	General Seismic Zoning Plan of 1987
HCCh	Hot Condensate Chamber
HETS	Head of Emergency Technical Service
HF	Hydraulic Facilities
HPP	Hydroelectric Power Plant
HTT	Hydraulic Test of Turbine
I&C	Instrumentation & Control Systems
IAEA	International Atomic Energy Agency
ICC	Intermediate Cooling Circuit
ICS	Information Computer System
INPP	Ignalina Nuclear Power Plant
INSC	International Nuclear Safety Centre

IRV	Isolating and Regulating Valves
ISFSF	Interim Spent Fuel Storage Facility
LEI	Lithuanian Energy Institute
LSW	Low Salted Water
LTC	Leak-Tight Confinement
LWL	Low Water Lines
MCC	Main Circulation Circuit
MCE	Maximum Calculated Earthquake, causing a concussion of maximum intensity at the construction site over a period of 1000 years
MCP	Main Circulation Pump
MCR	Main Control Room
MSK-64	12 degrees earthquake intensity scale, developed in 1964 in the USSR. Currently is still used in Russia and CIS States
MTC	Maintenance Cooling Tank
NFD	Nuclear Fuel Department
NFMW	Nuclear Fuel Management Workshop
NPP	Nuclear Power Plant
OEP	Organization of Emergency Preparedness
OM&ESD	Operational Management and Engineering Support Department
PD	Power Decay
PDMS	Power Density Monitoring Sensor
PGA	Peak Ground Acceleration
PH	Pressure Header
PHEU	Pump and Heat Exchanger Unit
PMD	Projects Management Department
PNAE	Russian acronym for Regulations and Standards in Nuclear Power Engineering
PNIIIS	Russian acronym for Industry and Research Institute for Construction Engineering Survey (Russia)
PSA	Probabilistic Safety Analysis
PSS	Plant Shift Supervisor
PSW	Power Supply Workshop
RBMK	Water-Cooled, Graphite-Moderated, Pressure-Tube-Type Boiling-Water Power Reactor
RC	Reactor Cavity
RCC	Reflector Cooling Channel
RMMS	Repair and Maintenance Management System
RP	Reactor Plant
RPAE	Relay Protection and Automatic Equipment
RPDMS	Radial Power Density Monitoring Sensor

RSM	Radiation Safety Monitoring
RSS	Radiation Safety Service
RUZA	Russian acronym for Beyond Design-Basis Accidents Management Procedure
RV	Reverse-Flow Valve
RWMS	Radioactive Waste Management Service
SA&MS	Seismic Alarm and Monitoring System
SAR	Safety Analysis Report
SAS	Seismic Alarm System
SAT	Startup Auxiliary Transformer
SCC	Specially Cleaned Condensate
SEFA	Spent Erbium Fuel Assembly
SFA	Spent Fuel Assembly
SFP	Spent Fuel Pool
SH	Suction Header
SMS	Seismic Monitoring System
SNF	Spent Nuclear Fuel
SOEI	System-Oriented Emergency Instruction
SOUSR	Single Operating Unit Safety Report
SPC	Specially Purified Condensate
SPH	Storage Pools Hall
Stress test	Target reassessment of NPP safety limits in the light of extreme natural events that occurred at the Japanese Fukushima NPP in March 2011
SWMS	Seismic Warning and Monitoring System
SWP	Steam-Water Piping
SWS	Service Water Supply
SY	Switch-Yard
TS	Technological Service
TS&QMD	Technical Surveillance and Quality Management Department
TSRCh	Top Steam Reception Chamber
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
VATESI	Lithuanian acronym for the State Nuclear Power Safety Inspectorate
VNIPIET	Russian acronym for Design and Research Institute of Complex Energy Technology
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators’ Association
WT	Working Transformer

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6 INTRODUCTION

Having considered the consequences of the accident at the Fukushima NPP in Japan, which occurred due to the earthquake and ensuing tsunami in March 2011, the Council of the European Union decided to undertake additional analysis of the safety limits of all nuclear facilities of the Member States of the European Union. On the basis of proposals made by the Western European Nuclear Regulators’ Association (WENRA), the European Committee and members of the European Nuclear Safety Regulators Group (ENSREG) agreed on the composition and volume of the additional analysis of the safety limits, presenting it in Annex 1 of the ENSREG Declaration dated 13 May 2011 “EU “stress tests” specifications”.

This report is intended to prepare the national report of Lithuania on the results of “stress tests” performed at the Ignalina NPP in accordance with the requirements of Annex 1 of the ENSREG Declaration dated 13 May 2011.

7 PART 1. GENERAL CHARACTERISTICS OF THE INPP

7.1 General Data

INPP Site Location

The Ignalina Nuclear Power Plant is located in the north-eastern Lithuania near the borders of Belarus and Latvia. The Plant was built on the southern shore of Lake Drukshiai at a distance of 39 km from Ignalina town. The nearest major cities are Vilnius (about 550 thousand inhabitants) at a distance of 130 km, and Daugavpils in Latvia (about 130 thousand inhabitants), at a distance of 30 km from the INPP. Visaginas town (about 30 thousand inhabitants), inhabited by the employees of the INPP is located at a distance of 6 km from the NPP. The location of the INPP is shown in Figure 7.1-1.

Figure 7.1-1. Location of the INPP

Roads

The existing road and railway system are shown in Figure 7.1-2. The nearest primary road is located 12 km west to the INPP. This road connects Vilnius city with Zarasai town, which borders Latvia, and enters the highway Kaunas-St Petersburg (A-6). The drive from the INPP to the highway is near the Dukshtas town. The length of the road from the INPP to Dukshtas is about 20 km.

The main railway line Vilnius-St Petersburg is 9 km west to the INPP. The INPP is connected with Dukshtas by the railway. Dukshtas railway station is used both for freight and passenger transport. Railway gauge is 1520 mm.

In Lithuania there are three areas aircraft flying over which is prohibited. One of them is the area within a radius of 5 km around the INPP.

Figure 7.1-2. Road and railway diagram

Climate

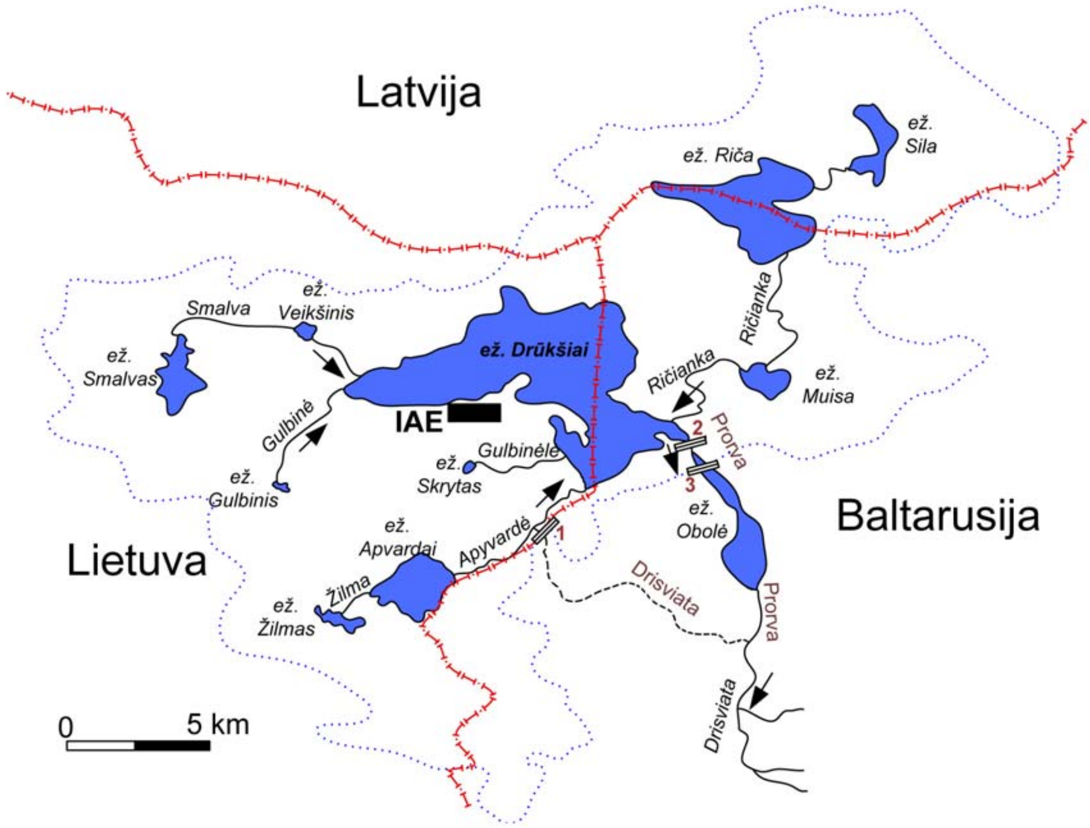
The Ignalina NPP site is located in the Eastern Europe, in the continental climate zone. One of the main features of the climate of the area is the fact that cyclones are not formed there. Cyclones in the majority are related to the polar front and determine the constant movement of air masses. They are formed in the middle latitudes of the Atlantic Ocean and they move from the West to the East over Eastern Europe, thus, the INPP region very often occurs on the crossroads of cyclones that bring moist sea air. Since the change of marine and continental air masses is frequent, the climate of the region can be considered as transitional - from the maritime climate of Western Europe to the continental climate of Eurasia. An average annual precipitation near the Ignalina NPP in 1988-2007 years was about 665 mm. A snow cover in the region rests for 100-110 days a year. An average snow depth is 16 cm.

The western and southern winds dominate in the region. The strongest winds blow westward and southeastward. The annual average wind speed is about 3.5 m/sec, and maximum speed (gust) may reach 28 m/sec. Extreme cases are rare in the vicinity of the Ignalina NPP site. During the storm in 1998 the wind speed of 33 m/sec was registered.

The average annual temperature is +5.5°C. The average calculated temperature of the coldest five-day period is –27°C. The absolute registered maximum is +36°C, the absolute minimum is –40°C.

Hidrology

Lake Drukshiai is the largest lake in Lithuania. The total surface area of the lake is about 49 km^2. The maximum depth of the lake is 33.3 m and average depth is 7.6 m. The length of the lake is 14.3 km, the maximum width is 5.3 km, the perimeter is 60.5 km. The average surface elevation of the lake is approximately 141.6 meters. Taking into account spring floods, the water level may reach the highest value of 142.3 m. The catchment area of Lake Drukshiai is not great and is 564 km^2. The lake is characterized by a relatively slow water exchange rate. The main outflow occurs by the river Prorva in the southern part of the lake. Water from Lake Drukshiai goes its way for about 550 km and enters the Bay of Riga of the Baltic Sea. The scheme of Lake Drukshiai basin is shown in Figure 7.1-3.



Legend:

River flow directions

Artificial dams

Country border

Basin contour

- - - - - dotted line indicates the old Drisvyaty river bed

Figure 7.1-3. The scheme of Lake Drukshiai basin

1 – blind earthen dam, building 501, 2 – water regulating building 500,
3 – hydroelectric power plant “Druzhba narodov” dam

Seismicity

The territory of Lithuania has traditionally been considered as non-seismic zone or zone of low seismic activity. This fact depends on the geological structure of the area and the large distance from the tectonically active regions. Historical and modern instrumental data indicate that cases of low and medium seismic activity have been registered in the Baltic countries.

Recent cases of seismic activity of magnitude 4.4 and 5.0 on the Richter scale have been recorded on 21 September, 2004 in the Kaliningrad region of Russia. They have been registered by the worldwide seismological network and a seismic station of the Ignalina NPP. According to the historical data, 19 earthquakes were registered within a radius of 250 km from the INPP site since 1616.

In 1999 four seismic monitoring stations were installed in the region of the INPP. Since then the Geological Survey of Lithuania, by the agreement with the INPP, processes and analyzes the data collected at these stations. According to the available data, the Geological Survey of Lithuania has estimated that the calculated earthquake magnitude on the INPP territory is 6 on MSK-64 scale, and maximum calculated earthquake magnitude is 7 on MSK-64 scale.

Industrial Site of the Plant

The INPP industrial site covers about 0.75 km^2. The buildings employ about 0.22 km^2. INPP has two identical nuclear power units with RBMK-1500 reactors.

Both power units have the following common structures: solid radioactive waste storage facilities, dry storage facility for spent nuclear fuel, switch-yard, nitrogen-oxygen plant, emergency diesel power plant and other auxiliary facilities. The pumping stations, which are used for continuous supply of cooling water, are constructed on the shore of Lake Drukshiai for each unit separately.

General Plan of the INPP is shown in Figure 7.1-4. The following buildings and facilities are shown in the plan:

101/1, 101/2, – main buildings of Unit 1 and Unit 2;
103 – transformer inspection tower;
109 – oil facility operating unit;
117/1, 117/2 –ECCS tanks buildings of Units 1 and 2;
119 – central heating plant building;
111 – emergency diesel power plant;
112 – diesel fuel tanks;
120/1, 120/2 – service water supply pumping stations;
129 – administrative building;
130 – integrated maintenance building;
130A – warehouse;
131 – building of chemical water preparation with electrolysis unit;
132 –makeup demineralizer tank system;
133 - outdoor plant of hydrogen and nitrogen receivers;
137 – nitrogen-oxygen plant;
138 – compressor and refrigerating station;
140/1, 140/2 – utility rooms building;
141 – tank for water leakages collection;
150 – liquid radioactive waste treatment and bituminization building;
151 – drainage waters filling tanks;
152/1, 152/2 – low salted water filling tanks;
154 – operational washings tanks;
155, 155/1 – low-level radioactive waste storage facilities;

156 – special laundry;
157, 157/1 – solid radioactive waste storage facilities;
158 – bitumen compound storage facility;
158/2 – radioactive waste cement grouting building and temporary storage facility;
159 – special transport vehicles garage and washer;
159A – boxes for special transport vehicles of decontamination department;
159B – industrial waste treatment facility;
161 – bitumen warehouse;
165 – integrated warehouse;
166 – warehouse of dry and liquid chemicals with the unit of sorbents preparation;
185 – administrative building;
186 – canteen;
270 – gases warehouse;
01, 02, 03, 04 – steam boiler house;
31 – Personnel Directorate administrative building;
31A – Power Plant security batallion administrative building;
31B – “Visagino poligrafija” enterprise building;
31V – administrative building;
36 – INPP archive building.

General INPP Site view from the air is shown in Figure 7.1-5.

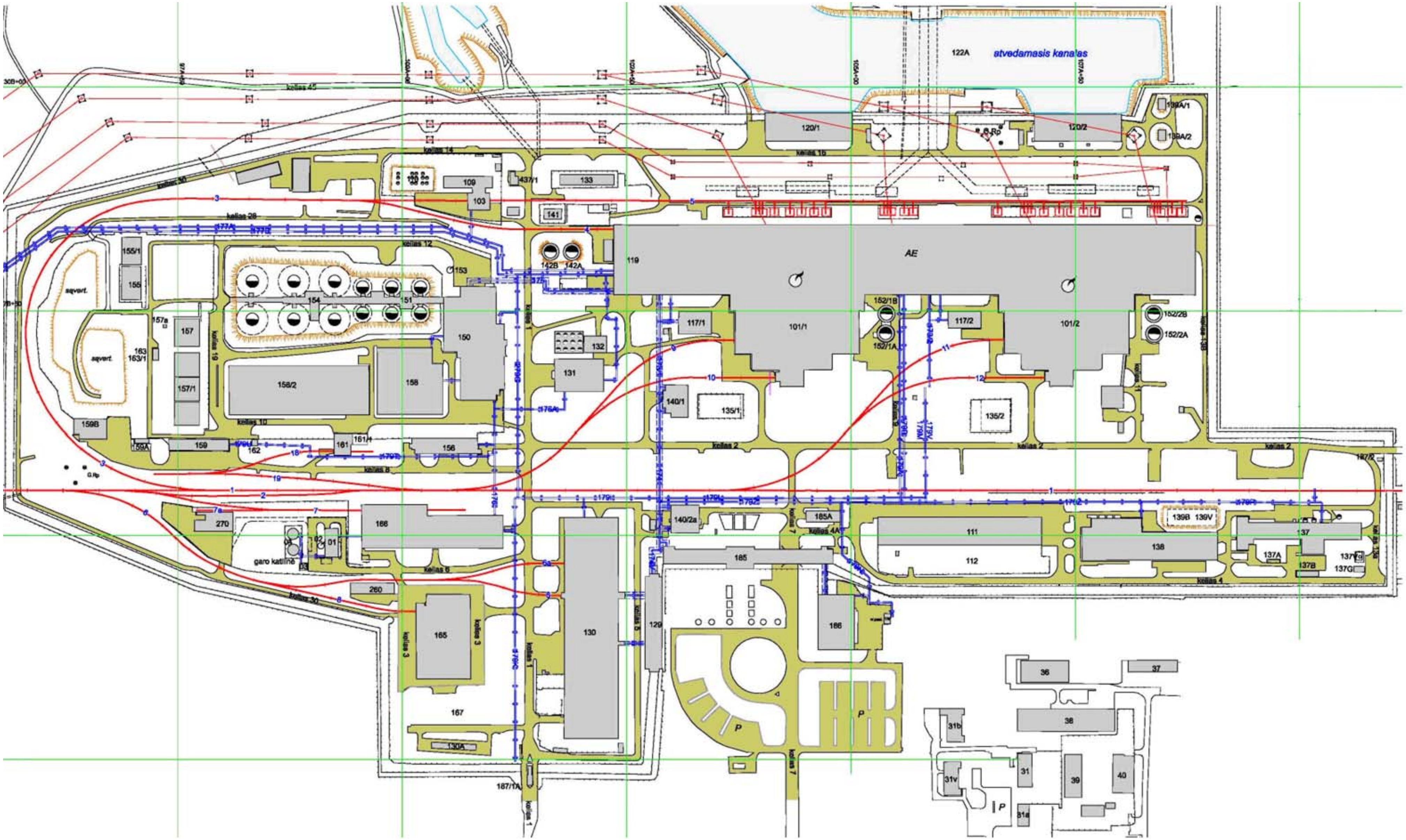


Figure 7.1-4. General Plan of the INPP

Figure 7.1-5. General INPP Site view from the air

1 – Unit 1, 2 – Unit 2, 3 – on-shore pump station, 4 – inlet canal, 5 - emergency diesel power plant, 6 - DSFSF, 7 – switch-yard, 8 – administrative building.

7.2 INPP Obtained Licenses

The State Enterprise Ignalina Nuclear Power Plant is an entity performing the decommissioning of the INPP. At the INPP two channel-type nuclear reactors RBMK-1500 of 1500 MW electric capacity were operated. INPP Unit 1 was commissioned in December 1983 and shut down in December 2004. Unit 2 was operated since August 1987 till December 2009. Currently reactors decommissioning works are performed at the INPP.

Currently the INPP has the following licenses issued by VATESI:

Operating licenses

- License No 12/99(P) for the operation of the INPP Unit 1, dated 29 July, 2004.
- License No 2/2004 for the operation of the INPP Unit 2, dated 15 September, 2004.
- License No 3/2000(P) for the operation of INPP DSFSF, dated 22 July, 2004.
- License No 1/2006 for the operation of the INPP cemented radioactive waste storage facility, dated 10 March, 2006.

Construction licenses

- License No 1/2009 for the construction of solid radioactive waste treatment facility, dated 27 August, 2009.
- License No 2/2009 for the construction of interim spent nuclear fuel storage facility – ISFSF, dated 13 October, 2010.
- License No 1/2010 for the construction of very low-level radioactive waste storage facility, dated 05 March, 2010.
- License No 1/2011 for the construction of solid radioactive waste retrieval and primary treatment facility, dated 27 April, 2011.

Design licenses

License No 2/2008 for the design of very low-level radioactive waste storage facility and repository, dated 23 July, 2008.

7.3 Main Characteristics of Each Unit

7.3.1 Reactor Type

At the Ignalina NPP the water-cooled, graphite-moderated, pressure-tube-type boiling-water power reactors RBMK-1500 are installed. The reactor thermal power is 4800 MW, the electric capacity is 1500 MW. There is a direct steam cycle at the INPP. Saturated steam with pressure of 6.5 MPa, introduced to the turbines, is formed directly in the reactor channels. Two turbine generators, each of 750 MW electric capacity, are installed at the power unit. The fuel assemblies are placed in the individual channels. Graphite blocks are used as the moderator. The reactors use low-enriched fuel and the reloading is performed during the operation process. The fuel of 4 types is currently being used at the INPP:

- Fuel assemblies with fuel of 2% enrichment by U^{235}. Uranium mass in FA is 111.2 kg;
- Erbium FA with Uranium-Erbium fuel of 2.4% enrichment by U^{235} with the concentration of burnable absorber of Erbium (Er_2O_3 dioxide) 0.41% weight. Uranium mass in FA is 111.2 kg;

- Erbium FA with Uranium-Erbium fuel of 2.6% enrichment by U^{235} with the concentration of burnable absorber of Erbium (Er_2O_3 dioxide) 0.5% weight. Uranium mass in FA is 111.08 kg;
- Erbium FA with Uranium-Erbium fuel of 2.8% enrichment by U^{235} with the concentration of burnable absorber of Erbium (Er_2O_3 dioxide) 0.6% weight. Uranium mass in FA is 110.92 kg.

The main heat cycle is identical to the boiling water (BWR) reactor cycle, which is used worldwide and is identical to the heat power plant cycle. However, in comparison with BWR reactors, the RBMK-1500 has a number of unique properties. One of such properties is the possibility to reload spent fuel by the reloading machine on the operating reactor without the power reduction.

Simplified INPP heat diagram is shown in Figure 7.3-1. Water, cooling the reactor (1), passes the core, boils and partially evapourates. Water-steam mixture enters the drum separators (3), located above the reactor. The steam from drum separators enters the turbines (4). Spent steam condensates in the condensers (6). The condensate returns to the drum separator through deaerators (8) via the feed-pumps (9). Water from drum separator is delivered for the core cooling by circulation pumps (10) and there it partially evapourates again.

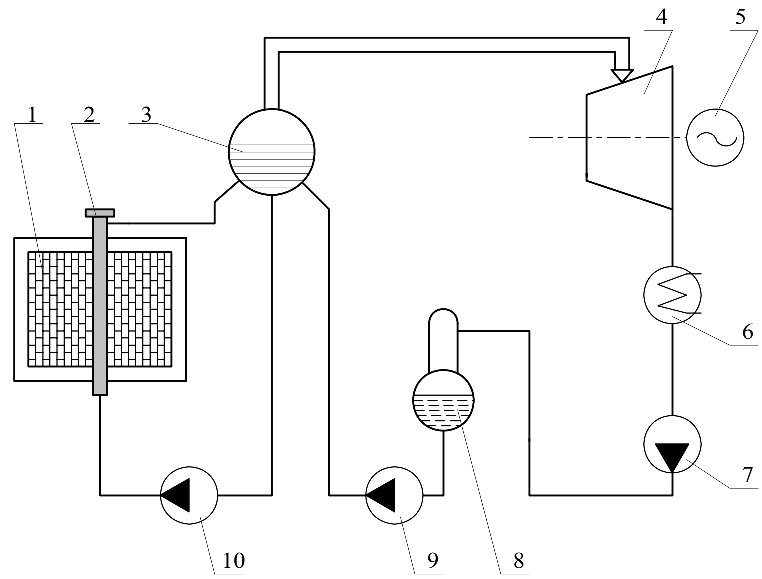


Figure 7.3-1. Simplified INPP heat diagram

1 – reactor, 2 – fuel channel with FA, 3 - drum separator, 4 – turbine, 5 – generator, 6 – condenser, 7 - condensate pump, 8 – deaerator, 9 - feed-pump, 10 – main circulation pump

Currently both Units are shut down, the Units decommissioning projects have been developed and the INPP decommissioning is being performed.

INPP Units life-cycle:

	Unit 1	Unit 2
Start of construction	1978	1980
First criticality condition	4 Oct 1983	11 Dec 1986
Synchronization with energy system	31 Dec 1983	18 Aug 1987
Commissioning	31 Dec 1983	31 Aug 1987
Shutdown (decommissioning)	31 Dec 2004	31 Dec 2009

7.3.2 Current Condition of the Units

According to the decisions of the Government of the Republic of Lithuania, INPP Unit 1 was shut down on 31 December, 2004 [5.84], and Unit 2 was shut down on 31 December, 2009 [5.85].

Condition of Unit 1

Currently the fuel has been completely removed from the reactor and moved to the spent fuel pools. As of 1 July, 2011, 7175 spent fuel assemblies are stored in the spent fuel pools. The works on the implementation of Unit 1 equipment dismantling and decontamination projects are being performed.

The operation of the Unit is being performed according to the INPP Unit 1 operation technological regulations [5.29]. The regulations describe the conditions of safe operation of safety related systems and equipment of Unit 1, which remain in operation at the stage of nuclear fuel removal from the storage pools, when the reactor is completely defuelled.

Condition of Unit 2

INPP Unit 2 was shut down on 31 December, 2009. At the time of the reactor shutdown there were 1653 FA (14 SFA of 2.0% enrichment, 30 SEFA of 2.4% enrichment, 804 SEFA of 2.6% enrichment, 805 SEFA of 2.8% enrichment). In early January 2010 the reactor was put 3 times in critical condition with full and empty CPS cooling circuit in order to evaluate effectiveness of FASS rods, reactor subcriticality, average effectiveness of manual control rods as well as to evaluate the effect of CPS cooling circuit void in conditions when cold reactor is depoisoned. The results of measurements of Unit 2 reactor parameters in condition of cold depoisoned reactor after the final shutdown for decommissioning has shown that all the measured characteristics of the reactor are within the limits set in the INPP Unit 2 reactor facility certificate. During 2010, 19 SFA were removed form Unit 2 reactor.

No cases of emergency removal of irradiated SFA with discharge of fission products from SFA during the removal, leading to excess of safe and normal operation limit for I^{131} activity in MCC water, were noted. Normal operation limit for Cs^{137} activity in MCC water, established by the technological regulations on operation of INPP Unit 2 at the stage of the reactor defuelling has not been exceeded, too.

During Unit 2 reactor defueling the requirements on subcriticality value, established in Nuclear Safety Regulations for NPP reactor units [5.82], were met. Unit 2 reactor subcriticality during the performance of nuclear hazardous works was at least 0.02 for the core condition with maximum fission factor. Neutron-physical properties of Unit 2 reactor are within the limits of designed values, set at the stage of the reactor defuelling.

Currently the fuel is partially removed from the reactor. As of 1 July, 2011, 1335 FA are in the reactor (14 SFA of 2.0% enrichment, 29 Spent Eribium FA of 2.4% enrichment, 710 Spent Eribium FA of 2.6% enrichment, 582 Spent Eribium FA of 2.8% enrichment), 7045 SFA are stored in the storage pools.

Operation of the unit is performed according to the technological regulations on operation of INPP Unit 2 [5.30]. The systems that are currently in operation at INPP Unit 1 and 2 are described in section 7.9.2 "Currently Existing Differences".

7.4 Main Characteristics of SNF Storage Places

Spent nuclear fuel storage at the INPP is performed only in the designed storage places. The SNF storage and management system may be represented in a form of several separate independent subsystems, each of them performs a separate function. The SNF storage and management system consists of the following subsystems:

- SNF transportation within the unit;
- SNF storage after the reactor defuelling;
- SNF cutting into separate fuel bundles and their loading into 102-place transport baskets;
- storage of 102-place transport baskets with the cut FA in the storage pools;
- transportation of SNF outside the unit to the plant's detached storage facility;
- storage of SNF casks in the storage facility for 50 years.

Spent nuclear fuel is stored in Unit 1 and Unit 2 storage pools and at the Dry Storage Facility for Spent Fuel (DSFSF) site.

The principles and conditions of nuclear fuel storage at the INPP correspond to the requirements of Nuclear Safety Regulations for NPP reactor units [5.82]. The continuous control of FA movements is performed. Periodically IAEA and VATESI inspections are carried out at the INPP. Physical inventory of nuclear materials is carried out every year. All nuclear materials are under IAEA safeguards.

7.5 Spent FA Storage Pools

Spent FA (SFA) storage pools are intended for temporary storage of SNF under a protective layer of water and decay heat removal before FA cut in the hot cell or at the long- length fragmentation facility.

Storage pools' compartments and transfer canyons are the concrete tanks, lined with stainless steel sheets of 5 mm thick and filled with chemically purified water. Under the bottom of every storage pool compartment a stainless drain pan, filled with aerated concrete, is mounted for the collection and removal of possible water leakages. Removal of leakages is being performed via drainage pipeline of 89 mm diameter. To prevent unauthorized emptying of storage pools compartments there are no drainage systems installed. Emptying is being performed using special submersible pumps. To prevent the breakdown of the lining in case of FA drop, the bottom of storage pools compartments and canyons is lined with stainless steel sheets of 10 mm thick (excluding compartment 236/1,2). Layout of compartments in the Storage Pools Hall is shown in Figure 7.5-1. Location of Unit 1 and Unit 2 storage pools is the same.

Total surface area of all storage pools compartments of one power unit is 467.7 m^2. Information on the storage pools compartments is presented in Table 7.5-1.

TABLE 7.5-1

Compartment number	Surface area, m^2	Volume, m^3	Pool floor level, m	Spill-over level, m
234	41.4	800	+7.70	+24.60
236/1	38.6	669	+7.20	+24.60
236/2	42.0	727	+7.20	+24.60
336	42.2	507	+13.00	+24.60
337/1	36.7	441	+13.00	+24.60
337/2	36.7	441	+13.00	+24.60
338/1	38.9	391	+13.00	+24.60
338/2	10.3	81	+17.00	+24.60
339/1	42.2	507	+13.00	+24.60
339/2	37.0	444	+13.00	+24.60
157	17.4	134	+7.20	+24.60
235	84.3	1024	+7.20	+24.60

Storage pools compartments have different purposes:

- compartment 234 is designed for SFA location before their delivery for the cutting in a hot cell;
- compartments 236/1, 236/2 are designed for SFA location after removal from the reactor;
- compartments 336, 337/1, 337/2, 339/1, 339/2 are designed for SFA storage in 32M baskets after their cutting;
- compartments 338/1, 338/2 are designed for the performance of operations of 32M baskets loading into the cask;
- compartment 157 - transfer canyon is designed for the transportation of FA from the Reactor Hall to the Storage Pools Hall;
- compartment 235 - transfer canyon is designed for the transportation of FA and 32M baskets with SFA between the pools.

There are three options to store individual SFA removed from the reactor. They are as follows:

- "A" (rectangular lattice) – SFA are stored in baskets with 250x160 mm spacing;
- "B" (rectangular lattice) – SFA are stored uncased and placed with 250x100 mm spacing;
- "C" (triangular lattice) – SFA are stored uncased and placed with 136x136x112 mm spacing.

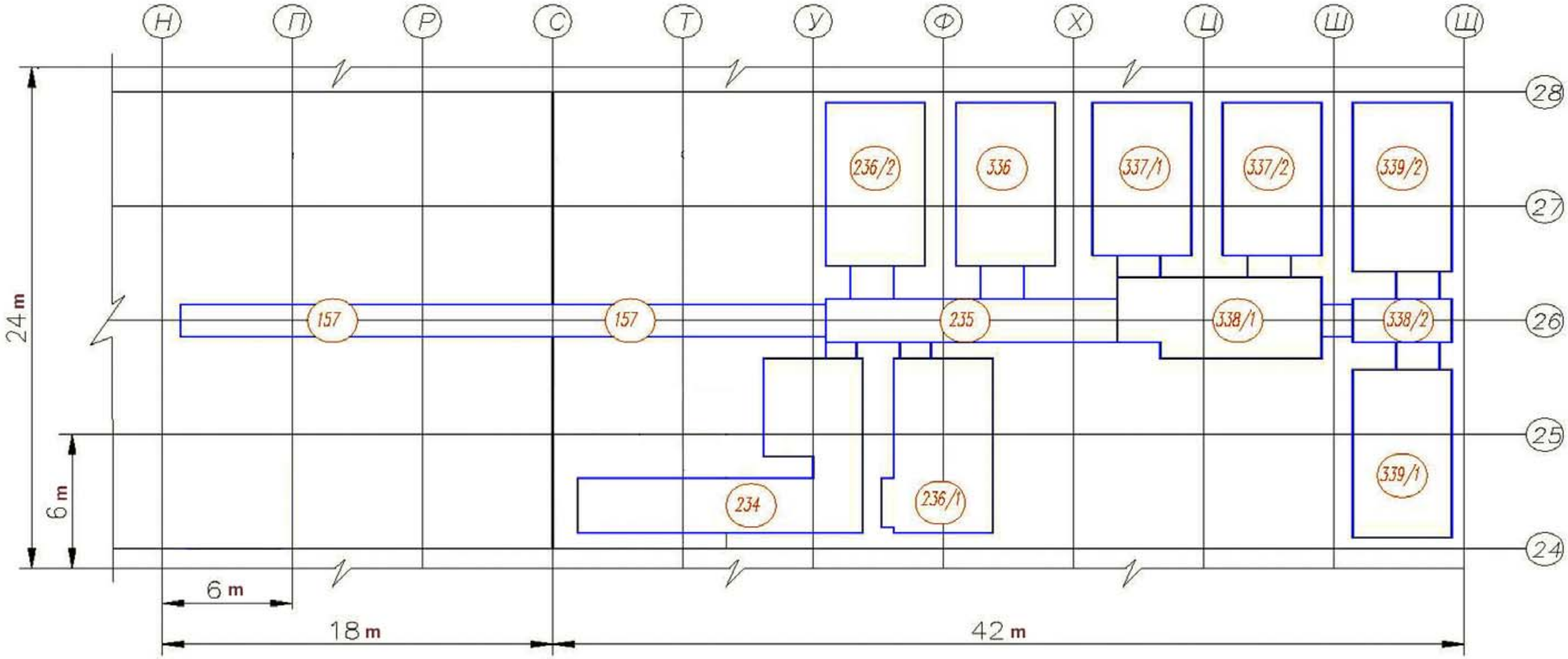


Figure 7.5-1. Layout of compartments in the Storage Pools Hall

Compartments 236/1,2 – SFA storage pools.
Compartments 336; 337/1,2; 339/1,2 – storage pools for the storage of cut SFA in 102-place baskets.
Compartments 338/1,2 – transport casks loading pools.
Compartments 157, 235 – transfer canyon.

7.6 Storage Pools SNF Cooling System

Water level and temperature in the Spent Fuel Pools (SFP) are maintained by Pump and Heat Exchanger Unit (PHEU).

Spent Fuel Pool's PHEU consists of four centrifugal circulation pumps AX 160/29, providing the discharge capacity of 44.4 kg/sec (160 m3/h), head pressure is 274.4 kPa (29 m water column), and three heat exchangers 400 ТНГ-10-М1/20Т4-1.

The main design characteristics of PHEU are the following:

- cooling capacity 4000 kW;
- cooled water flow rate 400 t/h;
- service water flow rate 480 t/h.

Water cooling in the pools is being performed as follows: water from the storage pools is poured into the pipes, tied-in to the pools at the level of +23.20 m and delivered by natural flow to the heat exchangers. After cooling up to +30°C, water is being returned via pumps through the regulation point to the bottom part of the storage pools.

Parameters and Conditions of Normal and Safe Operation of SFP and Permitted Deviations for Unit 1 and Unit 2

1. Subcriticality in the Fuel Storage Pools of at least 0.05 is provided by the designed arrangement of the fuel.
2. The temperature of the cooling water in the Spent Fuel Pools is maintained in the range of 20-50°C. Safe operation limit is 60°C. Storage pools cooling is performed by SFP PHEU consisting of 4 pumps, 3 heat exchangers, pipelines and fittings.
3. Water level in the Fuel Storage Pools is maintained in the range of 950-650 mm from the FSP floor. Safe operation limit is 1000 mm from the FSP floor (level +24.20 m). Periodical FSP makeup is provided by pumps through the filling and makeup point, which ensures automatic maintenance of water level in the pools in the range of 950-850 mm from the FSP floor.

Safe operation limits of FSP water temperature and level initially were not foreseen in the INPP design, and they were established in 2011 on the request of VATESI in accordance with the justification of safe operation limits of FSP during the fuel storage in the pools [5.39]. After that they were included into the requirements of technological regulations on Unit 1 [5.29] and Unit 2 [5.30] operation.

For the monitoring of these safe operation limits in case of beyond design-basis accident "Full Blackout of INPP Auxiliaries" (with a failure of all diesel generators (DG)) an additional modification has been planned at INPP and is being at the stage of implementation. It would ensure power supply from the mobile DG to the Unit 1 and Unit 2 storage pools level and temperature survey meters [5.74].

7.7 Dry Storage Facility for Spent Nuclear Fuel (DSFSF)

Dry Storage Facility for Spent Nuclear Fuel (DSFSF) is located on the INPP site about 1 km from the power units and about 400 m from Lake Drukshiai (see Figure 7.1-5). DSFSF site is fenced round the perimeter by protective reinforced concrete wall and three-row security fence with alarm system. Behind the protective reinforced concrete wall there are technological facilities, ensuring the safe operation of the storage facility. Between the rails of gantry crane there is an area for CASTOR RBMK and CONSTOR RBMK-1500 casks storage in an upright position.

SNF casks storage area is a monolithic reinforced concrete slab of 500 mm thick and 23.5×105 m size, which lies on a specially prepared base. Concrete of strength grade B25 and water resistance W6 is applied for the slab. The slab has a slope to drain precipitations into the wells. 300 mm thick drainage of no-fine concrete and waterproofing of the polyethylene-rubber bitumen composition is installed under the slab. Special supporting pedestals are made on the slab to set the casks, six pedestals for each cask.

Storage facility is designed for storage of 120 spent nuclear fuel casks (20 CASTOR RBMK casks and 100 CONSTOR RBMK-1500 casks) during 50 years. Currently 20 CASTOR RBMK casks and 98 CONSTOR RBMK-1500 casks with 51 fuel assemblies in each cask are stored at the DSFSF site.

Cartogram of SNF casks layout at the DSFSF site is shown in Figure 7.7-3.

Spent nuclear fuel stored in the storage pools at the end of 5 years since the date of the reactor defuelling can be transported outside the unit to the SNF storage facilities.

Spent nuclear fuel loaded into the casks and transported to the DSFSF site meets the following criteria:

- initial enrichment: 2.0% uranium isotope U^{235} by weight;
- burnup: not more than 20 MW day/kg of uranium from the mean cask burnup value;
- the criterion of tightness of SNF assemblies loaded into the cask: the growth of Cs^{137} activity in cask water shall not exceed 5×10^{-6} Ci/kg;
- total decay heat of SNF loaded into the casks is not more than 6.1 kW.

The INPP DSFSF meets the following requirements:

- provision of SNF safety for 50 years;
- provision of possibility of SNF transportation outside the storage facility at any time and from any storage place;
- elimination of ambient air impact on structural materials of fuel assemblies;
- provision of passive heat removal from SNF stored in the casks;
- provision of quick identification of source of detected radioactive contamination;
- stability of the casks against the external impacts (aircraft crash, air shock wave, flying objects, earthquakes, hurricanes, tornado).

7.7.1 CASTOR-RBMK Cask

CASTOR-RBMK cask is developed by GNB company (Germany) and is designed for long-term storage of RBMK-1500 reactor spent fuel assemblies, which are placed in 32M type transport basket. 102-place basket is designed to ensure the fixed position of SFA during their transportation and storage.

The cask is being filled with helium which ensures the corrosion protection, improves the passive heat removal and provides the possibility to perform periodical testing of cask sealing integrity during its long-term storage. The structure of the cask ensures the protection from gamma and neutron emissions. In case of dry storage of SNF in CASTOR cask, no active decay heat removal systems are required. Decay heat is removed through the body of the cask and spreads into the environment due to the natural convection and radiation.

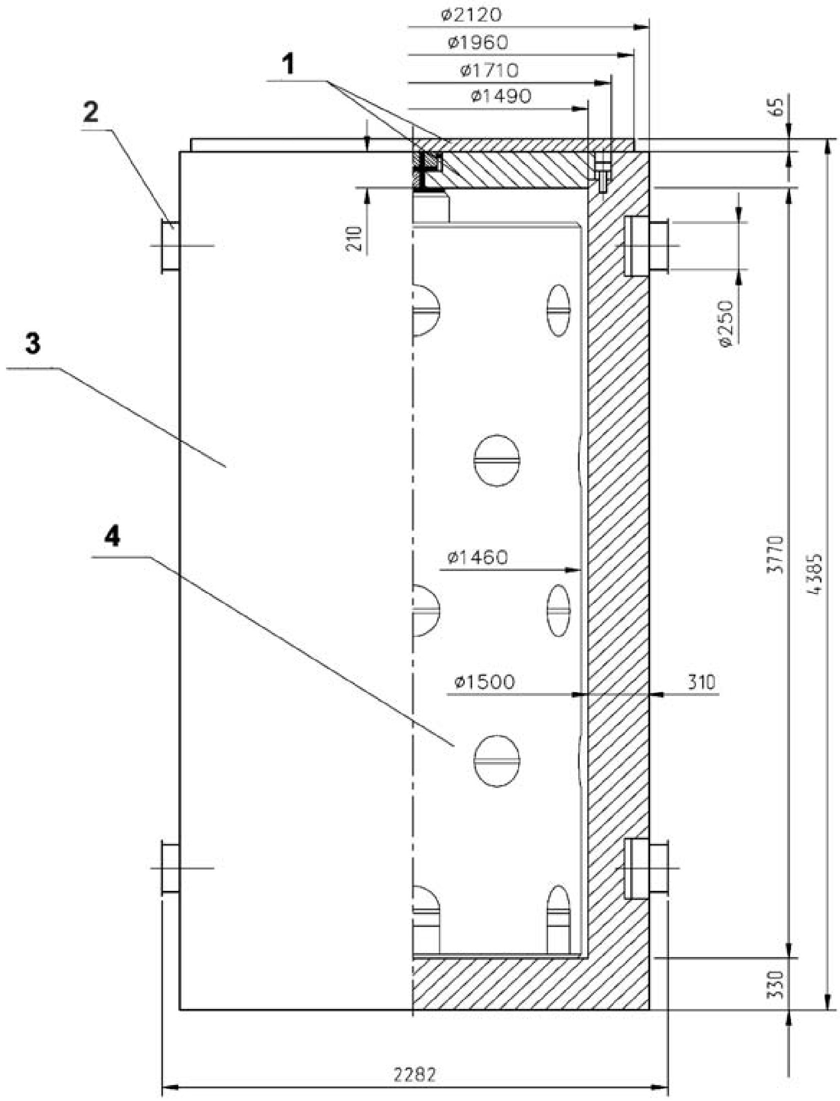


Figure 7.7-1. CASTOR-RBMK cask

1 – lids, 2 – strapping element, 3 – cask body, 4 – 102-place basket.

The cask body is casted entirely of malleable cast iron. The carbon, which is a good neutron moderator, distributed homogeneously in the material the cask is made of. Herewith an optimal mechanical strength and thermal conductivity of the walls of the casks are ensured. The cask is sealed by two lids made of carbon steel with anticorrosive protective coating. Outer and inner surfaces of the cask are protected against corrosion by several layers of paint. Outside the cask is protected by multi-layer epoxy coating.

Dimensions of the cask are: outer diameter – 2120 mm, height – 4385 mm. The weight of loaded cask is 71840 kg.

7.7.2 CONSTOR RBMK-1500 cask

CONSTOR RBMK-1500 cask is developed by GNB (Germany) and CKTI (Russia) companies and is designed for long-time storage of RBMK-1500 reactor spent nuclear fuel which is placed in 32M type transport basket. The cask is being filled with helium.

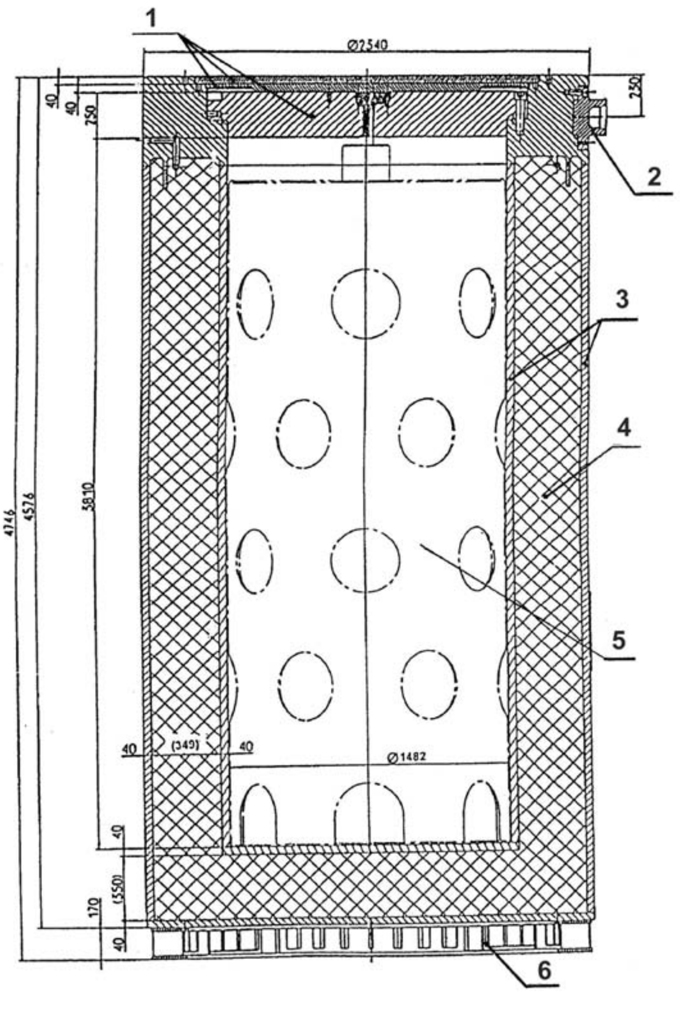


Figure 7.7-2. CONSTOR-RBMK-1500 cask

1 – lids, 2 – strapping element, 3 – inner and outer liners, 4 – concrete, 5 – 102-place basket, 6 – shock-absorber

The cask is a metal-concrete container with steel lids. Cask body is made of two metal cylindrical liners, placed one inside the other and welded to the upper forged ring. In the bottom part of the cylindrical liners there are welded bottom plates. Annular space between the liners of the body, as well as the space between the bottom plates is filled with heavy concrete with reinforcement steel bars. The concrete in the cask serves as a structural and protective material, and reinforcement strengthens the structure, increases the heat conduction of concrete walls and ensures more uniform heat removal through the entire cask body surface. The container is being sealed by three lids.

Dimensions of the cask are: outer diameter – 2340 mm, height – 4746 mm. Weight of the loaded cask is 84500 kg.

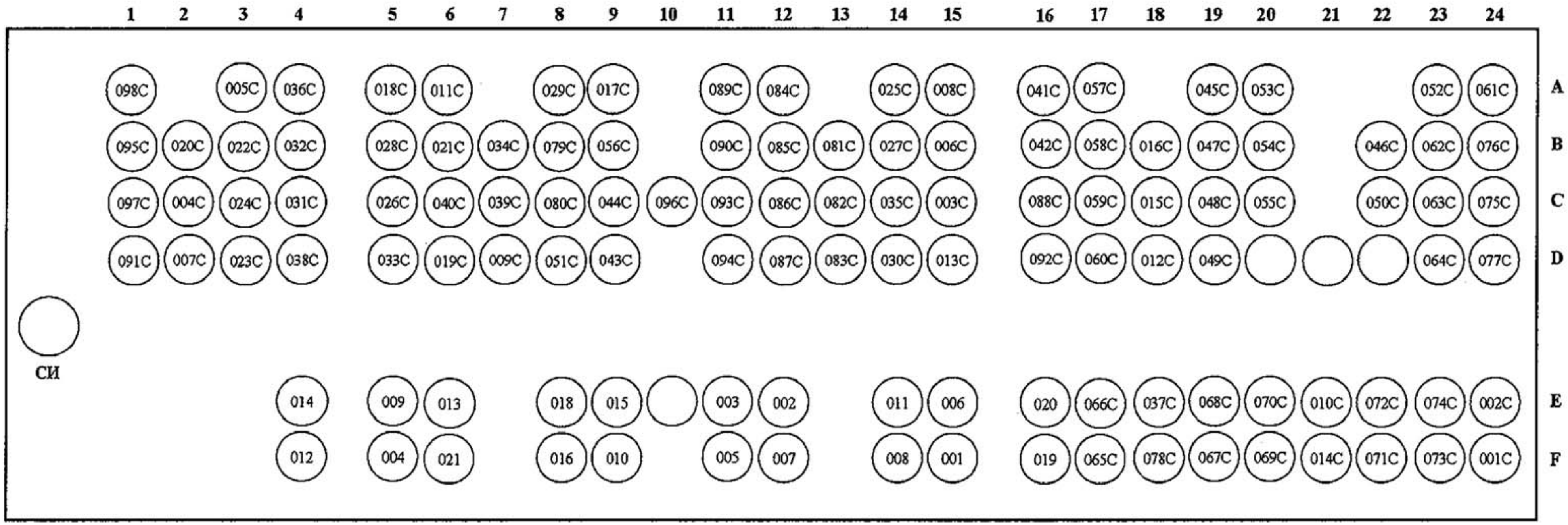


Figure 7.7-3. Cartogram of SNF casks layout at the DSFSF site as of 1 July, 2011

CONSTOR RBMK-1500 cask is marked at the diagram by number with letter “C” (098C, 095C, 097C)
CASTOR RBMK cask is marked at the diagram by number without letter (014, 012, 009)
СИ – cask inspection desk

7.8 Interim Spent Fuel Storage Facility

Interim Spent Fuel Storage Facility (ISFSF) is currently under construction at the INPP. The commissioning of ISFSF is scheduled in 2012. It is located at the INPP site, at a distance of approximately 550 m from the power units and is designed for long-term storage of 201 CONSTOR® RBMK-1500/M2 casks with spent nuclear fuel. The storage facility consists of several buildings:

- main processing building;
- administrative service and physical protection service building;
- motorway and railway transport control station.

The Main Processing Building is designed for the storage of 201 CONSTOR® RBMK-1500/M2 casks. There are three main areas in the building: Acceptance Hall, Storage Hall and Fuel Inspection Hot Cell**.** Casks transportation operations are being performed using the bridge crane of 125/25 t capacity with appropriate vertical traverse. ISFSF main building plan is shown in Figure 7.8-1.

Acceptance Hall consists of cask transportation corridor, cask service station, control room, personnel entrance and exit rooms, radiation monitoring room, IAEA inspections room and other auxiliary rooms.

Storage Hall is designed for CONSTOR® RBMK-1500/M2 casks storage and monitoring. Storage Hall is separated from the Acceptance Hall by biological shielding with sliding gates for casks transportation. Decay heat removal is achieved by natural ventilation through holes on the side and top of the Storage Hall. The arrangement of the casks will be performed with minimum space of 70 cm longwise between them. This will ensure necessary cooling. To protect the casks during transportation from the cask service station to the hot cell a special shock-absorber will be installed in the floor.

Fuel Inspection Hot Cell is designed for the inspection and repackaging of SNF to the new cask, in case it will be found out that the cask is damaged. Hot Cell is equipped with exhaust ventilation system with primary and secondary fine filters for air cleaning, preventing the spread of radionuclides to the environment.

The building was designed and built taking into account seismic loads and explosion shock wave impact. Preparation and calculations of the building resistance to blast loads were performed in accordance with existing regulations. Thereat initial data on seismic loads and explosion shock wave impact are taken into account. Blast load is understood as a dynamic load impact converted to a static load depending on the frequency of vibrations of building structures and duration of exposure.

The foundation of the storage facility is made of reinforced concrete slab of 135 cm thickness. Hot Cell, Acceptance Hall and Storage Hall are located on a single foundation slab.

Administration and Physical Protection Service Building will be combined with the motorway and railway transport entrance compartments and will be integrated into the safety protection system at the site, adjacent to the transport control area. The building was designed in accordance with the performed functions.

Motorway and Railway Transport Control Station will be made as an open structure made of steel columns. It will be lined with steel sheets outside and will have a roof for the protection from adverse weather conditions. Freight transport inspection box made of concrete elements will be installed at the area.

ISFSF meets the following requirements:

- provision of SNF safety for 50 years;
- provision of possibility of SNF transportation outside the storage facility at any time and from any storage place;
- elimination of ambient air impact on structural materials of fuel assemblies;
- provision of passive heat removal from SNF stored in the CONSTOR® RBMK-1500/M2 casks;
- stability of the casks to the external impacts (aircraft crash, air shock wave, flying objects, earthquakes, hurricanes, tornado);
- provision of quick identification of the source of detected radioactive contamination.

SNF may be sent for the storage to ISFSF only after 5 years of storage in the storage pools. CONSTOR® RBMK-1500/M2 protective SNF casks, meeting the established transportation criterion are being accepted for the storage at ISFSF site. Inspection of SNF compliance with the transportation criterion (casks lids leak tightness, surface contamination, etc.) is performed at the power units after SNF loading into the cask. Inspection results are recorded in the cask certificate.

When operating the storage facility following values shall be continuously monitored:

- gamma-radiation exposure dose rate and neutron-radiation dose rate in Cask Storage Hall and along the perimeter of the site, applying the radiation monitoring system;
- emissions of radioactive substances into the atmosphere through a ventilation stack and from the Cask Storage Hall;
- individual external dose of ISFSF maintenance personnel, using personal dosimeters.

Inventory of nuclear materials stored in the storage facility is being performed by ISFSF personnel in accordance with the procedures, determining nuclear materials inventory requirements. The monitoring of nuclear materials storage in the storage facility is provided by the following measures:

- storage facility physical protection system;
- sealing of each SNF cask with special seals by the IAEA inspectors when the cask is placed to the storage place.

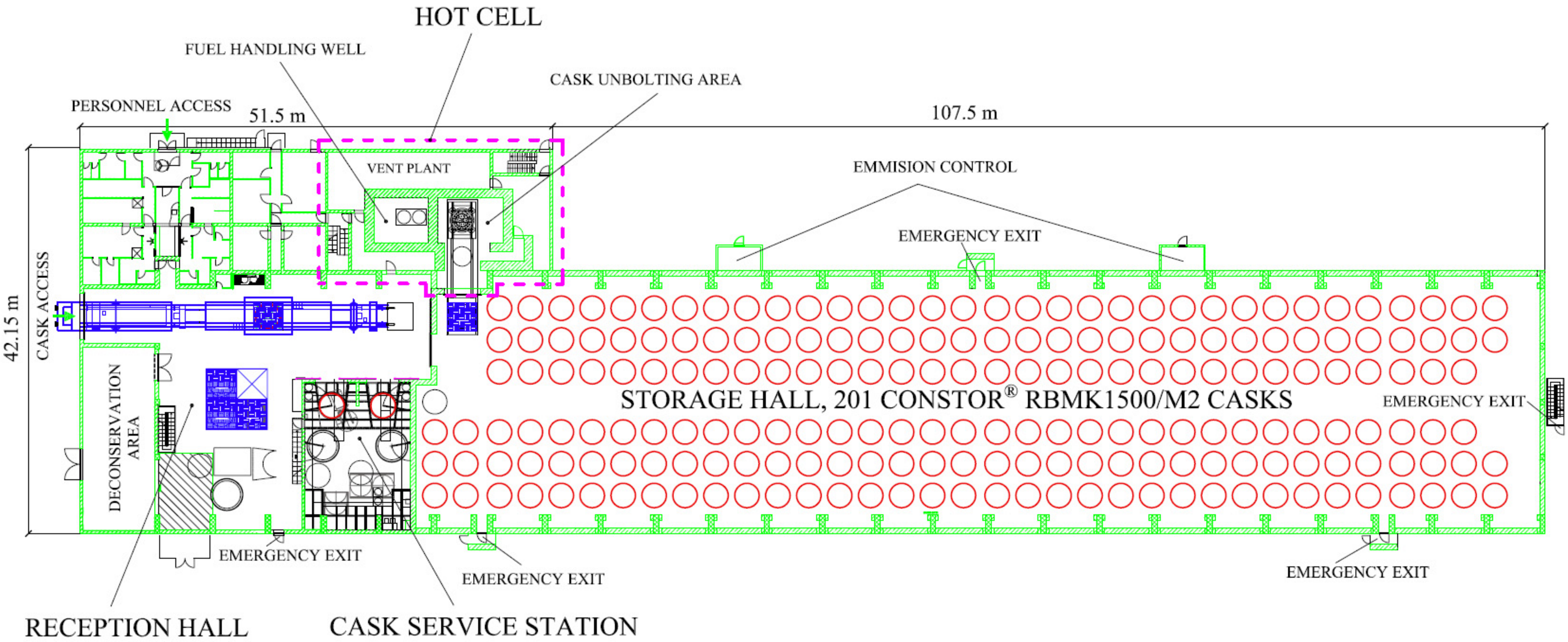


Figure 7.8-1. ISFSF main building plan

7.8.1 CONSTOR® RBMK-1500/M2 Cask

CONSTOR® RBMK-1500/M2 cask was developed by GNB company (Germany). It is designed to store RBMK-1500 reactor spent nuclear fuel in the Interim Spent Fuel Storage Facility within 50 years. The cask is designed for storage of undamaged, damaged and experimental FE. The cask is designed to accommodate 32M basket and ring basket, which are used to create geometric lattice, which provides nuclear safety under normal and emergency conditions of spent nuclear fuel storage as well as for adequate heat removal from the SFA.

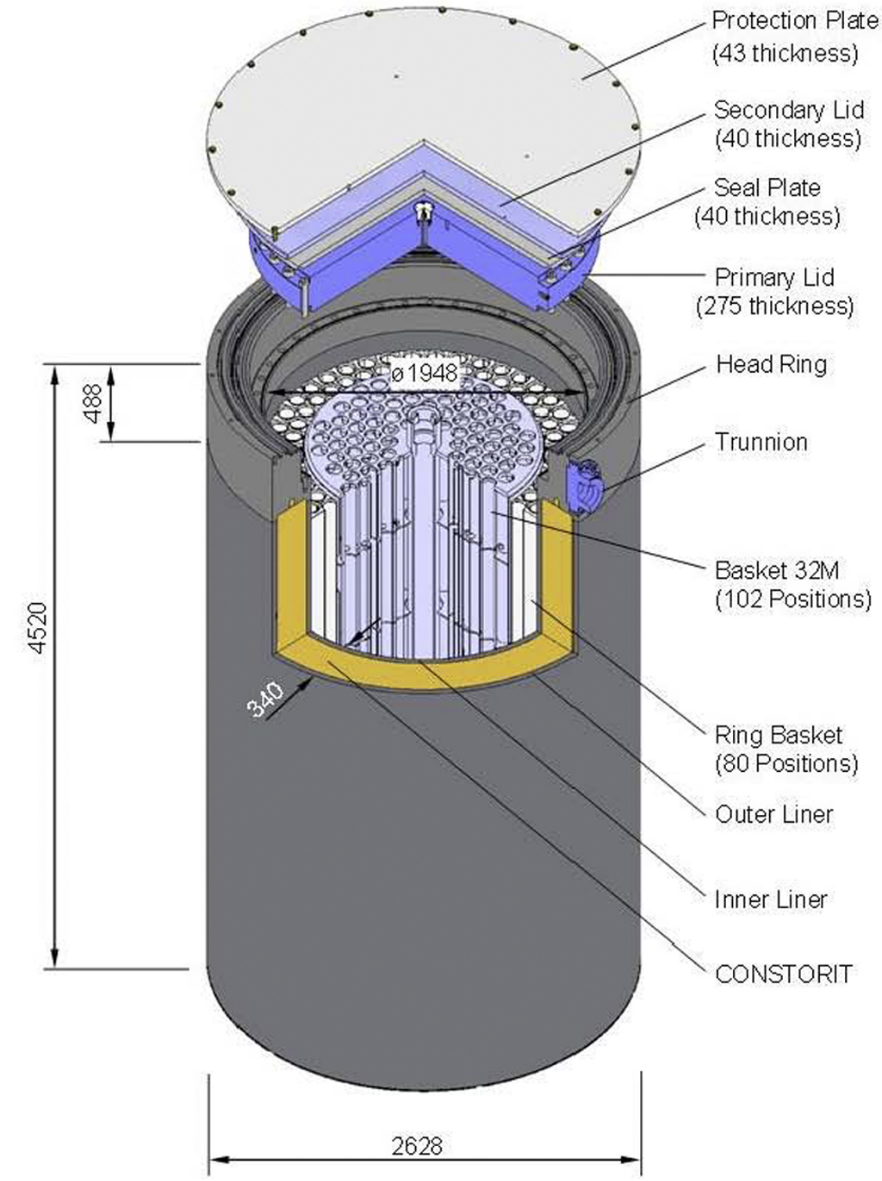


Figure 7.8-2. General view of CONSTOR® RBMK-1500/M2 cask with 32M basket and ring basket without concrete lid

Cask body is made of carbon steel and has a multilayer structure with inner and outer liners. A layer of shielding material CONSTORIT, made of granulated material with cement grout, fills the spaces between inner and outer liners and between the bottoms. Inner surface of the cask is covered with nonorganic two-component zinc-silicon layer. This cover is suitable for underwater loading. The outer surface of the cask is covered by the multilayer epoxy resin. The cask is being leak tightened using three lids. A protective plate, made of carbon steel, is being fastened to the upper part of the cask. To improve neutron-radiation shielding through the system of lids, a concrete plate is being installed at a distance of 25 mm over the protective plate.

At the factory, a ring basket is being installed into the cask body. Also the delivery of casks with modified ring basket and damaged fuel basket is foreseen.

After SNF loading and vacuum drying, the casks are being filled with noble gas. If the cask was loaded with leaking fuel bundles, the cask is loaded with over-pack sleeve with drying agent applied for collection of a small amount of water, which may stay after vacuum drying. Leak tightened cask containment prevents air access. Thus, a significant deterioration of materials due to corrosion is impossible. Cask integrity is ensured even after long storage phase.

The cask has following dimensions: outer diameter – 2628 mm, height – 4520 mm. Weight of loaded cask during the transportation from the power units to the ISFSF – 112 024 kg. Weight of loaded cask in the storage position in ISFSF after the installation of sealing lids and concrete plate is 115 656 kg.

7.9 Differences of the Units

7.9.1 Design Differences

Between 1978 and 1987 two identical power units with RBMK-1500 reactors were built at the Ignalina NPP. There were no significant differences in the design of the power units, but there are some differences in the safety related systems, which have been described in the INPP Unit 1 Safety Analysis Report [5.2]:

Accident Localization System (ALS)

All inner surfaces of ALS compartments at Unit 2 have leaktight metal lining. At Unit 1 only Low Water Lines compartments, Leak-Tight Confinement floors and Accident Localization Tower floors are lined with leaktight metal sheets.

Ventilation Systems

WZ52-55 exhaust ventilation systems of compartments, adjacent to the ALS compartments at Unit 1, besides regular power supply are provided with reliable power supply from diesel generators. This is done in order to locate possible gas leakages from ALS leak tight circuit during an accident and simultaneous loss of regular power supply.

WZ52-54 ventilation devices at Unit 2 do not have additional power supply from the sections of reliable power supply, because Unit 2 ALS gas leakiness value is significantly less than at Unit 1.

Reactor Control and Protection System (CPS)

There are some differences in Unit 1 and Unit 2 CPS rod drives electric power supply circuit.

Fuel Channels

There are some differences in the design of isolation plugs, sealing the fuel channels. At Unit 2 FA suspension RBMK-K15сб.15 with a fixing screw plug is used, while at Unit 1 suspension RBMK-K5сб.15 with ball locking plug is used.

GMCU, Decay Chamber, GDPS

There are some differences in Gas Discharge Purification Systems (GDPS) of Units 1 and 2. Unit 2 gas decay chamber is underground metal structure, of 3500 m^3 volume, made of pipes, diameter of 3500 mm. Unit 1 gas decay chamber is underground concrete structure, consisting of two compartments each of 1500 m^3 volume.

At Unit 2 gaseous mixture discharge pipeline after turbine generator GMCU (Ø 200 mm) is connected with gas discharge pipeline after reactor cavity (Ø 150 mm) and enters the general compartment of decay chamber by general pipeline (Ø 200 mm). As distinct from Unit 1, the separation of decay chamber into compartments for GMCU and for reactor cavity is not foreseen.

There are some differences in Unit 1 and 2 connection diagrams of activity suppression plant carbon adsorbers.

Unit 2 GDPS vacuum pump facility consists of 3 pumps VVN1-12, each of 750 nm^3/h capacity. The capacity of one of the pumps at Unit 1 is 1500 nm^3/h.

Reactor Service Water Supply

At Unit 2, as distinct from Unit 1, there is no shunt pipe (Ø 1000 mm) between pressure water conduits VF23 and VF24 in Unit D2 room 003. Also there is no connection of these water conduits with industrial site water conduits via pipeline (Ø 800 mm).

Reactor Intermediate Cooling Circuit (ICC)

The differences between the power units are in the position of valves on the heads of the pumps ICC 1,2. At Unit 1 each ICC 1,2 pump (operating and standby) pressure pipeline valves are open, electrical circuits are assembled.

At Unit 2 operating pump pressure pipeline valves are open, electrical circuits are assembled. Standby pumps pressure pipeline valves are closed. They open automatically in case pumps are actuated and close when pumps are disabled. Electrical circuits are assembled. This difference is determined by the different design solutions, i.e. at Unit 1 automatic operation of the valves was not foreseen.

7.9.2 Currently Existing Differences

Currently both power units are shut down. The differences of the power units are determined by their condition.

Unit 1 reactor was defuelled on 14 December, 2009. Part of the fuel (988 fuel assemblies) was transported to Unit 2 for afterburning in the reactor. Part of spent fuel was transported for storage to DSFSF, part of SFA is stored in Unit 1 storage pools. As of 1 July, 2011, 7175 SFA are stored in Unit 1 storage pools [5.76].

Currently the works on implementation of equipment dismantling and decontamination projects have been started at Unit 1. Safety related systems, performing the functions of radiation protection of the personnel and population remain in operation, as well as Unit 2 reactor and MCC makeup system transit pipelines.

At Unit 2 all the systems, necessary for the reactor and storage pools maintenance in a safe condition, are in operation. To date a part of spent fuel is unloaded from the reactor to the storage pools. As of 1 July, 2011, 1335 FA are still in Unit 2 reactor and 7045 SFA are stored in the storage pools. [5.76].

The main systems which are currently under operation at Unit 1 and 2 are shown in Table 7.9-1.

TABLE 7.9-1

No	Unit 2	Unit 1
1.	Reactor and system of steam and gas discharge from the reactor cavity	Reactor and system of steam and gas discharge from the reactor cavity
2.	Reactor power monitoring and control system	-
3.	Coolant flow through FC regulation system	-
4.	Fuel reloading system	Fuel reloading system
5.	Coolant main circulation circuit	Coolant main circulation circuit
6.	Live steam pipelines	Live steam pipelines
7.	Service water supply system	Service water supply system
8.	Reactor blowdown and cooling system	Reactor blow-down and cooling system
9.	MCC bypass purification system	MCC bypass purification system
10.	MCC and reactor makeup system	MCC and reactor makeup system
11.	-	Unit 2 reactor and MCC makeup system transit pipelines
12.	LSW, SCC, CPW consumers makeup system	LSW, SCC, CPW consumers makeup system
13.	Reactor maintenance cooling system	-
14.	Spent fuel storage system	Spent fuel storage system
15.	Protective casks handling system	Protective casks handling system
16.	Solid radioactive waste treatment system	Solid radioactive waste treatment system
17.	Spent ChWPS filtering materials acceptance and unloading system	Spent ChWPS filtering materials acceptance and unloading system
18.	Drainage waters acceptance and pumping out system including leaktight compartments	Drainage waters acceptance and pumping out system including leaktight compartments
19.	Radiation safety automated monitoring system	Radiation safety automated monitoring system
20.	Control of elements of systems important to safety	Control of elements of systems important to safety
21.	Centralized monitoring information computer system TITAN	Centralized monitoring information computer system TITAN
22.	System of ALS leaktight compartments	System of ALS leaktight compartments

No	Unit 2	Unit 1
23.	Compartments ventilation system	Compartments ventilation system
24.	Power supply system	Power supply system
25.	Power plant fire extinguishing system	Power plant fire extinguishing system
26.	Additional Hold-Down system	-

Comments to Table 7.9-1 are given below.

Unit 1 Reactor is completely defuelled, therefore the reactor power monitoring and control system, coolant flow through FC regulation system, the reactor maintenance cooling system do not operate.

Service Water Supply (SWS) is used for INPP mechanical equipment cooling by lake (service) water under normal operation and emergency operation mode. INPP SWS was designed independent for each power unit and consists of inlet and outlet canals, pumps, pressure and drain pipelines and fittings. After final shutdown of the power units, SWS of Unit 1 and 2 were merged.

Power Supply Systems of the Units have the following differences**:**

Unit 2 auxiliaries have operational and standby power supply system, fed from the external source – electrical power supply system (grid), and emergency power supply system, consisting of:

- reliable power supply subsystem of 6 channels, fed from diesel generators;
- uninterruptible power supply subsystem of 6 channels, fed from diesel generators and accumulator batteries.

Unit 1 auxiliaries have only operational and standby power supply system, fed from the external source – electrical power supply system (grid).

Unit 1 Compartments Ventilation System after its shutdown and the reactor defuelling was optimized, because technological equipment and pipelines decay heat reduced, leading to the change of parameters of ambient air in the serviced compartments. Currently ventilation of separate compartments in design volume is not necessary. Ventilation system optimization foresees the ventilation system equipment energy consumption decrease due to the reduction of air flow in separate compartments. After the monitoring in order to decrease energy consumption and unproductive expenses, new configurations of power units ventilation system were developed.

Additional Hold-Down System (AHS) is intended for absorber injection to the storage pools in order to maintain long-term subcriticality of spent fuel in the storage pools in case of threat of self-sustaining chain reaction. Gadolinium nitrate hexahydrate which has a large neutron-capture cross-section and high solubility in water is used as the absorber. There is AHS only at Unit 2. With a view to the possibility to use this system for Unit 1 a justification of use of S19 strategy “Supply of Absorber into Emergency Storage Pool” for Unit 1 has been developed [5.41].

7.10 Significant Results of PSA

INPP Probabilistic Safety Analysis (PSA) was started in 1991 within the frames of a joint project “Barselina” between Lithuania, Russia and Sweden. The ultimate goal of the project was to create a common vision and a common framework for the risk assessment of most serious accidents in RBMK-type reactors.

In the course of the project a full-scale model of INPP Unit 2 PSA was developed to identify opportunities to increase the Power Plant safety in terms of risk. The probabilistic analysis methodology has been applied to channel-type RBMK reactors. To approximate the model to reality, several deterministic tests were performed and database on nuclear power plants with RBMK reactors was developed and used. The overall concept of analysis of reactors of this type was developed.

Experience and information obtained at different phases of the implementation of INPP PSA were used as source material for other projects - SOEI, SAR and RMMS. According to the results of PSA [5.26] some modifications which are shown in Table 7.9-2 have been proposed to introduce.

TABLE 7.9-2

No	Description of proposed improvement	Proposal implementation date
1.	Reduction BRU-B capacity	1994
2.	Redundancy implementation in low salted water supply system	1995
3.	Change of the normal state of valves between EFWP and DS	1995
4.	Closure of MCP bypass lines	1996
5.	Increase of capacity of steam and gas discharge from the reactor cavity	1996
6.	EFWP/ECCSP pressure valves supply change	1997
7.	Improvement of procedure and decrease of intervals between the inspections of ECCS PH check valves and ECCS MCP check valves	1997
8.	Improvement of ECCS operation algorithm in order to reduce the necessity of operator intervention	1997
9.	Implementation of accident procedures with the provision of emergency water supply and pressure release from MCC	2000

The obtained quantitative results are partly based on the power plant data and partly on the generalized data, mostly the western ones. PSA results first of all are intended to show not the absolute risk level, but topography of the risk and provide a basis for identification of the dominant risk components and improve safety of the power plant.

The overall results show that the probability of the final condition "violation" is about 10^{-2}, herewith a single FC flow blocking is dominated. This probability is based on operational data. There were 3 similar cases on RBMK reactors. It shall be noted that according to the core damage condition classification, accepted in the PSA, the condition "violation" is a very modest effect.

The total probability of final conditions "damage" and "failure" is about 10^{-5}. This is the same level as for the condition "core damage" for the western reactors.

The results, obtained during the implementation of PSA, were used for the development of the INPP Unit 2 safety analysis reports, the INPP beyond design-basis accidents list and beyond design-basis accidents management procedures.

8 PART 2. ASSESSMENT OF EXTREME SITUATIONS

8.1 Chapter 1. Earthquake

8.1.1 Design Data

8.1.1.1 Power Plant Design-Basis Earthquake

Special researches on study of seismicity of the Ignalina Nuclear Power Plant site were carried out in 1988.

In 1988 after completion of design and construction of both power units, “Earthquake Resistant Nuclear Power Plants Design Standards”, PNAE G-5-006-87 [5.80] were brought into force. If the construction area was not specified on map OCP-87 (as an area of the Ignalina Nuclear Power Plant site), the values of intensity of the design-basis earthquake (DBE) shall be accepted as 5 points, while the maximum calculated earthquake (MCE) equal to 6 points for category II soils.

According to the results of the special researches the Instrumental Researches Report [5.9] was issued which includes summary data about the geological and tectonic structure as well as seismicity of the Ignalina Nuclear Power Plant site. In 1988 the governmental committee concerning the issue of seismicity was established where the leading seismicity and seismotectonic experts were involved. Historical records starting from the year 1616 were assessed and an attempt was made to assess these events according to scale MSK-64. The intensity of a number of events, to which the intensity of more than 6 points was previously attributed, was called in question. The conclusion of the committee was approved by the USSR Academy of Sciences. According to the conclusion of the committee, the estimates of possible maximum magnitudes of earthquakes and their maximum intensity, performed according to the determined regularities, are presented in the report. Thus, two concepts were taken into consideration: the concept of connection of the earthquakes focus with active tectonic zones, and especially, with their intersection nodes, and the concept of the earthquakes focuses diffusion. It was accepted that the earthquakes with magnitude M=4.5÷4.6 are referred to the fracture zones of the first rank in the territory of the Baltic countries, while the earthquakes with magnitude M = 4.75 refer to the intersection nodes of the first and second rank zones. As follows from the report [5.9], (Table 3.2., page 27), the consideration of the most unfavourable conditions (the assumption about the possibility of the focus directly under the site) in case of the conservative evaluation of values of maximum magnitudes leads to the conclusion that in case of local earthquakes their maximum intensity on the category II soils will be 6 points according to scale MSK-64.

Besides the local earthquakes, the Ignalina Nuclear Power Plant site can undergo shakings from the remote earthquakes of the Carpathian area (depth of the focuses is about 120 km, distances are about 1300÷1400 km) and the Scandinavian focuses like earthquake in Skagerak in 1904. The maximum force of shakings on the Ignalina Nuclear Power Plant site from the focuses of the remote Carpathian and Scandinavian earthquakes will not exceed 4÷5 points according to MSK-64 scale.

The engineering-geological works, researches of mechanical and physical properties of the soils, both dynamic and static penetration tests and tests by static loads using special devices were performed. The main part of the work consisted of the instrumental researches – seismic investigations, as well as seismological observations of microoscillations and earthquakes.

After completion of all investigation works, the calculated quantitative characteristics of expected seismic impacts were prepared. The calculated accelerograms and spectral characteristics of expected ground vibrations were obtained taking into account both actual records of strong earthquakes and by using synthetic accelerograms in accordance with the expected oscillations strengths at two levels of probability.

In order to assess possible seismic impacts of the local earthquakes on the soils of foundations of the Ignalina Nuclear Power Plant Units 1 and 2, the maps of distribution of categories II and III soils were compiled and appropriate calculations and modelling were carried out.

The main result of microzoning works is the conclusion that the expected intensity of seismic impacts on categories II÷III soils is 6.5 points (INPP Unit 1), while on category II soils it is 6.0 points (INPP Unit 2). The accelerograms and other characteristics corresponding to these conditions were prepared.

The data were transferred to the general designer of the Ignalina Nuclear Power Plant to carry out the calculations of the characteristics of separate facilities on the INPP site, taking into account that these characteristics can be applied to the nearby facilities by introducing of appropriate coefficients.

In 1991 the general designer of the Ignalina Nuclear Power Plant – VNIPIET took the data of PNIIIS institute as a basis and decided to use the following data specified in Table 8.1-1 [5.6] in order to perform the calculations of the main INPP structures floor accelerograms and floor response spectra.

TABLE 8.1-1

No	Number of Building, Structure	Intensity of Maximum Design Earthquake, point	Maximum Ground Acceleration, m/sec^2
1.	101/1, Units A1, B1, V1, D1, D0	6.5	0.75
2.	101/2, Units A2, B2, V2, D2	6.0	0.60
3.	Pumping station, build. 120/1,2	6.0	0.60
4.	Room for ECCS pressurized tanks, Bld. 117/1,2	7.0	1.00

According to the initial data the probabilistic characteristics of the Ignalina Nuclear Power Plant main structures floor response spectra in case of earthquakes were made. The results of the probabilistic processing of the design-basis spectra are provided in documents [5.10], [5.11], [5.12], [5.13].

According to the results of the analysis carried out, the probabilistic characteristics of available spectra correspond to the MCE. In case of PE the values of mathematical expectations and standards are 2 times less.

8.1.1.2 DSFSF Design-Basis Earthquake

While designing of the DSFSF the design-basis earthquake of 6 points according to MSK-64 scale was taken as a basis. The appropriate maximum acceleration on the ground surface

is
a = 0.6 m/sec^2 = 0.06g.

The following components of the DSFSF were designed taking the design-basis earthquake into account:

- base slab of the casks storage site;
- shielding wall;
- radiation monitoring system equipment.

CONSTOR RBMK-1500 and CASTOR RBMK casks are calculated and designed to bear the impact of significant loads acting on them in case of design-basis accidents occurring due to the drop of a cask during casks handling operations at the power units and transportation to the DSFSF site.

The structure of CONSTOR RBMK-1500 cask bears the overload of 87g [5.18], while CASTOR RBMK bears the overload of 110g [5.19]. 32M baskets and fuel bundles bear the overload up to 85g [5.18], [5.19]. Such overloads are possible in case of accidents during transportation of a loaded cask from the power units to the DSFSF. This considerably exceeds the overloads acting on the casks in case of design-basis and beyond design-basis seismic loads.

Moreover, the design justifies the stability of CONSTOR RBMK1500 and CASTOR RBMK casks to the tipover in case of simultaneous impact of horizontal acceleration a_H=±0.2g and vertical acceleration a_V=±0.1g (these accelerations exceed the values of the design-basis earthquake). It is shown that CONSTOR RBMK1500 and CASTOR RBMK casks do not tip over in case of such impact (the safety factor for CASTOR RBMK is equal to 2.07 [5.19], while the safety factor for CONSTOR RBMK1500 is equal to 2.14 [5.18]).

8.1.1.3 ISFSF Design-Basis Earthquake

While designing of the ISFSF the maximum design-basis earthquake of 7 points according to MSK-64 scale was taken as a basis with the maximum acceleration on the ground surface a = 1.0 m/sec^2 = 0.1g.

The ISFSF site components designed to bear the seismic loads impact in case of the design-basis earthquake are as follows:

- supporting structures of the storage facility including the base slab;
- internal and external shielding walls of the casks storage hall;
- reinforced concrete fence of the Fuel Inspection Hot Cell (FIHC);
- concrete lid of the casks;
- 125/25 t bridge crane of the building;
- load-lifting equipment including horizontal and vertical cross arms for cask lifting, tilting device;
- shock-absorbers of the ISFSF;
- FIHC equipment including FIHC hoisting crane for the spent fuel handling, crane beams, rails of the FIHC crane, fuel bundles gripping devices, gripping device for a

primary lid, gripping device for the damaged fuel cartridges, hatch of the hot cell, shielding plate, shielding windows, manipulators and shielding door;

- the trolley for cask transportation to FIHC;
- primary and secondary filters of the FIHC ventilation system;
- railway transporter M-2 (steel structure and equipment).

The equipment for cask loading at the power units estimated for the design-basis earthquake:

- reloading machine including the fuel bundle gripping device (for reloading of the fuel bundle from 32M basket to the ring basket);
- casks handling equipment including a lifting beam, a protective ring, gripping devices for lifting of a primary lid;
- shock-absorbers of type 1 and type 2 in storage pool 338/1, shock-absorber of type 3 in room 174 (a transportation corridor).

The case of CONSTOR® RBMK1500/M2 cask is calculated and designed to bear the significant overloads acting on it in case of design-basis accidents occurring due to the drop of a cask during the casks handling operations at the power units and during transportation of it to the ISFSF site.

CONSTOR® RBMK1500/M2 cask structures as well as 32M baskets and fuel bundles are designed for overloads up to 85g, which are possible in case of accidents during transportation of the loaded cask from the power units to ISFSF [5.20]. This considerably exceeds the overloads acting on the cask in case of an earthquake.

Moreover, the design justifies the stability of CONSTOR® RBMK1500/M2 cask to the tipover and sliding in case of simultaneous impact of horizontal acceleration $a_H = \pm 0.2g$ and vertical acceleration $a_V = \pm 0.1g$. It is shown that CONSTOR® RBMK1500/M2 cask in case of this impact does not slide (the safety factor is 1.35) and does not tip over (the safety factor is 2.27) [5.20].

8.1.1.4 Nuclear Power Plant Protective Measures in Case of the Design-Basis Earthquake

As of 1 July, 2011 there are 1335 fuel assemblies in Unit 2 reactor, while 7045 spent fuel assemblies are stored in the storage pools. In the storage pools of Unit 1 7175 spent fuel assemblies are stored [5.76].

In Unit 2 Reactor Safety Justification [5.43] it is indicated that during unloading of 110 spent FA from Unit 2 reactor core, the critical state becomes impossible even in case of withdrawal of all CPS rods. Currently, only FASS rods (24 pieces) have been withdrawn from Unit 2 reactor core, while the other CPS rods (187 pieces) are inserted into the reactor core.

In 2005 the International Nuclear Safety Centre carried out the assessment of the burden of the welded joints of pipelines Du 300 of the cooling systems of the Ignalina NPP Unit 2 RBMK-1500 reactor in the main operating modes and under external impacts [5.35]. The reactor cooling system includes the main circulation circuit and the blowdown and cooling system. According to the data of the report, the researches carried out enable to draw the following certain generalizing conclusions regarding preliminary conservative estimations of stresses and efforts in the welded joints of the INPP pipelines Du 300:

- provided stresses, obtained taking into account the operational and seismic loads from MCE do not exceed the permissible ones regulated by the Nuclear Power Facilities Equipment and Pipelines Strength Calculation Standards, PNAE G-7-002-86 [5.81];
- the most dynamically loaded amongst the pipelines Du 300 in case of possible seismic impact are downtake pipelines within the steam separators room;
- values of the provided seismic stresses in the pipelines are assessed by the values up to 110 MPa for rectilinear areas and up to 100 MPa for curvilinear areas;
- the maximum level of bending stresses due to the seismic impact for any welded joint of the pipeline 325x15 mm of INPP Unit 2 can be assessed as 12 MPa.

It is stated in the INPP Detailed Design [5.22] that the equipment applied for the operations with SF in the storage pools hall (a crane, gripping devices for SFA and cartridges) prevents the possibility of spontaneous unhooking and drop of SFA or cartridges with SFA to the bottom of the pools while hanging them on the slot floor beams. In order to avoid the possibility of drop in case of design-basis seismic impacts, the slot floor beams and suspension brackets applied for compacted storage of SFA are designed for the strength taking into account seismic loads. Thus the drop of the cartridges with SFA or SFA themselves is possible only due to erroneous actions of the personnel or in case of beyond design-basis seismic impact (more than 6.5 points).

According to the design the bottom of the spent fuel pools of INPP Units 1, 2 is made of the double liner and the space between the liners is filled with 90 mm thick concrete of high drainability. The internal liner is made of 5 mm thick corrosion-resistant steel, while the outer liner is made of 3 mm thick carbon steel. The high-drainability concrete is foreseen with special drain channels, ensuring the drainage of the possible water leakages from the space between the liners to the drainage pipelines 89 mm in diameter, which remove the leakages in an orderly way to the contaminated drain waters tank with the air gap. The maximum water flow rate may be not higher than 76 m^3/h.

After the analysis of the emergency situation and the storage pools liner strength calculation carried out by VNIPIET [5.22] it was determined that the drop of the 102-place basket with SFA from height of 4.5 m in compartments 336 and 337/1,2 causes the rupture of the internal liner of the bottom of the pool with 10 mm penetration depth and makes a hole of the equivalent area of 88.7 cm^2.

In the amendment to the INPP project [5.15] it is stated that the drop of SFA, cartridges with SFA or baskets with fuel elements bundles will not result in the rupture of the liner, since according to the recommendations of the Detailed Design [5.22] 10 mm thick armour plates were laid in compartments 157, 234, 235, 339/1,2, 337/2, 338/2 of Units A1, A2 and in compartment 337/1 of Unit A2 at the INPP.

So far the armour plates, preventing the rupture of the bottom in case of the drop of the 102-place baskets with SFA at their lifting/lowering places, were not laid in the following baskets storage pools: 336, 337/1 of Unit A1 and 336 of Unit A2. Those works were not carried out earlier since the mentioned pools are loaded with the baskets with SFA and it is not possible to transpose the baskets in order to lay the armour plates.

Currently, in the compartment 336 of Unit A1 there are 23 baskets with SFA, in compartment 337/1 of Unit A1 there are 21 baskets with SFA and in compartment 336 of Unit A2 there are 28 baskets with SFA. Moreover, the armour plates were not laid in the SFA storage pools 236/1,2 of both units due to their filling with SFA.

It is possible to lay the armour plates in the baskets lifting/lowering area in compartments 337/1 of Unit A1 and 336 of Unit A2 only after the beginning of fuel removal from the storage pools. The laying of the armour plates in the baskets lifting/lowering area in compartment 336 of Unit A1 is possible prior the fuel removal from the unit on condition that the baskets with SFA are temporary transferred to the free places in compartment 235 of Unit A1.

Currently, lifting of a cartridge with SFA to the height of 8 m during process operations is possible only in the compartments 234 of Units A1, A2 and 157 of Unit A1, where the armour plates have already been laid. At present the transportation and processing operations performed with SFA or cartridges with SFA in compartments 236/1,2 of both units foresee the lifting of SFA or cartridge with SFA to the height not more than one meter above the bottom that, in case of their drop, will not cause the liner seal failure.

Strength and stability of walls and bottoms of the storage pools are ensured up to the seismic impact with intensity of the maximum calculated earthquake of 6.5 points inclusive according to MSK-64. Thus at normal operation and at design-basis accidents there are no leakages from the storage pools, which can cause radiation-dangerous decrease of the water level.

Actions of the personnel in case of the initiating events leading to the design-basis and beyond design-basis accidents are described in the Fuel Storage and Handling Facilities Operational Manual [5.47]. In case of the pools bottoms rupture due to the drop of SFA or SFA with a basket, the sealing device “Plaster” is foreseen. Its area is 180 cm2 and it is intended for sealing of all types of damage of the liner. The operations including the lifting of the dropped item are carried out according to the special programme of works performance.

Since August 2008 till January 2011 LEI was performing the works on the analysis of Building 101/2 Unit A2 reaction to the seismic impact. In the final report [5.27] the results of the strength calculations of the rooms, the functions of which are related to the storage of the spent nuclear fuel, are provided.

The following was obtained on the basis of the strength assessment results:

- the most dangerous is a combination of static and tensile seismic loads;
- the floor and the walls of pools have the least safety factor; 92 % of the bearing capacity of the floor and 83 % of the bearing capacity of the walls shall resist the seismic impact;
- the cracks can emerge, but their width will not exceed the admissible size.

The walls and floors of Ignalina NPP Building 101/2 Unit A2 meet the criteria of strength and are able to sustain the seismic impact.

The crane equipment of Unit 1 and Unit 2 was designed not taking seismic loads into account. In the amendment to the INPP design [5.14] it is indicated that the the cranes drop in case of the maximum calculated earthquake is impossible. The failures in the cranes operation can lead to a break in the work, i.e. to the suspension of SFA, cartridges with SFA or baskets with the fuel elements bundles during transportation and processing operations. Since all the operations are carried out under the water layer, the mentioned emergency conditions do not lead to an accident. The grabs for cartridges, SFA and baskets keep their durability in case of the MCE up to 6.5 points.

Failures of the cooling, makeup and ventilation systems equipment due to the seismic impacts as well are considered in sections 15.1÷15.5 of the Safety Justification of the Second Stage of INPP Unit 1 Decommissioning [5.46].

INPP Seismic Alarm and Monitoring System

Purpose of the Seismic Alarm and Monitoring System

The seismic alarm and monitoring system (SA&MS) consists of two independent subsystems performing different functions:

- seismic alarm system (SAS);
- seismic monitoring system (SMS).

The SAS system is intended for warning about an earthquake prior to its waves reaching the INPP. Upon a seismic wave reaches one or several external seismic stations, the system sends alarm signals to the MCR of Unit 2. The time from the beginning of alarm signals till the seismic waves come to INPP is approximately 10 seconds. When the alarms are received at MCR, no actions of operating personnel are foreseen, except for the repair service warning.

The SMS system is intended for obtainment of the data about stability of the INPP structures, systems and components in case of earthquakes. The analysis of the data will enable to evaluate, whether Unit 2 structures and equipment are damaged after the earthquake. The information received by SMS system is transferred to the Lithuanian Geological Survey for the analysis.

The information received from seismic sensors is saved in the archive for:

- further analysis of the data about the dynamic behaviour of the INPP structures, systems and components;
- assessment of the degree of applicability of the analytical techniques applied in the seismic system and qualification of the buildings and equipment;
- post-earthquake assessment – whether the safety related structures and/or equipment are damaged.

Structure of the Seismic Alarm and Monitoring System

The seismic alarm and monitoring system consists of the following components:

- 4 external seismic stations installed at 30 km distance from the power plant (No 1 – No 4) and station No 7 installed at INPP (see Figure 8.1-1);
- equipment of Unit 2 system located in Bld. 101/2 and on the INPP site.

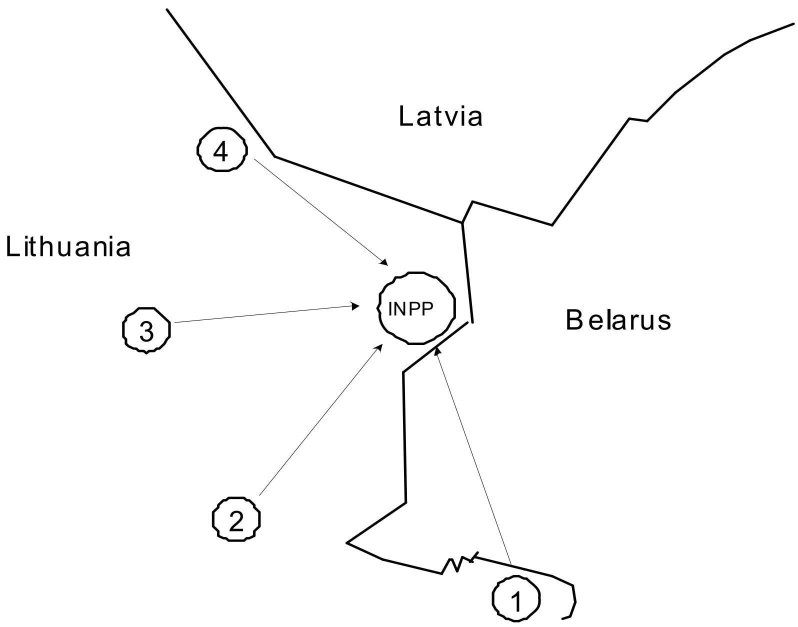


Figure 8.1-1. Layout of the Seismic Stations

1 – station in Didzhiasalis (Navikai village), 2 – station near Ignalina (Azhushile village), 3 – station in Salakas, 4 – station in Zarasai (Dimitrishkes village)

Equipment of seismic alarm and monitoring system at INPP includes:

- three sensors of SAS-320 system on the INPP site;
- 7 sensors of SSA-320 type of SMS system to monitor buildings on the INPP site and at Unit A2;
- sensor of CA-164 type of SMS system installed on BS-12 of Unit A2;
- 16 receiving aerials of the SAS and SMS systems on the roof of Unit A2;
- GPS system of exact time reception on the roof of Unit A2;
- 3 data reception and conversion cabinets in room 1404\1 of Unit A2;
- central control panels of the system with computers for data recording at the MCR of Unit 2.

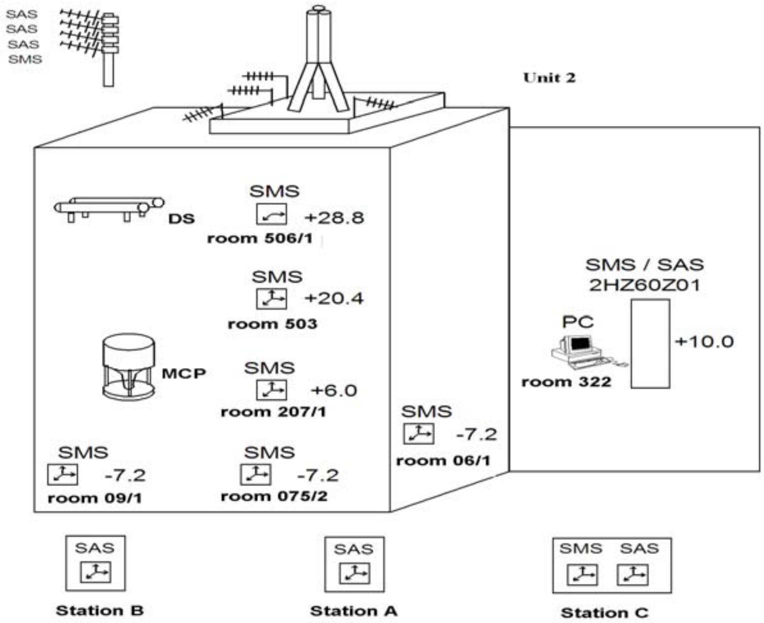


Figure 8.1-2. Layout of the Seismic Equipment at the INPP

The service life appointed for the seismic alarm and monitoring system is10 years.

8.1.1.5 DSFSF and ISFSF Protection Measures in Case of the Design-Basis Earthquake

The measure of protection of DSFSF and ISFSF in case of the design-basis earthquake is designing and calculation for the impact of loads during the seismic impact on the equipment important to safety, the damage of which can cause the infringement of the established nuclear and radiation safety limits.

8.1.2 Compliance of the Power Plant with Current Requirements Related to Earthquakes

8.1.2.1 Compliance of the Units with Current Requirements Related to Earthquakes

The final report of LEI on analysis of Building 101/2 Unit A2 reaction to the seismic impact [5.27] provides the following conclusions:

- The available geological, geophysical and geodetic data related to the territory of the Baltic States and Ignalina region enables to detect the high seismic hazard zone. Nearby the nuclear power plant the Drukshiai seismogenic area is defined.
- The seismic potential of the Ignalina Nuclear Power Plant site is estimated PGA=130 cm/sec^2 on the basis of the probabilistic analysis that corresponds to the value of the maximum possible earthquake with a magnitude of 5.0 points at a depth of 15 km nearby the power plant in Drukshiai fracture.
- The analysis of the Ignalina Nuclear Power Plant Building 101/2 Unit A2 structures strength is carried out to find out the impact of the seismic load on the rooms important not only during the operation of the reactor, but also after its shutdown.
- The calculation of the strength and crack resistance of Building 101/2 Unit A2 reinforced concrete structures is performed in accordance with standard STR 2.05.05:2005 "Regulations on the Design of Concrete and Reinforced Concrete Structures". The results of the analysis of the structures strength show that the analyzed

reinforced concrete walls and floors meet the criteria of strength and crack resistance, specified in STR 2.05.05:2005, and are capable to sustain maximum design-basis earthquake. The analysis of operational capability shows that cracks can appear, but their width will not exceed admissible size.

The seismic impact does not have an iinfluence on the casks handling system since the newly developed and updated equipment having an impact on the safety is applied in a seismic design basis. Such design basis ensures preservation of operability of this equipment during MCE and avoids the possibility to exceed the dose loads above the admissible annual ones in case of seismic impact.

The struggle against the consequences of possible beyond design-basis earthquakes at INPP Units is described in section 8.4 "Severe Accidents Management" of the present report.

8.1.2.2 Compliance of DSFSF with Current Requirements Related to Earthquakes

DSFSF corresponds to the requirements related to the ability to sustain the impact of a design-basis earthquake. As it is shown in the section of the analysis of DSFSF permissible limits (8.1.3.2), the safety criteria for storage of the fuel in casks will not be infringed in case of the impact of beyond design-basis seismic loads as well.

The DSFSF is operated within the framework of the effective VATESI license. According to the results of the DSFSF operation in 2010 the following conclusions are made in the report [5.80]:

- all casks with the spent fuel stored on the DSFSF site are leak-tight;
- radiation environment meets the set requirements;
- dose loads of the DSFSF personnel are considerably below the design values;
- there is no negative impact on the environment;
- nuclear safety during operation of the DSFSF is ensured;
- there were no failures having an impact on the safety of DSFSF.

Due to the implementation of new nuclear safety requirements BSR-3.1.1-2010 "General Requirements for DSFSF" [5.83] in 2010, INPP has analyzed the compliance of DSFSF and ISFSF with the new requirements. According to the results of the analysis the Plan of Activities on elimination of non-conformities with the VATESI document "General Requirements for DSFSF, BSR-3.1.1-2010" No MtDPl-75 (3.67.22) dated 3 December, 2010 was issued. Non-conformities related to the requirements concerning seismic stability of DSFSF and ISFSF have not been revealed.

8.1.2.3 Compliance of ISFSF with Current Requirements Related to Earthquakes

Currently ISFSF is under construction. The Technical Design and Preliminary Safety Analysis have undergone the state expert examination in the institutions of the Republic of Lithuania concerning compliance with the requirements of normative documents of the Republic of Lithuania, IAEA, etc. The VATESI license for construction of ISFSF has been received. After completion of construction and performance of complex cold tests, the INPP shall have to receive the VATESI license for ISFSF operation prior to the beginning of the hot tests. According to the results of the hot tests VATESI will issue the permission for industrial operation of ISFSF.

As it is shown in section of the ISFSF permissible limits analysis (8.1.3.3), safety criteria for storage of the spent fuel in shielding casks will not be violated in case of the impact of beyond design-basis seismic loads.

8.1.3 Permissible Limits Assessment

8.1.3.1 Assessment of Permissible Limits of the Units

The primary task of the INPP units safety is to ensure nuclear and radiation safety during normal operation and in case of accident, as well as limitation of impact on the personnel, population and environment, regulated by sanitary rules and safety norms.

All the structures, systems and elements that ensure safe storage of the spent fuel in the spent fuel pools at both units as well as in the reactor core of Unit 2, sustain the maximum design-basis earthquake.

The struggle against the consequences of the possible beyond design-basis earthquakes at INPP units is described in section 8.4 “Severe Accidents Management” of the present report.

8.1.3.2 Assessment of Permissible Limits of DSFSF

DSFSF, including protective CONSTOR RBMK1500 and CASTOR RBMK casks, is a part of the spent fuel storage and handling system referring to the safety related normal operation system.

The primary task of DSFSF safety is to ensure nuclear and radiation safety during normal operation and in case of accident, as well as to limit the impact on the personnel, population and environment, regulated by sanitary rules and safety norms.

The basic safety criteria during the storage and handling of the spent fuel in DSFSF in the CONSTOR RBMK1500 and CASTOR RBMK casks are:

- spent fuel subcriticality assurance, coefficient ≤0.95;
- non-exceedance of the maximum temperature of fuel cladding 300°C;
- ensuring of radiation protection of the personnel and population – non-exceedance of a dose rate of 1 mSv/h value on any surface of a cask, non-exceedance of annual dose on a physical protection barrier of DSFSF 5 mSv/year.

In the appropriate safety analysis reports it is indicated that these criteria are observed under the established conservative conditions for normal operation and for design-basis emergency scenarios [5.16], [5.18], [5.19].

Spent Fuel Subcriticality Assurance

Subcriticality of the spent fuel loaded into the CONSTOR RBMK1500 and CASTOR RBMK casks is ensured by geometrical arrangement of the spent fuel inside 32M basket. Moreover, the subcriticality is justified for the conservative case of loading of the CONSTOR RBMK1500 and CASTOR RBMK casks with the fresh nuclear fuel with the maximum enrichment on U^{235} 2.4 %, with flooding of a cavity of the cask with water having the density corresponding to the optimum moderation of neutrons and for an infinite lattice of the casks placed at the storage site [5.18], [5.19].

The cases of CONSTOR RBMK1500 and CASTOR RBMK casks, 32M and spent fuel baskets can stand overloads which are considerably higher than the overloads influencing the cask in case of the design-basis and beyond design-basis earthquakes. Therefore, there are no conditions for violation of geometry of the spent fuel arrangement in the casks in case of the impact of seismic loads.

The CONSTOR RBMK1500 and CASTOR RBMK casks also can be exposed to the applied shock of the drop of the fragments of construction structures (shielding walls, roof) and the equipment of a collapsing building in case of beyond design-basis seismic loads. Integrity of the CONSTOR RBMK1500 and CASTOR RBMK casks and, accordingly, absence of violation of the spent fuel arrangement geometry, is justified for a case of applied shock impact on a cask containment by an item weighing 1000 kg and the velocity of which before the shock is 300 m/sec [5.18], [5.19]. The shock load in this case is estimated at 26 MH, there is no loss of structural integrity and leak-tightness of the casks.

Moreover, the analysis of the drop of fragments of crane GK100 on the casks loaded with spent fuel from 23 meter height has been carried out [5.16]:

- drop of the trolley weighing 7 t on a detached cask;
- drop of a crane crossbar weighing 86 t on a row (6 casks);
- drop of a crane crossbar on a detached cask.

The carried out analysis has shown that the values of maximum loads from the dropped fragments is less than the maximum load for which the CASTOR RBMK and CONSTOR RBMK1500 casks are designed.

Non-Exceedance of the Maximum Temperature of Fuel Cladding

The CONSTOR RBMK1500 and CASTOR RBMK casks are stored on the open DSFSF site. Heat removal is carried out passively from an external surface of the cask by means of natural air circulation. Non-exceedance of the established criterion for the spent fuel cladding 300°C is justified both for normal conditions of storage and for a case of fire (temperature 600°C for 1 hour) [5.18], [5.19].

In case of impact of beyond design-basis seismic loads the collapse of a shielding wall and partial blockage of the first row of the casks by the shivers with partial malfunction of heat removal path by means of natural air circulation. The thermal analysis of the casks for this case was not carried out. However, this event will not cause the excess of the value of admissible temperature of the fuel cladding within the time necessary for clearing of blockages. This conclusion was made taking into account the following aspects:

- the results of calculations received for a case of the blockage by the shivers of the new CONSTOR® RBMK1500/M2 cask having a greater design-basis thermal load (12.57 kW) in comparison with CONSTOR RBMK1500 and CASTOR RBMK (6.1 kW) [5.19].
- the calculated maximum temperature for a surface of CASTOR RBMK cask is 100°C, that corresponds to the temperature inside the cask 212°C [5.19]. For a surface of CONSTOR RBMK1500 cask 72°C corresponds to the temperature inside the cask 271°C [5.18]. The actual measured temperatures of a surface of the casks on the DSFSF site are considerably lower and their values change only depending on the change of the temperature of the ambient air [5.80].

Assurance of Radiation Protection of the Personnel and Population

The limit dose rate on any surface of the cask of 1 mSv/h set by the cask designer is ensured by the biological shielding of the cask which is formed by the walls and system of lids (see the description of casks in section 7.7). The calculated justification is presented in the reports [5.18], [5.19]. The casks operation experience shows that the actual dose rate is below the calculated value [5.80].

For emergency loads in case of seismic impacts the limit dose rate on any surface of the cask is 1 mSv/h, as well as retention of radioactive fission products is ensured by integrity of biological shielding and leak-tightness of the cavity of the casks as it is shown above including external impacts on the case of a cask.

In case of the postulated loss of leak-tightness of a cask, this cask will be sent from DSFSF to the Unit in accordance with operational procedures in force at INPP for performance of actions on repacking of the spent fuel to the other cask.

In case of the impact of beyond design-basis seismic loads there is a possibility of cracks formation or collapse of a shielding concrete wall (in case of a high-magnitude earthquake) and increase of the dose rate of direct gamma- and neutron-irradiation behind the protective fence of DSFSF. The Organization of Emergency Preparedness (OEP) of the INPP in order to develop further personnel and population protection measures according to the Plan of Emergency Preparedness (PEP) in force will require the OEP Radiation Protection Service to carry out the measurement of the dose rates in the area depending on the place and character of a damage of the DSFSF shielding wall.

DSFSF is located within the existing sanitary protection area of INPP. The distance from a protective fence to the borders of the sanitary protection area is 2 km. In order to develop further actions on protection of the personnel and population, the measurement of the dose rates in the area depending on the place and character of a damage of the shielding wall of DSFSF will be needed in this case.

Failures of Support Systems

In case of impact of beyond design-basis seismic loads, the postulated failure of all support systems (radiation monitoring systems, power supply system, fire protection system, physical security system) does not cause violation of safety limits since the safety of storage of the spent fuel in protective casks is based on the passive principles:

- reliable assurance of the spent fuel arrangement geometry;
- heat removal from the walls of casks by means of natural air circulation;
- leak-tightness of a cask containment with application of the double-barrier system and absence of need for maintenance of the inert ambient of storage (helium).

In accordance with the PEP appropriate OEP services will perform the works and actions aimed at facility safety assurance (radiation environment monitoring by mobile means, arrangement of the temporary physical protection, etc.) for the period of recovery or maintenance of the design systems important to safety. No restrictions on application of such actions are determined.

8.1.3.3 Assessment of Permissible Limits of ISFSF

ISFSF, including protective CONSTOR® RBMK1500/M2 casks, is a part of the spent fuel storage and handling system referring to the safety related normal operation system.

The primary task of ISFSF safety is to ensure nuclear and radiation safety during normal operation and in case of accident as well as to limit the impact on the personnel, population and environment regulated by sanitary rules and safety norms. The basic safety criteria during the storage and handling of the spent fuel in ISFSF in the CONSTOR® RBMK1500/M2 casks are as follows:

- spent fuel subcriticality assurance, coefficien ≤0.95;
- non-exceedance of the maximum temperature of fuel cladding 300°C;
- ensuring of radiation protection of the personnel and population – non-exceedance of a dose rate of 1 mSv/h value on any surface of a cask, non-exceedance of annual dose on a physical protection barrier 0.2 mSv/year [5.20].

The preliminary SAR [5.20] shows that these criteria are observed under the adopted conservative conditions for normal operation conditions and for emergency scenarios.

Assurance of Subcriticality

Subcriticality of the spent fuel loaded into CONSTOR® RBMK1500/M2 cask is ensured by geometrical arrangement of the spent fuel inside 32M basket and ring basket (see Figure 7.8-2). The subcriticality of the cask is justified by the design for the conservative case of loading of the cask with the fresh nuclear fuel with the maximum enrichment on U^{235} 2.8 %, with flooding of a cavity of the cask with water having the density corresponding to the optimum moderation of neutrons and for an infinite lattice of the casks placed at ISFSF storage area. The burnable absorber erbium, which is in the composition of the nuclear fuel, is not conservatively taken into account as well [5.20].

The case of CONSTOR® RBMK1500/M2 casks, baskets and spent fuel can sustain the overloads up to 85g, which considerably exceed the overloads influencing the cask during design-basis and beyond design-basis earthquakes. Therefore, there are no conditions which can cause violation of geometry of the spent fuel arrangement in the casks during the impact of design-basis and beyond design-basis seismic loads.

Integrity of the cask and, accordingly, the spent fuel arrangement geometry, is also justified for a case of applied shock impact on the cask containment by an item weighing 1012 kg and the velocity of which before the shock is 300 m/sec. The calculation shows that for this beyond design-basis emergency scenario the maximum stresses in weld seams of the welded lids of the cask are 25 % of a material yield point limit and 44 % for bolts. Integrity of the top ring of the cask as well as the primary, secondary and sealing lids will not be violated. The maximum plastic deformations do not exceed 1 % and there will not be violation of geometry of the spent fuel arrangement in the cask [5.20]. This analysis covers the case of the shock impact of dropped shivers of the construction structures (shielding walls and roof) and the equipment of the collapsed building in case of beyond design-basis earthquake.

Non-Exceedance of the Maximum Temperature of Fuel Cladding

Since in the new ISFSF the CONSTOR® RBMK1500/M2 casks will be stored in the closed building, for the beyond design-basis emergency scenario related to destruction of the construction structures of the storage hall subjected to the beyond design-basis impacts (including the seismic ones), within the framework of the PSAR the case of blockage of a cask by shivers and the resulting failure of heat removal from the external surface of the cask by means of natural air circulation has been analysed. The cask blockage cases due to which the surface of the cask closed by shivers makes 60 %, 40 % and 20 % from the whole area of the surface of the cask have been analysed. The following results have been obtained:

- For the coefficient of blockage by construction shivers equal to 60 % the maximum temperature of the cladding reaches the permissible temperature value of 300°C after 3.75 days;
- For the coefficient of blockage by construction shivers equal to 40 % the maximum temperature of the cladding reaches the permissible temperature after 5.5 days;
- For the coefficient of blockage by construction shivers equal to 20 % the maximum temperature of the cladding after 7 days is 8°C lower than the permissible temperature.

Thus it is shown that the temperature of the spent fuel cladding does not rise above 300°C over a period of time sufficient for acceptance of emergency actions on removing of blockages [5.20].

Assurance of Radiation Protection of the Personnel and Population

The limit dose rate on any surface of the cask of 1 mSv/h determined by the cask designer is ensured by the biological protection of the cask which is formed by the walls of the cask and the system of lids (see the description in section 7.8.1).

For emergency loads in case of the seismic impacts the limit dose rate on any surface of the cask is 1 mSv/h, as well as retention of radioactive fission products is ensured due to the integrity of biological protection and leak-tightness of the cavity of the casks as it is indicated above including shock impacts on the case of the cask.

In case of the postulated loss of leak-tightness of the cask (for any reason), this cask will be directed from the storage hall to the fuel inspection hot cell in order to perform the actions on repacking of the spent fuel to the other cask.

The following scenarios of impact of the beyond design-basis seismic loads during ISFSF operation have been analysed:

- coincidence of the beyond design-basis seismic impact and transportation of CONSTOR® RBMK1500/M2 cask with non-leaktight spent fuel from the power units to ISFSF;
- formation of through cracks or collapse of a shielding fence of the casks storage hall;
- coincidence of the beyond design-basis seismic impact and temporary being of the spent fuel in the baskets of the FIHC during repackaging of the spent fuel in the FIHC for the purpose of inspection of the spent fuel or in case of damage and loss of leak-tightness of the protective cask for some reasons.

In case of coincidence of the beyond design-basis seismic impact and transportation of CONSTOR® RBMK1500/M2 cask from the power units to ISFSF using the special railway transporter there is a possibility of tipover of a cask in such a configuration, in case of which leak-tightness of the cask is ensured by elastomeric sealing of the primary lid (see Figure 7.8-2). The tipover of the cask can cause disruption of the sealing and emission of gaseous fission products into the atmosphere. The issue of additional calculations of the cask tipover scenario during transportation from the power units to ISFSF in order to assess the possibility of the seal failure of the cask in case of its tipover in the aforementioned configuration is under discussion with the Contractor of the ISFSF Project. It will be necessary to study the impact on the environment, population and personnel with respect to this emergency scenario after the results of the calculations are obtained, and if needed, to introduce changes or supplements to the appropriate INPP emergency preparedness documents.

In case of the impact of the beyond design-basis seismic loads there is a possibility of the through cracks formation or collapse the shielding fence of the casks storage hall and increase of the dose rate at the physical protection fence and at the border of the sanitary protective area of ISFSF determined at 500 m distance from the fence (the sanitary protective area of ISFSF is determined inside the sanitary protective area of INPP which is 3km).

For the development of further actions on protection of the personnel and population (including the measures of restoration of design barriers) in accordance with PEP in force the dose rates shall be measured on the site by the Radiation Protection Service depending on the place and character of damage of the reinforced concrete fence of the storage hall of the ISFSF.

For the development of further actions on protection of the personnel and population in case of impacts of the seismic loads, the dose rates shall be measured on the site depending on the place and character of damage of the shielding wall of the ISFSF.

In case of coincidence of the beyond design-basis seismic impact and temporary storage of the spent fuel in the baskets of the FIHC during repackaging of the spent fuel in the FIHC for the purpose of inspection of the spent fuel or in case of damage and loss of leak-tightness of the shielding cask for some reasons. The FIHC represents the massive reinforced concrete fence with 1250 mm thick walls installed directly on a massive base slab. The probability of coincidence of beyond design-basis seismic impacts and performance of operations on cask repackaging is very small (ISFSF Project supposes very conservatively that repackaging of 10 casks will be required during ISFSF operating life of 50 years). The pit of the FIHC in which the baskets for temporary storage of the spent fuel are located is closed on top by the massive metal sliding plate which will protect the SF against the drop of shivers of the building structures and equipment of the FIHC. However, in case of seal failure of the reinforced concrete fence of the FIHC there is a possibility of emission of fission products from non-leaktight claddings of the fuel elements into the atmosphere, passing the aerosol filters installed in the FIHC exhaust ventilation system.

For elaboration of further actions on protection of the personnel and population in case of impacts of the seismic loads and loss of containment of the fence of FIHC when the SF is simultaneously located there (including the recovery of the design barriers) in accordance with the PEP in force the dose rates shall be measured on the site depending on the place and character of damage of the protective fence of the FIHC.

Failures of Support Systems

In case of the impact of the beyond design-basis seismic loads the postulated failure of all support systems (radiation monitoring systems, power supply systems, fire protection system, physical security system) does not result in violation of safety limits since the SF storage safety in protective casks is based on the passive principles:

- reliable assurance of SF arrangement geometry;
- heat removal from the walls of the casks by means of natural air circulation;
- leak-tightness of a cask containment with application of double-barrier system and absence of need in the maintenance of the inert ambient of SF storage (helium).

In PSAR of ISFSF [5.20] the additional analysis of the emergency scenario according to which shutters on incoming air channels in the cask storage hall remain completely closed. Hereat it is shown that the maximum permissible temperature of the fuel cladding 300°C reaches in 5 days. Thus, there is a sufficient reserve of time for restoration of a normal mode of heat removal by natural ventilation in the casks storage hall (ISFSF project also stipulates manual adjustment of the shutters installed on incoming and exhaust air channels).

Due to the character of the layout of the ISFSF site there are no restrictions on application of mobile means of the radiation environment monitoring and arrangement of temporary physical protection for the period of design means restoration.

8.2 Chapter 2. Flood

8.2.1 Design Data

8.2.1.1 Power Plant Design-Basis Flood

The lake Drukshiai serves as a natural water tank which is the source of the cooling water for the power plant. The length of the lake is 14.3 km, the maximum width – 5.3 km, perimeter is 60.5 km. The total lake area is 49.32 km^2. The maximum depth of the lake is 33.3 m, the average – 7.6 m, dominant – 12 m. The total amount of water in the lake is about 369 million m^3. The area of filtration (drainage) of the lake is 564 km^2 (Figure 7.1-3). There are a lot of lakes in the neighbourhood of the Ignalina NPP. The total surface of water (without lake Drukshiai) makes 48.4 km^2. The lakes occupy 15 %, bogs – 15 %, the cultivated land - 40 %, woods about 30 %. The density of the rivers is about 0.3 km/km^2 [5.4, 5.8].

The data indicated above are provided for the normal level of the lake - 141.6 m in relation to the Baltic system of levels. The permissible deviations are presented in the manual INPP hydraulic engineering constructions manual [5.54] and in the technical description of INPP hydraulic engineering constructions [5.66] and are as follows:

- maximum affluent level – 142.3 m.
- maximum emptying level – 140.7 m.

Heat removal to the ultimate heat sink (lake Drukshiai) is implemented by the service water supply system (service water pumps are located in the on-shore pump station Bld. 120/1,2). Power supply is carried out from the power supply system of Lithuania via the Switch-Yard (SY). In case of failure of external power supply sources, power supply of safety related service water pumps and systems is carried out from the reliable power supply system - standby diesel power plant (SDPP).

The on-shore pump station (Bld. 120/2) located at level 144.0 [5.87] is exposed to the maximum danger due to the possible floods with regard to INPP operation safety assurance.

Increase of the water level in lake Drukshiai above 144.0 m leads to the flood of the on-shore pump station as well as to:

- failure of service water pumps of the service water supply system and hence, of the heat transfer system to the ultimate heat sink (lake Drukshiai);
- failure of the standby diesel power plant and, hence, the systems reliable power supply to the safety related equipment and systems including the service water pumps;
- termination of liquid radioactive waste processing.

All other buildings and structures of the Ignalina NPP (see Figure 7.1-5) are located at levels above 144.0 m [5.91] (the DSFSF site – 149.0 m, SDPP – 149.5 m, OPDS – 153.15 m, the ISFSF site – 155.5 m).

The flood (increase of a water level in lake Drukshiai) which could lead to flooding of the equipment of INPP safety related buildings and structures is not stipulated by the design.

The designs of DSFSF and ISFSF do not stipulate any special measures aimed at protection against the design-basis flood. The design of the casks and the system of barriers ensure the necessary leak-tightness. Subcriticality of the SF in a cask is justified for a case of complete flooding of all internal cavities of the cask.

Water relationships of the lake [5.54, 5.66] are regulated by a system of hydraulic structures (Figure 7.1-3, Figure 8.2-1).

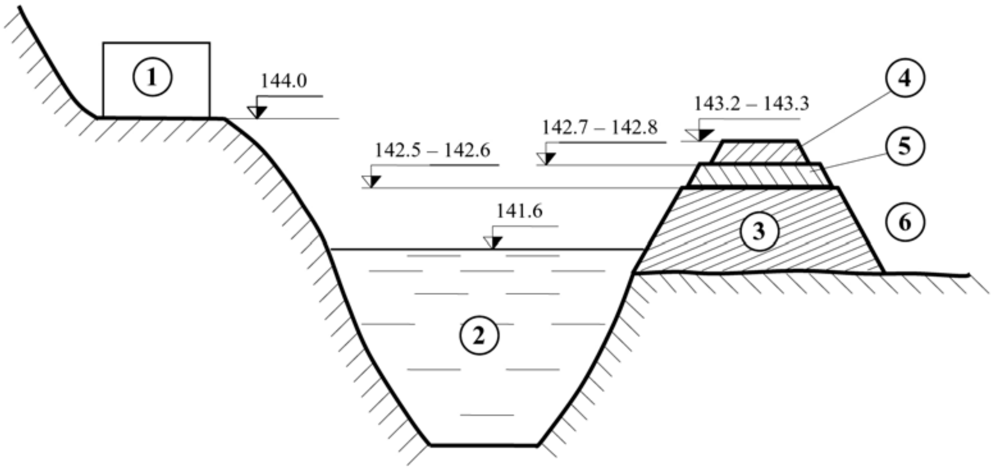


Figure 8.2-1. INPP Hydraulic Structures

1 – on-shore pump station, 2 – Lake Drukshiai, 3 – hydroelectric power plant “Druzhba narodov” dam, 4 - water regulating building 500, 5 - blind earthen dam, structure 501, 6- high-water bed of river Drysviaty

The top marks of INPP hydraulic structures in meters in relation to the Baltic System of Heights are presented in Table 8.2-1.

Table 8.2-1

Level, m	Place of location	Document Number
143.2 – 143.3	Slope and concrete platform of the water regulating structure 500	PVS-7081, dated 16-09-2011
142.7 – 142.8	Blind earthen dam (dike), structure 501	PVS-7081, dated 16-09-2011
142.5 – 142.6	Dam of hydroelectric power station “Druzhba Narodov”	PVS-8007, dated 17-10-2011

In case of uncontrollable emergency rise of a water level in lake Drukshiai in case of the most negative development of flood according to any scenario regardless of the reasons of its occurrence, the water level will even never reach a mark of a building of the on-shore pump station (144.0 m) since the top marks of hydraulic structures are located below 144 m (see Table 8.2-1). As a last resort, water of lake Drukshiai will flow over the dams of hydraulic structures (Figure 8.2-1) in the rivers Prorva and Drysviaty and then will fall into river Zapadnaya Dvina (Daugava) and to the Gulf of Riga of the Baltic sea (Figure 7.1-3)

DSFSF and ISFSF Design-Basis Flood

In the designs of DSFSF and ISFSF the design-basis flood is not included to the list of design accidents since flooding of the DSFSF site which is at level of 149.0 m above sea level and ISFSF site which is at level of 155.5 m above sea level due to raising of the water level in lake Drukshiai is impossible.

8.2.1.2 Power Plant Protection Measures in Case of Design-Basis Flood

The top levels of the hydraulic structures are located below the potentially dangerous buildings and structures (Table 8.2-1, Figure 8.2-1).

8.2.2 Power Plant Compliance with the Current Requirements related to Flood

The potentially dangerous buildings and structures of INPP including DSFSF and ISFSF comply with the current requirements related to flood.

8.2.3 Assessment of Permissible Limits

Limits of flood of potentially dangerous buildings and structures of INPP including DSFSF and ISFSF cannot be reached since the water level of lake Drukshiai cannot rise up to a level of their inundation at any flood.

8.3 Chapter 3. Loss of Power Supply and Heat Removal to the Ultimate Heat Sink

8.3.1 Loss of External Power Supply

8.3.1.1 External Power Supply

According to the design Ignalina NPP is connected with the external power system via 330/110 kV switch-yard.

The connection at voltage 330 kV is carried out:

- with a power supply system of Lithuania via two lines L-454, L-453;
- with a power supply system of Belarus via three lines L-450, L-452 and L-705;
- with a power supply system of Latvia via line L-451.

Connection at 110 kV voltage via the power supply lines is carried out via the first section of the 110 kV switch-yard:

- with a power supply system of Lithuania through “Zarasai” line;
- with a power supply system of Latvia through “Daugavpils” line L-632.

Connection between 330 kV switch-yard and 110 kV switch-yard is carried out via two coupling autotransformers AT-1, AT-2. Power of each autotransformer is 200 MVA

8.3.1.2 Internal Power Supply

According to the design, INPP 6 kV voltage auxiliaries can be powered:

- from 330 kV network via block transformers and from them via operation auxiliary transformers;

- from a 110 kV network via start-up transformers.

At each unit two block transformers, 4 operation transformers and 4 start-up auxiliary transformers are installed. At present the auxiliaries are powered via start-up auxiliary transformers from the 110 kV network. Block transformers and operation transformers of auxiliaries are at present in standby mode.

Unit 2 auxiliaries according to the technical regulations [5.30] have:

The system of normal power supply and standby power supply of auxiliaries consisting of:

- 4 start-up transformers;
- 6 sections of 6 kV normal power supply.

The system of emergency power system:

The system of reliable power supply consisting of 6 channels each of which includes the following main devices:

- sections of 6 and 0.4 kV reliable power supply;
- diesel generator;
- 6/0.4 kV reliable power supply transformer of a reliable supply;
- secondary sections and assemblies of reliable power supply which supplies power to the equipment of systems important to safety remaining in operation.

The uninterruptable power supply subsystem consisting of 6 channels each of which includes the following devices:

- accumulator battery;
- direct-current board;
- uninterruptible power supply units;
- uninterruptible power supply assemblies which ensure power supply to the equipment of systems important to safety remaining in operation,
- trickle chargers and transformers.

Structural Data Regarding Redundancy and Physical Separation of Channels

The structural principle of arrangement of emergency power systems is chosen on the basis of the general principle of redundancy of consumers of safety systems which lays in the process engineering part of the design. Since the major part of consumers of the safety systems are chosen with redundancy of two units (4 operating of 6 determined or 2 operating of 4 determined), the similar principle also is used during designing of circuits of reliable and uninterruptable power supply.

Reliable and uninterruptable power supply circuits are designed according to the channel structure and each of them consists of six independent channels. Normal functioning of the circuits is ensured in case of failures (damages) in any two channels including failures of process loads of the safety systems. Each section of 6 kV reliable power supply is located in a separate room. Sections of reliable power supply are located at level +4.2 in two rooms of unit G2 (axes 24-25 and 48-49) and in four rooms at level 0.0 of unit D2 (axes 2-6, 6-10, 41-45, 45-48).

Diesel generators are installed in building 111. Independence of each diesel generator from the other ones is ensured by installation of a diesel generator and its systems in a separate box. Each box is separated from another by the chief wall and has a separate entrance. There are no passes from one box to another one.

Each direct-current board is located in a separate room at level 0.0 of unit D2 together with the uninterrupted power supply units which belong to the same safety channel. In the same room there is an appropriate 0.4 kV reliable power supply section.

Three accumulator batteries are located in separate rooms at level 0.0 of unit D2. The accumulator batteries of another three channels are located in separate rooms at level 6.0 of unit D2.

Cable paths of different channels of the safety systems usually pass in different rooms. In exceptional cases the design allows passage of cables of no more than two different safety channels in one room: Unit A2-03/1,4; 092/1,2; 095/1,2; 099/1,2; 027/1,2; cable tunnels 2KT 21, 22, 23 on a building 120/2; cable paths CP19, 20 to diesel generators DG-7 and DG-8. In this case the cables are laid on cable racks located on different walls of the rooms.

In order to prevent rise and spread of fire the cables are coated by flame retardants. Cable rooms and shafts are equipped with the automatic fire alarm and automatic fire-extinguisher system. The fire resistance of the doors of electrical rooms is not less than REI45. Above the equipment the visors protecting from water ingress are installed.

Electrical Equipment Maintenance

The equipment is maintained by the trained and certified personnel according to the procedures executed in the appropriate order.

The scope and periodicity of tests and inspections of the equipment of emergency power supply system is performed according to the technical regulations [5.30]. The testing is carried out according to the schedule of inspections of the power supply systems equipment functioning [5.67] and a note on performance is made in the schedule at CCR. The remedial maintenance is carried out according to the standard of electrical equipment maintenance types [5.70].

Maintenance of RPAE is carried out according to:

- standards of types and periodicity of maintenance of electrical equipment RPAE devices [5.68];
- standards of types and periodicity of maintenance of the measuring devices and periodicity of electrical equipment insulation control [5.69].

The round check and examination of equipment of Unit 1 and Unit 2 is carried out in accordance with:

- Power Supply Workshop rooms and equipment round check and examination instruction [5.64];
- schedule and routes of the round checks and examinations of rooms and equipment by the operating personnel of the Power Supply Workshop [5.72];
- schedule of the round checks of the Power Supply Workshop safety systems equipment by engineering technicians for 2011 [5.71].

The results of the round checks are recorded by the operating personnel in the operating log books and in round checks worksheets at the equipment installation place. The results of the round checks performed by engineering technicians (ET) are recorded in the log books of equipment round checks and examinations by the management and ET [5.73].

Unit 1 auxiliaries have only the system of normal and standby power supply of auxiliaries. At present the auxiliaries are powered through the startup auxiliary transformers from 110 kV network. Four operation transformers can be connected to 330 kV network through two block transformers. Currently, all they are in the standby mode. The arrangement and design of the equipment remained in operation is similar to Unit 2. The maintenance of the equipment of Unit 1 is performed by the same personnel, as in Unit 2.

8.3.1.3 Loss of INPP Power Supply from External Source

In case of loss of external power supply, no more than in 15 seconds the diesel generators are automatically connected to the sections of reliable power supply of Unit 2. They ensure power supply to the 6 kV voltage consumers and through the step-down transformers to the 0.4 kV voltage consumers.

When there is power supply on the section of reliable 6 kV power supply through the transformer 6/0.4kV and the rectifier powered from it, the accumulator battery is recharged and power supply of the uninterruptible power supply is performed.

The power of each diesel generator is 5600 kW. The reserve of fuel without refuelling will be enough for operation of each diesel generator within 72 hours provided operation at design power required for emergency shutdown of the reactor and cooling of the Unit. Since the Unit 2 reactor is shut down and is at a stage of defuelling, a part of 6 kV consumers powered from the diesel generators is decommissioned and the reserve of fuel will suffice more than for 72 hours. Engineering assessment of time for which the available reserve of fuel will be enough in order to ensure power supply to the consumers of the Unit with the shut down reactor. The data is presented in Table 8.3-1.

TABLE 8.3-1

Diesel Generator	DG-7	DG-8	DG-9	DG-10	DG-11	DG-12
Load, kW	2670	2650	3150	1800	2650	2100
Load reduction factor (increase of DG operation time)	2.1	2.1	1.8	3.1	2.1	2.7
DG operation time without refueling, hour	151.2	151.2	129.6	223.2	151.2	194.4

Thus the minimum DG operation time without refueling is not less than 5 day. With the refueling the operation time of a diesel generator is not limited. In order to carry out the refueling it is necessary to conclude the fuel supply contract.

Since modifications on putting diesel generators and uninterruptible supply units out of operation have been carried out, Unit 1 auxiliaries, in case of the loss of external power supply, lose the power supply of alternating current, except for the RSM board which is common for two units, but located at Unit 1. The design power supply of the board remains from DG-7 of Unit 2. The direct current power supply from the battery 1AB-7 of the general unit consumers and emergency lighting of Unit 1 also remains. The assessment of Unit 1 safe operation is performed in the report [5.36].

The loss of external power supply results in the loss of DSFSF and ISFSF power supply, however, it does not result in the violation of the safety limit. See the analysis in section 8.1.3.2 for the DSFSF and section 8.1.3.3 for the ISFSF.

In order to ensure power supply to I&C of temperature and water level in the storage pools of Building 101/1,2 according to the decision [5.74] design No 10.2501.00 EM was developed. The terms of its implementation are roughly determined for December 2011. After laying of cables according to this design, power supply of I&C of temperature and water level in SFP of Building 101/1 will be ensured from the assemblies that are supplied with power from DG-7 of Unit 2. Then power supply of I&C of temperature and water level in SFP of Building 101/2 in case of loss of external power supply, according to the initial design is already ensured from the assemblies that are supplied with power from DG-11 and DG-12 of Unit 2.

In case of loss of external power supply the consumers of service water of Unit 1 are provided with operating service water pumps of Unit 2. The service water system of Unit 1 and Unit 2 is operated according to INPP Units 1,2 service water supply system operation manual [5.51].

Unit 1 water and foam extinguishing systems in case of the loss of external power supply are powered in accordance with the Units 1,2 stationary fire extinguishing systems operation manual [5.65] from the systems of Unit 2 which elecrtic motors are powered from the DG.

Actions on restoration of external power supply are presented in the instruction on breakdown elimination and technological malfunctions No 10210-1 of the Lithuanian power supply system [5.86] and in INPP electrical annex breakdown elimination instruction [5.58].

In the instruction on breakdown elimination and technological malfunctions No 10210-1 of the Lithuanian power supply system the time needed for possible restoration of power supply of INPP auxiliaries after complete shutdown of the power system (total failure) is indicated. It will be approximately 30 minutes. Some variants of voltage supply to INPP auxiliaries by 110 kV and 330 kV network are developed for this purpose. 4 diagrams of voltage supply to the startup auxiliary transformers are provided. The supply of voltage is foreseen from the hydropower plants as well – Pļaviņas Hydro Power Plant of the Republic of Latvia and the Kruonis Pumped Storage Plant of the Republic of Lithuania.

In INPP electrical annex breakdown elimination instruction the personnel operating procedure during elimination of malfunctions and breakdown in INPP electrical annex is defined. The order of cooperation of various sections of operating personnel is determined: Power Supply Workshop operating personnel, DPSS, PSS, the system services supervisory control group supervisor, the East supervisory control group supervisor.

8.3.2 Loss of External Power Supply and Standby Power Supply Sources at the Site

During the beyond design-basis accident in case of blackout of the INPP auxiliaries and failure of all diesel generators the consumers of 6 uninterruptible power supply channels powered from the accumulator batteries (AB) remain alive. At each Unit there is also the general unit battery 1,2AB-7 intended for the power supply of the general unit consumers and emergency lighting.

In case of the lack of voltage at a reliable 6 kV power supply section, the time of discharge of the battery for the full design load required for the emergency shutdown of the reactor and cooling of the Unit is **not less than one hour**.

Since Unit 2 reactor is shut down and is at the stage of defueling and the part of consumers is taken out of operation, the batteries discharge time will be considerably more. The engineering evaluation of battery discharge time for the power supply of the consumers of the Unit with the shut-down reactor was performed. The data are presented in Table 8.3-2. The evaluation is performed according to the value of the load of the transformers which power the rectifiers applying the conservative approach.

During calculation neither losses in a transformer and rectifier nor the accumulator batteries charging rates in a normal mode were subtracted from the load. The rated capacity of the Vb2421 VARTA type battery is 2100 A×h at a 10 hour rate current 210 A.

Table 8.3-2

Battery	2AB-1	2AB-2	2AB-3	2AB-4	2AB-5	2AB-6
Load, kW	8	18	15	38	33	22
Discharge Current, A	36.4	81.8	68.2	172.7	150	100
Discharge Time for Real Load, hour	57.7	25.7	30.8	12.2	14	21

Thus the minimum battery discharge time for the load at the shut-down reactor is **not less than 12 hours.** For general unit batteries the discharge currents taking into account the powering of emergency lighting are as follows: 1AB-7 223 A, 2AB-7 109 A. The discharge time for the real load will be: 1AB-7 9.4 hours, 2AB-7 19.3 hours.

During a beyond design-basis accident in case of blackout of the INPP auxiliaries and failure of all standard diesel generators, power supply to I&C, communications and RSM board is ensured from the portable diesel generators. For this purpose, according to technical decision TACмод-1632-681 (revision МОД-05-02-723) [5.75], the connecting points are installed on a wall of building 101/2 and wall of building 185. From them the cables are laid to distribution cabinet 2HZ201Z01 in room 2UPS-1 of building 101/1 and to cabinet 0DP201Z05 in room 110 of bld.185.

Actions of the personnel on the voltage supply from the portable diesel generators are provided in the Instructions on Assurance of Emergency Heat Removal from the Unit 2 Reactor in case of Blackout of the Auxiliaries [5.62] and in the INPP electrical Annex Breakdown Elimination Instruction [5.58]. Operating personnel of PSW was acquainted with instructions and signed for it. The last testing of the portable diesel generators energizing 2HZ201Z01 and 0DP201Z05 was carried out on 14 April 2011.

The detailed description of actions on breakdown elimination in process part of INPP is provided in Chapter 8.4 "Severe Accidents Management" of this report.

In order to ensure power supply of I&C of temperature and water level of the storage pools of bld. 101/1,2 in case of loss of external power supply and standby power supply sources on the site according to the decision [5.74], design 10.2501.00 EM has been developed. The term of its implementation is roughly planned for December 2011. Under this design the cables are laid to the assemblies connected to the portable diesel generators.

8.3.3 Loss of Heat Removal to the Ultimate Heat Sink

The main ultimate heat sink (UHS) for Unit 2 reactor and SFP of Unit 1 and Unit 2 is lake Drukshiai.

Heat removal from Unit 2 reactor to the main UHS is carried out using the systems related to the reactor according to the diagram: SFP→SFP PHEU→SWS→ lake Drukshiai.

An alternative ultimate heat sink for the reactor is the environment (atmosphere). In this case diffusion of heat to the environment occurs during ventilation of rooms where the equipment and pipelines are located, during the reactor space blowdown with compressed air, during evaporation of water from the coolant circuit in ALT and periodic makeup of the MCC.

An alternative ultimate decay heat sink of SFA located in SFP of Unit 1 and Unit 2 is the environment (atmosphere). In this case diffusion of heat to the environment occurs during evaporation of water and periodic makeup of SFP, during water exchange in SFP using the drain waters and contaminated LSW collection and pumping system and makeup system.

8.3.3.1 Heat Removal from the Reactor

In accordance with the technical regulations [5.30] the heat removal from Unit 2 reactor is carried out in the following reactor cooling modes:

- mode of natural circulation of the coolant;
- mode of forced circulation of the coolant;
- mode of disrupt natural circulation of the coolant;
- mode of the coolant bubbling.

Mode of Natural Circulation of the Coolant

In a mode of natural circulation of the coolant the cooling of any half of the reactor core is ensured within unlimited time if the following requirements are fulfilled:

- the water level in the MCC is above the levels of tie-in of SWP pipes in DS (not below level +29.7 m that correspond to "- 600 mm" according to the device with a scale -1200÷ +400 mm);
- pressure in the DS – any, but not above the limiting pressure permitted for hydraulic pressure testing;
- all IRV and gate valves on an inlet to DGH are open;
- gate valves on connecting pipes of MCP PH-SH are open;
- suction gate valves and pressure gate valves not less than on two MCP are open.

In the natural circulation mode the decay heat removal is carried out:

- in the non-boiling mode by means of BCS operation;
- in the boiling mode by means of steam removal from the DS through BRU-B to ALT and periodical makeup of MCC.

Mode of Forced Circulation of the Coolant by BCS Pumps

In the mode of forced circulation of the coolant, the cooling of any half of the reactor core is ensured within unlimited time if the following requirements are fulfilled:

- water level in the MCC is above the level of tie-in of SWP pipes in the DS (not below level +29.7 m that correspond to "- 600 mm" according to the device with a scale -1200÷ +400 mm);
- pressure in the DS – atmospheric;

- pressure and/or suction gate valves of all MCP, at all connecting pipes of MCP PH-SH and the gate valves at inlet of each DGH can be closed.

Cooling of the reactor core in the non-boiling mode is carried out by the forced circulation of the coolant in FC and operation of BCS.

Mode of the Disrupt Natural Circulation

In the mode of the disrupt natural circulation, the cooling of any half of the reactor core is ensured within unlimited time if the following requirements are fulfilled:

- water level in the MCC is lowered to the level which is no more than 1 meter below the plugs of FC (not below level +23.7 m);
- pressure in the DS - atmospheric;
- all IRV are open;
- MCT with a nominal water level are connected to all DGH of this half through the pipelines of ECCS;
- all bypass lines of DGH RV of this half are open;
- closure of all gate valves of DGH of this half is permitted.

The decay heat is removed by means of steam removal from the DS and water makeup of the reactor core by gravity flow from MCT. In this case the level in the MCC is maintained at the level indicated above by means of the periodic makeup, while in MCT – by means of the automatic makeup system.

Mode of the Coolant Bubbling

In the mode of the coolant bubbling the cooling of any half of the reactor core is ensured within the unlimited period of time if the following requirements are fulfilled:

- water level in MCC is above the levels of tie-in of SWP pipes in DS (not below level +29.7 m that corresponds to "-600 mm" according to the level gauge with a scale -1200÷ +400 mm);
- pressure in the DS – atmospheric;
- gate valves on all connecting pipes of MCP PH-SH are closed;
- any amount of IRV and the gate valves on inlet to each DGH are closed or all IRV with the gate valves open on the inlet to DGH are closed.

The decay heat is removed in the bubbling mode by means of steam removal from the DS and periodic makeup of the MCC.

Combination of any modes of cooling of the shut-down and cooled-down reactor at different halves of the MCC is allowed without time limitation in case the following conditions are fulfilled:

- difference of water temperatures in FC at different halves of the reactor does not exceed 30°C;
- difference of water temperatures in FC at different halves of the MCC up to 50°C is allowed for no more than 1 hour.

Reactor Makeup

Makeup of the MCC is carried out from one of the following sources:

- tank 2TD52B01 with a reserve of water not less than 1000 m^3 and two LSW pumps which are a part of 2TD61÷63D01 and which are ready for switching on;
- tanks TW15B01, TW41B01 and TW32B01 with the total amount of water not less than 3000 m^3 and two pumps as part of TW16D01÷03 ready for switching on;

- in case MCT are connected to the MCC – by pumps 2TD61÷63D01 or 2TD83÷87D01 in case of the switched on automatic equipment of MCT makeup.
- one HCCh (500 m^3 of water) with two ECCS pumps is the first channel of makeup, another HCCh (500 m^3 of water) with two ECCS pumps is the second channel of makeup.

In all Reactor Cooling Modes

The control of the temperature of water in the channels of each half of the reactor core is carried out by means of the thermocouples placed into the central tubes of FA instead of RPDS up to the level of the reactor core centre (+11.4÷13.4 m).

The control of the water level in each half of the MCC and FC of the reactor is carried out, at least, in two ways:

- in case there is a level in the DS – according to two standard level gauges;
- in case of the DS are emptied– according to the standard level gauge connected to the drain header of the ECCS pipelines;
- in case of connection of the repair tanks to the MCC – according to the level in tanks;
- if needed – according to the water level in not leak-tight FC from the CH.

8.3.3.2 Heat Removal from the SFP

Heat is removed from SFA located in the SFP of each Unit by means of cooling of water in pools when the appropriate SFP PHEU is operating or when PHEU is idle, as well as by exchange of water in the SFP. The SFP PHEU ensures the forced circulation of water in the following pools of unit A:

- compartment 235 – the transfer canyon;
- compartment 236/1,2 – spent fuel storage pool;
- compartment 234 – pool of transfer of an item to the cutting department;
- compartment 339/1,2 – spent fuel storage pool;
- compartment 336 – spent fuel storage pool;
- compartment 337/1,2 – spent fuel storage pool.

Water from SFP flows under gravity through the pipelines tied in at level +23.20 in each pool to the heat exchangers where it is cooled down by the service (lake) water to 30°C. After the heat exchangers the water flows to suction inlets of pumps and by the operating pumps returns through the regulation unit to the lower part of the SFP.

The temperature of water in the SFP is maintained within the range of 20÷50°C. The limit of safe operation is 60°C [5.29, 5.30, 5.39].

The PHEU temperature regime is determined by the quantity of heat exchangers connected to the service water, quantity of the switched on pumps, the flow rate of the circuit water and flow rate of the service water through the heat exchangers. In case of the maximum values of the decay heat in the pools, two pumps and three heat exchangers are constantly in operation. The SFP PHEU can be switched-off without time limitations if the temperature of water in all the storage pools is below 45°C. If the SFP PHEU is switched-off, the temperature of water in any SFP is reduced by the water exchange in this SFP.

The main source of Unit 1 (Unit 2) SFP makeup is pure LSW tank 1TD52B01 (2TD52B01). The periodic makeup is carried out by pumps 1TD61÷63D01 (2TD61÷63D01) through the regulation unit which ensures the automatic maintenance of water levels in the pools within the range of 950÷850 mm from the SFP floor. The CPW and SPC tanks serve as a standby source of the SFP makeup. Water is supplied by: pumps

1TD83÷87D01 (2TD83÷87D01) from tanks 1TD81, 82B01 (2TD81, 82B01) or pumps TW17D01÷03 from tanks TW15B01, TW32B01, and TW41B01.

Since the decay heat of SFA located in Unit 1 SFP is low, the Unit 1 SFP PHEU is switched off. Thus the temperature and chemical conditions of water in the SFP are maintained by the periodic water exchange.

The Unit 2 SFP PHEU is constantly operating in a nominal mode (2 pumps, 2 heat exchangers) and ensures the operational values of the water temperature in the SFP.

The correspondence of the ultimate heat sinks to the various modes of heat removal from Unit 2 reactor and from the SFP of both Units is presented in Table 8.3-3.

TABLE 8.3-3

Mode		Ultimate Heat Sink
Heat removal from the reactor		
Natural circulation of the coolant	non-boiling mode	main + alternative
	boiling mode	alternative
Forced circulation of the coolant		main + alternative
Disrupt natural circulation of the coolant		alternative
Coolant bubbling		alternative
Heat removal from the SFP		
Operating PHEU		main
Switched-off PHEU		alternative

8.3.3.3 Assessment of the Decay Heat Value in the Reactor with 1335 FA (as of 1 July 2011)

By July 2011 a year and a half has passed since the shutdown of Unit 2. Partial defueling of the reactor is carried out according to the “Working Program of Unloading of 500 Pieces of SFA from INPP Unit 2 Reactor”, code EPg-17(3.67.7). As of 1 July 2011 312 FA were unloaded and currently there are 1335 SFA in the reactor. In March 2012 it is planned to complete unloading of 500 FA.

Figure 8.3-1 represents the distribution of SFA located in the reactor according to the initial enrichment. It is obvious from the distribution that the enrichment of the basic amount of SFA is 2.6 % and 2.8 %.

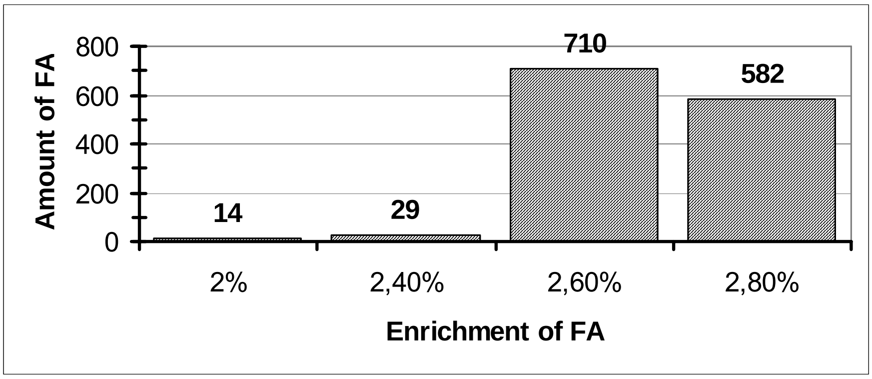


Figure 8.3-1. Distribution of SFA in the reactor according to the initial enrichment

The value of the decay heat of SFA depends on the irradiation rate, time of location of this assembly in the reactor core during power operation of the reactor as well as on time of storage.

Figure 8.3-2 indicates the distribution of SFA located in the reactor according to the energy production.

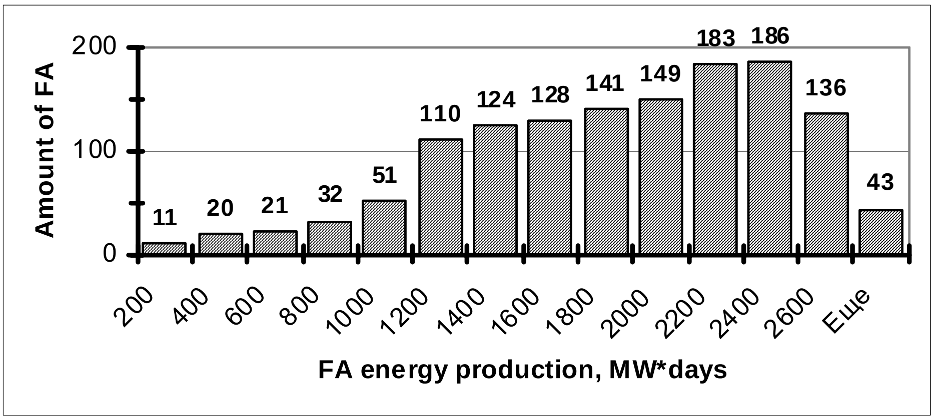


Figure 8.3-2. Distribution of the SFA in the Reactor according to the Energy Production

It is obvious from the distribution that there are the SFA in the reactor that have a full spectrum of working burnouts practically from 0 up to 3000 MW*days. Therefore, in the calculations of the decay heat in the reactor the same algorithm was used as in the calculation of the decay heat of non-cut SFA in the storage pools. The algorithm is described in the radioactive materials accounting system in protective casks of the DSFSF. As a result of the calculation it was obtained that the decay heat W_{dec} of 1335 SFA in the reactor as of 1 July 2011 is 443 kW.

Residual fuel rating will be:

$$W_{dec}/V_{r.c.} = 443\ 000 / 765.5 \sim 579\ W/m^3$$

8.3.3.4 Assessment Calculation of the Unit 2 Reactor Heating Up Process with 1335 SFA in the Reactor Core

In section 8.3.3.3 of this report it is determined that the decay heat W_{dec} in the Unit 2 reactor with 1335 SFA in the reactor core is 443 kW. Below there is the assessment calculation of the necessary time for heating of the reactor core up to the temperature of the graphite cladding 100°C in case of hypothetic loss of all ways of heat removal from the reactor.

From considerations of conservatism in the calculations it is accepted that water in the FC does not circulate and the heat is not transferred through the MCC, the reactor space is not blown down, CPS CC is switched off and there is no heat dissipation from the reactor to the environment.

In order to simplify the pattern of the calculations it is accepted that the decay heat source is evenly distributed throughout the volume of the reactor core. Processes of the heat transfer from the FA to the graphite cladding are not considered. Taking into account small diameters of FE, small thickness of claddings and expected relative duration of the graphite cladding temperature increase process it was accepted that the temperature increment occurs simultaneously for the FE, water and graphite cladding.

As of July 2011 taking into account that there were 1335 SFA in the Unit 2 reactor core and that cooling of the reactor core was carried out in the natural circulation mode the average measured temperature of the graphite cladding was 45°C. The temperature of water in the MCC within the range of ±3°C corresponds to the average temperature of the graphite cladding.

The initial data for the calculation of time of heating up of the graphite cladding are presented in Table 8.3-4.

TABLE 8.3-4

Description	Notation	Value
Decay heat in the reactor core, kW	W_{dec}	443
Number of fuel channels, pieces	N_{FC}	1661
Number of CPS channels within the reactor core, pieces	N_{CPS}	235
Number of graphite columns in the reactor core, pieces	N_{gr}	1896
Number of FA in the reactor core, pieces	N_{FA}	1335
Reactor core height, m	$H_{r.c.}$	7
Cell dimensions, m	l	0.25
External diameter of FC and CPS channels, m	d_{ext}	0.088
Internal diameter of FC, m	d_{int}	0.080
Internal diameter of CPS channels, m	d_{CPS}	0.082
External diameter of FE, cm	d_{FE}	1.36
External diameter of the bearing central tube, cm	d_{ct}	1.50
Initial temperature of the reactor core, °C	$t_{initial}$	45
Final temperature of the reactor core, °C	t_{final}	100
Specific density of reactor graphite, g/cm^3	γ_{gr}	1.70
Specific density of zirconium, g/cm^3	γ_{zr}	6.5
Specific density of water, g/cm^3	γ_{w}	0.98
Mass of fuel in FA, kg	M_f	125
Specific heat of graphite, kJ/kg°C	C_{gr}	0.982
Specific heat of zirconium, kJ/kg°C	C_{zr}	0.293
Specific heat of water, kJ/kg°C	C_w	4.186
Specific heat of uranium-erbium fuel, kJ/kg°C	C_f	0.287

Volume of the graphite cladding within the reactor core:

$$V_{gr} = N_{gr} \cdot H_{r.c.} \cdot (l^2 - \pi \cdot d_{ext}^2 / 4) \qquad (8.1.)$$

$$V_{gr} = 1896 \cdot 7 \cdot (0.25^2 – 3.1416 \cdot 0.088^2 / 4) = 749\ m^3;$$

Mass of graphite:

$$M_{gr} = V_{gr} \cdot \gamma_{gr} \qquad (8.2.)$$

M_{gr} = 749 · 1700 = 1.273·10^6 kg;

Heat accumulated by the graphite cladding:

$$Q_{gr} = M_{gr} \cdot C_{gr} \cdot (t_{final} - t_{initial}) \quad (8.3.)$$

Q_{gr} = 1.273·10^6 · 0.982 · (100 - 45) = 68.75·10^6 kJ

Volume of zirconium in the FC and CPS channels:

$$V_{zr} = H_{r.c.} \cdot \pi \cdot (N_{FC} \cdot (d_{ext}^2 - d_{int}^2) + N_{CPS} \cdot (d_{int}^2 - d_{CPS}^2)) / 4 \quad (8.4.)$$

V_{zr} = 7 · 3.1416 · (1661 · (0.088^2 – 0.080^2) + 235 · (0.088^2 – 0.082^2)) / 4 = 13.6 m^3;

Mass of zirconium in the reactor core:

$$M_{zr} = V_{zr} \cdot \gamma_{zr} \quad (8.5.)$$

M_{zr} = 13.6 · 6500 = 8.84·10^4 kg;

Heat accumulated by zirconium in the FC and CPS channels:

$$Q_{zr} = M_{zr} \cdot C_{zr} \cdot (t_{final} - t_{initial}) \quad (8.6.)$$

Q_{zr} = 8.84·10^4 · 0.293 · (100 - 45) = 1.42·10^6 kJ

Volume of water in the FC within the reactor core:

$$V_w = H_{r.c.} \cdot \pi \cdot (N_{FC} \cdot d_{int}^2 - N_{FA} \cdot (18 \cdot d_{FE}^2 + d_{ct}^2)) / 4 \quad (8.7.)$$

V_w = 7 · 3.1416 · (1661 · 0.080^2 - 1335 · (18 · 0.0136^2 + 0.0150^2)) / 4 = 32.36 m^3;

Mass of water within the reactor core:

$$M_w = V_w \cdot \gamma_w \quad (8.8.)$$

M_w = 32.36 · 980 = 3.17·10^4 kg;

Heat accumulated by water in the FC:

$$Q_w = M_w \cdot C_w \cdot (t_{final} - t_{initial}) \quad (8.9.)$$

Q_w = 3.17·10^4 · 4.186 · (100 - 45) = 7.3·10^6 kJ

Heat accumulated by fuel:

$$Q_f = N_{FA} \cdot M_f \cdot C_f \cdot (t_{final} - t_{initial}) \quad (8.10.)$$

Q_w = 1335 · 125 · 0.287 · (100 - 45) = 2.63·10^6 kJ

The time required for the rise of the reactor core temperature from initial temperature 45°C up to temperature of boiling of water 100°C in FC, if there is no any heat removal, will be:

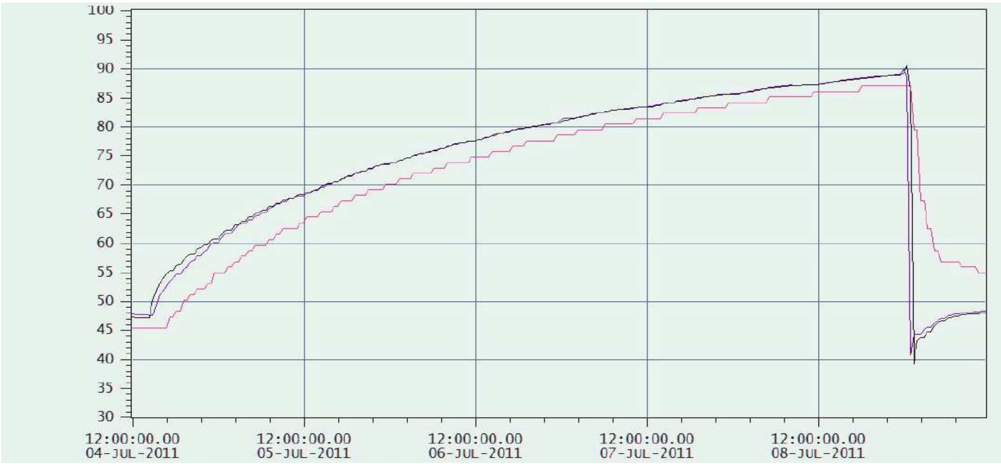
$$t = (Q_{gr} + Q_{zr} + Q_w + Q_f) / W_{dec} \quad (8.11.)$$

t = (68.75·10^6 + 1.42·10^6 + 7.3·10^6 + 2.63·10^6) / 443 = 1.80·10^5 sec = 50.2 hours = 2 days.

According to the INPP electrical annex breakdown elimination instruction [5.58] the power system restoration is possible in approximately 30 minutes after the system breakdown.

Thus, the time span available is more than sufficient for restoration of the heat removal from the reactor plant. In this case it also shall be taken into account that the calculation is carried out under unduly conservative conditions. In the calculations neither heat dissipation from the RP into the environment nor heat removal from the RP by forced ventilation of the CH and rooms of the DS and SWP were taken into consideration. Restoration of natural circulation of the coolant will allow effectively and quickly reducing of the temperature of the reactor core to a reference value.

The fixed real process of the heating up of the Unit 2 reactor graphite cladding in case of changing of the reactor cooling mode from the natural circulation mode to the disrupt natural circulation mode for SFA unloading can serve as an indirect confirmation of undue conservatism of the accepted assumptions. In this case the drum separators are emptied, the level in the reactor core is controlled according to the MTC level gauges and repair level gauges connected to the ECCS headers (see Figure 8.3-3).



	2YH16T01	MCC TEMPERATURE (LEFT HALF)	GOOD48.02	DEGREE
	2YH26T01	MCC TEMPERATURE (RIGHT HALF)	GOOD48.27	DEGREE
	2ZN00T01.300	MAXIMUM MEASURED TEMPERATURE OF GRAPHITE IN FC	GOOD54.9	DEGREE

Figure 8.3-3. Graphite Cladding Heating Up in case of the Reactor Cooling Transfer from the Natural Circulation Mode to the Disrupt Natural Circulation Mode.

Over 4.5 days from 4 July 2011 till 9 July 2011 the maximum temperature of the graphite cladding raised in total by 41.5°C (from 45.5°C up to 87°C). The maximum heating up rate during the first day was 19°C/day that corresponds to the calculations presented above. During the presented process of the graphite cladding heating up in the Unit 2 reactor core there were from 1332 up to 1317 pieces of FA.

Considering that during the defueling of Unit 2 reactor the amount of FA in the reactor core will decrease gradually, and also in view of gradual decrease of the SFA decay heat and increase of storage time it can be guaranteed that safety of Unit 2 reactor will be ensured in any emergency situation related to the loss of heat removal from the reactor.

It also shall be noted that according to the technical regulations [5.30] the maximum measured temperature of graphite of the reactor cladding shall not exceed 150°C. It practically doubles all available time reserves for restoration of the heat removal from the reactor.

8.3.3.5 Calculation of the Temperature Regime of Water in the Unit 1 SFP

Calculation of the temperature regime of water in Unit 1 SFP was performed after the full defueling of the Unit 1 reactor core in 2010 within the framework of the Unit 1 safe operation assessment at a stage of unloading of fuel from the SFA storage pools during INPP Unit 1 decommissioning. [5.36]

The calculations have shown that as of 1 January 2010 the rate of rise of temperature of water in the most heat-stressed compartment (236/2) in Unit 1 SFP, in case there is no heat removal by PHEU, is about 0.13°C/h, while the time during which the temperature of water there can reach the value close to the temperature of boiling (95°C), is approximately 16 days. In fact, the operation experience shows that in the period from May till July 2011 the increase of temperature of water in SFP (236/2) was 5°C (increase from 33 up to 38°C).

8.3.3.6 Calculation of the Temperature Regime of Water in the Unit 2 SFP

Under normal operation conditions the temperature of water in SFP is constantly controlled and maintained below 50°C by the Pump and Heat Exchanger Unit. The time permissible for switching-off of PHEU is determined in view of the condition of non-admission of water boiling up in the storage pools compartments.

In order to define the time permissible for switching-off of PHEU the calculation of the decay heat of SFA in each SFP compartment has been carried out as of 1 July 2011. The calculation of the decay heat of the cut SFA loaded into 32M baskets was performed using the system of accounting of radioactive materials in protection casks of the DSFSF for each 32M basket and then was summarized according to all 32M baskets installed in the compartment. The system of accounting of radioactive materials in the protection casks of the DSFSF calculates the decay heat only for the SFA loaded into the 32M baskets. Therefore, the decay heat of non-cut SFA stored in compartments 236/1, 236/2 has been calculated using the Excel spreadsheet. The calculations were carried out using the same algorithm as in the System of Accounting of Radioactive Materials in protection casks of the DSFSF [5.91]. The calculation was carried out by the method of interpolation. Interpolation [5.90] was performed in two stages. At the first stage the decay heat interpolation y_i was performed for enrichment and storage time set for SFA for five points with burnout z_1=5MW×day/kg, z_2=15MW×day/kg, z_3=20MW×day/kg, z_4=25MW×day/kg, z_5=30MW×day/kg. The combination of two exponents was used as a functional basis:

$$y_i = y_0(z_i) + A_1(z_i) \times \exp(-(x-x_0(z_i))/t_1(z_i)) + A_2(z_i) \times \exp(-(x-x_0(z_i))/t_2(z_i)) \qquad (8.12.)$$

where, y means the decay heat at the moment in time x;

$x_0(z_i)$, $y_0(z_i)$, $A_1(z_i)$, $A_2(z_i)$, $t_1(z_i)$, $t_2(z_i)$ are the parameters depending on enrichment of the fuel and its burnout.

At the second stage the Lagrange interpolation formula was used in order to calculate the value of the SFA decay heat at burnout z:

$$y = \frac{(z-z_2)\times(z-z_3)\times(z-z_4)\times(z-z_5)}{(z_1-z_2)\times(z_1-z_3)\times(z_1-z_4)\times(z_1-z_5)}\times y_1 + \frac{(z-z_1)\times(z-z_3)\times(z-z_4)\times(z-z_5)}{(z_2-z_1)\times(z_2-z_3)\times(z_2-z_4)\times(z_2-z_5)}\times y_2 + \frac{(z-z_1)\times(z-z_2)\times(z-z_4)\times(z-z_5)}{(z_3-z_1)\times(z_3-z_2)\times(z_3-z_4)\times(z_3-z_5)}\times y_3 + \frac{(z-z_1)\times(z-z_2)\times(z-z_3)\times(z-z_5)}{(z_4-z_1)\times(z_4-z_2)\times(z_4-z_3)\times(z_4-z_5)}\times y_4 + \frac{(z-z_1)\times(z-z_2)\times(z-z_3)\times(z-z_4)}{(z_5-z_1)\times(z_5-z_2)\times(z_5-z_3)\times(z_5-z_4)}\times y_5 \qquad (8.13.)$$

The results of the calculations are presented in Tables 8.3-5 and 8.3-6.

TABLE 8.3-5

SNF Decay Heat in Compartments with Cut Fuel

Number of SFP Compartment	Quantity of Baskets in Compartment, pieces	Decay Heat, kW
234	1	13.5
235	2	25.3
336	28	147.3
337/1	21	141.2
337/2	21	143.7
338/2	1	9.8
339/1	16	79.9
339/2	16	60.7
Total	106	607.9

TABLE 8.3-6

SNF Decay Heat in Compartments with Non-Cut Fuel

Number of SFP Compartment	Quantity of SFA, pieces	Decay Heat, kW
236/1	2% enrichment - 59	8.9
	2.4% enrichment - 15	3.8
	2.6% enrichment - 407	109.2
	2.8% enrichment - 225	53.3
Total in 236/1	706	**175.1**
236/2	2% enrichment - 126	6.3
	2.4% enrichment - 56	7.9
	2.6% enrichment -512	127.4
	2.8% enrichment -239	54.9
Total in 236/2	933	**196.5**
Total	1638	371.6

It can be seen from the provided calculations that the most temperature stressed SFP compartments are 236/2 (196.5 kW) and 336 (147.3 kW). The time permissible for switching-off of PHEU was calculated for these compartments.

The accumulating capacity of water, SFA and cartridges metal, fuel and of the metal cladding of the compartment was taken into account in the calculation. The heat removal from the water plane to the air ventilating the above-water space under the slot floor is not taken into account since in case of the full loss of power supply the ventilation system can fail.

In view of rather low level of temperature of water in the storage pools, massiveness of its building structures, and consequently, plenty of time for heating up of the building structures, the accumulating capacity of them is not taken into account that goes to a stock of calculation.

Initial data for calculation of time permissible for PHEU switching-off are presented in Table 8.3-7. In order to simplify the calculations the zirconium elements of SFA were considered as steel elements that practically has no impact on the final result.

TABLE 8.3-7

Characteristics	**Notation**	**Compartment 236/2**	**Compartment 336**
Width of compartment, m	a	5.2	5.2
Length of compartment, m	b	8.6	8.6
Height of compartment, m	h	16.85	11.55
Thickness of the liner of compartment walls, m	δ_p	0.003	0.003
Thickness of the compartment tray, m	δ_d	0.005	0.005
Quantity of SFA in cartridges, pieces	n_1	325	-
Quantity of SFA without cartridges, pieces	n_2	608	-
Quantity of the cut SFA, pieces	n_{cut}	-	1428
Quantity of 32M baskets, pieces	n_b	-	28
SFA decay heat, kW	W	196.5	147.3
Initial temperature of water in compartment, °C	$t_{initial}$	50	
Final temperature of water in compartment, °C	t_{final}	95	
Mass of SFA without suspension, kg	m_1	185	
Mass of SFA with suspension, kg	m_2	280	
Mass of fuel in SFA, kg	m_f	125	
Mass of cartridge, kg	m_c	145	
Mass of 32M basket, kg	m_{32Mb}	3780	
Specific density of steel, kg/m^3	γ_{steel}	7800	
Specific density of fuel, kg/m^3	γ_f	10400	
Specific density of water, kg/m^3	γ_w	978	
Specific heat of metal, kcal/kg°C	C_m	0.12	
Specific heat of water, kcal/kg°C	C_w	1.0	
Specific heat of fuel, kcal/kg°C	C_f	0.065	

Geometrical volume of the compartment from the tray to the water plane:

$$V_{comprt} = a \times b \times h \tag{8.14.}$$

$V_{comprt}(236/2) = 753.5$ m^3, $V_{comprt}(336) = 516.5$ m^3

Mass of metal of the walls liner from the tray to the water plane in the compartment:

$$M_{steel} = V_{steel} \times \gamma_{steel} = 2(a+b) \times h \times \delta_P \times \gamma_{steel} \tag{8.15.}$$

$M_{steel}(236/2) = 10880$ kg, $M_{steel}(336) = 7459$ kg

Mass of metal of the tray of the compartment:

$$M_{tray} = V_{tray} \times \gamma_{steel} = a \times b \times \delta_D \times \gamma_{steel} \quad (8.16.)$$

$M_{tray}(236/2) = M_{tray}(336)$ = 1740 kg.

Quantity of SFA in the compartment:

$$n(236/2)=n_1+n_2 \quad (8.17.)$$

$$n(336)=n_{cut} \quad (8.18.)$$

$n(236/2)$= 933 pieces $n(336)$= 1428 pieces.

Mass of cartridges (compartment 236/2):

$$M_c=m_c \times n_1 \quad (8.19.)$$

$M_c(236/2)$= 47125 kg.

Mass of 32M basket (compartment 336):

$$M_{32Mb}=m_b \times n_b \quad (8.20.)$$

$M_{32Mb}(336)$= 105840 kg.

Mass of the SFA with suspensions (compartment 236/2):

$$M_{FA}=m_2 \times n \quad (8.21.)$$

$M_{FA}(236/2)$= 261240 kg.

Mass of the SFA without suspensions (compartment 336):

$$M^{Cut}{}_{FA}=m_1 \times n \quad (8.22.)$$

$M^{Cut}{}_{FA}(336)$= 264180 kg.

Mass of fuel in the compartment:

$$M_F=m_F \times n \quad (8.23.)$$

$M_F(236/2)$= 116625 kg. $M_F(336)$= 178500 kg.

Mass of the SFA metal in the compartment:

$$M^M_{FA}(236/2)= M_{FA} - M_F \quad (8.24.)$$

$$M^M_{FA}(336)= M^{Cut}{}_{FA} - M_F \quad (8.25.)$$

$M^M_{FA}(236/2)$= 144615 kg, $M^M_{FA}(336)$= 85680 kg

Volume in the compartment occupied by metal:

$$V_M(236/2) = \frac{M_c + M^M_{FA}}{\gamma_{steel}} \quad (8.26.)$$

$$V_M(336) = \frac{M_{32Mb} + M_{FA}^M}{\gamma_{steel}} \qquad (8.27.)$$

$V_M(236/2)$= 21 m^3, $V_M(336)$= 24.6 m^3

Volume in the compartment occupied by fuel:

$$V_F = \frac{M_F}{\gamma_F} \qquad (8.28.)$$

$V_F(236/2)$= 11.2m^3 $V_F(236/2)$= 17.2 m^3

Volume of water in the compartment:

$$V_W = V_{comprt} - (V_M + V_F) \qquad (8.29.)$$

$V_W(236/2) \approx$ 721 m^3 $V_W(336) \approx$ 475 m^3

Temperature increment in the compartment:

$$\Delta t = t_{final} - t_{initial} \qquad (8.30.)$$

Δt= 45 °C.

In view of thin walls of the metal cladding and cartridges, small diameters of the FE, suspensions and expected relative duration of the temperature rise process in the compartment, it is accepted that an increment of temperature for the FE, water and metal occurs simultaneously, i.e. there is no a delay process.

Heat accumulated by water of the compartment:

$$Q_W = V_W \times \gamma_W \times C_W \times \Delta t \qquad (8.31.)$$

$Q_W(236/2)$= 3.1×10^7 kcal. $Q_W(336)$= 2.04×10^7 kcal.

Heat accumulated by metal of the compartment:

$$Q_M(236/2) = M_M \times C_M \times \Delta t = \left(M_{steel} + M_{tray} + M_c + M_{FA}^M\right) \times C_M \times \Delta t \qquad (8.32.)$$

$$Q_M(336) = M_M \times C_M \times \Delta t = \left(M_{steel} + M_{tray} + M_{32Mb} + M_{FA}^M\right) \times C_M \times \Delta t \qquad (8.33.)$$

$Q_M(236/2)$= 1.1×10^6 kcal, $Q_M(236/2)$= 1.2×10^6 kcal.

Heat accumulated by fuel of the compartment:

$$Q_F = M_F \cdot C_F \cdot \Delta t \qquad (8.34.)$$

$Q_F(236/2)$= 3.4×10^5 kcal, $Q_F(336)$= 5.2×10^5 kcal.

Time during which the temperature of water in the SFP compartment will rise from the initial temperature (50°C) up to the temperature close to boiling up (95°C) in case there is no heat removal by the PHEU and there is no heat transfer to the water of the other compartments:

$$T = \frac{Q_W + Q_M + Q_F}{W} \qquad (8.35.)$$

T(236/2)=189 h ≈8 days *T(336)*= 174 h ≈7 days.

The results of the calculation show that the rate of rise of water temperature in the compartment, in case there is no heat removal by the PHEU, is low and is about 0.26 °C/h, while the time during which the water in the most temperature stressed compartment can reach the value close to boiling up (95°C) is approximately 7 days. This time is quite sufficient to eliminate the malfunction of the PHEU and to recover its functionality.

8.3.3.7 Loss of the Main Ultimate Heat Sink

The analysis of design (operational) modes of the reactor cooling (Table 8.3-3) shows that the loss of the main UHS can have an impact on heat removal from the reactor in the following modes: in the non-boiling mode of natural circulation due to BCS operation and in the mode of forced circulation of the coolant by the cooling pumps. In these modes the loss of the main UHS will not cause any malfunctions in heat removal from the reactor since in both cases the reactor cooling mode will switch over (can be transferred by an operator) to the boiling mode of natural circulation. In the second case, if the pressure gate valves and/or suction gate valves of all MCP and on all connecting pipes of MCP PH-SH and the gate valves on the inlet to each DGH are closed, the reactor cooling mode will switch over to the coolant bubbling mode, i.e. to the operational modes in which the heat is removed to the alternative UHS and in which the cooling of any half of the reactor core is ensured within the unlimited time by means of steam removal from the DS through BRU-B in ALT and periodic makeup of the MCC [5.30].

In case of loss of the main UHS the heat removal from the SFA located in SFP can be carried out by water exchange in the SFP: discharge of water from SFP to tank TZ50B01 and further to TD51B01 and makeup from TD52B01 (CPW and SPC tanks). Thus the heat removal can be carried out with the flow rate up to 100 m^3/hour during not less than 48 hours [5.49].

8.3.3.8 Loss of the Main and Alternative Ultimate Heat Sink

Situations in which the loss of both main and alternative ultimate heat sinks is possible are completely covered by situations which can occur only in case of blackout of the power plant. Therefore, they are not considered in this section.

8.3.4 Loss of Heat Removal to the Ultimate Heat Sink with the Loss of External Power Supply

At full blackout of the power plant all the design means of heat removal to UHS (main and alternative) and the means of MCC makeup are not available (do not function). As a result of disbalance between heat emission and heat removal there is a loss of the coolant due to its evaporation and discharge through BRU-B to ALT. The warming up of the core components can begin after evaporation of water stocks from DS and MCC pipelines located above the core of the reactor.

At the analysis of the core cooling support conditions in various modes of heat removal from the reactor (see section 8.3.3.1 of the given report) it is seen that owing to the lowered water level in MCC not more than to 1 meter below FC plugs (not lower than level

+23.7 m) the most critical in time mode for probable damage of the fuel is the mode of the disrupt natural circulation.

As it is specified above, the removal of decay heat in this mode is carried out by removal of the steam from DS and makeup of the core by the natural water flow, from MCT; thus the water level in MCC is maintained at the level specified above by periodic MCT makeup from system CPW and SPC. Besides, a part of decay heat is removed as a result of process ventilation 2WZ51 operation in Rooms 210 and 506/1,2 of Unit A2. Thus, cooling of DS, SWP and pipe guides, condensation of the steam and return of condensate to MCC (FC) occur.

Loss of MCT makeup and process ventilation will lead to decrease in a water level in FC due to evaporation. Growth of FE temperature can begin only after decrease of a water level below the top of the reactor core [5.93].

In the report [5.1] the results of the conservative (at calculation of FE claddings temperature the presence of water in SWP and dispersion of the heat into the environment were not considered) analysis of water evaporation from FC and heating of dehydrated FC are provided. For the case, when circulation of water in FC stops in 365 days after the reactor shutdown, it is established that the time, which will be needed for evaporation of the water from FC and when it is possible to expect the achievement of the acceptance criterion for FE claddings 700^{o}C, is more than 140 hours (6 days). The tendency of change of FE claddings temperature at emptying of FC in 365 days after the reactor shutdown is presented in Figure 8.3-4.

The water stock in SWP and FC above the core (for two halves of MCC) is roughly from 145 up to 152 m^3 [5.46]. Conservatively, full evaporation of the specified amount of water, taking into account the decay heat of 1335 SFA located in the reactor, will take approximately 8.5 days.

For prevention of fuel degradation, in accordance with the instruction [5.61] developed at INPP, the possibility of supply of the artesian water to MCC from the water intake area from DPW system is foreseen, for which realization of modification МОД-05-02-718 “Supply of Domestic Potable Water to Fuel Channels through MCT Pipelines” is required. Thus, DPW pumps have the possibility of powering from the own diesel generator, therefore the given system is considered to be enough reliable.

At full blackout of the power plant there is a loss of the main and alternative ultimate heat sinks for SFP that will inevitably lead to the gradual growth of the water temperature in SFP. As it is shown in sections 8.3.3.5, 8.3.3.6, in the most heat-stressed SFP the water temperature can reach the value close to the temperature of boiling for Unit 1 in 16 days and for Unit 2 in 7 days. At the further non-availability of heat removal from SFA the evaporation of water and decrease of a water level in SFP will occur.

The level of 3940 mm from the SFP floor (at level +21.26 m) is considered to be critical for the water level in SFP, which corresponds to the top of fuel at its placement in 102-place baskets (the top and bottom tiers). Nevertheless, no damage of fuel occurs at that, it occurs in 1005 hours after the beginning of SFP emptying and exposure of fuel in 102-place baskets (the top tier) – already in 361 hours after the beginning of SFP emptying [5.93].

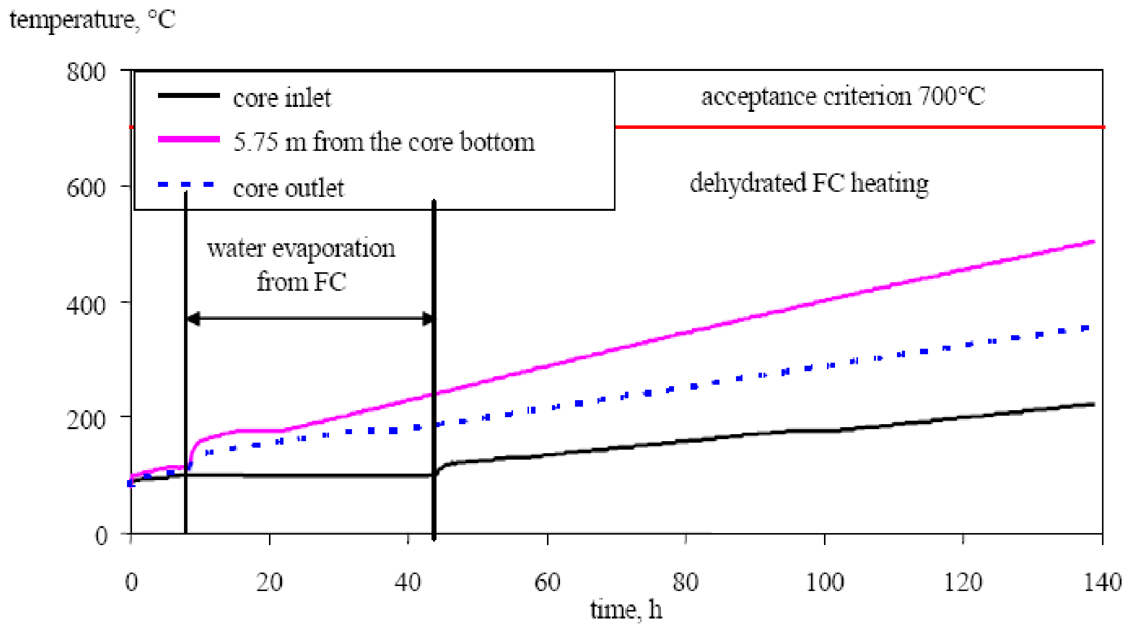


Figure 8.3-4. Change of FE claddings temperature at emptying of FC in 365 days after the reactor shutdown

With the purpose of prevention of the mass damage of SFA during the accidents concerned with the drop of the water level in SFP, the instruction [5.59] has been developed at INPP. According to the given instruction, the supply of artesian water to SFP from the water intake area from DPW system through the fire hydrants is foreseen as the strategy of restoration of the level in SFP.

8.4 Chapter 4. Severe Accidents Management

8.4.1 Assessment of the Current State of Fuel

The assessment of the current state of the fuel in the reactor of power Unit 2 and storage pools of power Units 1 and 2 according to the report [5.40] and section 8.3 of the present report is presented below.

The Ignalina Nuclear Power Plant has been finally shutdown for decommissioning and has been being in such mode already for a long time (power Unit 1 for 6.5 years, power Unit 2 for 1.5 years). Mass repair works are not carried out at the power units. Only the required minimum of the equipment maintenance is carried out. For the period of the power units being in a shutdown mode such critical safety functions as reactivity management (the reactor subcriticality has increased) and heat removal from the fuel (the decay heat of FA located both in the reactor and in SFP has decreased) have considerably improved. This considerably reduces the risks of nuclear fuel damage and radioactive substances discharge into the environment in the case of an event occurrence.

In Figure 8.4-1 the change of the value of effective multiplication factor (K_{eff}) during SFA unloading [5.43] is presented. As it is seen in the figure, after 110 FA unloading the reactor transition to a critical state is impossible, even at retrieval of all CPS rods. Thus, at the loss of the reactivity control at present there is no risk of uncontrolled transition to a critical state.

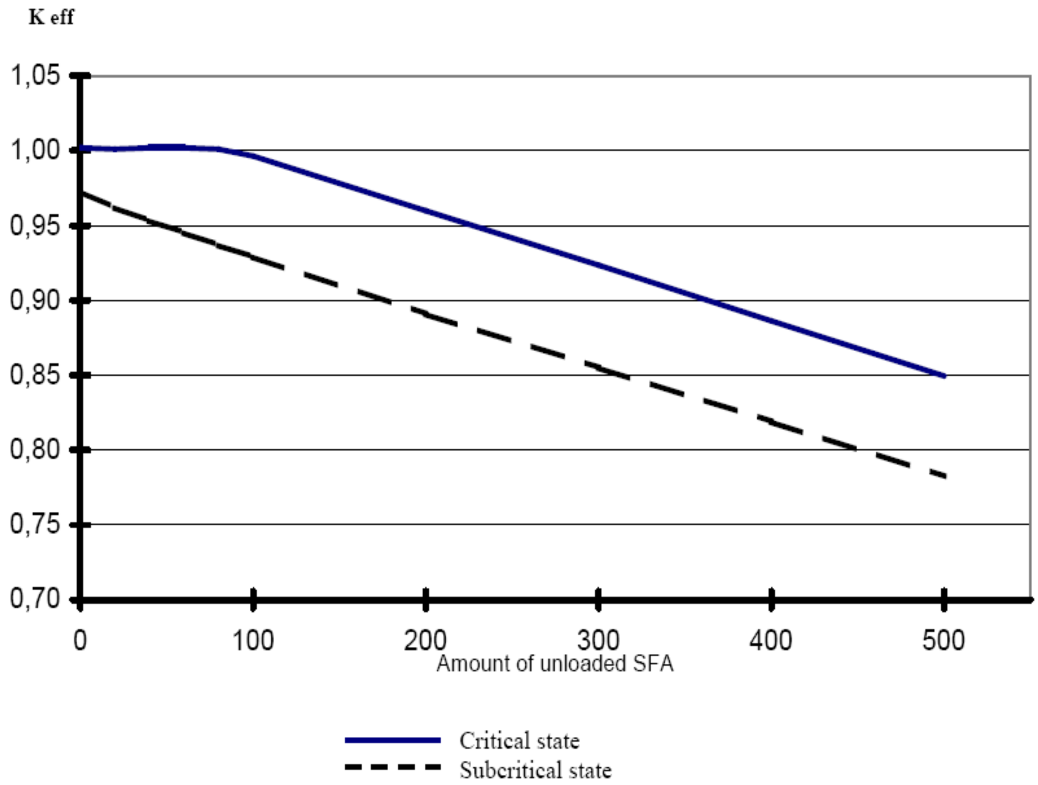


Figure 8.4-1. Change of K_{eff} value during 500 SFA unloading

The longtime being of the reactor of power Unit 2 in the shutdown mode has led to the significant decrease of decay heat of FA located in the reactor. In Figure 8.3-3 of section 8.3 of the present report the schedule of the MCC left and right halves coolant heating (points 2YH16, 26T01 - the centre of the reactor core) in the mode of the disrupt natural circulation is presented. As it is seen in the presented schedule, boiling up of the coolant in the core of the reactor is possible not earlier than in 92 hours after emptying of drum-separators.

The longtime shutdown mode of the reactors has also led to significant decrease of decay heat in the FA storage pools. Conservative (without taking into account the dispersion of the heat into the environment) assessment of the heating of the most energy intensive SFP compartment with SFA in case of the cooling loss, executed in 2010 [5.1], showed that the increase of water temperature from 50ºC up to 95ºC at power Unit 1 lasted for more than 400 hours, at power Unit 2 - for more than 110 hours.

According to the results of the analysis provided in section 8.3 of the present report and executed for the current state of the storage pools of power Unit 2, the increase of water temperature from 50ºC up to 95ºC will last for more than 174 hours, and in the process of reduction of SFA energy release in the storage pools this time will constantly increase.

Thus, power Unit 2 reactor partial defuelling and the longtime shutdown mode of the reactor have led to significant decrease in the risk of FA damage in the reactor of power Unit 2 and the storage pools of power Units 1 and 2.

8.4.2 Assessment of Prevention of Fast Combustion or Detonation of Hydrogen

The assessment of prevention of fast combustion or detonation of hydrogen for the reactor of power Unit 2 and the storage pools of power Units 1 and 2 is presented according to the report [5.45].

For RBMK reactors the processes (sources) leading to generation of hydrogen, except for the processes in which hydrogen is applied at normal operation, according to the “Rules of Hydrogen Explosion Proofing at Nuclear Power Plant HP-040-02” (Moscow, 2002) are:

- water radiolysis in the main circulation circuit and in the reactor control and protection system circuit at power operation of the reactor and at shutdown;
- water radiolysis in the nuclear fuel storage pools;
- corrosion of constructional materials.

For the current INPP state the actual processes (sources) leading to generation of hydrogen are:

- water radiolysis in Units 1 and 2 nuclear fuel storage pools;
- water radiolysis in Unit 2 main circulation circuit.

Generation of hydrogen in power Unit 1 MCC and CPS CC is impossible, since they are in the emptied state, and the reactor is defuelled.

During Unit 1 reactor defuelling, after CPS CC emptying the monitoring of CPS CC longtime being in the emptied state was performed under the “Programme of Monitoring for Estimation of the Possibility of CPS CC Longtime Being in the Emptied State after Power Unit 1 Final Shutdown” (ПТОпр-1110-1075). Practical non-availability of hydrogen in the emptied CPS CC was confirmed.

Recommendations of the “Act on the Results of Monitoring of CPS CC Longtime Being in the Emptied State after Power Unit 1 Final Shutdown” (ПТOота-1147-13) based on the results of Unit 1 CPS CC monitoring were considered at emptying of Unit 2 CPS CC and during the development of the instruction [5.55].

The hydrogen generated as a result of water radiolysis in SPH of Units 1 and 2 is diluted and removed from the storage pools hall by the supply-and-exhaust ventilation systems. The extraction of air from SPH is carried out by the intake of air from the above-water area of SFP compartments. As it is shown in the report [5.1], even at the full loading of all SNF storage compartments, both at normal operation and at continuous emergency disconnection of ventilation, generation of explosive concentration of hydrogen in the above-water volumes and in the storage pools hall does not occur.

Generation of radiolytic hydrogen in Unit 2 MCC is related to the fact that nuclear fuel is still in the reactor core. The hydrogen generated in the reactor FC (core) as a result of water radiolysis comes to DS and steam lines. In order to avoid accumulation of radiolytic hydrogen in MCC (DS, steam lines) during Unit 2 reactor defuelling, ventilation of DS and steam lines via open air vents and via BRU-B to the top steam reception chambers ALT is carried out. For prevention of hydrogen accumulation in the top part of TSRCh, blowdowns from TSRCh in the ALT decay chamber through Du 6 holes in the top part of a branch pipe of the vacuum breaking action valve are stipulated by the design. From the decay chamber the gases are discharged into the environment through the corresponding ALS exhaust ventilation systems.

In 2002 modification МОД-01-12-328 "Plugging of Pipelines Du 600 of Blowdown from TSRCh ALT to Power Units No 1, 2 FC" was implemented at power Units 1 and 2, concerning the plugging of not applied pipelines Du 600 of the steam blowdown from TSRCh to the process condenser that excluded the presence of not ventilated volumes related to TSRCh, where accumulation of hydrogen is probable.

According to the design [5.24], for the period of Unit 2 reactor defuelling the design systems of hydrogen concentration in ALS rooms monitoring and registration maintenance, the compressed air system intended for decrease of hydrogen concentration in ALT and ventilation systems providing design conditions of environment for ALS remain in operation.

During INPP decommissioning the design solutions, directed on prevention of hydrogen accumulation, remain constant till the final elimination of the process (source) of hydrogen release [5.24].

Thus, hydrogen monitoring and registration in the places of its probable accumulation, impossibility of hydrogen explosive concentration generation are provided by the design; thereof the development of additional actions is not required.

8.4.3 Description of Beyond Design-Basis Accident "Full Blackout of INPP Auxiliaries"

The description of beyond design-basis accident "Full Blackout of INPP Auxiliaries" for power Unit 2 reactor is presented below according to the instruction [5.62] and the report [5.95]. The calculations were made without taking into account the operator's interference, for power Unit 2 reactor state right after its shutdown on 31 December, 2009. At simulation it was accepted that for the FA of 2.6 % U^{235} with the depth of burnout 25 MWday/kgU, the decay heat after 1 day of the reactor shutdown is 13.99 kW/FA.

Full blackout of INPP auxiliaries refers to the switching-off of electric power supply of the consumers due to the loss of a power supply of all normal and reliable power supply 6 kV sections from all power Unit 2 SAT, WT and DGs. Full blackout of the power unit is the beyond design-basis accident and causes the loss of water supply to MCC from all active power plant makeup sources (pumps).

At the initial moment of the beyond design-basis accident conservatively the following state is accepted - MCC is filled with water, MCPs are stopped, the temperature of water in DS is equal to 98°C, the pressure in DS is atmospheric, the temperature of water in FC and of graphite is 115÷120°C. The decay heat from the reactor is removed in a mode of natural circulation of the coolant by means of DS and SWP rooms ventilation.

After switching-off of the ventilation system the water in SWP pipelines will not be cooled. The water with temperature 115-120°C comes to DS, where the pressure is atmospheric. The water in DS boils, and the steam is discharged via the open BRU-B to the condensate plates CP-15, CP-25 of ALT-1,2. Non-operability of the ALT pump and heat exchanger unit after the beginning of the steam discharge to ALT-1,2 causes boiling up of the water in CP-15, CP-25, increase of the pressure in the ALT decay chamber and the steam discharge via the swing panels into the environment.

ATS of the reliable power supply 6 kV sections is unsuccessful (failure of all DG-7 - DG-12 for the common cause, and absence of the voltage at auxiliary busbars of the startup auxiliary transformer). MCC makeup is not performed due to the absence of a power supply of electric motors of all pumps.

The volume of the water in DS decreases from the beginning of the accident. The coolant circulates through the fuel channels unless the water level in DS decreases to a level of the SWP bottom lines tie-in to DS, approximately in 8 hours (28800 sec), after that natural circulation stops, and fuel channels are cooled by a steam-and-water mix until evaporation of all the water from the core.

After the beginning of intensive water boiling in FC (in 8 hours after the loss of electric power supply) the pressure of FC - DS water column starts to decrease, since a part of the water is evaporated. After the beginning of water boiling in FC the heat-transfer factor from FA to the coolant decreases a little, but remains sufficient for reliable cooling of FA - the core components temperatures remain within the limits of 100 - 130^{o}C.

The sharp increase of the core temperatures in 18 hours (64800 sec) after the beginning of the accident is related to the core emptying. Prior to that FA are reliably cooled in a mode of the coolant natural circulation with the steam discharge via BRU-B and in a mode of the water boiling in FC.

At blackout of the INPP auxiliaries, if not to undertake reactive actions, in 18 hours the fuel channels with located FAs will be emptied, and further the FAs will be heated up above 700ºC, as a result FE claddings will be damaged and fission products, which are contained inside the cladding, will discharge into the environment via ALS.

At actuation of emergency protection on disappearance of the voltage at all 6 kV buses sections of the power unit auxiliaries electric power supply, according to the requirements of the instruction [5.63], the following actions on ensuring of the heat removal from the reactor are carried out:

- during heat removal from the reactor in a mode of forced circulation of the coolant by the cooling pumps, after completion of the step-by-step load connection automatics operation, the cooling pumps are put into operation, BCS and PHEU ICC-1 operation parameters are restored;
- during heat removal from the reactor in a mode of the disrupt natural circulation of the coolant MCT makeup operation is provided;
- reliable power supply sections power supply is restored via the section switches from the normal power supply sections.

In case of the unsuccessful actions under the instruction [5.63] at occurrence of the beyond design-basis accident “Full Blackout of INPP Auxiliaries” - the loss of a power supply at all normal and reliable power supply 6 kV sections from all power Unit 2 SAT, WT and DGs, being in operation, the instruction [5.62] is applied. Thus, the following actions are carried out:

- for ensuring of the rector core cooling, the supply of DPW is carried out via DGH (through MCT pipelines) to both MCC halves after realization of modification МОД-05-02-718 “Supply of Domestic Potable Water to Fuel Channels through MCT Pipelines”. While that the supply of DPW to FC shall be provided not later, than in 18 hours after the beginning of the beyond design-basis accident (if it is impossible to supply water to MCC from CPW pumps of Bld. 150);
- for emergency power supply to MCR devices, communication, RSM board, realization of modification МОД-05-02-723 “Bld.101/2, 185 Consumers Supply at Full Auxiliaries Blackout” is carried out (power supply connection from mobile DG to electric assemblies and consumers of Bld. 101/2 and Bld. 185);

- for reception of the steam from the core all ALS water and air volumes are applied - prior to the beginning of water boiling at condensate plates CP-15, CP-25 the swing panels of ALT-2 decay chamber are locked and BRU-B No 1 are closed. Non-contaminated steam discharge from MCC is carried out via BRU-B No 2 to CP-25 and further via emptied CP-21 ÷ CP-24 → BSRCh-2 → LTC → BSRCh-1 → CP-11 ÷ 14 and via the swing panels of ALT-1 decay chamber into the atmosphere. DPW supply is carried out from the fire-fighting vehicle via the fire hydrants located on the roof of Unit A2, in the area of ALT-1 open swing panels (for this purpose the call of the fire-fighting vehicle to Unit A2 from ALT-1 side is required).

8.4.4 Unauthorized Decrease of a Water Level in Any Half of MCC

According to the requirements of the instruction [5.63], in case of unauthorized decrease of a water level in any half of MCC the following actions for ensuring the heat removal from the reactor are carried out: the water level in MCC, required for the given mode of the reactor cooling, is restored, using the increase of MCT makeup if MCC MCT are connected, cooling pumps and BCS if a stock of water at another half of MCC is avalaible, one of three independent sources of MCC makeup if a stock of water at another half of MCC is unavailable:

- tank 2TD52B01 with a stock of water not less than 1000 m^3, two of three pumps 2TD61-63D01 are ready for operation;
- tanks TW15B01, TW41B01, TW32B01 with a total stock of water not less than 5000 m^3, two of three pumps TW16D01-03 are ready for operation;
- one HCCh with two ECCS pumps - the first channel of makeup; one HCCh with one ECCS pump - the second channel of makeup.

In case of the unsuccessful actions under the instruction [5.63] at unauthorized decrease of a water level in any half of MCC the instruction [5.61] is applied. Thus, the following actions are carried out:

- strategies S2 are applied - water supply to MCC (S2.1 - water supply to MCC from the high pressure sources, and S2.2 - water supply to MCC from the low pressure sources), as well as strategy S4 - isolation of MCC leakage;
- for application of RUZA-R1 strategies the following modifications are realized, which shall be executed by the repair personnel on demand of the Head of the Technical Support Centre according to the works performance projects prepared in advance: МОД-05-02-717 "Water Supply to Room 125 of Unit A-2", МОД-05-02-718 "Supply of Domestic Potable Water to Fuel Channels through MCT Pipelines", and МОД-05-02-723 "Bld.101/2, 185 Consumers Supply at Full Auxiliaries Blackout".

8.4.5 Procedures Intended for Beyond Design-Basis Accidents Management at INPP

According to the instruction [5.57] the set of beyond design-basis accidents management procedures includes five instructions developed for INPP power Units 1 and 2:

- INPP Beyond Design-Basis Accidents Management Procedures User's Manual [5.57].
- Instruction. Beyond Design-Basis Accidents Management Guideline RUZA-R1. Provision of Heat Removal from INPP Unit 2 Reactor [5.61].
- Instruction. Beyond Design-Basis Accidents Management Guideline RUZA-RB. Reduction of INPP Units 1 and 2 Fission Products Emission [5.60].

- Instruction. Beyond Design-Basis Accidents Management Guideline RUZA-B. INPP Units 1 and 2 Storage Pools Condition Management [5.59].
- Instruction on the Provision of Emergency Heat Removal from Unit 2 Reactor in Case of INPP Full Auxiliaries Blackout [5.62].

All the mentioned above procedures were completely reviewed after power Unit 2 reactor shutdown, and in the process of the decommissioning works performance on the systems and equipment the appropriate amendments are inserted into them according to the order established at INPP.

These procedures contain the description of application at INPP of 10 beyond design-basis accidents management strategies, which are shown in figure 8.4-2.

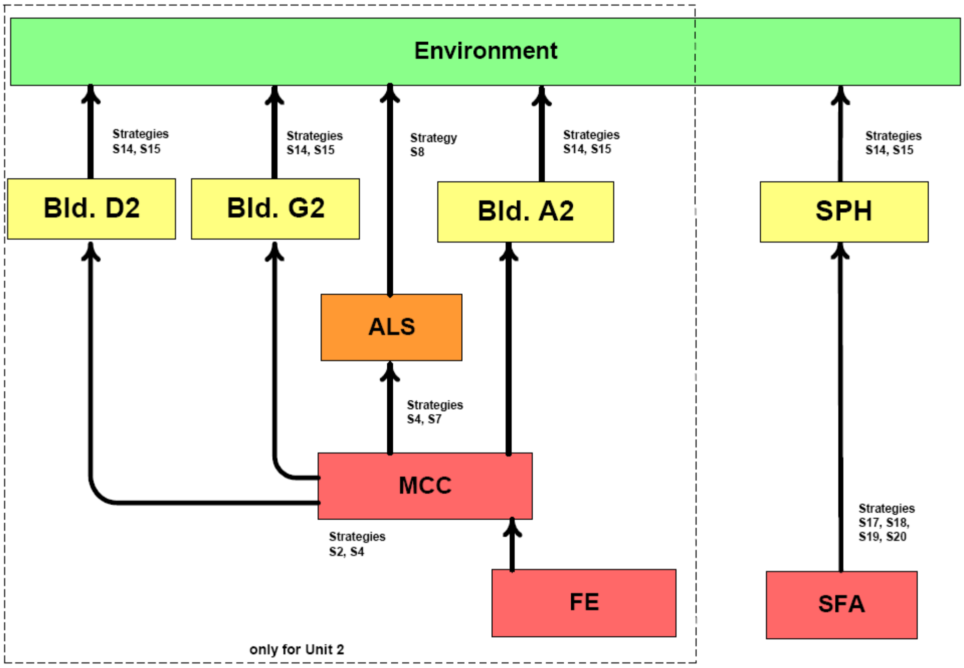


Figure 8.4-2. Diagram of the possible ways of the radioactive fission products emission and beyond design-basis accidents management strategies at INPP

In figure 8.4-2 the following beyond design-basis accidents management strategies are presented:

- S2 - water supply to MCC;
- S4 - isolation of MCC leakage;
- S7 - restoration of ALT CP cooling;
- S8 - ALT ventilation;
- S14 - isolation of the unit emergency rooms;
- S15 - DPW supply via fire hydrants;
- S17 - water supply to SFP;
- S18 - isolation of SFP leakage;

- S19 - supply of absorber to emergency SFP;
- S20 - isolation of emergency SFP from the other pools.

At that, for beyond design-basis accidents related to the failure of heat removal from power Unit 2 reactor, and abnormality of power Units 1 and 2 storage pools state, the application of the following high and low pressure pumps is stipulated in the beyond design-basis accidents management procedures (in brackets the corresponding RUZA, in which they are applied, are indicated):

- ECCS pumps 2TH52,53,61,62D01 (RUZA-R1);
- DEM pumps 2TD66-69D01 (RUZA-R1);
- LSW pumps 1TD61-65D01 (RUZA-R1, RUZA-B);
- LSW pumps 2TD61-65D01 (RUZA-R1, RUZA-B);
- CPW, SPC pumps TW16D01-03 (RUZA-R1).
- CPW pumps TW17D01-03 (RUZA-R1, RUZA-B);
- HTT, SPC pumps TW19D01-03, TW33D01,02 (RUZA-R1);
- Pumps of drain waters pumping out from LTC 2TZ11-13D01 (RUZA-R1);
- Pumps of drain waters pumping out from rooms 2TZ21-23D01 (RUZA-R1);
- Pumps of contaminated LSW pumping out 2TZ51-53D01 (RUZA-R1);
- Service water pumps 2VF11,12,15,16D01 (RUZA-B);
- SFP PHEU pumps 1TG11-14 D01 (RUZA-B);
- SFP PHEU pumps 2TG11-14 D01 (RUZA-B);
- DPW pumps - supply of artesian water from the water intake area from DPW system, for realization of which modification МОД-05-02-718 “Supply of Domestic Potable Water to Fuel Channels through MCT Pipelines” (RUZA-R1, RUZA-B) is required.

Characteristics of low pressure MCC and SFP makeup pumps are presented in Table 8.4-1.

TABLE 8.4-1

Pump name and tagging	2TD66-69D01	0TW16D01-03	1TD61-65D01, 2TD64,65D01/ 2TD61-63D01	0TW17D01-03	0TW19D01-03	0TW33D01, 02	DPW pump
Pump head, kgf/cm^2	22	15	8.5	8.5	8.5	8	6.5
Pump capacity, m^3/h	500	500	90	90	90	100	600
Powering from DG	+	+	-/+	-	-	-	Own DG

According to the design pumps 1,2TG11-14D01, 1,2TD64-65D01, TW17D01-03, TW19D01-03, TW33D01,02, 2TZ51-53D01 are powered from the system of normal electric power supply. At present pumps 1TD61-63D01 are powered from the system of normal electric power supply. DPW pumps (artesian water) have their own diesel generator. According to the design the other pumps are powered from the system of emergency electric power supply.

As it is apparent from the presented list of pumps, there is a sufficient redundancy of power Unit 2 reactor and power Units 1 and 2 storage pools makeup sources, and at that DPW pumps (artesian water) have their own diesel generator and are the last of the makeup sources, when both normal and emergency, from regular DG, electric power supply of all the other pumps is lost.

According to instructions [5.61] and [5.62] there is a possibility of supply of artesian water from the water intake area from DPW system, for which realization of modification МОД-05-02-718 “Supply of Domestic Potable Water to Fuel Channels through MCT Pipelines” is required. At that DPW pumps have the possibility of powering from their own diesel generator, therefore the given system is considered to be enough reliable.

The main purposes of the beyond design-basis accidents management procedures (RUZA) are the following:

- termination of radioactive fission products emissions from the power plant into the environment, and limitation of radioactive fission products emissions while achieving RUZA main purposes;
- support or return of nuclear fuel in the reactor core in a controlled, stable state;
- support or return of nuclear fuel in the storage pools in a controlled, stable state.

Realization of RUZA strategy has a priority over the actions under the other procedures, being a part of INPP operational-engineering documentation. After the beginning of any RUZA strategies realization, the actions under these procedures can be carried out only after reception of the sanction from the Head of Emergency Technical Service (HETS), after an assessment that these actions will not contradict RUZA strategy realization.

At application of RUZA instructions the actions which are not applied at normal operation are admitted - removal of protections and blockings of the operating equipment, apparent damage of the ancillary equipment, the limited radioactive fission products emission into the environment, etc.

According to the definition of the beyond design-basis accident, the reduction of the beyond design-basis accident consequences is achieved by the accident management and/or realization of the plans of measures on the personnel and the population protection if the accident management becomes impossible or inefficient.

Implementation of the INPP Emergency Preparedness Plan [5.34] is carried out after classification of the state “preparedness” at the INPP.

Realization of RUZA strategies related to the actions of the personnel in the places, where deterioration of the radiation situation is possible, is carried out according to the requirements of the INPP radiation safety procedures and the requirements of the plan [5.34], where the potential effective doses for the personnel and the order of their estimation are indicated, as well as the ways of the personnel protection from the radiation impact.

8.4.6 Rules of the Beyond Design-Basis Accidents Management Procedures Application

Rules of the beyond design-basis accidents management procedures application at decision-making concerning the necessity of any RUZA strategy realization beginning, actions on its realization are carried out in the following order:

- HETS gives an order to PSS for application of the strategy;
- realization of RUZA strategies, leading to the radioactive fission products emission and to the excess of the irradiation dose limits for the personnel carrying out the actions, is carried out after the sanction of the Head of OEP in accordance with recommendations of the HETS, the Head of Radiation Protection Service, the Head of OEP operations at realization of continuous radiation monitoring;
- PSS organizes the performance of required actions by the shift personnel, if necessary, the attraction of the Emergency Technical Service personnel via HETS is possible;
- if for realization of RUZA strategies the implementation of modification is required, it shall be executed according to the works performance projects by the Emergency Technical Service personnel, under PSS application and with HETS sanction;
- PSS supervises via DPSS and the Heads of the workshops (departments) shifts, that no actions directed to liquidation of the beyond design-basis accident would be carried out at the same time (in parallel) without the HETS permission;
- PSS, receiving the information from DPSS and the Heads of the workshops (departments) shifts, provides the required information to the Head of the Technical Support Centre and HETS, who direct it to the Head of OEP operations;
- in case of absolute obstacles occurrence in strategy realization PSS in due time informs HETS about this, for him to reassess the current state of the power plant and to choose another strategy or another way of the given strategy realization;
- realization of RUZA strategies related to the actions of the personnel in the places, where deterioration of the radiation situation is possible, is carried out according to the requirements of the INPP radiation safety procedures and the requirements of the INPP Emergency Preparedness Plan [5.34] taking into account non-exceedance of effective doses for the personnel.

Distribution of the INPP personnel responsibility at RUZA application is presented in Table 8.4-2.

TABLE 8.4-2

No	Actions	Monitoring (assessment)	Development of recommendations	Decision-making	Solution realization
1.	Accident management till the moment of the state classification as “preparedness”	shift operational personnel	-	PSS	shift operational personnel
2.	Classification of the state “preparedness” and the beginning of RUZA DF application	PSS	-	PSS	PSS

No	Actions	Monitoring (assessment)	Development of recommendations	Decision-making	Solution realization
3.	State "preparedness" announcement (start of the emergency preparedness plan and gathering of the Technical Support Centre)	-	-	Head of OEP (PSS)	Head of OEP operations, Head of the Technical Support Centre HETS
4.	DF control and estimation of availability of input conditions in separate RUZA, the beginning of their application	Technical SupportCentre (PSS)	-	Head of the Technical Support Centre (PSS)	Head of the Technical Support Centre (PSS)
5.	Assessment of other operational procedures actions results, and release of sanctions for the further actions under these procedures in parallel with RUZA	Technical SupportCentre (PSS)	Head of the Technical Support Centre	HETS	PSS
6.	Estimation of contradictions between separate RUZA requirements and setting of priorities between them at their implementation, application of RUZA support procedures	Technical SupportCentre	Head of the Technical Support Centre	HETS	PSS
7.	Restoration of the equipment which lost the working capacity during beyond design-basis accidents, preparation and realization of RUZA modifications	Technical SupportCentre	Head of the Technical Support Centre	HETS	HETS, Emergency Technical Service personnel
8.	RUZA strategies realization	Technical SupportCentre	Head of the Technical Support Centre	HETS	PSS, Shift operational personnel
9.	Realization of RUZA strategies, leading to the radioactive fission products emission beyond ALS limits and to the excess of the irradiation dose limits for the personnel carrying out the actions	-	Head of the Technical Support Centre, Head of Radiation Protection Service, HETS	Head of OEP	PSS, Shift operational personnel

No	Actions	Monitoring (assessment)	Development of recommendations	Decision-making	Solution realization
10.	Estimation of availability of output conditions from separate RUZA and completion of the actions on them (exit from RUZA	Technical SupportCentre	Head of the Technical Support Centre	HETS	Head of the Technical Support Centre
11.	RUZA DF actions completion, execution of the check-list of exit from the set of RUZA procedures	Technical SupportCentre	Head of the Technical Support Centre	HETS	Head of the Technical Support Centre
12.	Development and realization of the plan of the accident consequences liquidation, elimination of the long-term negative consequences of RUZA strategies application	Head of the Technical Support Centre, OEP Headquarters	Head of OEP operations, HETS	Head of OEP	Head of OEP operations HETS, Emergency Technical Service personnel

8.4.7 Technical Means and Resources for Beyond Design-Basis Accidents Management

For beyond design-basis accidents management one permanent modification, МОД-05-02-691 "Installation of Thermocouples into Power Unit 2 FA Bearing Tubes Ass.9", was implemented at INPP, as well as five more modifications were prepared for implementation in the conditions of the beyond design-basis accidents occurrence:

- МОД-05-02-716 "Water Supply to SFP from Service Water Supply System";
- МОД-05-02-717 "Water Supply to Room 125 of Unit A-2";
- МОД-05-02-718 "Supply of Domestic Potable Water to Fuel Channels through MCT Pipelines";
- МОД-05-02-723 "Bld.101/2, 185 Consumers Supply at Full Auxiliaries Blackout";
- МОД-05-02-732 "Supply of the Absorber to SFP".

Modification МОД-05-02-691 "Installation of Thermocouples into Power Unit 2 FA Bearing Tubes Ass.49"

The temperature measurement system in the FA bearing tubes Ass.49 provides continuous registration of the measured value, indication of the current value and archiving of data, the range of measured temperatures is 0-1200°C, that provides the temperature measurement in the operating mode 0-300°C, in the emergency mode 0-1100°C.

Two-region thermocouples (2YK21T121,122, and 2YK21T361,362) are installed in the bearing tubes of FAs 21-12, 21-36 for the temperature measurement task. The distance between the sensitive elements of the thermocouple is 1 meter; they are located at levels 14.4 m and 13.4 m.

Indications of the thermocouples are displayed at ICS and MCR-2 recording display devices, panel 2HZ01Z23,24 control points 2YB12,22P04B1. Archived data are kept for the last 6 days. Dimensions of the thermal converter correspond to the radial power density monitoring sensor.

The analysis executed within the limits of RUZA development project showed that the thermocouples installed for the temperature measurement in the FA bearing tubes enable to fix authentically the moment of the reactor core emptying. At decrease of the water level in MCC up to the level, equal approximately to the middle of the core, the growth of FE temperature values begins.

Modification МОД-05-02-716 “Water Supply to SFP from Service Water Supply System”

RUZA-B, Strategy 17.2 - Service water supply to SFP, support procedure SP-15 “Service Water Supply to SFP from Service Water Supply System”.

Modification includes manufacturing and installation of pipe spool piece Du 200 installed at one of heat-exchangers 2TG21-23W01 for connection of the service water supply pipeline with the circuit water pipeline in Room 331 of Unit A2 (Figure 8.4-3).

During beyond design-basis accident at SFP in case of non-recoverable by design means loss of water in SFP, pipe spool piece Du 200 installation at one of heat-exchangers 2TG21-23W01 and service water supply to SFP shall be executed.

Figure 8.4-3. Pipe units DN 200 for modification МОД-05-02-716

At present an additional modification on the possibility of service water supply to the storage pools of power Unit 1 MOD-11-01-1147 “Development of the Circuit of Service Water Supply to Power Unit 1 SFP at Beyond Design-Basis Accidents” has been planned at INPP and is being at the stage of implementation (technical solution ПТОмод-1632-372 dated 06 June, 2011, the term of modification implementation is 30 June, 2012).

Modification МОД-05-02-717 “Water Supply to Room 125 of Unit A-2”

During beyond design-basis accident in case of the insufficient heat removal from the reactor, the water supply to Room 125 of Unit A2 via tank 2TZ40B01 shall be provided. Room 125 of Unit A2 shall be filled by the water from ALT CP and from DPW system or from the tanks of contaminated drain waters and LSW from pump stations 2TZ11-13D01 (2TZ21-23D01 or 2TZ51-53D01):

- way 1 - ALT CP ⇒ CP emptying pipelines ⇒ pipelines of drain waters pumping out ⇒ short pipe instead of return valve 2TZ43S02 ⇒ pump 2TZ43D01 ⇒ tank 2TZ40B01 ⇒ floor drains of Room 125 A2 ⇒ Room 125 A2. This option is applied at non-availability of the power unit auxiliaries' electric power supply. The place of short pipe DN200 storage is Room 040 of Unit A2, Building 101/2.
- way 2 - DPW system ⇒ fire hydrants in Rooms 188/1,2, 147/2 of Unit A2 ⇒ fire hoses ⇒ adapters - connection heads ⇒ EWP scavengings in Rooms 126, 132, 134 of Unit A2 ⇒ tank 2TZ40B01 ⇒ floor drains in Room 125 of Unit A2 ⇒ Room 125 of Unit A2. This option is applied at non-availability of the power unit auxiliaries' electric power supply.
- way 3 - tanks 2TZ10B01 or 2TZ20B01 or 2TZ50B01 ⇒ pump stations 2TZ11-13D01 or 2TZ21-23D01 or 2TZ51-53D01 ⇒ the pipeline of emergency pumping out to Bld. 150 ⇒ short pipe instead of return valve 2TZ43S02 ⇒ pump 2TZ43D01 ⇒ 2TZ40B01 ⇒ floor drains in Room 125 of Unit A2 ⇒ Room 125 of Unit A2. This option is applied at availability of the power unit auxiliaries' electric power supply (Figure 8.4-4).

Figure 8.4-4. Short pipe DN 200 for modification МОД-05-02-717

Modification МОД-05-02-718 “Supply of Domestic Potable Water to Fuel Channels through MCT Pipelines”

During beyond design-basis accident at non-availability of the other makeup sources (full INPP auxiliaries blackout), the connection of pipeline Du 200 to process pipelines shall be executed and water supply to the reactor FCs from DPW system with the flow not less than 200 m^3/h to a side of MCC shall be ensured.

The modification shall ensure water supply from DPW header (in Room 003 of Unit D2) into water supply pipelines from MCT to ECCS header (Room 112 of Unit A2) according to the circuit: DPW header in Room 003 of Units D1, D2 ⇒ connection pipe Du 200 between DPW header in Room 003 of Unit D2 and the water supply pipeline to DGH from MCT in Room 112 of Unit A2 ⇒ valves 2TQ13,14S01,02 of ECCS header ⇒ DGH ⇒ the reactor core.

Thus the possibility of DPW supply to MCT pipelines and further via ECCS pipelines to DGH through the connections between valves 2TQ13S01 and 02, and between valves 2TQ14S01 and 02 is provided.

Assemblies 1, 2, 3 and plug DN 200 will be installed between DPW header in Room 003 of Unit D2, Building 101/2 and the water supply pipeline to DGH from MCT in Room 112 of Unit A2 for DPW supply to the reactor FC during power Unit 2 beyond design-basis accident. The storage place of assemblies 1, 2, 3 and plug DN 200 is Room 112 of Unit A2, Building 101/2 (Figure 8.4-5).

Figure 8.4-5. Pipe units DN 200 for modification МОД-05-02-718

Modification МОД-05-02-723 “Bld.101/2, 185 Consumers Supply at Full Auxiliaries Blackout”

During beyond design-basis accident at full INPP auxiliaries blackout (failure of all DGs) it will be necessary in 1 hour (minimal time of regular accumulator batteries discharge) to power the required equipment (MCR, ECR, communication facilities, portable accumulators charging boards for portable lighting fixtures, etc.) from an independent source - mobile diesel generator DES-60R with capacity 60 kVa for support of the main technological parameters and processes control and monitoring (Figures 8.4-6 and 8.4-7).

Figure 8.4-6. Portable accumulators charging board 2DP201Z04 and emergency switch cabinet 2HZ201Z01 in Room 2UPS-1

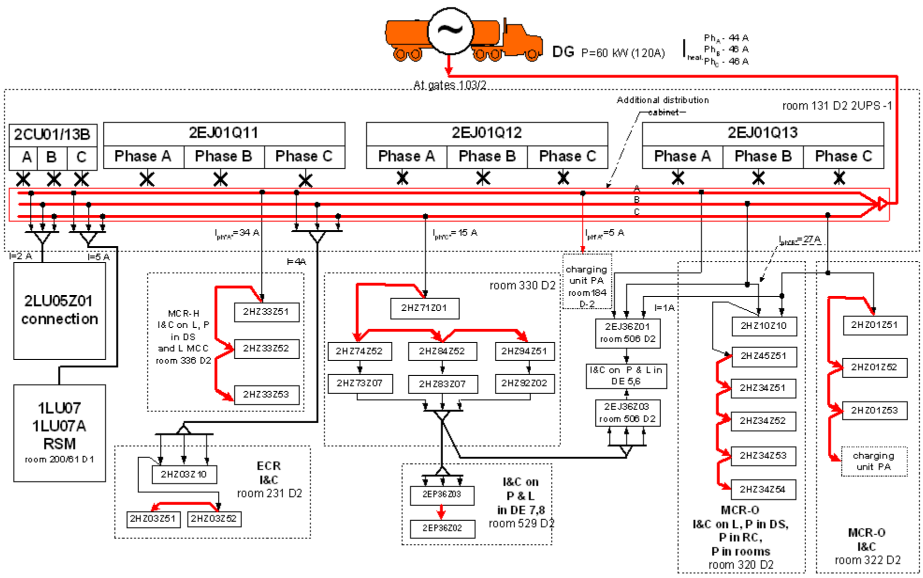


Figure 8.4-7. Emergency power supply 0.4 kV circuit from mobile diesel generator DES-60R to Building 101/2

The power supply of consumers during beyond design-basis accident with full INPP auxiliaries blackout is performed from mobile electric power supply sources - diesel generator VS-184E with capacity 20 kVa, installed on the emergency service vehicle "KAMAZ", as well as from diesel generator DES-60R with capacity 60 kVa, installed on the automobile trailer and belonging to the Heat Supply, Transport and Utilities Workshop. Thus, the possibility of I&C, RSM and communication facilities consumers' powering, required for beyond design-basis accident management, is provided. In Figures 8.4-8 and

8.4-9 the emergency power supply circuit and the power supply circuit of communication facilities from mobile diesel generators are presented.

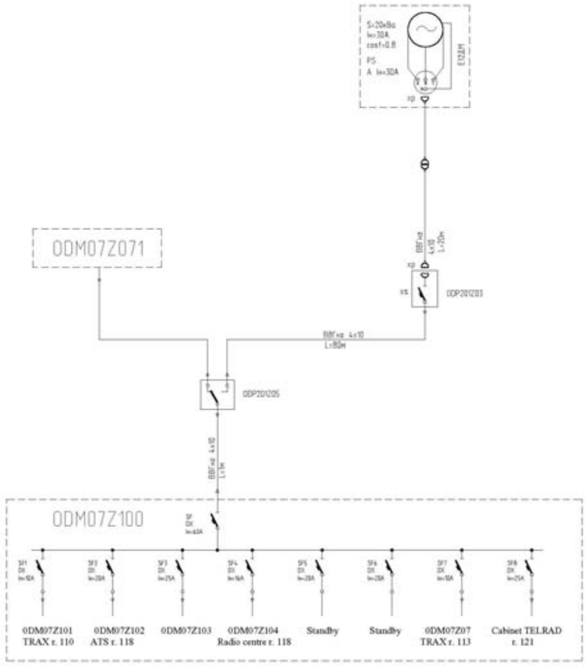


Figure 8.4-8. Emergency power supply circuit from mobile diesel generator VS-184E to Building 185

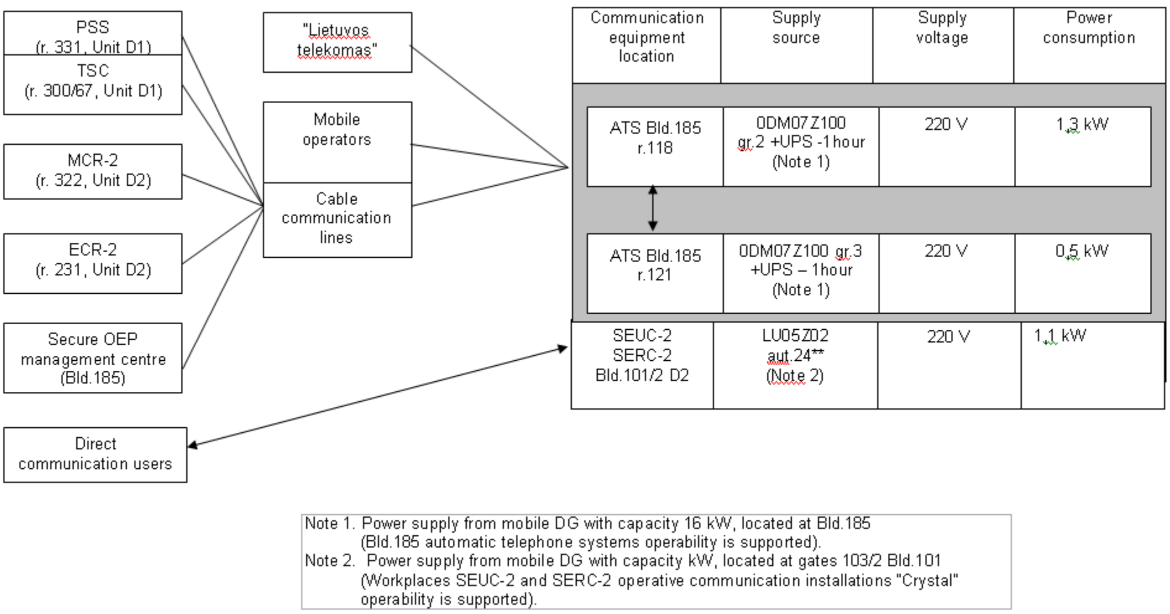


Figure 8.4-9. Power supply circuit of communication facilities from mobile diesel generator VS-184E and DES-60R

At present an additional modification on the possibility of power supply of the devices of the level and temperature monitoring in the storage pools of power Units 1 and 2 from mobile diesel generator DES-60R MOD-11-12-1143 “Modification of the Water

Temperature Measurement Circuit” has been planned at INPP and is being at the stage of implementation (technical solution ПТОмод-1632-370 dated 08 June, 2011). The conclusions of the report regarding safe operation limits justification on temperature and water level at storage of fuel in the storage pools are considered in the modification [5.39].

The solution “Electric Power Supply of Building 101/1,2 SFP Temperature and Water Level I&C” [5.74] was accepted for a possibility of power supply of Building 101/1,2 SFP I&C from mobile diesel generator at full power plant blackout. The planned term of modification implementation is 31 December, 2011.

Modification МОД-05-02-732 “Supply of the Absorber to SFP during Beyond Design-Basis Accidents”

Modification consists of the development of the design for manufacturing and installation of the additional connection pipe Du 50 between the pipeline of LSW supply to the liquid absorber tank and the pipeline of the absorber supply to CPS CC from 2YS51B01. The fire hose for the additional connection pipe, tee Du 50 with the connection head and flange Du 80 with the connection head are located in Room 908/1 of Unit A2.

During the beyond design-basis accident at SFP for prevention of self-sustained chain reaction occurrence and absence of the requirement for the absorber supply to CPS CC (it is carried out at AZ-BSM failure, at present according to the project [5.24] AZ-BSM is taken out from operation) it is required to dismantle the simulator of the “visible break” pipe spool piece in Room 908/1 of Unit A2, to install connection pipe Du 50 between valve 2YS51S01 and LSW pipeline in Room 908/1 of Unit A2. (Figure 8.4-10).

According to the AHS operational manual, it is required to fill tank 2YS51B01 with low salted or chemically purified water, prepare a solution of the absorber in the tank and according to the prepared procedure (RUZA-B support procedure SP-16 “Supply of the Absorber to SFP”) supply the absorber to emergency SFP.

Figure 8.4-10. Pipe units DN 50 for modification МОД-05-02-732

According to the report [5.41] the justification of strategy S19 “Supply of the Absorber to Emergency SFP” was executed at INPP for power Unit 1 SFP as well.

For realization of the listed above modifications the developed projects of the works performance, the required property of emergency preparedness (equipment and tools), and the trained personnel of the Organization of Emergency Preparedness, presented in Table 8.4-3, are available at INPP.

TABLE 8.4-3

<table>
<tr><th rowspan="2">No</th><th rowspan="2">Modification name</th><th rowspan="2">Working documentation for works performance</th><th colspan="4">Storage places (availability)</th><th rowspan="2">Availability of the trained personnel</th></tr>
<tr><th>Documentation</th><th>Tools accessories</th><th colspan="2">Equipment</th></tr>
<tr><td>1</td><td>МОД-05-02-716 Water Supply to SFP from SWS</td><td>ИТ01.4573.00.00Р</td><td>Bld. 101/2, Unit V2, Room 117</td><td rowspan="2">Bld. 101/2, Unit V2, Room 117 (PHEU Division Workshop)</td><td>Pipe unit ИТ14.186.00.00</td><td>Bld. 101/2, Unit A2, Room 331</td><td rowspan="2">Available unit No3 of SWS, ventilation system and PHEU repair group 4 men, working place Bld. 101/2, Unit V2, Room 117</td></tr>
<tr><td rowspan="3">2</td><td rowspan="3">МОД-05-02-717 Water Supply to Room 125 of Unit A2</td><td rowspan="3">ИТ14.217.00.00СБ</td><td rowspan="3">Bld. 101/2, Unit V2, Room 322</td><td>Short pipe DN200 ИТ14.205.00.00</td><td>Bld. 101/2, Unit A2, Room 040</td></tr>
<tr><td rowspan="2">Bld. 101/2, Unit V2, Room 325 (ICC Division Workshop)</td><td>Adapter DN 50 ИТ14.207.00.00</td><td>Bld. 101/2, Unit A2, Room 908/1</td><td rowspan="6">Available unit No1 of reactor-turbine department process systems repair group, 4 men, working place Bld. 101/2, Unit V2, Room 322</td></tr>
<tr><td>Adapter DN 50 ИТ14.217.00.00</td><td>Bld. 101/2, Unit A2, Room 132</td></tr>
<tr><td rowspan="3">3</td><td rowspan="3">МОД-05-02-718 Supply of Domestic Potable Water to Fuel Channels through MCT Pipelines</td><td rowspan="3">ИТ01.4677.00.00</td><td rowspan="3">Bld. 101/2, Unit V2, Room 322</td><td rowspan="3">Bld. 101/2, Unit V2, Room 325 (ICC Division Workshop)</td><td>Assembly 1 ИТ14.229.00.00</td><td rowspan="3">Bld. 101/2, Unit A2, Room 112</td></tr>
<tr><td>Assembly 2 ИТ14.230.00.00</td></tr>
<tr><td>Assembly 3 ИТ14.231.00.00</td></tr>
<tr><td>4</td><td>МОД-05-02-732 Supply of the Absorber to SFP</td><td>ИТ14.225.00.00РМЧ</td><td>Bld. 101/2, Unit V2, Room 322</td><td>Bld. 101/2, Unit V2, Room 325 (ICC Division Workshop)</td><td>Flange tee DN 50 ИТ14.225.00.00</td><td>Bld. 101/2, Unit A2, Room 908/1</td></tr>
<tr><td>5</td><td>МОД-05-02-723 Bld.101/2, 185 Consumers Supply at Full Auxiliaries Blackout</td><td>DVSed-0812-8 (form SP-3) (project ArchPD-1859-73027)</td><td>Bld. 101/1, Unit D1, Room 337</td><td>Bld. 101/1,2, Unit B1,2, Room 411 Bld. 101/1, Unit D1, Room 708 Bld. 101/2, Unit D2, Room 261, 308, 311, 319</td><td>Cable connection pipes</td><td>Bld. 101/2, Unit D2, Room 284</td><td>Available 2 groups and 2 units of I&C equipment failures elimination team, 12 men, working place Bld. 101/1, Unit D1, Room 308</td></tr>
</table>

Modification МОД-09-02-936 “Repowering of Electric Motors of Pumps TW16D02, 03 Bld. 154”

In addition to the modifications listed above at present the modification on the possibility of CPW and SPC supply to the reactor of power Unit 2 and the storage pools of power Units 1 and 2 from the pumps powered from the system of emergency electric power supply МОД-09-02-936 “Repowering of Electric Motors of Pumps TW16D02, 03 Building 154” has been planned at INPP and is being at the stage of implementation (technical solution ПТОмод-1632-163 dated 15 April, 2009).

For reliable electric power supply support in case of beyond design-basis accident with imposition of mechanical failures of pumps TW16D01,03 the solution "Electric Power Supply of Pumps TW16D01,02,03" SPr-150 (3.67.19) dated 23 May, 2011 for the possibility of CPW pump and SCC TW16D02 electric powering from DG-7 or DG-9 was accepted.

In the states of beyond design-basis accidents the given modification will enable to apply the flexible circuit of the emergency power supply sources and the reactor and the storage pools makeup system mechanisms connection to each other, so that during the failure of DG safety system in one channel, and the failure of the pump in the other one, it would be possible to connect the remained efficient DG and pumps, belonging to different channels according to the design circuit, to each other.

Realization of the listed above modifications during beyond design-basis accidents management shall be carried out by the personnel of the Emergency Technical Service: the team of accidents elimination of the Nuclear Fuel Management Workshop equipment and facilities (56 persons in the structure of 5 groups and 12 units), the team of emergency recovery works (33 persons in the structure of 4 groups and 4 parts), and the team of I&C equipment failures elimination (12 persons in the structure of 2 groups and 2 units). The Structural Diagram of accidents elimination team is presented in Figure 8.4-11.

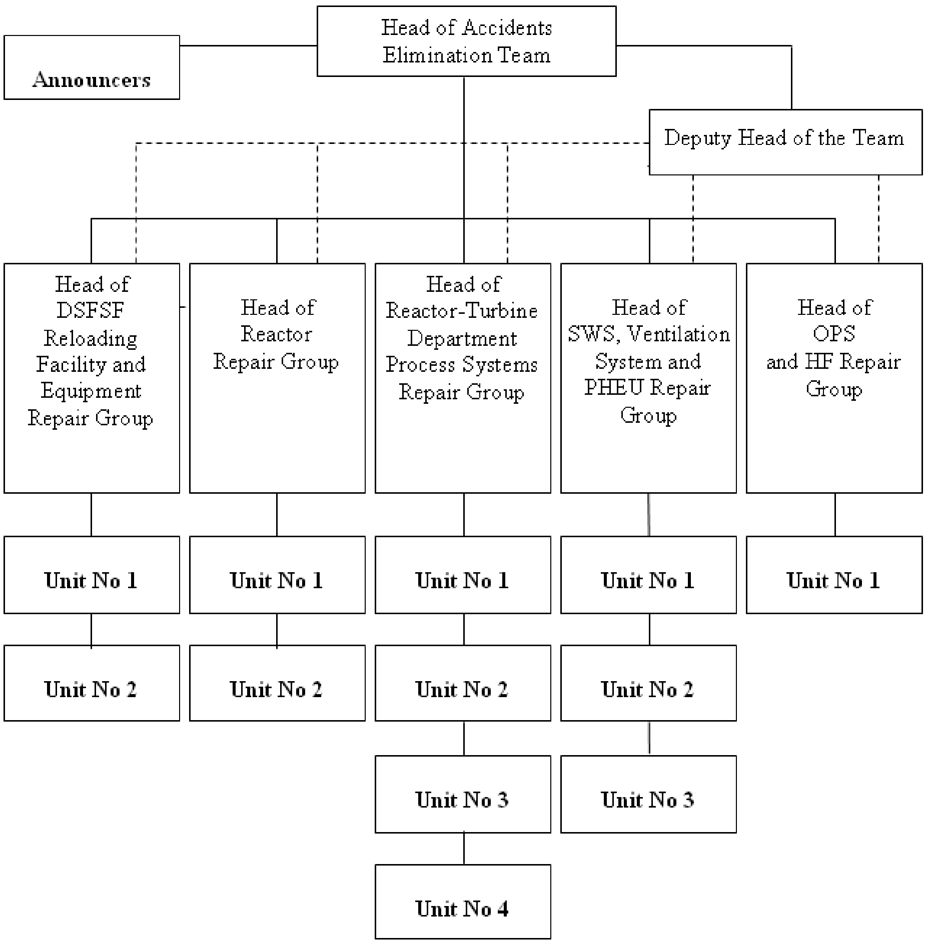


Figure 8.4-11. Structural Diagram of Accidents Elimination Team

8.4.8 Organization of Emergency Preparedness at INPP According to the Plan of Emergency Preparedness

8.4.8.1 Structure of the INPP Organization of Emergency Preparedness

The new plan of emergency preparedness [5.34] and the emergency preparedness working procedures developed for the shutdown state of power Unit 2 reactor have been commissioned at INPP.

The structure of the INPP Organization of Emergency Preparedness (OEP), the Organization of Emergency Preparedness Headquarters are created, and the Heads of the services and teams of emergency preparedness and the subordinated personnel are formed on the basis of industrial principle from the personnel of the shops and departments, considering specific tasks which the subdivisions of the enterprise solve in the state of normal operation.

INPP OEP structure includes:

- INPP Director General - the Head of the Organization of Emergency Preparedness (in case of his non-availability his duties are executed by the Decommissioning Director);
- Decommissioning Director - the Head of the Organization of Emergency Preparedness operations (in case of his non-availability his duties are executed by the Head of OEP Emergency Technical Service);
- Economics and Finance Director - the Head of the OEP Financial and Material Resources Provision Service (in case of his non-availability his duties are executed by the Economics and Finance Deputy Director);
- the Head of the Technological Service - the Head of OEP Emergency Technical Service (in case of his non-availability his duties are executed by the Head of the OEP Technical Support Centre);
- the Head of the Operational Management and Engineering Support Department - the Head of the OEP Technical Support Centre (in case of his non-availability his duties are executed by the Senior Operational Engineer of the Operational Management and Engineering Support Department);
- the Senior Operational Engineer of the Operational Management and Engineering Support Department – the Deputy Head of the OEP Technical Support Centre;
- the Head of the Radiation Safety Service - the Head of the OEP Radiation Protection Service (in case of his non-availability his duties are executed by the Head of the Radiation Safety Department);
- Personnel Director - the Head of the OEP Medical and Evacuation Activities Organization Service (in case of his non-availability his duties are executed by the Head of the Personnel Department);
- the Head of the Physical Security Organization Service - the Head of the OEP Physical Security Service (in case of his non-availability his duties are executed by the Head of the Physical Security Organization Department);
- the Head of the TS&QMD Fire Supervision and Civil Protection Group - the Head of the Organization of Emergency Preparedness Headquarters (in case of his non-

availability his duties are executed by the civil protection engineer of the TS&QMD Fire Supervision and Civil Protection Group);

- the civil protection engineer of the TS&QMD Fire Supervision and Civil Protection Group - the assistant of the Head of the OEP Headquarters (in case of his non-availability his duties are executed by the Head of the TS&QMD Fire Supervision and Civil Protection Group);
- the Head on Communications - the press agent - the Head of the Support Group under OEP Headquarters (in case of his non-availability his duties are executed by the Information Centre specialist on communications);
- the Head of the Documents Management Department - the Head of the Documents Management Subgroup, being a part of the Support Group under OEP Headquarters (in case of his non-availability his duties are executed by the Head of the Secretariat);
- the specialist on communications - the Head of the Communications and Mass Media Subgroup, being a part of the Support Group under OEP Headquarters (in case of her non-availability her duties are executed by the Personnel Director Assistant);
- the Head of the Information Technologies and Fire Automatic Equipment Department - the Head of the Computer Equipment Maintenance, Warning System and Communication Facilities Subgroup, being a part of the Support Group under OEP Headquarters (in case of his non-availability his duties are executed by the Deputy Head of the Information Technologies and Fire Automatic Equipment Department);
- the Head of the Economy Management Department - the Head of the Accidents Management Centre Functioning Support Subgroup, being a part of the Support Group under OEP Headquarters (in case of her non-availability her duties are executed by the Deputy Head of the Economy Management Department.

In order to ensure constant readiness of the Organization of Emergency Preparedness, substitutional persons (not less than 2 persons per one OEP position), meeting the requirements demanded for these positions, are foreseen for the top management of the Headquarters, as well as the personnel of all the OEP services and teams.

The INPP personnel physically qualified for the work with ionizing radiation sources can be involved in the OEP. It is forbidden to involve pregnant women, invalids and persons previously received a single exposure dose - more than 50 mSv, in the OEP.

8.4.8.2 Training of OEP Heads and Personnel

Decommissioning Director, as an authorized person of the Director General regarding emergency preparedness and civil protection, once per 5 years is trained at the civil protection training centre of the Fire and Rescue Department under the Ministry of the Interior on the civil protection introduction training programme for the heads, or the authorized by them persons, of the state importance facilities included in the register of the state importance facilities and hazardous facilities.

The senior engineer, fire supervision and civil protection inspector, the Head of TS&QMD Group (as the Head of the Organization of Emergency Preparedness Headquarters), as well as the civil protection engineer of the TS&QMD Fire Supervision and Civil Protection Group (as the assistant of the Head of the Organization of Emergency Preparedness Headquarters) once per three years are trained at the civil protection training centre of the

Fire and Rescue Department under the Ministry of the Interior on the civil protection programme for the permanent members.

Training of the personnel provides the initial training in the scope of requirements to the position at the employment, and development of the practical skills during trainings and exercises.

The Head of the Fire Supervision and Civil Protection Group gives annual classes in the educational groups of the OEP top management:

- the schedule includes educational themes on PEP, actual issues of emergency preparedness and civil protection in the concrete educational year, as well as recommendations of VATESI and of the Fire and Rescue Department under the Ministry of the Interior;
- not less than once per year the Head of the TS&QMD Fire Supervision and Civil Protection Group organizes and conducts group exercises with the Heads of the Organization of Emergency Preparedness Headquarters.

The civil protection engineer of the TS&QMD Fire Supervision and Civil Protection Group conducts classes with group No 3, which includes the heads of the INPP structural subdivisions, which are not members of the Organization of Emergency Preparedness.

Not less than once per three years INPP Director General organizes complex training of the Organization of Emergency Preparedness. In Figures 8.4-12 and 8.4-13 the photos of the work of the INPP Organization of Emergency Preparedness Headquarters and of the Fire-Rescue Management Team from the last complex training conducted on 24 February, 2011 are presented. The results of training are provided in the report [5.44].

Figure 8.4-12. Work of the INPP Organization of Emergency Preparedness Headquarters

Figure 8.4-13. Work of the Fire-Rescue Management Team

The Heads of the OEP teams and groups are responsible for development of the training programmes according to the Plan of emergency preparedness activities and their agreement with the Head of the TS&QMD Fire Supervision and Civil Protection Group. The Heads of the teams and groups are responsible for organization of training of the subordinated personnel of corresponding teams and groups, as well as for preparation and implementation of functional trainings in groups, teams.

The assistant of the Head of the OEP Headquarters together with the Heads of the OEP Services organize functional trainings in the services. Functional trainings are assessed by the Head of the OEP Headquarters and his assistant, the development of practical skills of the subordinated teams and groups management for performance of the tasks directed towards liquidation of probable accidents is carried out at such trainings.

Once per three years the specified staff of the OEP teams and services participates in complex trainings, where the degree of the personnel emergency preparedness and his ability to work in complicated conditions at performance of assigned tasks is checked.

8.4.8.3 OEP Accidents Management Headquarters Activities

Accidents management activities are directed towards achievement of the following safety objectives:

- prevent accident progressing at the reactor core damage;
- ensure continuous cooling of the reactor core;
- if possible, ensure integrity of the accidents localization system.

For achievement of the stated above safety objectives the top management of OEP Headquarters shall implement the following tasks:

- develop and realize the plan of accident liquidation activities and the INPP reversion to the normal operation condition;
- develop and realize the plan of accidents consequences mitigation regarding exclusion of radioactive materials discharges into the environment or reduction of these discharges;

- develop and realize protective activities against radiation exposure of the workers and the population or reduction of it;
- develop and realize protective activities against ionizing irradiation for the workers liquidating the accident;
- cooperate with emergency services and institutions of the state management and surveillance, as well as with municipalities;
- provide duly medical aid for the victims;
- ensure cooperation with mass media.

8.4.8.4 Arrangement of Protective Activities

In the case of radiation accident the protective activities for the INPP personnel are the following:

- termination of all scheduled works and evacuation of the personnel from the workplaces;
- evacuation of the personnel from the controlled area;
- evacuation of the personnel beyond the sanitary protective area boundaries;
- iodine prophylaxis;
- decontamination of the personnel.

These protective activities are carried out on the basis of the protective activities application criteria. The main protective activities application criteria are:

- gamma-radiation dose rate in the areas, where the constant staying of the personnel for the safety systems maintenance is required;
- gamma-radiation dose rate in the areas, where the periodic staying of the personnel for safety systems maintenance is required;
- gamma-radiation dose rate within the limits of the sanitary protective area;
- concentration of radioactive iodine in the air;
- contamination of skin surfaces and personal clothes of the personnel by radionuclides.

The protective activities application criteria established at the INPP in the case of accident are presented in Table 8.4-4.

TABLE 8.4-4

No	Protective activities application criteria	Criterion meaning	Protective activities
1.	Gamma-radiation dose rate in the rooms of the personnel constant staying	0.012 mSv/hour	Termination of all scheduled works and evacuation of the personnel from the workplaces
2.	Gamma-radiation dose rate in the rooms of the personnel periodic staying	0.056 mSv/hour	
3.	Gamma-radiation dose rate in the rooms of the personnel constant staying	from 0.012 up to 1.0 mSv/hour	Evacuation of the personnel from the controlled area

No	Protective activities application criteria	Criterion meaning	Protective activities
4.	Gamma-radiation dose rate within the limits of the SPA	from 0.012 up to 0.1 mSv/hour	
5.	Gamma-radiation dose rate within the limits of the SPA	more than 0.1 mSv/hour	Evacuation of the personnel beyond the SPA boundaries
6.	Concentration of radioactive iodine in the air	110 Bq/m^3	Iodine prophylaxis
7.	Contamination of skin surfaces and personal clothes of the personnel by radioactive substances	more than 0.4 Bq/cm^2	Decontamination of the personnel

In the case of radiation accident at the INPP the radiation impact limits on the INPP personnel and OEP personnel, presented in Table 8.4-5, are established.

TABLE 8.4-5

Personnel	Radiation impact limits
INPP personnel (category A personnel)	Effective dose - 50 mSv/year (5 Rem/year), but not more than 100 mSv/year during 5 years in succession
OEP personnel	Effective dose - 100 mSv/year (10 Rem/year) (In a special case during the works on the people rescue)

After the INPP plan of emergency preparedness is started to be carried out, limit 50 mSv/year is checked by:

- PSS - for the subordinated operational personnel;
- Director General and Decommissioning Director - for all INPP personnel.

Meeting the requirements of the plan of emergency preparedness, Director General and Decommissioning Director are obliged to supervise (via the Head of the Radiation Safety Service) all the activities, so that the dose of irradiation received by the personnel, participating in localization of the accident, does not exceed 50 mSv (in a special case 100 mSv), and in the case of the works on prevention of the accident or during the works on the people rescue - does not exceed 500 mSv.

When irradiation doses can exceed the annual effective dose, the INPP category "A" personnel shall be involved in rescue operations; it shall be in details and precisely informed beforehand about the danger menacing to their health. Prior to the commencement of the works performance the written agreement of the worker shall be received.

8.4.8.5 Interaction with the External Organizations

Interaction of the Organization of Emergency Preparedness with local, territorial, state institutions, relations with the local organizations involved in the works in the case of the accident at the INPP, is executed by preliminary contracts-agreements between the INPP and the appropriate local organizations.

Interaction with the state institutions is carried out according to the requirements of the Law on Civil Protection of the Republic of Lithuania.

8.4.8.6 Main Technical Facilities, Resources, Rooms and Communication Systems

For successful performance of the assigned tasks INPP OEP has the OEP Accidents Management Centre in Bld. 185, equipped by all required equipment for accidents management and transfer of the information (Figure 8.4-14), as well as by the special room for the OEP Technical Support Centre, which also has everything required for the work of the experts.

The following rooms are provided for the operational personnel to manage the emergency situation:

- MCR-2 - power Unit 2 main control room;
- ECR-2 - power Unit 2 emergency control room;
- CCR - INPP central electric control room;
- RSMR - INPP main radiation safety monitoring room;
- working rooms of the Emergency Preparedness Services.

OEP Services are supplied with the required equipment, outfit, protective equipment, reagents, etc., required for liquidation of arisen beyond design-basis accidents, which list and places of storage are specified in the emergency preparedness instructions of the corresponding OEP Services.

Figure 8.4-14. A stationary emergency diesel generator providing the OEP Accidents Management Centre in Bld. 185 functioning during the auxiliaries' blackout

Lists of personal protective equipment, dosimetric control devices, radiation survey devices, equipment, tools, notification and communication means are also specified in the emergency preparedness instructions of the OEP Services.

Besides, the centralized storage of personal protective equipment is foreseen at the INPP OEP, intended for the personnel of the OEP Radiation Protection Service and Emergency Technical Service (in Room 163 of Unit A1).

The required special protective equipment, intended for the personnel of the OEP Radiation Protection Service and Emergency Technical Service, is centrally stored in the Radiation Safety Department.

The list of additional emergency preparedness property, vehicles and special OEP equipment is presented in the instruction [5.56].

INPP OEP has the monitoring system, which includes:

- the monitoring system of discharges into the ventilation pipe;
- the automated radiation safety monitoring system (radiation state monitoring inside the power plant);
- the automated radiation monitoring system (discharges monitoring, drains monitoring, radiation state in the district monitoring via the stationary posts, also gamma-background in 30 km area monitoring), Figure 8.4-15.

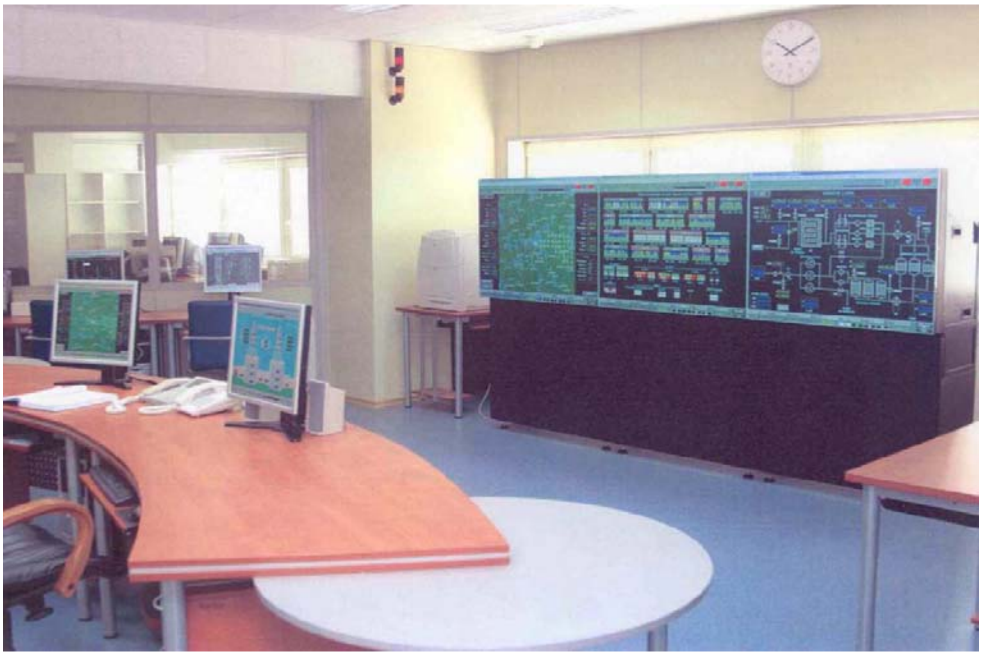


Figure 8.4-15. Radiation safety monitoring room with the automated radiation monitoring system means

For assessment of radiation consequences of the accident, the hardware and software of the computer system "NOSTRADAMUS", intended for operative forecasting of the radiation situation at the radioactive materials discharges during the accidents at the nuclear power plant and other nuclear facilities, are foreseen. In Figure 8.4-16 the INPP surroundings map is presented with the plotted lines of the level of the district radioactive contamination from the radioactive emissions, received as a result of system "NOSTRADAMUS" calculation during the OEP exercises.

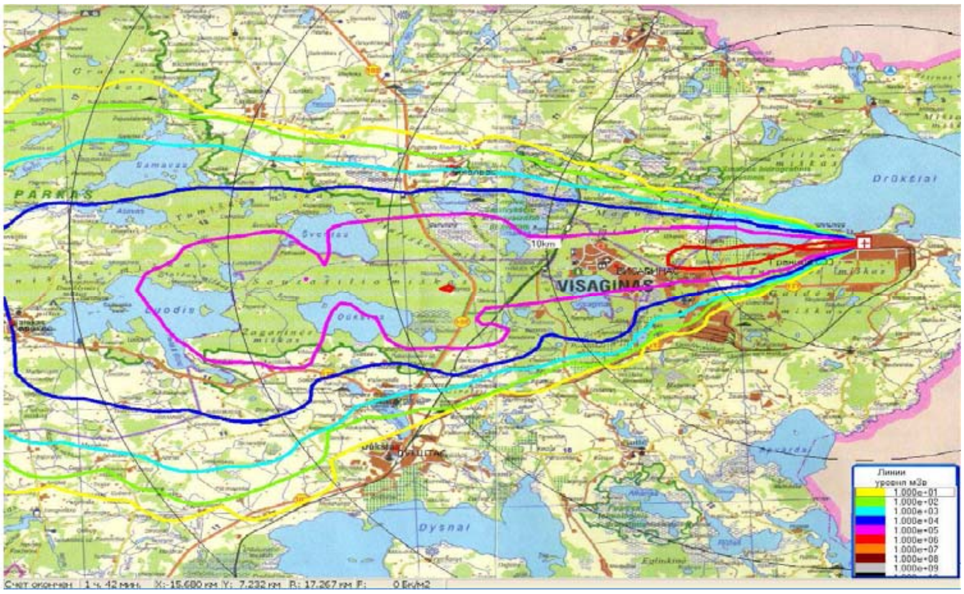


Figure 8.4-16. The SE INPP surroundings map with the plotted lines of the level of the district radioactive contamination from the radioactive emissions

The INPP possesses all the required resources and technical facilities for liquidation of beyond design-basis accidents, which can occur at the enterprise. The resources and technical facilities of other state institutions and departments for liquidation of beyond design-basis accidents at the INPP are not applied.

Besides the complex training of the Organization of Emergency Preparedness, which took place at the INPP on 24 February, 2011, according to the plan of additional INPP safety inspection and analysis (MnDPl-293 (3.67.22) dated 30 March, 2011), the Programme of Organization and Implementation of Emergency Training “Decrease of Water Level in INPP Unit 2 MCC and SFP” [5.78] has been developed at INPP. The purpose of this training is inspection of the knowledge and skills of the works performance by the operational personnel, and the inspection of the skill of interaction in the shift and with the personnel of the INPP Organization of Emergency Preparedness at occurrence of beyond design-basis accident, which causes the decrease of a level in INPP power Unit 2 MCC and SFP, with impossibility of its restoration by regular makeup sources. At present the trainings have begun at the INPP under the Programme specified above with the shifts on-duty operational personnel according to the annual schedule of the general power plant emergency trainings for the INPP operational personnel [5.79].

9 CONCLUSIONS AND RECOMMENDATIONS

9.1 Section “Earthquake”

9.1.1 To consider the necessity of the INPP emergency preparedness procedures amendments or addenda after reception and studying of the SNF casks tipover calculation results at its transportation by railway transporter from the power units to ISFSF and the environmental, personnel and population impact assessment.

9.1.2 To consider the possibility of the seismic alarm and monitoring system application for formalization of the emergency preparedness announcement criterion at the INPP and the subsequent inclusion of the given criterion in the operational manual of the seismic warning and monitoring system, code DVSed-0912-240V1.

9.1.3 The results of the analysis of Building 101/2 Unit A2 structures strength show that the analyzed reinforced concrete walls and floors meet the criteria of strength and crack resistance, specified in STR 2.05.05:2005, and are capable to sustain maximum design-basis earthquake. The analysis of operational capability shows that cracks can appear, but their width will not exceed admissible size.

9.2 Section “Flooding”

9.2.1 During uncontrollable abnormal rise of water level in lake Drukshiai, at the most negative development of flooding under any scenario, irrespective of the cause of its occurrence, the water level in lake Drukshiai cannot reach the marks, which could lead to the flooding of the INPP buildings and facilities safety related equipment.

9.3 Section “Loss of Electric Power Supply and Heat Removal to the Ultimate Heat Sink”

9.3.1 For provision of diesel generators refuelling at their operation over a long period of time it is required to conclude a contract on fuel delivery.

9.3.2 For provision of electric power supply of Bld. 101/1,2 storage pools temperature and water level I&C it is required to realize design No 10.2501.00 ЭМ and to insert additions to the corresponding procedures.

9.3.3 Additional activities at the loss of external electric power supply are not required.

9.3.4 Additional activities at the loss of external electric power supply and standby sources of electric power supply on the site are not required.

9.3.5 At occurrence of the deviations related to the loss of the ultimate heat sink, the INPP personnel have sufficient time and required means for prevention of cliff edge effects occurrence.

9.4 Section “Severe Accidents Management”

9.4.1 The performed analysis, including safety analysis reports conclusions (section 5 of the present document) concerning probable external and internal events, has not revealed the necessity for additional measures on compensation of the specified events impact on the INPP systems, including MCC, the reactor and the storage pools makeup systems, besides already planned at the INPP and presented in the report [5.42].

9.4.2 350 SFA unloading from power Unit 2 reactor under the “Working Programme of 500 SFA Pieces Unloading from INPP Unit 2 Reactor”, code EPg-17 (3.67.7), and the longtime shutdown mode of the reactor has led to significant decrease of the risk of FA damage in power Unit 2 reactor and in power Units 1 and 2 SFP in the case of cooling loss.

9.4.3 The reactor and SFP makeup sources redundancy, existing at the INPP, as well as availability of the developed procedures on beyond design-basis accidents management enable to carry out power Unit 2 reactor and power Units 1 and 2 storage pools makeup with sufficient redundancy of makeup sources, including DPW supply to the reactor and the storage pools from DPW pumps having their own diesel generator.

9.4.4 For implementation of modifications on beyond design-basis accidents management the industrial design and technological documentation (IDTD) has been developed at INPP, which is required for manufacturing and installation of corresponding parts of the pipelines, there is the required property of emergency preparedness (the equipment and the tools), and the trained personnel of the Organization of Emergency Preparedness Services.

9.4.5 All the required elements of the pipelines have been produced, tagged according to IDTD, are available and stored directly on the places of supposed works performance. Places of storage are designated by special plates and are located in the locked rooms, keys from which are accessible at the NFMW shift supervisor, or are equipped with attachments (supports, arms with clamps, chains, etc.) excluding the possibility of their damage or casual moving to another place.

9.4.6 Qualification of the personnel, who shall participate in realization of the modifications, is sufficient for performance of corresponding works, since they do not differ from the works carried out by the given personnel daily and are included into their official duties. The amount of the personnel in the units is sufficient for performance of corresponding works during the time established for this.

ENT vertėja
Marina RUDKOVSKAJA
2011-10-25