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List of abbreviations

AB	Accumulating Battery
AC	Alternate Current
ALS	Accident Localization System
ALT	Accident Localization Tower
BCS	Blowdown and Cooling System
Bld	Building
BRU-B	Russian acronym for "Fast-Acting Pressure-Reducing Valve"
ChWPS	Chemical Water Purification System
CPS	Control and Protection System
CPW	Chemically Purified Water
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DBF	Design Basis Flood
DC	Direct Current
DG	Diesel Generator
DGH	Distributing Group Header
DS	Drum Separator
ECCS	Emergency Core Cooling System
ECCSP	Emergency Core Cooling System Pump
ECR	Emergency Control Room
EFWP	Emergency Feedwater Pump
EMS-98	European Macro-seismic Scale
ENSREG	European Nuclear Safety Regulators Group
EU	European Union
FA	Fuel Assembly
FASS	Fast Acting Scram System
FC	Fuel Channel
FE	Fuel Element
FIHC	Fuel Inspection Hot Cell
GPS	Global Positioning System
HN	Hygiene Norm

IAEA	International Atomic Energy Agency
ICS	Information Computer System
Ignalina NPP	Ignalina Nuclear Power Plant
INSC	International Nuclear Safety Centre
IRV	Isolating and Regulating Valve
LEI	Lithuanian Energy Institute
LSW	Low Salted Water
MCC	Main Circulation Circuit
MCP	Main Circulation Pump
MCR	Main Control Room
MCT	Maintenance Cooling Tank
MSK-64	Medvedev-Sponheuer-Karnik seismic event classification scale
NPP	Nuclear Power Plant
NSFISF	New Spent Fuel Interim Storage Facility
OEP	Organization of Emergency Preparedness
PEP	Plan of Emergency Preparedness
PH	Pressure Header
PNAE	Russian acronym for “Rules and Norms in Atomic Energy”
PNIIIS	Russian acronym for “Industry and Research Institute for Construction Engineering Survey”
PSA	Probabilistic Safety Analysis
PSAR	Preliminary Safety Analysis Report
RBMK	Russian acronym for “Channelized Large Power Reactor”
RUZA	Russian acronym for “Beyond Design Basis Accidents Management Procedure”
RV	Reverse-Flow Valve
SAMS	Seismic Alarm and Monitoring System
SAS	Seismic Alarm System
SCC	Specially Cleaned Condensate
SF	Spent Fuel
SFA	Spent Fuel Assembly
SFISF	Spent Fuel Interim Storage Facility
SFP	Spent Fuel Pool

SFSF	Spent Fuel Storage Facility (SFISF or NSFISF)
SH	Suction Header
SL	Seismic Level
SMS	Seismic Monitoring System
SNF	Spent Nuclear Fuel
SWP	Steam-Water Piping
UHS	Ultimate Heat Sink
VATESI	Lithuanian acronym for "State Nuclear Power Safety Inspectorate"
VNIPIET	Russian acronym for "Design and Research Institute of Complex Energy Technology"

List of measurement units

%	Percent
A	Ampere
cm^2	Square centimeter
g	Gram
h	Hour
kcal	Kilo calorie
kg	Kilogram
kJ	Kilo Joule
km	Kilometer
km^2	Square kilometer
kV	Kilovolt
kW	Kilowatt
m	Meter
m^3	Cubic meter
mm	Millimeter
MPa	Megapascal
mSv	Millisievert
MW	Megawatt
^oC	Degree Celsius
s	Second
s^2	Second squared
t	Ton
MN	Meganewton
MVA	Megavoltamper

Introduction

The European Council of 24/25 March 2011 stressed the need to fully draw the lessons from recent events related to the accident at Fukushima Daiichi Nuclear Power Plant, and to provide all necessary information to the public. The European Council decided that all EU nuclear power plants should be reviewed, on the basis of a comprehensive and transparent risk and safety assessment ("stress tests"). European Commission and the European Nuclear Safety Regulators Group (ENSREG) on 24 May 2011 confirmed the specification of declaration which defines technical scope and the process to perform the "stress tests" and their review [1].

There is only one nuclear installation in Republic of Lithuania – Ignalina Nuclear Power Plant (Ignalina NPP) – corresponding the scope of European Commission and ENSREG declaration. In response to the declaration regulatory authority of Lithuania – State Nuclear Power Safety Inspectorate (VATESI) – on 27 May 2011 obligated Ignalina NPP to perform "stress tests" for two final shutdown power units, Spent Fuel Interim Storage and New Spent Fuel Interim Storage facilities, which are in operation and construction respectively.

This position paper represents the National Final Report of "stress tests" for Ignalina NPP in accordance with requirements stated in ENSREG declaration.

The National Final Report is based on the Licensee's "stress tests" Final Report, which was prepared by State Enterprise Ignalina NPP and presented to VATESI in 25 October 2011.

Chapter 1 of this report describes general data about the Ignalina NPP site, main characteristics and significant differences of the Units, current status of the Ignalina NPP units and main characteristics of Spent Fuel Storage disposed on the Ignalina NPP site. Also Chapter 1 presents relevant PSA results of Ignalina NPP. Chapters 2 to 5 address the assessment of extreme situations referred in specification of ENSREG declaration, namely those are earthquake, flooding, extreme weather conditions, loss of electrical power and loss of the ultimate heat sink. Chapter 6 reports the severe accident management at Ignalina NPP. Chapter 7 represents general conclusion and recommendations of this National Final Report.

1. General data about the site and nuclear power plant

1.1. Brief description of the site characteristics

The Ignalina NPP site is located in the north-eastern part of Lithuania, close to the borders of Belarus and Latvia. The plant is built on the southern shores of Lake Drūkšiai and 39 km from the Ignalina town. The biggest cities located near to the Ignalina NPP are the capital of Lithuania Vilnius (130 km) with about 550 thousands of habitants and Daugavpils in Latvia (30 km) − about 126 thousands of habitants. The staff of Ignalina NPP lives in Visaginas town which have about 29 thousands of habitants and is at distance of 6 km from Ignalina NPP. Location of the Ignalina NPP site is shown in Figure 1.1-1.

Figure 1.1-1. Location of the Ignalina NPP site

There are two Units at Ignalina NPP site. Both Units are permanently shut down and under decommissioning process now. For both Units separate operation licenses are valid as nuclear fuel is in Units buildings. Licenses holder is State Enterprise Ignalina NPP. Ignalina NPP has the following valid licenses:

- License for operation of Unit 1;
- License for operation of Unit 2;
- License for operation of Spent Fuel Interim Storage Facility;
- License for operation of Cemented Waste Storage Facility;

- Four licenses for construction of various radioactive waste management nuclear facilities including license for construction of New Spent Fuel Interim Storage Facility;
- License for design of Disposal Facility for Very Low Level Waste.

1.1.1. Main characteristics of the Units

Both Units of Ignalina NPP have the reactors of RBMK-1500 type. "RBMK" is the Russian acronym for "High Power Channel-type Reactor". It is boiling-water reactor with graphite moderator. Uranium fuel is located inside separately cooled fuel channels (pressure tubes). Designed thermal power of the RBMK-1500 reactor is 4800 MW, what corresponds to 1500 MW electrical power. Authorised power was 4200 MW and 1350 MW accordingly. Two turbine generators, each of 750 MW electric capacities, are installed at each Unit.

Life cycle of the Ignalina NPP is presented in Table 1.1-1.

Table 1.1-1. Life cycle of Ignalina NPP

	Unit 1	Unit 2
Start of construction	1978	1980
First criticality	4 Oct 1983	11 Dec 1986
Synchronization with energy system	31 Dec 1983	18 Aug 1987
Commissioning	31 Dec 1983	31 Aug 1987
Permanent shutdown	31 Dec 2004	31 Dec 2009

Spent fuel is stored in storage pools located in Unit buildings and in common Spent Fuel Storage Facilities described below. The New Spent Fuel Interim Storage Facility is under construction to cover all remaining in both Units spent fuel.

1.1.2. Description of the systems for conduction of main safety functions

Reactor

At the Ignalina NPP the water-cooled, thermal neutron with graphite moderator, pressure-tube-type boiling-water power reactors RBMK-1500 are installed. Fuel assemblies are placed in the individual channels and refuelling was performed during reactor power operation. The reactors used low-enriched fuel. The fuel of 5 types was used at the Ignalina NPP:

- Fuel assemblies (FA) with fuel of 2% enrichment by U^{235}. Uranium mass in FA is 111.2 kg;
- Fuel assemblies (FA) with fuel of 2.1% enrichment by U^{235} from reprocessed uranium. 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.

Each nuclear fuel assembly is located in a separately cooled fuel channel (pressure tube). There are a total of 1661 of such channels and the cooling water flow rate is equally divided among associated feeder pipes. After passing the core, pipes are brought together to feed the steam-water mixture to the separator drums.

Steam cycle

RBMK-1500 is one coolant loop unit. Saturated steam with pressure of 6.5 MPa, diverted to the turbines, is generated directly in the reactor channels and separated in drum separators. Simplified Ignalina NPP heat diagram is shown in Figure 1.1-2. Water, cooling the reactor (1), passes the core, boils and partially evaporates. Water-steam mixture enters the drum separators (3), located above the reactor. The separated steam from drum separators enters the turbines (4). Spent steam condensates in the condensers (6). The condensate is fed by condensate pumps (7) to deaerators (8) and returns to the drum separator by the feed-water pumps (9). Water from drum separator is delivered for the core cooling by the main circulation pumps (10) and there it partially evaporates again.

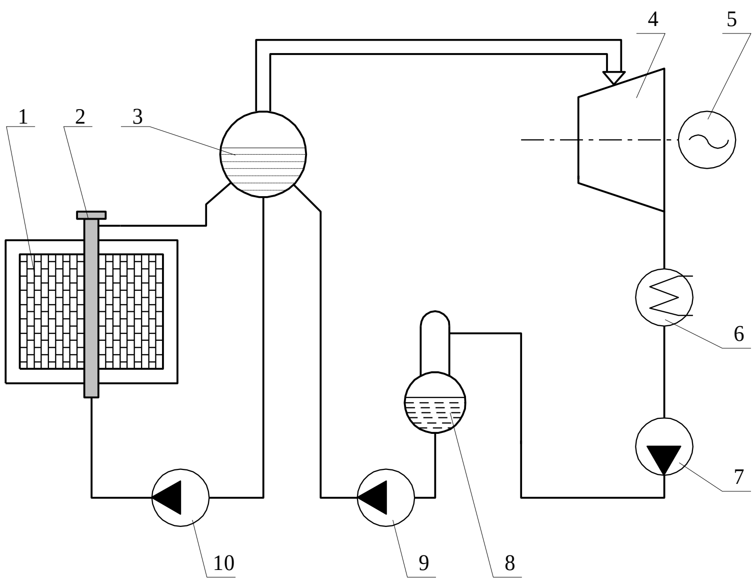


Figure 1.1-2. Simplified Ignalina NPP heat diagram

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

NPP layout

Each Unit consists of five main buildings. Reactor buildings A1 and A2 are adjacent to a common building D1 and D2 housing the control rooms, electric instrumentation rooms and deaerator rooms. D buildings are adjacent to a common turbine hall G. The main buildings of the plant are situated about 400-500 m from the banks of Lake Drūkšiai. Each Unit has spent nuclear fuel storage pool, located in neighbouring hall to central (reactor) hall. Both Units have the following common facilities: low-activity solid waste storage, medium- and high-activity solid waste storage, liquid waste storage, storage facility for bitumen compound, 110/330 kV switchyard, nitrogen and oxygen production facility and other auxiliary systems. 12 diesel-generators (six diesel-generators per Unit) for emergency power supply are housed in common building and physically separated from each other by walls. Currently all diesel generators at Unit 1 are put out of operation and isolated, 3 of them are conserved and 3 under dismantling process. All 6 diesel generators at Unit 2 are ready for operation. A separate water-pump service stations are built for each Unit, serving the needs of uninterrupted supply of water.

Current state of the Units

Unit 1 of Ignalina NPP was shut down by 31 December 2004; Unit 2 was shut down by 31 December 2009. Both Units are under decommissioning process now.

The reactor of Ignalina NPP Unit 1 was defueled at the end of 2009. Part of withdrawn fuel assemblies were transported to Unit 2 for re-use, other are placed in spent fuel pools. State on 1 July 2011: 7175 fuel assemblies are stored in spent fuel pools of Unit 1. Taking into account safety justification documentation some of mechanical and electrical equipment, which are no more required for ensuring safety functions, are put out of operation, isolated and dismantled.

The reactor of Ignalina NPP Unit 2 is partly defueled. State on 27 of December, 2011: 1278 fuel assemblies are still in reactor, and 7045 fuel assemblies are stored in spent fuel pools. The results of measurements of Unit 2 reactor parameters in condition of cold unpoisoned reactor after the final shutdown for decommissioning has shown that all the measured characteristics of the reactor are within the limits set in the Ignalina NPP Unit 2 reactor specification (passport of reactor unit). Taking into account safety justification documentation some of mechanical and electrical equipment, which are no more required for ensuring safety functions, are put out of operation and isolated.

Spent Fuel Storage

There are few systems of spent fuel handling and storage that perform the following functions:

- To transport the fuel assembly (FA) within the reactor building;
- To store FAs in the pool;
- To cut FAs and to put into transport cover having 102 places;
- To store transport covers with spent fuel in the pool;
- To load transport covers into cask;
- To transport casks with spent fuel to the Spent Fuel Storage;
- To store casks with spent fuel in the Spent Fuel Storage during 50 years.

Spent fuel is stored in both Units storage pools and in Spent Fuel Storage Facility.

Fuel storage pools

Storage pools are intended for temporary storage of spent fuel in water, screening radiation and removing heat release. There are 8 storage pools and 4 pools for handling operations. All pools for each Unit separately are situated in the reactor buildings in Storage Pool Halls. The total water surface of all pools at each Unit is 467.7 m^2. Cooling of storage pools carried out by Pump-Cooling Plant, which consists of 4 pumps and 3 heat exchangers. Each pump provides flow rate of 160 m^3/h. Main characteristics of Pump-Cooling Plant are:

- Heat removal capacity 4000 kW;
- Total water flow rate in pools 400 t/h;
- Flow rate of cooling water in heat exchangers 480 t/h.

Water temperature in pools is maintained in the range of 20^{o}C to 50^{o}C. The temperature safety operation limit is 60^{o}C.

Spent Fuel Interim Storage Facility

Spent Fuel Interim Storage Facility (SFISF) of dry type is situated at distance of about 1 km from Unit 2 and of 400 m from Lake Drūkšiai. SFISF consists of operation buildings and reinforced concrete platform where dry casks with spent fuel are placed vertically. Two types of casks are used: CASTOR RBMK (steel) and CONSTOR RBMK-1500 (reinforced concrete). The SFISF is surrounded by guarding concrete fence.

The SFISF is designed to store 120 casks during 50 years. At present time 20 CASTOR RBMK casks and 98 CONSTOR RBMK-1500 casks are stored; places for 2 casks are reserved for unforeseeable operations.

New Spent Fuel Interim Storage Facility

New Spent Fuel Interim Storage Facility (NSFISF) is under construction near Ignalina NPP at distance of about 550 m. Commissioning of the NSFISF is planned in 2012. This storage facility is intended for handling and long-term storage in special building of 201 casks of CONSTOR® RBMK-1500/M2 type with spent fuel. Storage building will be equipped with facilities to handle containers and spent fuel. There will be Reception Hall, Storage Hall, Cask Service Station and Hot Cell in the building. Design of the NSFISF takes into account possible seismic, aircraft crash and air-blast wave loadings.

1.2. Significant differences between Units

Design differences

There are some design differences between the first and second Units of Ignalina NPP. These differences are described in the Safety Analysis Report of Ignalina NPP Unit 1 [2]. List of most important design differences follows:

- All inner walls of Accident Localization System (ALS) compartments of Unit 2 have a tight proof steel liner. ALS compartments of Unit 1 have partly only steel liner.

- Ventilation systems of rooms adjacent to ALS compartments of Unit 1 have backup power supply from diesel-generators. There is no backup power supply of such systems of Unit 2 because ALS compartments are much more waterproof at Unit 2.
- Power supplies of control rod drives are different at Unit 1 and Unit 2.
- Sealing/locking devices of fuel assemblies are different at Unit 1 and Unit 2.
- Gas Release Cleaning Systems are different at Unit 1 and Unit 2.
- There is insignificant difference between Service Water Systems of Unit 1 and Unit 2.
- Some valves of Interim Circuit at Unit 1 are controlled manually only, whereas these valves at Unit 2 are controlled automatically.

Current differences

Now there are additional differences between Units caused by different decommissioning stages of Unit 1 and Unit 2. Main safety significant difference is that Unit 1 reactor is fully defueled whereas in Unit 2 reactor 1335 fuel assemblies remain. So, systems important to safety of spent fuel pools are in operation at Unit 1. Safety systems and systems important to safety of reactor and spent fuel pools are in operation at Unit 2. The main systems, which are currently under operation at Unit 1 and 2, are shown in Table 1.2-1.

Table 1.2-1. Currently operating systems

No	System	Unit 1	Unit 2
1.	Reactor and system of steam and gas discharge from the reactor cavity	Yes	Yes
2.	Reactor power monitoring and control system	No	Yes
3.	Coolant flow through FC regulation system	No	Yes
4.	Fuel reloading system	Yes	Yes
5.	Main circulation circuit (MCC)	No	Yes
6.	Live steam pipelines	Yes	Yes
7.	Service water supply system	Yes	Yes
8.	Reactor blowdown and cooling system	Yes	Yes
9.	MCC bypass purification system	Yes	Yes
10.	MCC and reactor makeup system	Yes	Yes
11.	Unit 2 reactor and MCC makeup system transit pipelines	Yes	No
12.	LSW, SCC, CPW consumers makeup system	Yes	Yes
13.	Reactor maintenance cooling system	No	Yes
14.	Spent fuel storage handling system	Yes	Yes
15.	Casks handling system	Yes	Yes

No	System	Unit 1	Unit 2
16.	Solid radioactive waste treatment system	Yes	Yes
17.	Spent ChWPS filtering materials acceptance and unloading system	Yes	Yes
18.	Drainage waters acceptance and pumping out system including leaktight compartments	Yes	Yes
19.	Radiation safety automated monitoring system	Yes	Yes
20.	Control of elements of systems important to safety	Yes	Yes
21.	Centralized monitoring information computer system TITAN	Yes	Yes
22.	System of ALS leaktight compartments	Yes	Yes
23.	Compartments ventilation system	Yes	Yes
24.	Power supply system including: Diesel generators Batteries	Yes 3 (conserved) 1	Yes 6 7
25.	Power plant fire extinguishing system	Yes	Yes
26.	Additional Hold-Down system	No (not installed)	Yes

1.3. Use of PSA as part of the safety assessment

Ignalina NPP Probabilistic safety analysis (PSA) was started in 1991 in the frame of "Barselina" project performed by joint team of specialists from Lithuania, Russia and Sweden. Project goal was to elaborate the line of development and the common base for risk assessment of severe accidents at RBMK reactors. The final PSA report [3] was issued in 2001.

The full power PSA and shutdown PSA models of the Ignalina NPP Unit 2 were developed and the method of probabilistic analysis was applied to the RBMK reactor. A number of deterministic analyses were performed to make the model realistic. Data base for NPPs with RBMK reactor was elaborated and used. The general conception of RBMK reactor analysis was developed.

Full power PSA results showed that core damage frequency is less than 1.0E-5 per reactor year.

Experience and information obtained at different phases of Ignalina NPP PSA were used for the development of the Ignalina NPP Unit 2 safety analysis reports, the Ignalina NPP beyond design-basis accidents list, beyond design-basis accidents management procedures and as input for other safety improving projects. As a result of PSA some modifications were proposed and implemented. Most important modifications are shown in Table 1.3-1.

Table 1.3-1. Modifications caused by PSA

No	Description of proposed improvement	Implementation
1.	Reduction of 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.	Change of EFWP/ECCSP pressure valves supply	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

Shutdown PSA confirmed the results of deterministic analysis of potential events (reactivity and heat removal accidents) at shutdown reactor that at observance of operating procedures all works at shutdown reactor can be executed with the minimal risk.

2. Earthquakes

2.1. Design basis

2.1.1. Earthquake against which the plant is designed

Characteristics of the design basis earthquake (DBE)

Ignalina NPP site is situated in the area of Eastern Europe platform, which is considered as less active area, seismic activity is low here. On the base of instrumental investigations and assessment of historical records design basis earthquake (DBE) for the Ignalina NPP area was assumed of intensity of 6 points on the MSK-64 scale (maximum ground acceleration is 0.5 m/s^2 = 0.05g). The beyond design basis earthquake (BDBE) for the Ignalina NPP area is the intensity of 7 points on the MSK-64 scale (maximum ground acceleration is 1.0 m/s^2 = 0.1g). The DBE with the intensity of 6 points on the MSK-64 scale corresponds to the seismic level SL-1 of the European Macroseismic Scale EMS-98.

The calculations of reaction to the seismic impact were performed for Ignalina NPP buildings and heavy equipment. The results of strength analysis of Unit 2 Reactor Building (including spent fuel pools) structures show that the analyzed reinforced concrete walls and floors are capable to sustain DBE and meet the criteria of strength and crack resistance, specified in national regulation for construction.

Dry Spent Fuel Storage Facility is designed taking into account the intensity of 6 grades on the MSK-64 scale and New Spent Fuel Interim Storage – intensity of 7 grades. Casks of CASTOR RBMK, CONSTOR RBMK-1500 and CONSTOR® RBMK-1500/M2 types are designed to withstand vertical acceleration of 110g, 87g and 85g correspondingly. This considerably exceeds acceleration acting on the casks in case of DBE and BDBE.

Methodology used to evaluate the design basis earthquake

Special researches on study of seismicity of the Ignalina Nuclear Power Plant site were carried out in 1988. According to the results of these researches the Instrumental Researches Report [4] was issued which includes summary data about the geological and tectonic structure as well as seismicity of the Ignalina Nuclear Power Plant site.

To assess the region seismicity, historical records since the year 1616 were observed and an attempt was made to assess these events according to scale MSK-64. 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. The intensity of a number of events, to which the intensity of more than 6 points was previously attributed, was called in question. Investigation results are described in the report [4]. Taking into account the most unfavourable conditions (the focus directly under the site), 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 the MSK-64 scale.

Besides the local earthquakes, the Ignalina NPP site can undergo shakings from the remote earthquakes of the Carpathian area (depth of the focuses is about 120 km, distances are about 1300 km) and the Scandinavian focuses like earthquake in Skagerak in 1904. The maximum force of shakings on the Ignalina NPP site from the focuses of the remote Carpathian and Scandinavian earthquakes will not exceed intensity of 5 points on the 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 micro oscillations 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.

Conclusion on the adequacy of the design basis for the earthquake

In order to assess possible seismic impacts of the local earthquakes on the soils of foundations of the Ignalina NPP 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 micro zoning works is presented in Table 2.1-1. The conclusion is that the expected intensity of seismic impacts on categories II÷III soils is 6.5 points (for Unit 1) while on category II soils it is 6.0 points (for Unit 2). The accelerograms and other characteristics corresponding to these conditions were prepared. In 1991 VNIPIET (the general designer of the Ignalina NPP) took the data of PNIIIS institute as a basis and used these data to calculate floor accelerograms and floor response spectra of main Ignalina NPP structures.

Table 2.1-1. Micro zoning result

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

The probabilistic characteristics of the Ignalina NPP main structures floor response spectra in case of earthquakes were calculated. The results of the probabilistic processing of the design-basis spectra are provided in documents [5], [6], [7], [8]. According to the results of the analysis carried out, the probabilistic characteristics of available spectra correspond to the beyond design basis earthquake. In case of DBE the average of distribution is 2 times less.

DBE of 6 points according to MSK-64 scale was taken as a design basis for the Spent Fuel Interim Storage Facility (SFISF). The appropriate maximum acceleration on the ground surface is 0.6 m/s^2 = 0.06g. The following components of the SFISF were designed taking the DBE into account:

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

CONSTOR RBMK-1500 and CASTOR RBMK casks are designed to bear the impact of significant loads acting on them in case of the drop of a cask during handling operations or transportation to the SFISF.

The structure of CONSTOR RBMK-1500 cask bears the overload of 87g, while CASTOR RBMK bears the overload of 110g. 32M baskets and fuel bundles bear the overload up to 85g. Such overloads are possible in case of accidents during transportation of a loaded cask to the SFISF. 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 tip over 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, while the safety factor for CONSTOR RBMK1500 is equal to 2.14).

New Spent Fuel Interim Storage Facility (NSFISF) is designed to withstand the DBE of 7 points according to the MSK-64 scale with the maximum acceleration on the ground surface of 1.0 m/s^2 = 0.1g. Safety significant structures, systems and components of NSFISF are designed to bear the impact of DBE. The equipment for cask loading at the Unit 1 and Unit 2 are designed to bear the impact of DBE.

The case of CONSTOR® RBMK1500/M2 cask is 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 or transportation to the NSFISF.

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. 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 tip over and sliding in case of simultaneous impact of horizontal acceleration a_H = ±0.2g and vertical acceleration a_V = ±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).

2.1.2. Provisions to protect the plant against the DBE

Reactor and other systems

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

Safety Justification of Unit 2 Reactor indicates that after unloading of 110 fuel assemblies from the reactor core, the critical state becomes impossible even in case of withdrawal of all CPS rods. Currently, only 24 FASS rods have been withdrawn from the core of Unit 2 reactor, while the other 187 CPS rods 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 [9]. The reactor cooling system includes the main circulation circuit and the blow-down 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 pipelines Du 300:

- stresses applied taking into account the operational and seismic loads under DBE do not exceed the permissible ones regulated by PNAE G-7-002-86 [10];
- amongst the pipelines Du 300 the most dynamically loaded in case of possible seismic impact are downtake pipelines in the steam separator rooms;
- the applied 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 may be assessed as 12 MPa.

Fuel storage pools

It is stated in the Ignalina NPP Detailed Design [11] that the equipment used 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 used 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 BDBE.

According to the design the bottom of the spent fuel pools of Ignalina NPP Units 1, 2 is made of the double liner and the space between the liners is filled with 90 mm thick concrete of high drain-ability. 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 drain-ability concrete is provided with special drain channels, ensuring the drainage of the possible water leakages from the space between the liners to the drainage pipelines of 89 mm 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 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. According to the recommendations of the VNIPIET report, 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. So far the armour plates are not laid in the 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. 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. 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. 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 cm^2 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.

Buildings and cranes

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 [12] 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 may 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 Ignalina NPP design it is indicated that 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 hand-up of SFA, cartridges with SFA or baskets with the bundles of fuel elements 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 strength in case of the DBE.

Spent Fuel Storage Facilities

The basic safety criteria during the storage and handling of the spent fuel in Spent Fuel Storage Facilities (SFSF) in the CONSTOR RBMK1500, CASTOR RBMK and CONSTOR® RBMK1500/M2 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 SFSF 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 accident scenarios [13], [14], [15], [16], [17].

Subcriticality of the spent fuel loaded into the CONSTOR RBMK150, CASTOR RBMK and CONSTOR® RBMK1500/M2 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 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.

The cases of casks and 32M 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 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/s. The shock load in this case is estimated at 26 MN, 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:

- 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 casks are designed.

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/s. The calculation shows that for this beyond design-basis accident 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. 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.

The CONSTOR RBMK1500 and CASTOR RBMK casks are stored on the open SFISF 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).

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).
- The calculated maximum temperature for a surface of CASTOR RBMK cask is 100°C, that corresponds to the temperature inside the cask 212°C. For a surface of CONSTOR RBMK1500 cask 72°C corresponds to the temperature inside the cask 271°C. The actual measured temperatures of a surface of the casks on the SFSF site are considerably lower and their values change only depending on the change of the temperature of the ambient air.

Since in the new NSFISF 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.

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. The calculated justification is presented in the reports [15], [16]. The casks operation experience shows that the actual dose rate is below the calculated value.

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, the spent fuel will be repacked 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 SFISF. The Organization of Emergency Preparedness (OEP) of the Ignalina NPP in order to develop further personnel and population protection measures according to the Plan of Emergency Preparedness (PEP) in force will require the Radiation Protection Service of OEP to carry out the measurement of the dose rates in the area depending on the place and character of a damage of the SFSF shielding wall.

SFISF is located within the existing sanitary protection area of Ignalina NPP. 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 SFISF will be needed in this case.

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

- coincidence of the beyond design-basis seismic impact and transportation of CONSTOR® RBMK1500/M2 cask with non-leaktight spent fuel to NSFISF;
- 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 SFISF using the special railway transporter there is a possibility of tip-over 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. The tip over 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 tip-over scenario during transportation from the power units to NSFISF in order to assess the possibility of the seal failure of the cask in case of its tip-over in the aforementioned configuration is under discussion with the Contractor of the NSFISF 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 Ignalina NPP 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 NSFISF determined at 500 m distance from the fence (the sanitary protective area of NSFISF is determined inside the sanitary protective area of Ignalina NPP site which is 3 km).

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 Centre depending on the place and character of damage of the reinforced concrete fence of the storage hall of the NSFISF.

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 NSFISF.

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 (NSFISF Project supposes very conservatively that repackaging of 10 casks will be required during NSFISF 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 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.

Seismic Alarm and Monitoring System

Ignalina NPP has the Seismic Alarm and Monitoring System (SAMS) that intended to inform operators of Main Control Rooms about the coming earthquake and to record data of reactor building and main equipment reaction during earthquake.

SAMS consists of four external seismic stations at distance about 30 km from Ignalina NPP and one station on the Ignalina NPP site, see Figure 2.1-1. Data are transferred from external stations using radio link. Besides, 18 acceleration sensors are installed in the reactor buildings and on steam drum separators.

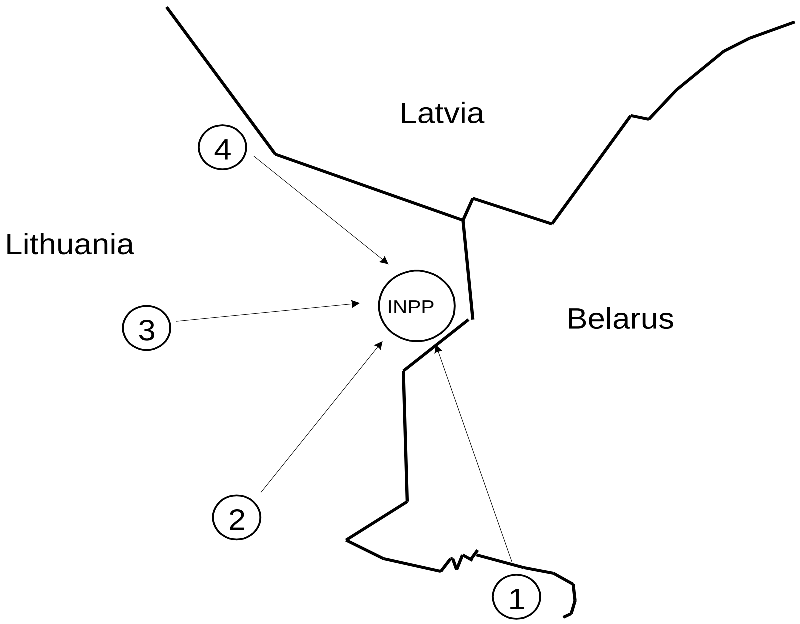


Figure 2.1-1. Layout of the seismic stations

1 – station in Didžiasalis (Navikai village), 2 – station near Ignalina (Ažušilė village),
3 – station in Salakas, 4 – station in Zarasai (Dimitriškės village)

Equipment of seismic alarm and monitoring system installed directly at Ignalina NPP site includes:

- three sensors of SAS-320 system in boreholes on the NPP site;
- 1 sensor of SSA-320 type of SMS system in borehole on the NPP site;
- 6 sensors of SSA-320 type of SMS system to monitor Unit 2 Building A;
- sensor of CA-164 type of SMS system installed on one drum separator of Unit 2;
- 16 receiving aerials of the SAS and SMS systems on the roof of Unit 2 Building A;
- GPS system of exact time reception on the roof of Unit 2 Building A;
- 3 data reception and conversion cabinets in room 1404/1 of Unit 2 Building A;
- central control panels of the system with recording computers at the MCR of Unit 2 Building D.

Schematic layout of the seismic equipment at Ignalina NPP site is shown in Figure 2.1-2.

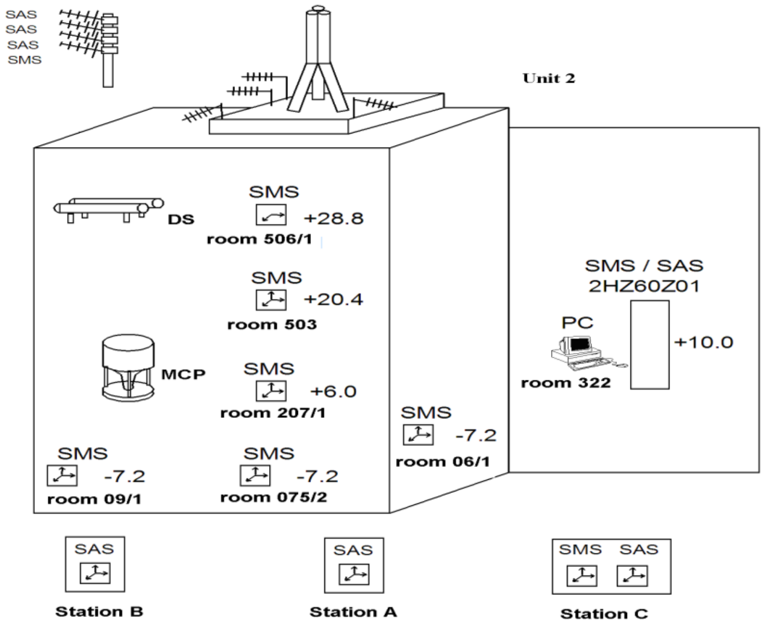


Figure 2.1-2. Layout of the seismic equipment at Ignalina NPP

Indirect effects of the earthquake

Possible loss of external power supply and loss of ultimate heat sink caused by any circumstance including an earthquake is discussed in Section 5 below.

An earthquake may not prevent access of personnel, diesel fuel and additional equipment to the NPP site. Access delay no more than 8 hours is possible; this time is uncritical for NPP safety.

No other external effects impact the Ignalina NPP safety.

2.1.3. Compliance of the plant with current licensing basis

Licensee ensures that plant systems, structures, and components that are needed for maintaining safe shutdown after earthquake, or that might cause indirect effects, remain in operable conditions.

The design measures are foreseen to protect spent fuel storage facilities in case of the design-basis earthquake and to prevent exceeding radiation safety limits.

Protective measures, interactions with external organizations, technical facilities, resources, rooms and means of communication are defined in the Emergency Preparedness Plan and Emergency Preparedness Operational Procedures that put in force at Ignalina NPP.

2.2. Evaluation of safety margins

2.2.1. Range of earthquake leading to severe fuel damage

Reactor building structures, systems and components that ensure the safety of fuel storage in the Unit 2 reactor and in pools of both Units are capable to withstand the design basis earthquake (DBE) taking into account possible failures of supporting systems for the time period sufficient for repair works. DBE is specified in Chapter 2.1.1 of this report. If an earthquake occurs with the force more than DBE, the design does not guarantee against spent fuel damage of different severity. In that case fission products will discharge to the environment. Such a situation is foreseen in the

Emergency Preparedness Plan and Emergency Preparedness Operational Procedures that put in force at Ignalina NPP and described in Section 6 of this report.

Spent Fuel Interim Storage Facility and designed New Spent Fuel Interim Storage Facility including casks of all types are capable to withstand the DBE. Safety limits of fuel sub-criticality, fuel temperature and cask external radiation will be not exceeded during and after beyond design basis earthquake taking into account possible failures of supporting systems (e.g. total long-term loss of power supply).

2.2.2. Range of earthquake leading to integrity loss of leak-tight compartments

Design-basis earthquake (DBE), which do not cause the loss of integrity of NPP leak-tight compartments that function as a containment, is specified in Chapter 2.1.1 of this report. If an earthquake occurs with the force more than DBE, the design does not guarantee integrity of leak-tight compartments. In that case the increased radiation level is expected at the NPP site. Such a situation is foreseen in the Emergency Preparedness Plan and Emergency Preparedness Operational Procedures that put in force at Ignalina NPP.

2.2.3. Earthquake exceeding the design basis earthquake for the plants and consequent flooding exceeding design basis flood

Design-basis earthquake (DBE) or earthquake exceeding the DBE cannot provoke any external flooding at Ignalina NPP site, see Section 3 below.

2.2.4. Measures which can be envisaged to increase robustness of the plants against earthquakes

As a result of “Stress Tests” carried out, the following measures are proposed by Licensee, which could be envisaged to increase plants robustness against seismic phenomena and would enhance plants safety:

- To consider the necessity of the emergency preparedness procedures amendments or addenda after reception and studying calculation results of the SF cask tip over during transportation and assessment of the consequent environment, personnel and population impact.

- To consider the possibility of the seismic alarm and monitoring system application for formalization of the emergency preparedness announcement criterion and the subsequent inclusion of this criterion in the operational manual of the seismic warning and monitoring system.

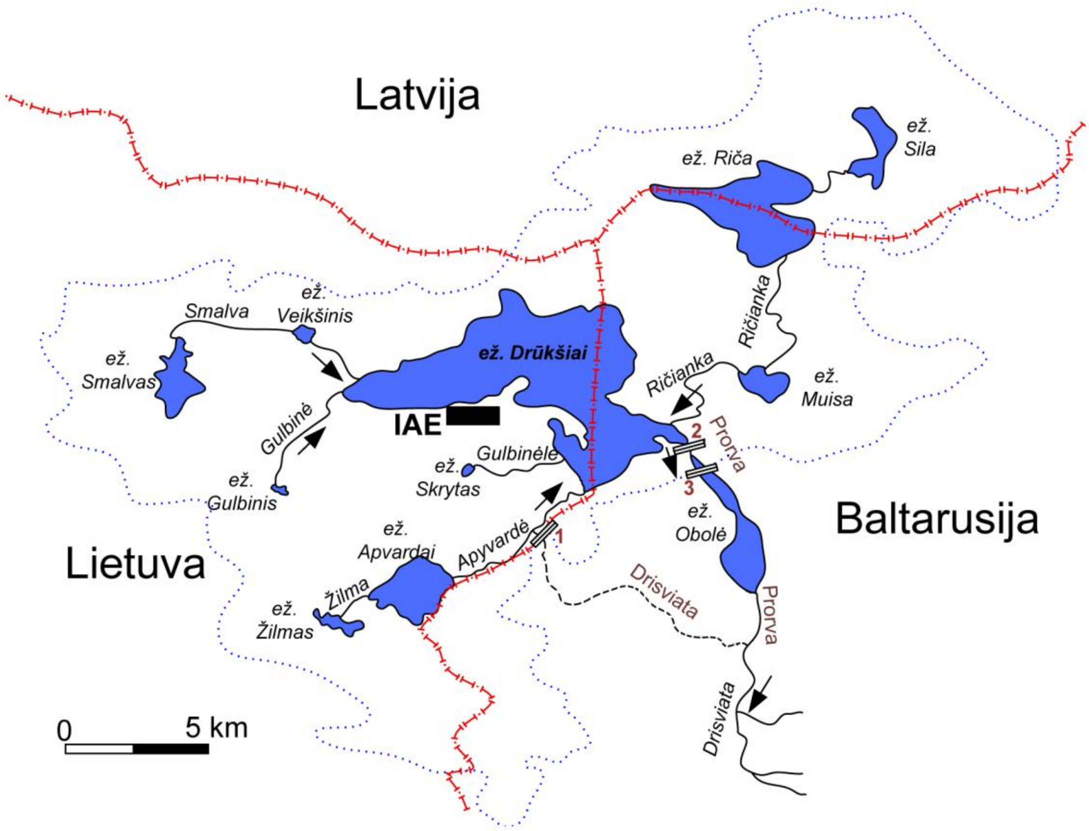
3. Flooding

3.1. Design basis

3.1.1. Flooding against which the plant is designed

Characteristics of the design basis flood (DBF)

The lake Drūkšiai serves as a natural water 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 3.1-1). There are a lot of lakes in the neighbourhood of the Ignalina NPP. The total surface of water (without Lake Drūkšiai) makes 48.4 km^2. The density of rivers is about 0.3 km/km^2.



Legend:

River flow directions

Artificial dams

Country border

Basin contour

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

Figure 3.1-1. Scheme of Lake Drūkšiai basin

1 – blind earthen dam, Structure 501, 2 – water regulating Structure 500,
3 – hydroelectric power plant "Druzhba Narodov" dam

Water levels in the Lake Drūkšiai relatively the Baltic Sea level are specified in Table 3.1-1.

Table 3.1-1. Water levels in the Lake Drūkšiai

	Level, m
Normal	141.6
Minimal	140.7
Maximal	142.3

There are three hydro-engineering structures regulating the Lake Drūkšiai water discharge: the water regulating Structure 500, Blind earthen dam (dike) of River Drisviata (Structure 501), and Dam of hydroelectric power station "Druzhba Narodov". Levels of all these structures are specified in Table 3.1-2. The levels were rechecked and documented in the period since 16 September till 17 October, 2011.

Table 3.1-2. Levels of hydro-engineering structures in the Lake Drūkšiai

	Level, m
Slope and concrete platform of the water regulating Structure 500	143.2 – 143.3
Blind earthen dam (dike), Structure 501	142.7 – 142.8
Dam of hydroelectric power station "Druzhba Narodov"	142.5 – 142.6

Ignalina NPP buildings and structures of interest are situated at levels indicated in Table 3.1-3.

Table 3.1-3. Levels of Ignalina NPP buildings and structures

	Level, m
Service water pump stations (lowest level)	144.0
Spent Fuel Storage Facility	149.0
Building of diesel generators	149.5
330/110 kV switchyard	153.15
New Spent Fuel Interim Storage Facility	155.5

Comparison of all levels is presented in Figure 3.1-2.

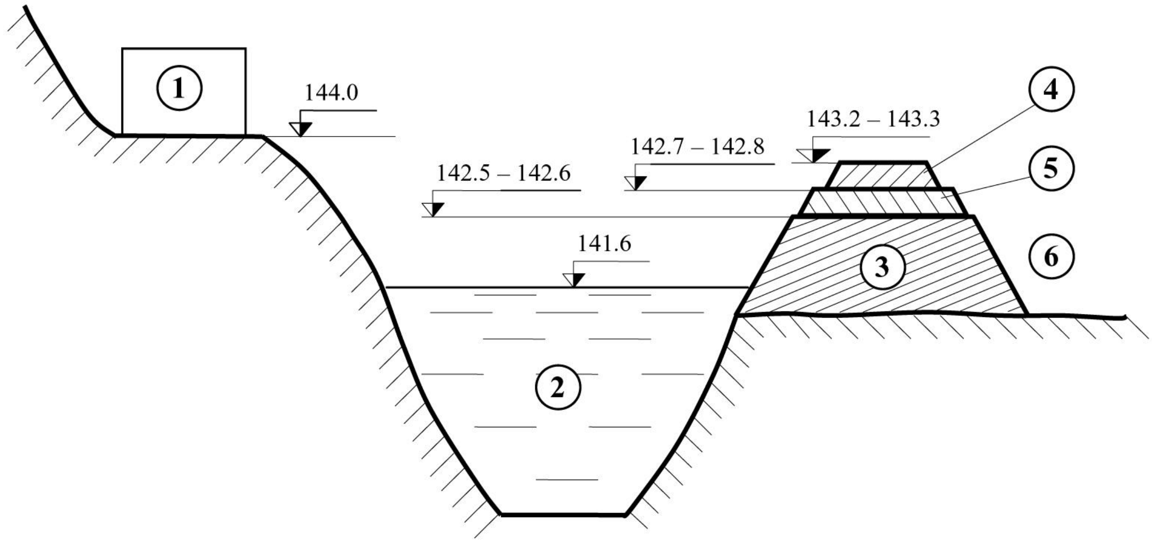


Figure 3.1-2. Levels of Lake Drūkšiai and hydro-engineering structures

1 – Service water pump station, 2 – Lake Drūkšiai, 3 – dam of hydroelectric power plant “Druzhba narodov”, 4 - water regulating Structure 500, 5 - blind earthen dam, Structure 501, 6- high-water bed of river Drysviaty

Methodology used to evaluate the design basis flood.

Tsunami is impossible at the Lake Drūkšiai. Taking this into account, the methodology to evaluate the design basis flood is based on the comparison of theoretically possible the highest level of Lake Drūkšiai (the level of hydroelectric power station “Druzhba Narodov” dam) and levels of Ignalina NPP buildings and structures given in Table 3.1-3. This evaluation is conservative: historical data confirm that the maximal level of water indicated in Table 3.1-1 was never exceeded.

Conclusion on the adequacy of protection against external flooding

Comparing the levels of Lake Drūkšiai and of Ignalina NPP buildings and structures, the conclusion may be made that external flooding of Ignalina NPP buildings and structures is impossible. In the worst case theoretically possible the highest level of the lake is no more than the level of hydroelectric power station “Druzhba Narodov” dam i.e. lower than levels of all structures and buildings of Ignalina NPP.

3.1.2. Provisions to protect the plant against the design basis flood

The top levels of the hydro-engineering structures are located below NPP buildings and structures. No special provisions are foreseen to prevent the design basis flood impact because it is impossible.

There is no flooding threat outside the plant, including preventing or delaying access of personnel and equipment to the site.

3.1.3. Plants compliance with its current licensing basis

During uncontrollable abnormal rise of water level in Lake Drūkšiai, at the most negative flooding scenario, irrespective of the cause of its occurrence, the water level in Lake Drūkšiai cannot reach the marks, which could lead to the flooding of the Ignalina NPP buildings and facilities. Licensee does not need any additional measures to ensure that plants systems, structures, and components that are needed for achieving and maintaining the safe shutdown state, as well as systems and structures designed for flood protection, remain in operable condition.

3.2. Evaluation of safety margins

3.2.1. Estimation of safety margin against flooding

Comparing the levels of Lake Drūkšiai and of Ignalina NPP buildings and structures, the conclusion may be made that in the worst case the level margin for service water pump stations is at least 1.4 m. Levels of spent fuel storage facilities 149.0 m and 155.5 m provide margins of at least 6.4 m and 12.9 m above the level of hydroelectric power station “Druzhba Narodov” dam (theoretically possible the highest level of the lake).

3.2.2. Measures which can be envisaged to increase robustness of the plant against flooding.

Ignalina NPP design ensures the adequate protection against an external flooding. No additional measures are needed to increase robustness of the plant against flooding. The only measure was carried out during “Stress Tests” course is re-checking of levels of all three hydro-technical Structures regulating the level of the Lake Drūkšiai.

4. Extreme weather conditions

4.1. Design basis

4.1.1. Reassessment of weather conditions used as design basis

Verification of weather conditions

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 NPP 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 annual average wind speed is about 3.5 m/s, the average annual temperature is +5.5°C. The average calculated temperature of the coldest five-day period is –27°C.

Specifications for extreme weather conditions

Extreme weather conditions are rare in the vicinity of the Ignalina NPP site. During the storm in 1998 the wind speed of 33 m/s was registered. The absolute registered temperature maximum is +36°C, the absolute minimum is –40°C.

Assessment of the design basis conditions

Weather conditions used as the design basis of Ignalina NPP are based on the area climate conditions taking into account necessary margins. Extreme external temperature, wind speed and atmospheric precipitates, including their combinations, are considered in the plant design in accordance with construction regulations. Design basis conditions correspond to the real weather conditions in the area of the Ignalina NPP site.

Conclusion on the adequacy of protection against extreme weather conditions

Ignalina NPP operation during 26 years and additional 3 years of post-operational shutdown state confirm the adequacy of the plant protection against extreme weather conditions.

4.2. Evaluation of safety margins

4.2.1. Estimation of safety margin against extreme weather conditions

Design basis conditions of Ignalina NPP including spent fuel storage facilities taken into account the extreme weather conditions, which are possible in the area of NPP site. These conditions were comprehensively analysed in Technical Safety Justification, in Safety Analysis Reports of Unit 1 and 2, in Single Operating Unit Safety Report for INPP Unit 2 and no significant nonconformities were found.

4.2.2. Measures which can be envisaged to increase robustness of the plant against extreme weather conditions

No measures required which could be envisaged to increase plant robustness against extreme weather conditions and would enhance plant safety.

5. Loss of electrical power and loss of ultimate heat sink

5.1. Loss of electrical power

External power supply

Ignalina NPP is linked with external power supply via 110/330 kV switchyard (open distributive system): with grid of 330 kV using 6 power lines and with grid of 110 kV using 2 power lines. Off-site AC power supply may be provided from any power line of 330 kV or 110 kV. Connection between 330 kV switch-yard and 110 kV switch-yard is carried out via two coupling autotransformers AT-1, AT-2. Power rating of each autotransformer is 200 MVA.

At each Unit two block transformers, 4 operation transformers and 4 start-up auxiliary transformers are installed. At present the consumers are powered via start-up auxiliary transformers from the 110 kV grid. Block transformers and operation transformers are in standby mode.

Internal power supply

Each Unit of Ignalina NPP is equipped with 6 diesel generators of 5600 kW each. Currently all diesel generators at Unit 1 are put out of operation and isolated, 3 of them are conserved and 3 under dismantling process. All 6 diesel generators at Unit 2 are ready for operation.

Each Unit of Ignalina NPP is equipped with 7 accumulating batteries. 6 batteries provide power supply for instrumentation, communication and radioactivity monitoring systems and the seventh battery mostly for emergency lighting. Currently 6 batteries at Unit 1 are put out of operation and one battery still in operation. All 7 batteries at Unit 2 are in operation. Capacity of instrumentation batteries is enough for at least 12 hours and lighting battery for at least 9 hours without recharging.

Communication facilities and computers of the Accident Management Centre can be powered by the independent stationary diesel generator, which is installed in the OEP auxiliary room (see 6.12 below).

Two additional mobile diesel generators and special connecting points are foreseen.

5.1.1. Loss of off-site power

If the off-site power supply is lost, all diesel generators are starting automatically and provide consumers important to safety with power supply. The 6 kV voltage consumers and the 0.4 kV voltage consumers (through the step-down transformers) will be powered with interruption of no more 15 seconds.

The power rating of each diesel generator is 5600 kW. The designed volume of fuel is enough for operation of each diesel generator during 72 hours without refuelling to ensure emergency shutdown and cooling of the reactor. Since the Unit 2 reactor is shut down and is at a stage of defuelling, some consumers important to safety are taken out of operation and the fuel volume will suffice for more than 72 hours. The time, for which the available reserve of fuel will be enough to ensure power supply to the remaining consumers of Unit 2, was assessed. The data is presented in Table 5.1-1.

Table 5.1-1. Time reserve of Unit 2 diesel generators operation

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

Thus the minimum operation time of Unit 2 all 6 diesel generators without refueling is at least 5 days. This time is much more than needed for restoration of the off-site power supply. With the refueling the operation time is not limited. In order to carry out the refueling it is necessary to conclude the fuel supply contract.

Diesel generators are qualified for the design basis earthquake with intensity of 6 points.

All diesel generators and 6 out of 7 batteries of Unit 1 are taken out of operation. If the external power supply is lost, all Unit 1 AC power consumers will be de-energized except the radiation monitoring system, which is common for two units, located at Unit 1 but powered from DG-7 of Unit 2. General DC consumers and emergency lighting of Unit 1 will be powered from the battery 1AB-7 that still in operation. Power supply of Unit 1 instruments of water temperature and level in the storage pools will be lost. Now the power supply of this instrumentation is re-designed to provide power from DG-7 of Unit 2 or from mobile diesel generator connected to Unit 2 (see paragraph 5.1.2 below). December 2011 is set as the term to implement the new design.

Spent Fuel Storage Facilities (SFSF) will be de-energized in case of loss of off-site power. However it will not violate the safety limits because the spent fuel in casks is cooled using natural convection without any power supply. Radiation monitoring and security systems of SFSF may be powered from own independent sources.

In case of loss of external power supply the consumers of service water of Unit 1 are provided with service water by operating pumps of Unit 2. Unit 1 water and foam extinguishing systems are operated using Unit 2 motors which are powered from diesel generators.

Actions and interactions on restoration of Ignalina NPP external power supply are prescribed in proper instructions of the Lithuanian Energy System [18] and NPP [19]. In the Lithuanian Energy System instruction [18], the time needed for restoration of NPP power supply after possible total shutdown of the Lithuanian Energy System is approximately 30 minutes. Various variants of power supply restoration are foreseen including start-up of Pļaviņas Hydro Power Plant in Latvia and Kruonis Pumped Storage Plant in Lithuania.

5.1.2. Loss of off-site power and loss of the ordinary back-up AC power source

If the off-site power supply and all diesel generators are lost (station total blackout), instrumentation, communication and radioactivity monitoring systems and emergency lighting of Unit 2 will be powered from 7 batteries without interruption. General consumers and emergency lighting of Unit 1 will be powered from one battery. The rated capacity of the Vb2421 VARTA type battery is 2100 A×h at the 10 hour rate current 210 A. The discharge time of each battery for the full design load required for the emergency shutdown and cooling of the reactor is not less than one hour. Since Unit 2 reactor is shut down and is at the stage of defueling and some consumers are taken out of operation, the batteries discharge time will be considerably more. The engineering evaluation of discharge time for 6 main batteries of Unit 2 was performed; results are presented in Table 5.1-2. The evaluation is performed applying the conservative approach.

Table 5.1-2. Unit 2 batteries discharge time

Battery	2AB-1	2AB-2	2AB-3	2AB-4	2AB-5	2AB-6
Actual load, kW	8	18	15	38	33	22
Discharge current, A	36.4	81.8	68.2	172.7	150	100
Discharge time for actual load, hours	57.7	25.7	30.8	12.2	14	21

Thus, the minimum battery discharge time for the load at the Unit 2 is not less than 12 hours. The discharge currents of Unit1 and Unit 2 seventh batteries powering general consumers and emergency lighting are as follows: 223 A for 1AB-7, 109 A for 2AB-7. The discharge time for the actual load will be: 9.4 hours for 1AB-7, 19.3 hours for 2AB-7. This time is enough for restoration of the off-site power supply.

Batteries are qualified for the design basis earthquake with intensity of 6 points.

5.1.3. Loss of off-site power and loss of the ordinary back-up AC power sources, and loss of permanently installed diverse back-up power sources

If all power supply sources (i.e. all external power lines, all diesel generators and all batteries) are lost, two additional mobile diesel generators will be connected and started manually. One of them will provide power supply for instrumentation and radioactivity monitoring systems, other one for communication system. Connecting points for those diesel generators are installed on walls of the Unit 2 building and the administrative building. Operations with mobile diesel generators are described in instructions [19] and [20], estimated time of connection and start-up is one hour. The involved personnel are trained. The last testing of these diesel generators was carried out on 14 April 2011.

5.1.4. Conclusion on the adequacy of protection against loss of electrical power.

Existing diesel generators are capable to provide backup power supply of remaining systems important to safety at Ignalina NPP for the needed time.

Existing batteries are capable to provide diverse backup power supply of vitally important systems at Ignalina NPP for the needed time.

Mobile diesel generators provide additional diversity of backup power supply.

5.1.5. Measures which can be envisaged to increase robustness of the plant in case of loss of electrical power

- Contract for supply of fuel shall be negotiated to ensure refuelling of diesel generators during operation over a long period of time.
- To ensure power supply of temperature and level instrumentation of spent fuel pools it is necessary to implement new design of backup power supply from mobile diesel generator and to include addenda to corresponding procedures.

5.2. Loss of the decay heat removal capability / ultimate heat sink

The main ultimate heat sink for the Unit 2 reactor and for spent fuel pools (SFP) of both Units is Lake Drūkšiai. Heat abstraction to the lake is provided by the following supporting systems:

- Blow-down and Cooling System,
- Intermediate Circuit,
- Service Water Supply System,
- Pump-Cooling Plant of Spent Fuel Pools.

The alternative ultimate heat sink for the Unit 2 reactor and for spent fuel pools of both Units is the environment (atmosphere). In the case of the Unit 2 reactor, 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 accident localisation system and periodic makeup of the main circulation circuit.

In the case of the alternative ultimate decay heat sink for spent fuel pools, diffusion of heat to the environment occurs during evaporation of water from the surface of pools and periodic makeup of SFP, during water exchange in SFP using the drain waters and contaminated LSW collection and pumping system, and makeup system.

Heat Removal from the Reactor

Different modes of residual heat removal from reactor are used:

- Mode of cooling water natural circulation;
- Mode of cooling water forced circulation;
- Mode of cooling water broken natural circulation;
- Mode of cooling water bubbling.

Each of above listed mode is described below.

Mode of Natural Circulation of the Coolant

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

- the water level in the main circulation circuit (MCC) is above the levels of tie-in of SWP pipes in DS (not below level +29.7 m);
- any pressure in the DS, but not above the limiting pressure permitted for hydraulic pressure testing;
- all IRV and gate valves on the inlet to DGH are open;
- gate valves on connecting pipes of MCP PH and 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 the reactor core is ensured during 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);
- atmospheric pressure in the DS;
- 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 fuel channels and operation of BCS.

Mode of the Broken Natural Circulation

In the mode of the broken natural circulation, the cooling of the reactor core is ensured during 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);
- atmospheric pressure in the DS;
- all IRV are open;
- MCT with a nominal water level are connected to all DGH through the pipelines of ECCS;

- all bypass lines of DGH RV are open;
- closure of all gate valves of DGH 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 the reactor core is ensured during 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 +29.7m);
- atmospheric pressure in the DS;
- gate valves on all connecting pipes of MCP PH and SH are closed;
- a number of IRV and gate valves on inlet to each DGH are closed or all IRV are closed when the gate valves open on the inlet to DGH.

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.

The correspondence of the ultimate heat sinks to the various modes of heat removal from Unit 2 reactor is presented in Table 5.2-1.

Table 5.2-1. Ultimate heat sink from the reactor

Mode of heat removal from the reactor	Ultimate Heat Sink
Non-boiling mode of coolant natural circulation	main + alternative
Boiling mode of coolant natural circulation	alternative
Forced circulation of the coolant	main + alternative
Broken natural circulation of the coolant	alternative
Coolant bubbling	alternative

Monitoring of water temperature in reactor is carried out using thermocouples installed in the central tubes of some fuel assemblies. Monitoring of water level in reactor is carried out by at least two out of possible four different methods using design and additional level meters.

Assessment of the decay heat value in the Unit 2 reactor is provided in Appendix A. Calculation of the Unit 2 reactor heating-up process is provided in Appendix B.

The main result of the assessment of the decay heat value and of the calculation of the Unit 2 reactor heating-up process is as follows: if the offsite power supply and all diesel generators are lost, the critical temperature of the fuel cladding (700^{o}C) in the Unit 2 reactor will be reached after 6 days.

Heat Removal from Spent Fuel Pools

Heat is removed from spent fuel assemblies located in the spent fuel pools (SFP) of each Unit by means of cooling of water in pools using the operating pump-cooling plants. If for any reason it is impossible to use pump-cooling plants, the alternative mode provides heat removal during a limited period of time. In this case, diffusion of heat to the environment occurs via evaporation of water from the surface of pools and periodic makeup of SFP, and by means of water exchange in SFP using the drain waters and contaminated LSW collection and pumping system, and makeup system.

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 to 50°C. The limit of safe operation is 60°C. The temperature regime is determined by the quantity of heat exchangers connected to the service water, quantity of the operating pumps, the flow rate of the pool 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 pump-cooling plants can be switched-off without time limitations if the temperature of water in all the storage pools is below 45°C. If the pump-cooling plant is switched-off, the temperature of water in any SFP is reduced by the water exchange in this SFP.

Since the decay heat in Unit 1 SFP is low, the Unit 1 SFP pump-cooling plant 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 pump-cooling plant 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 the SFP of both Units is presented in Table 5.2-2.

Table 5.2-2. Ultimate heat sink from spent fuel pools

Mode of heat removal from the reactor	Ultimate Heat Sink
Operating Pump-Cooling Plant	main
Non-operating Pump-Cooling Plant	alternative

Calculation of the temperature regime of water in the Unit 1 SFP is provided in Appendix C. Calculation of the temperature regime of water in the Unit 2 SFP is provided in Appendix D.

If the offsite power supply and all diesel generators are lost, main results of temperature and level calculations are:

- The critical temperature of water (100^{o}C) in the Unit 1 spent fuel pools will be reached after 16 days;
- The critical temperature of water (100^{o}C) in the Unit 2 spent fuel pools will be reached after 7 days;
- The critical low level of water in the Unit 2 spent fuel pools corresponding of top of the fuel in assemblies will be reached after 40 days;
- The critical low level of water in the Unit 2 spent fuel pools corresponding of top of the fuel in transport 102-places covers will be reached after 15 days.

5.2.1. Design provisions to prevent the loss of the primary ultimate heat sink

To prevent the loss of the primary ultimate heat sink and the subsequent fuel degradation, the appropriate design modification is foreseen at Ignalina NPP that provides an additional diverse source of cooling water. After implementation of this modification, the possibility will appear to supply the artesian water to the main circulation circuit of NPP Unit 2 from the water intake area of the domestic potable water system. There is the possibility to power the domestic potable water pumps from their own diesel generator, therefore this system is independent and considered to be enough reliable.

5.2.2. Loss of the primary ultimate heat sink

The analysis of design modes of the reactor cooling (see Table 5.2-1) shows that the loss of the primary ultimate heat sink (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 operation of Blowdown and Cooling System 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 be transferred by operator to the boiling mode of natural circulation. In the second case, if the pressure gate valves and/or suction gate valves of all main circulation pumps (MCP) and on all connecting pipes of MCP pressure header and suction header and the gate valves on the inlet to each DGH are closed, the reactor cooling mode will switch over to the coolant bubbling mode. In both cases the heat will be removed to the alternative UHS and the cooling of the reactor core is ensured during the unlimited time by means of steam removal from the drum separators and periodic makeup of the main circulation circuit.

In case of loss of the primary UHS, the heat removal from the spent fuel assemblies located in pools can be carried out by water exchange in the spent fuel pools (SFP): discharge of water from SFP to tank TZ50B01 and further to TD51B01 and makeup from TD52B01 (tanks of Chemically Purified Water and Specially Purified Condensate). Thus the heat removal can be carried out with the flow rate up to 100 m^3/h during not less than 48 hours.

5.2.3. Loss of the primary ultimate heat sink and the alternate heat sink

Situations of loss of both primary and alternate ultimate heat sinks are completely covered by the case of blackout of the power plant. Therefore, they are considered in Section 5.3.

5.2.4. Conclusion on the adequacy of protection against loss of ultimate heat sink

If the ultimate heat sink is lost, Ignalina NPP staff has enough time and necessary means to prevent cliff edge effects. In case of total loss of ultimate heat sink (UHS) to prevent the subsequent fuel degradation the appropriate design modification is foreseen is developed at Ignalina NPP that provides an additional diverse source of cooling water, see 5.2.1. In accordance with the instruction [25] developed at Ignalina NPP, the supply of the artesian water to Ignalina NPP Unit 2 from the domestic potable water system is foreseen that increases the reliability of protection against loss of UHS.

5.2.5. Measures which can be envisaged to increase robustness of the plant in case of loss of ultimate heat sink

No additional measures are required to increase robustness of Ignalina NPP in case of loss of ultimate heat sink.

5.3. Loss of the primary ultimate heat sink, combined with station blackout

If the total blackout of the power plant occurs, all the design means of heat removal to UHS and the means of MCC makeup will be not available (out of operation). As a result there will be a loss of the coolant due to its evaporation. The warming up of the core components can begin after evaporation of water stocks from DS and MCC pipelines located above the Unit 2 reactor core.

Analysis results (see section 5.2) shows that due to the lowered water level in MCC not more than to 1 meter below fuel channel plugs (not lower than level +23.7 m) the most critical mode for probable damage of the fuel is the mode of the broken natural circulation.

As it is specified above, the removal of decay heat in this mode is carried out by removal of the steam from drum separators and makeup of the core by the natural water flow; thus the water level in MCC is maintained above the level of periodic maintenance cooling tank makeup. Besides, a part of decay heat is removed as a result of ventilation in MCC rooms. Thus, cooling of drum separators and steam-water pipelines, condensation of the steam and return of condensate to MCC (fuel channels) occurs.

Loss of maintenance cooling tank makeup and ventilation will lead to decrease in a water level in fuel channels due to evaporation. Temperature rise of fuel will begin after decrease of the water level below the top of the reactor core.

The results of the conservative analysis of water evaporation from fuel channels and heating of void fuel channels are provided in the report [27]. For the case when water circulation in fuel channels stops at day 365 after the reactor shutdown, it is determined that the time of water evaporation from fuel channels, when fuel element cladding temperature rises up to 700°C, is more than 140 hours (6 days). The temperature change tendency of fuel element claddings for this case is presented in Figure 5.3-1.

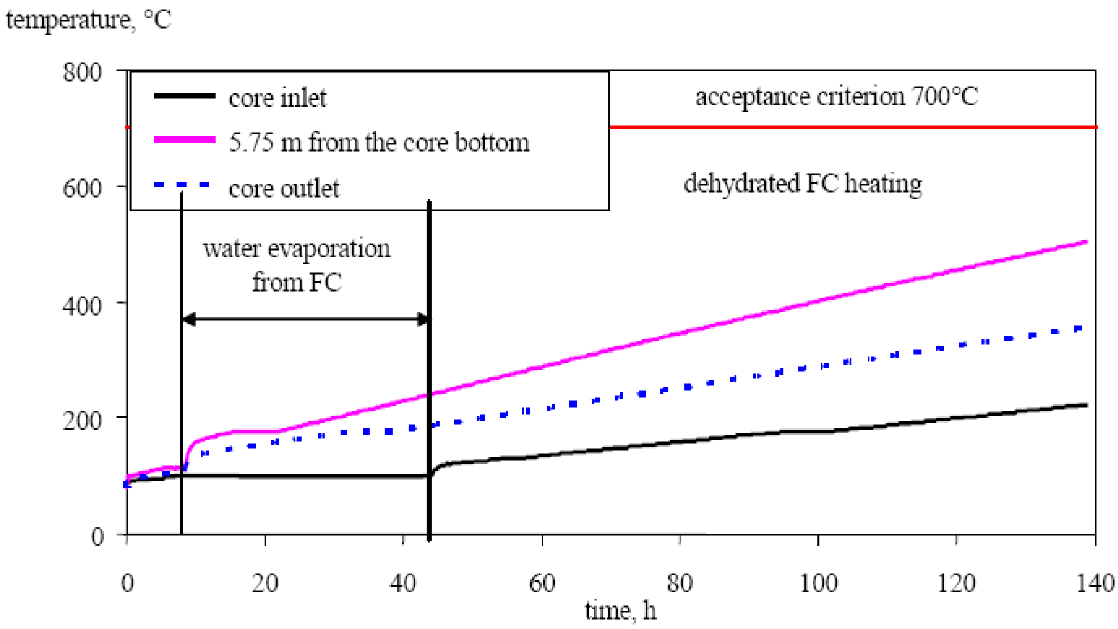


Figure 5.3-1. Change of fuel element cladding temperature for void fuel channel at day 365 after the reactor shutdown

The water volume in steam-water piping and fuel channels above the core is roughly 145 to 152 m^3. 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.

At total blackout of the power plant the spent fuel pools (SFP) primary and alternate ultimate heat sinks will be lost; this will inevitably lead to the gradual growth of the water temperature in pools. In the most heat-stressed SFPs, 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. Further the evaporation of water and decrease of a water level in SFP will occur. Damage of fuel occurs in 1005 hours after the beginning of SFP emptying [22].

5.3.1. Time of autonomy of the site before loss of normal cooling condition of the reactor core and spent fuel pool

Time of autonomy of the Ignalina NPP site before loss of normal cooling condition of the Unit 2 reactor core and spent fuel pools of Units 1 and 2 is defined by reserve amount of fuel for diesel generators. This time is estimated as at least 5 days (see 5.1.1). Only regular diesel generators are taken into account because additional mobile diesel generators are not capable to provide power supply for cooling pumps.

5.3.2. External actions foreseen to prevent fuel degradation.

The following external actions are foreseen to prevent fuel degradation in case of loss of the primary ultimate heat sink resulted in Ignalina NPP blackout:

- Restoration of Ignalina NPP external power supply by the Lithuanian Energy System in accordance with instruction [18]. The time needed for restoration of NPP power supply after possible total shutdown of the Lithuanian energy system is approximately 30 minutes.
- Prompt delivery of fuel for refuelling of diesel generators. Estimated delivery time is about one day; supplier and the exact obligation of supplier including the delivery time will be determined in contract.

5.3.3. Measures, which can be envisaged to increase robustness of the plant in case of loss of primary ultimate heat sink combined with station blackout

The measures, which can be envisaged to increase robustness of Ignalina NPP in case of loss of primary ultimate heat sink (UHS) combined with station blackout, are the same as for the case of only loss of electrical power, namely:

- Contract for supply of fuel shall be negotiated to ensure refuelling of diesel generators during operation over a long period of time.
- To ensure power supply of temperature and level instrumentation of spent fuel pools it is necessary to implement new design of backup power supply from mobile diesel generator and to include addenda to corresponding procedures.

The last measure is not concerned directly with the loss of primary UHS but is important for monitoring of the water condition in spent fuel pools after loss of primary UHS.

6. Severe accident management

6.1. Organization and arrangements of the licensee to manage accidents

6.1.1. Organisation of the licensee to manage the accident

Staffing and shift management in normal operation

Director General is the head of Ignalina NPP. He manages of 3 directions, 2 departments, Security Service and Group of Jurists. The most important direction is the Decommissioning Direction; number of personnel in this direction is more than 1600 people. The Decommissioning Direction is divided into 5 service offices: Operation Service, Decommissioning Projects Service, Radiation Safety Service, Radioactive Waste Handling Service and Dismantling and Decontamination Service. Each service office has 3 to 6 departments.

Measures taken to enable optimum intervention by personnel

Emergency Preparedness Plan and Emergency Preparedness Operational Procedures were updated and put in force at Ignalina NPP taking into account the shutdown state of both Units. The special structures – Organization of Emergency Preparedness (OEP) and Emergency Preparedness Headquarters – were established at Ignalina NPP. The OEP is staffed by the personnel of all NPP directions, departments and service offices on the professional basis and will work only if a beyond design-basis accident occurs. Headquarters of OEP consists of NPP high level managers:

- NPP Director General – the Head of the Organization of Emergency Preparedness (in case of his non-availability his duties are executed by the Decommissioning Director);
- NPP 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);
- NPP 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 NPP Operation 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 NPP 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 NPP Operational Management and Engineering Support Department – the Deputy Head of the OEP Technical Support Centre;
- the Head of the NPP 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);

- NPP 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 NPP Security 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 Department);
- the Head of the NPP 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 Fire Supervision and Civil Protection Group);
- the civil protection engineer of the NPP 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 Fire Supervision and Civil Protection Group);
- the Head on Communications – the NPP 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 NPP 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 NPP 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 NPP 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 NPP 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.

The Emergency Technical Service was established in frames of OEP. This service office is temporarily staffed by the personnel of all NPP directions, departments and service offices on the professional basis and will work only if a beyond design-basis accident occurs. There are three brigades in Emergency Technical Service divided into groups and units:

- Brigade of Damage Repair at Nuclear Facilities consists of 56 persons in 5 groups and 12 units,
- Brigade of Emergency Recovery Works consists of 33 persons in 4 groups and 4 units,
- Brigade of I&C Equipment consists of 12 persons in 2 groups and 2 units.

As an example, the structure of the Brigade of Damage Repair at Nuclear Facilities is presented in Figure 6.1-1.

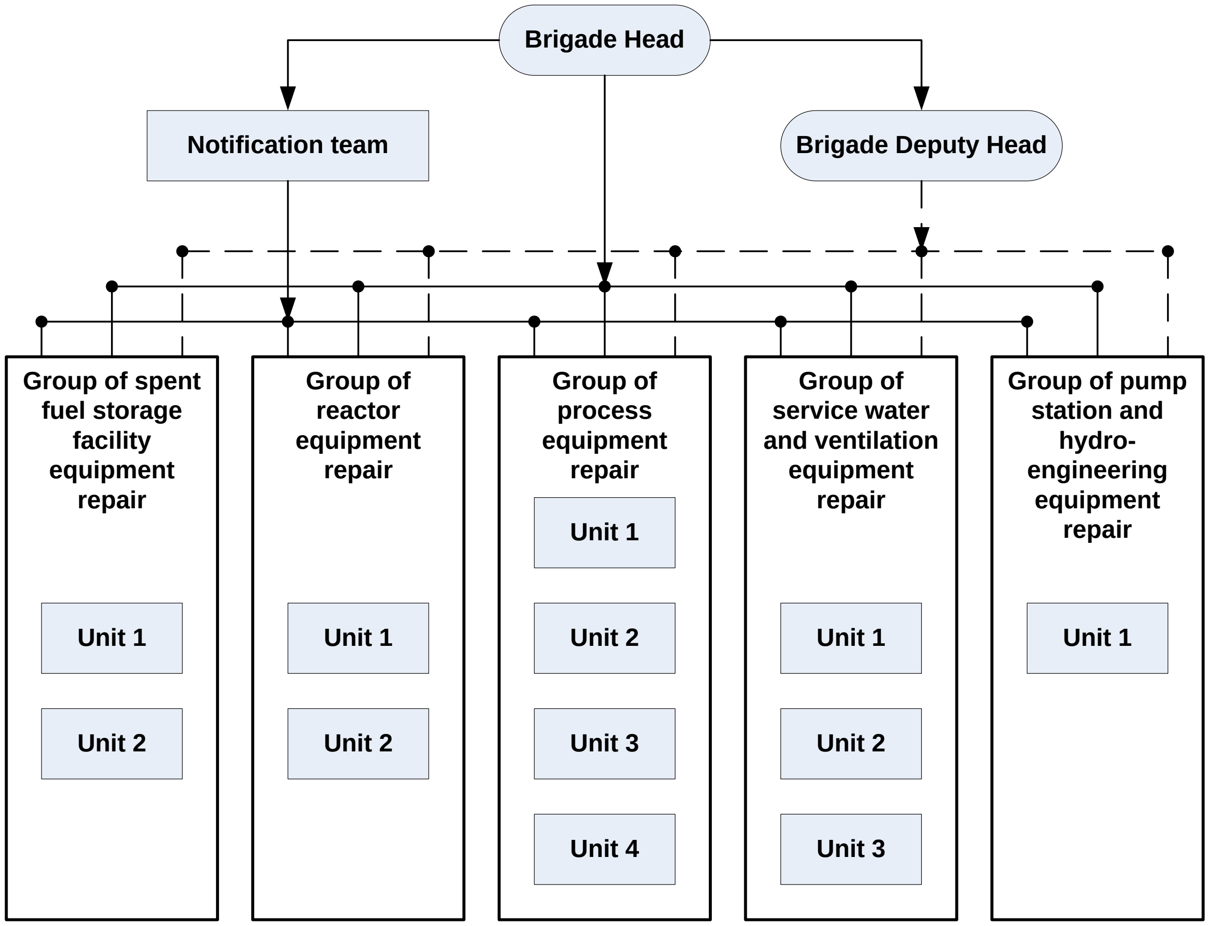


Figure 6.1-1. Structure of the Brigade of Damage Repair at Nuclear Facilities

Accident management activities of EOP 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 accident localization system.

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

- develop and realize the plan of accident liquidation activities and return the Ignalina NPP to the normal operation condition;
- develop and realize the plan of accident 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.

Use of off-site technical support for accident management

A lot of local, territorial and state institutions will provide resource, technical, scientific and human support in the case of the accident at the Ignalina NPP. Interaction of the Organization of Emergency Preparedness with those organizations is based on preliminary contracts/agreements between the Ignalina NPP and the appropriate off-site organizations. Interaction with the state institutions is carried out according to the requirements of the Law on Civil Protection of the Republic of Lithuania.

Dependence on the functions of other reactors on the same site

OEP responsibilities, which cover Unit 1 and Unit 2, are the same for both units.

Procedures

Mitigation of beyond design basis accident consequences is reached by accident control and/or by fulfilment of plans of personnel and population protection if the accident control is impossible or ineffective.

Ignalina NPP five instructions are part of procedures intended to control beyond design basis accidents:

- Instruction for user of procedures to control beyond design basis accidents;
- Manual on control of beyond design basis accidents RUZA-R1. Cooling of Ignalina NPP Unit 2 reactor;
- Manual on control of beyond design basis accidents RUZA-RB. Decreasing of release of fission products from Ignalina NPP Units 1, 2;
- Manual on control of beyond design basis accidents RUZA-B. Control of state of Ignalina NPP Units 1, 2 spent fuel pools;
- Instruction on emergency cooling of Unit 2 reactor under total loss of Ignalina NPP service power supply.

The listed instructions contain a description of 10 strategies to control beyond design basis accidents:

- Strategy C2 – water supply to MCC;
- Strategy C4 – elimination of MCC leakage;
- Strategy C7 – restoration of ALS cooling;

- Strategy C8 – ALS ventilation;
- Strategy C14 – isolation of Unit damaged rooms;
- Strategy C15 – feeding of water via fire cocks;
- Strategy C17 – feeding of water to spent fuel pools;
- Strategy C18 – elimination of spent fuel pool leakage;
- Strategy C19 – supply of neutron absorber into spent fuel pools;
- Strategy C20 – isolation of damaged spent fuel pool from other pools.

Manuals on control of beyond design basis accidents RUZA have the priority against all other procedures and instructions. During execution of RUZA procedures, actions are allowed, which are not allowed during normal operation, such as cut off of protection functions and interlocks, obvious damage of minor equipment, limited release of radioactive products in the environment etc.

Training and exercises

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 training programme for the heads, or the authorized 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 the Organization of Emergency Preparedness Headquarters, as well as the civil protection engineer of the 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 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 Fire Supervision and Civil Protection Group conducts classes with group No 3, which includes the heads of the Ignalina NPP subdivisions, which are not members of the Organization of Emergency Preparedness.

The Heads of the OEP brigades and groups are responsible for development of the training programmes according to the Plan of Emergency Preparedness activities and agreement of these programmes with the Head of the Fire Supervision and Civil Protection Group. The Heads of the units and groups are responsible for organization of training of the subordinated personnel, as well as for preparation and implementation of functional trainings.

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.

Not less than once per three years Ignalina NPP Director General organizes complex training of the Organization of Emergency Preparedness. The photos of the work of the Ignalina NPP Organization of Emergency Preparedness Headquarters and of the Fire-Rescue Team during the last complex training conducted on 24 February, 2011 are presented in Figures 6.1-2 and 6.1-3.

Figure 6.1-2. Work of the Ignalina NPP Organization of Emergency Preparedness Headquarters

Figure 6.1-3. Work of the Fire-Rescue Team

Besides the complex training of the Organization of Emergency Preparedness, which took place at the Ignalina NPP on 24 February, 2011, according to the plan of additional Ignalina NPP safety inspection and analysis dated 30 March, 2011, the Programme of Organization and Implementation of Emergency Training "Decrease of Water Level in Ignalina NPP Unit 2 MCC and SFP" has been developed. The purpose of this training is inspection of the knowledge and skills of the operational personnel to perform work, and the inspection of the skill of interaction in the shift and with the personnel of the Ignalina NPP Organization of Emergency Preparedness at occurrence of beyond design-basis accident, which causes the decrease of a level in MCC and SFP of Ignalina NPP Unit 2, with impossibility of its restoration by regular makeup sources. At present the trainings have begun at the Ignlina NPP according to the annual schedule of the general power plant emergency trainings for the Ignalina NPP operational personnel.

Plans for strengthening the site organisation for accident management

Organization and arrangements of the licensee to manage accidents are adequate. No plans for additional strengthening the site organisation for accident management are needed.

6.1.2. Possibility to use existing equipment

Provisions to use mobile devices

Two mobile diesel generators are available at Ignalina NPP, see 5.1.3 above. Time to bring them on site and put in operation is about one hour.

Provisions for and management of supplies

The minimum operation time of Unit 2 all 6 diesel generators without refueling is at least 5 days (see 5.1.1 above). With the refueling the operation time is not limited. In order to carry out the refueling the fuel supply contract will be concluded.

Management of radioactive releases, provisions to limit them

The Ignalina NPP possesses all the required resources and technical facilities for monitoring and consequences mitigation of radioactive releases caused by beyond design-basis accidents. The resources and technical facilities of other state institutions and departments are not used at the Ignalina NPP.

Ignalina NPP OEP has the monitoring system, which includes:

- the monitoring system of discharges into the ventilation stack;
- the automated radiation safety monitoring system (monitoring of radiation condition inside the power plant);
- the automated radiation monitoring system (monitoring of discharges, drains, radiation condition in the district using the stationary posts, also monitoring of gamma-background in 30 km area), Figure 6.1-4.

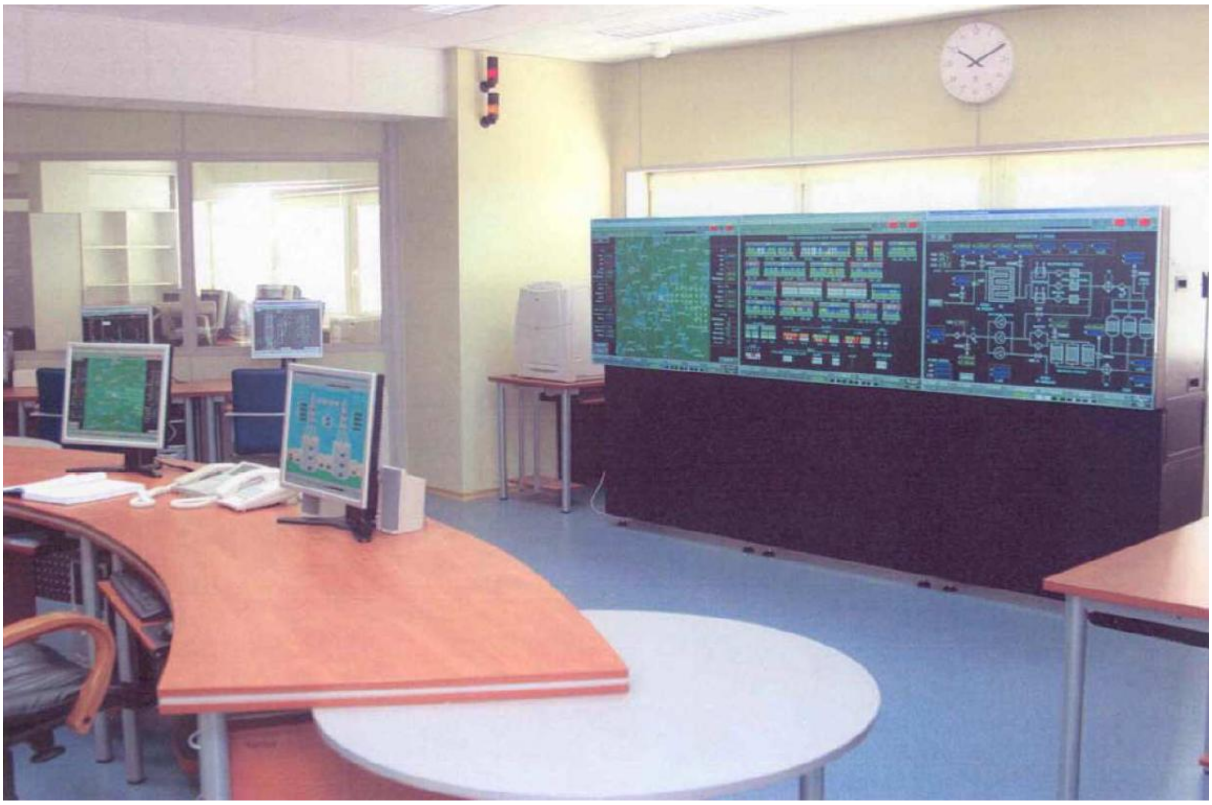


Figure 6.1-4. Radiation safety monitoring room with the automated radiation monitoring system

For assessment of radiation consequences of the accident, the hardware and software of the computer system “NOSTRADAMUS” is used. This system is intended for operative forecasting of the radiation situation caused by the discharge of radioactive materials during the accident. The Ignalina NPP surroundings map is presented in Figure 6.1-5 with the plotted lines of the level of the district radioactive contamination from the radioactive emissions. Figure 6.1-5 was obtained during the OEP exercises of system “NOSTRADAMUS”.

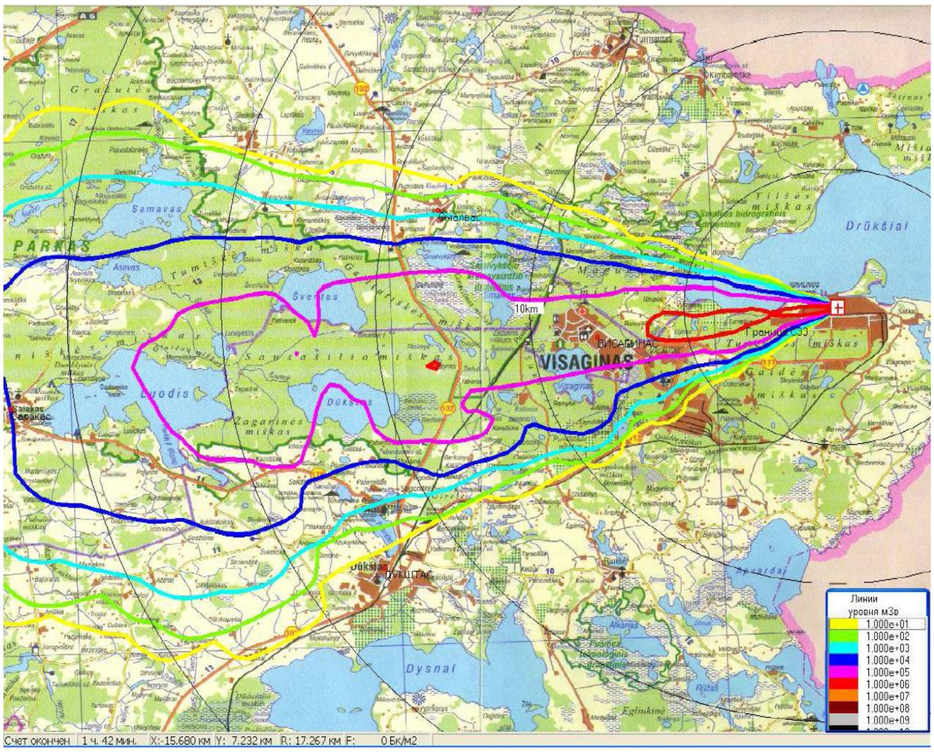


Figure 6.1-5. The Ignalina NPP surroundings map with the plotted lines of the level of the district radioactive contamination caused by the radioactive emission

Communication and information systems

Organization of Emergency Preparedness has the OEP Accident Management Centre in the administrative building equipped with all required facilities for accident management and communication. Besides, there is the special room for the OEP Technical Support Centre, which also has everything required for the work of the experts. Communication facilities ensure the reliable communication between any key points of the NPP such as Main Control Room, Emergency Control Room, Central Electric Control Room, Accident Management Centre, Technical Support Centre, Information Centre, local control points and many others. Communication facilities and computers of the OEP Accident Management Centre can be powered by the independent stationary diesel generator, which is installed in the OEP auxiliary room (see Figure 6.1-6). As well the OEP Accident Management Centre can be powered by the mobile diesel generator using connection point on the wall of administrative building (see 5.1.3 above).

Along with the internal communication, the Main Control Room and OEP Accident Management Centre operators have the possibility to communicate with external institutions such as government, regulator, local municipalities, energy system dispatchers, mass media etc. External communications are provided with few redundant communication lines.

There is the internal announcement system used loud-speakers connected with the Main Control Room and OEP Accident Management Centre. Any external institutions may be provided with all needed information, first of all concerning the radiation situation, from the Main Control Room, OEP Accident Management Centre and Ignalina NPP Information Centre.

Figure 6.1-6. Stationary emergency diesel generator providing the OEP Accident Management Centre with power supply during NPP blackout

6.1.3. Evaluation of factors that may impede accident management and respective contingencies

Some factors may impede accident management; these factors are evaluated below.

Destruction or flooding of infrastructure

Extensive destruction of infrastructure or flooding around the Ignalina NPP might hinder from access to the NPP site. Access to the Ignalina NPP is provided using two motor roads and one railway line. Intensity of earthquakes in the NPP area is not enough to destroy motor road or railway. In any case the destruction of all three access paths is improbable.

Flooding on the Ignalina NPP site is impossible, see 3.1.1 above. Taking into account that NPP outskirts are higher comparing the NPP site level, the flooding of the neighbour infrastructure is impossible too.

Loss of communication facilities

There are diverse communication facilities at Ignalina NPP: stationary telephone, cell phone, speakerphone and radio communication. The independent stationary diesel generator is installed in the OEP Accident Management Centre, which provides power supply for all communication facilities of the Accident Management Centre and recharging of batteries for all mobile communication facilities. Loss of all redundant and diverse communication facilities is improbable.

High dose rates, radioactive contamination

Work conditions may be impaired due to high local dose rates and/or radioactive contamination. In this case the safe conditions and the permissible time of work will be determined on the base of

Lithuanian radiation safety norms and regulations [28], [29], [30]. The real dose rates and levels of radioactive contamination will be measured by the automated radiation safety monitoring system (see 6.1.2 above) and manually using portable instruments.

Destruction of facilities on site

Some facilities on site, which are not seismically qualified, may be destroyed during DBE. In this case ruins and debris will make worse restoration work conditions, particularly accessibility to some buildings or rooms. The comparison of building height and distances between buildings and roads shows that all roads between buildings will be available. An access to some buildings or rooms may be difficult or impossible; this factor will be evaluated during restoration works and the needed measures will be taken.

Accessibility and habitability of the main and emergency control rooms

The main control rooms (MCR) of both Units have three entrances, so the accessibility to MCR will be provided if one or even two entrances are blocked. The Emergency Control Room (ECR) has only one entrance. The importance of ECR is very low now because the main function of the ECR is to shut the reactor down and to provide cooling, but both Unit 1 and Unit 2 are in shutdown cold condition.

The habitability of the MCR and ECR is ensured by existing ventilation system, which is in operation in the part providing MCR and ECR ventilation at both Units. If personnel are forced to leave control rooms, it does not impact on the safety at least for few hours. This time is enough to normalise the situation. No measures shall be taken.

Impact on the different premises

Premises used by the crisis teams for management of the beyond design basis accident are the OEP Accident Management Centre and Technical Support Centre. These premises are located in the administrative building, which possibly is not seismically qualified. So, sufficiently strong earthquake may impact on this building and make difficult or impossible the operation of those centres.

This issue shall be further analysed by the licensee.

Feasibility and effectiveness of accident management measures under the conditions of external hazards

External hazards such as earthquakes or floods may hamper the operation of the OEP Accident Management Centre and Technical Support Centre. Floods are impossible on the Ignalina NPP site, see 3.1.1 above. The possibility of operation of OEP after strong earthquake shall be confirmed or compensatory measures shall be proposed by licensee.

Unavailability of power supply

Unavailability of power supply does not impact on operation of the OEP Accident Management Centre because the Centre can be powered by the independent stationary diesel generator, which is installed in the OEP auxiliary room (see 6.1-2 above). As well the OEP Accident Management

Centre can be powered by the mobile diesel generator using connection point on the wall of administrative building (see 5.1.3 above).

Potential failure of instrumentation

The proper operation of instrumentation is very important to estimate the accident correctly and to take an adequate decision. To reduce the risk of information loss caused by failure of instrumentation, redundant and diverse instrumentation is used at Ignalina NPP. This design feature is enough to protect against single failure and common cause failure.

Potential effects from the other neighbouring installations at site

The most hazardous neighbouring installations at the Ignalina NPP site are the heating and boiling stations needed after shutdown of both reactors. The natural gas is used by these stations as a fuel. Some probability of gas escape and explosion is exists.

It was demonstrated in the Safety Analysis Report of both heating and boiling stations that gas explosion would not influence on the other neighbouring installations at the site.

Availability of trained staff

Ignalina NPP has enough manning level to cope with accidents in any or both Units or in the Spent Fuel Storage Facility. The highly qualified and especially trained personnel are included in the Organization of Emergency Preparedness (see 6.1.1 above). Besides, additional personnel may be involved to deal with extended accidents.

6.1.4. Conclusion on the adequacy of organisational issues for accident management

Organization of Emergency Preparedness (OEP) established by licensee is capable to manage beyond design basis accidents. OEP is structured taking into account the specificity of the Ignalina NPP. OEP is staffed by highly qualified and trained personnel. OEP Accident Management Centre and Technical Support Centre are created and equipped.

Ignalina NPP has developed all needed instructions and procedures intended to control beyond design basis accidents. 10 special strategies to cope with the most probable accidents are developed and included into the instructions.

There are all needed systems, equipment, devices, tools and materials to support the accident management. Communication systems are redundant and diverse. Additional independent diesel generators provide emergency power supply.

Accident Management Centre and Technical Support Centre are located in the administrative building, which is not qualified seismically.

6.1.5. Measures which can be envisaged to enhance accident management capabilities

No additional measures are required concerning the accident management at Ignalina NPP.

The seismic stability of the OEP Accident Management Centre and of the Technical Support Centre shall be analysed and appropriate measures shall be proposed and implemented by the licensee.

6.2. Accident management measures in place at the various stages of a scenario of loss of the core cooling function

This chapter including subchapters below is applicable to the Unit 2 only because the Unit 1 is fully defueled.

6.2.1. Before occurrence of fuel damage in the reactor

During the period of shutdown state of the Unit 2 reactor, part of fuel assemblies were unloaded. As a result the criticality of the reactor is impossible now. The residual heat release in the fuel remaining in the reactor is significantly reduced during the shutdown period.

In case of stop cooling water flow in the reactor channels (as a result of station total blackout) there is at least 18 hours to restore power supply. There are all necessary procedures and instructions. If the power supply restoration is unsuccessful, the cooling water will be provided from independent source – borehole – using borehole pumps powered by independent diesel generator. The corresponding modification is carrying out now and all necessary procedures and instructions are under preparation.

In this situation the shift operational personnel will follow the Instruction on the Provision of Emergency Heat Removal from Unit 2 Reactor in Case of Ignalina NPP Total Blackout [20]. No another management measures are needed.

6.2.2. After occurrence of fuel damage in the reactor

If fuel cooling in the reactor is not restored, the residual heat release will cause overheating of fuel elements and consequent loss of tightness and more serious damage of clad of fuel elements. The radiation level in the reactor hall and some rooms of Unit 2 will increase due to escape of gaseous fission products to the voided circulation circuit.

In this situation the following management measures will be taken:

- Plant Shift Supervisor classifies the NPP state as “Emergency Preparedness”.
- Head of OEP announces the NPP state as “Emergency Preparedness”.
- Gathering of the Technical Support Centre.
- The emergency preparedness plan is fulfilled.
- Shift operational personnel follow “Beyond Design-Basis Accidents Management Guideline RUZA-RB. Reduction of Ignalina NPP Units 1 and 2 Fission Products Emission” [24].
- Shift operational personnel follow “Beyond Design-Basis Accidents Management Guideline RUZA-R1. Provision of Heat Removal from Ignalina NPP Unit 2 Reactor” [25].

There measures are covered by existing instructions and procedures.

6.2.3. After failure of a number of pressure tubes in the reactor

If fuel cooling in the reactor is still not restored, the development of accident will result in failure of a number of fuel channels (pressure tubes) in the Unit 2 reactor. The radiation level will increase up to inadmissible levels.

In this situation the following management measures will be taken:

- OEP and the Technical Support Centre continue to work in accordance with the NPP state “Emergency Preparedness”.
- The emergency preparedness plan is continued to be fulfilled.
- The plan of the accident consequences liquidation and elimination of the long-term negative consequences of RUZA strategies is developed and fulfilling.
- Shift operational personnel follow “Instruction on the Elimination of Emergency Situations at the Ignalina NPP Unit 2” [26].
- Shift operational personnel continue to follow “Beyond Design-Basis Accidents Management Guideline RUZA-RB. Reduction of Ignalina NPP Units 1 and 2 Fission Products Emission” [24].
- Shift operational personnel continue to follow “Beyond Design-Basis Accidents Management Guideline RUZA-R1. Provision of Heat Removal from Ignalina NPP Unit 2 Reactor” [25].

6.3. Maintaining the containment integrity after occurrence of significant fuel damage (up to core meltdown) in the reactor core

Reactors of RBMK type do not have containments in the commonly accepted interpretation. Leak-tight compartments and Accident Localization Tower (ALT) act as the containment providing the functions of localisation and retention of steam and fission products in case of design basis accident (DBA) or beyond DBA. This chapter including subchapters below takes this into account.

6.3.1. Elimination of fuel damage / meltdown in high pressure

Fuel damage or meltdown in high pressure is impossible in the Ignalina NPP reactors because both reactors are shut down and there is no high pressure in fuel channels.

6.3.2. Management of hydrogen risks

Actual source of hydrogen in the current state of Ignalina NPP is water radiolysis in Unit 2 reactor channels.

Hydrogen generated in the reactor channels may be accumulated in drum separators, steam lines and Accident Localization System (ALS). To prevent accumulation of hydrogen, the ventilation of drum separators and steam lines through open air taps and blowing of top part of ALS is performed. The design systems of hydrogen monitoring, concentration reducing and removing are still in operation during all time of reactor defueling.

Thus, hydrogen monitoring and prevention of explosive concentration is provided by design. No additional measures are needed.

6.3.3. Prevention of overpressure of the Leak-tight compartments and Accident Localization Tower

Overpressure is impossible in the Ignalina NPP Leak-tight compartments and Accident Localization Tower because both reactors are shut down and there is no source of high pressure.

6.3.4. Prevention of re-criticality

The reactor of Ignalina NPP Unit 1 was fully defueled in the end of 2009. A part of fuel assemblies are removed from the reactor of Ignalina NPP Unit 2 and only 1335 fuel assemblies are still in reactor. This number of fuel assemblies is not enough to obtain the critical state of the reactor. No additional measures are needed to prevent re-criticality.

6.3.5. Prevention of basemat melt through

Fuel meltdown in high pressure is impossible in the Ignalina NPP reactors because both reactors are shut down and there is no high pressure in fuel channels. No additional measures are needed to prevent the basemat melt through.

6.3.6. Need for and supply of electrical AC and DC power and compressed air to equipment used for protecting integrity of Leak-tight compartments and Accident Localization Tower

Integrity of Leak-tight compartments and Accident Localization Tower is ensured by the robustness of building structures and leak tightness of liners. No special equipment is used for the integrity protection and therefore no supply of electrical power and compressed air is needed.

6.3.7. Measuring and control instrumentation needed for protecting integrity of Leak-tight compartments and Accident Localization Tower

Integrity of Leak-tight compartments and Accident Localization Tower is ensured by the robustness of building structures and leak tightness of liners. No measuring and control instrumentation is needed for the integrity protection.

6.3.8. Capability for severe accident management in case of simultaneous core melt/fuel damage accidents at different units on the same site

Fuel damage would be possible in the Unit 2 reactor only if fuel assemblies were not cooled during a long time. Fuel meltdown is impossible because the reactor is shut down and there is no high pressure in fuel channels. Simultaneous core melt/fuel damage accidents at different units of Ignalina NPP are impossible because the Unit 1 reactor is fully defueled.

6.3.9. Conclusion on the adequacy of severe accident management systems for protection of integrity of Leak-tight compartments and Accident Localization Tower

Integrity of Leak-tight compartments and Accident Localization Tower is ensured by the robustness of building structures and leak tightness of liners. Ignalina NPP design provides the adequate assurance of the integrity.

6.3.10. Measures which can be envisaged to enhance capability to maintain integrity of Leak-tight compartments and Accident Localization Tower after occurrence of severe fuel damage

No additional measures can be envisaged to enhance the capability to maintain integrity of Leak-tight compartments and Accident Localization Tower after occurrence of severe fuel damage.

6.4. Accident management measures to restrict the radioactive releases

6.4.1. Radioactive releases after loss of integrity of Leak-tight compartments and Accident Localization Tower

Design provisions

Accident Localisation System was designed to cope with design basis accidents of operation unit. After permanent shutdown of units this system remains in operation.

Operational provisions

Accident Localisation System will remain in operation time defined by decommissioning plans. Moreover in case of beyond design basis accident the management of radioactive releases will be performed according to "Manual on control of beyond design basis accidents RUZA-RB. Decreasing of release of fission products from Ignalina NPP Units 1, 2".

6.4.2. Accident management after uncovering of the top of fuel in the fuel pool

Hydrogen management

Hydrogen generated in spent fuel pools is removed by ventilation system. The concentration of hydrogen is very low and deficient to form an explosive mixture even if ventilation system is out of operation for a long time. No special management measures are needed.

Providing adequate shielding against radiation

Adequate shielding against radiation is provided by water covering fuel assemblies in the spent fuel pools (SFP). At total blackout of the power plant the SFP primary and alternate ultimate heat sinks will be lost; this will inevitably lead to the gradual growth of the water temperature in pools. Calculation of the temperature regime of water in the Unit 1 SFP is provided in Appendix C. Calculation of the temperature regime of water in the Unit 2 SFP is provided in Appendix D. In the most heat-stressed SFPs, 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. Further the evaporation of water and decrease of a water level in SFP will occur. The critical low level of water in the Unit 2 spent fuel pools corresponding of top of the fuel in assemblies will be reached after 40 days and of top of the fuel in transport 102-places covers after 15 days. This time is enough for restoration of normal water regime of SFP or at least to deliver water using fire-engine. Nevertheless, if all actions to restore the SFP water regime are unsuccessful, the radiation level in the SFP hall will increase.

A plan and program of shielding against radiation will be developed and implemented at least few days beforehand depending on actual situation. The level of radiation is registered by the automated radiation safety monitoring system, which monitors the radiation condition inside the Ignalina NPP, see 6.1.2 above.

Restricting releases after severe damage of spent fuel in the fuel storage pools

If the water level in SFP continues to decrease, the residual heat release will cause overheating of fuel elements and consequent loss of tightness and more serious damage of clad of fuel elements; as a result gaseous fission products will go out to air of the SFP hall.

A plan and program of restricting releases after severe damage of spent fuel in the fuel storage pools will be developed and implemented at least few days beforehand depending on actual situation.

Instrumentation needed to monitor the spent fuel state and to manage the accident

Each spent fuel pool is equipped with water temperature and level instruments. Data obtained by instruments are transferred to the ICS “TITAN” and to the MCR. The data are available in the Accident Management Centre and Technical Support Centre via the Ignalina NPP computer network.

In case of loss of off-site power:

- Power supply of instruments of water temperature and level in the Unit 1 storage pools will be lost. Now the power supply of this instrumentation is re-designed to provide power from DG-7 of Unit 2. December 2011 is set as the term to implement the new design.

- Power supply of instruments of water temperature and level in the Unit 2 storage pools will automatically switch-over to one of 6 diesel generators.

In case of Ignalina NPP total blackout (off-site power supply and all diesel generators are lost), to ensure power supply of temperature and level instrumentation of Unit 1 and Unit 2 spent fuel pools it is necessary to implement new design of backup power supply from mobile diesel generator and to include addenda to corresponding procedures. December 2011 is set as the term to implement this design.

Availability and habitability of the control room

The main control rooms (MCR) of both Units have three entrances, so the availability of MCR will be provided if one or even two entrances are blocked. The habitability of the MCR is ensured by existing ventilation system, which is in operation at both Units. If personnel are forced to leave control rooms, it does not impact on the safety at least for few hours. This time is enough to normalise the situation.

6.4.3. Conclusion on the adequacy of measures to restrict the radioactive releases

Measures to restrict the radioactive releases are adequate.

7. General conclusion

7.1. Key provisions enhancing robustness (already implemented)

7.1.1. Earthquake

Reactor building structures, systems and components that ensure the safety of fuel storage in the Unit 2 reactor and in pools of both Units, as well as Spent Fuel Storage Facilities are capable to withstand the design basis earthquake (DBE) taking into account possible failures of supporting systems for the time period sufficient for repair works.

The presence and operation of the Seismic Alarm and Monitoring System at Ignalina NPP should be appreciated as a good practice.

7.1.2. Flooding

In the worst case of increase of Lake Drūkšiai level, it does not culminate the level of Ignalina NPP safety related buildings and structures. Ignalina NPP design ensures the adequate protection against an external flooding. The flooding is impossible; the level margin is at least 1.4 m. The levels of hydro-engineering structures were rechecked and documented in the period since 16 September till 17 October 2011.

7.1.3. Extreme weather conditions

Ignalina NPP Design basis conditions correspond to the extreme weather conditions possible in the area of the Ignalina NPP site.

7.1.4. Loss of electrical power and loss of the ultimate heat sink

The time needed for restoration of NPP power supply after possible total shutdown of the Lithuanian energy system is approximately 30 minutes. Existing diesel generators are capable to provide backup power supply of remaining systems important to safety at Ignalina NPP for the needed time much more than 30 minutes.

Existing batteries are capable to provide diverse backup power supply of vitally important systems at Ignalina NPP for the needed time.

Mobile diesel generators provide additional diversity of backup power supply.

Water feeding of the Unit 2 reactor and of spent fuel pools at both Units is carried out using sufficient redundancy of feed sources.

As a good practice, the possibility to use the domestic potable water system should be noted. The domestic potable water system has independent pumps with own diesel generator.

Unloading of 350 fuel assemblies from Unit 2 reactor and shutdown state of the reactor during long time period significantly reduced the risk of fuel damage in the reactor and pools in case of loss of cooling.

If the electrical power supply and ultimate heat sink is lost, Ignalina NPP staff has enough time and necessary means to prevent cliff edge effects.

7.1.5. Severe accident management

Organization of Emergency Preparedness (OEP) established by licensee is capable to manage beyond design basis accidents. OEP is staffed by highly qualified and trained personnel to cope with accidents in any or both Units or in the Spent Fuel Storage Facility. OEP Accident Management Centre and Technical Support Centre are created and equipped with all required facilities for accident management and communication.

Ignalina NPP has developed Emergency Preparedness Plan and Emergency Preparedness Operational Procedures, all needed instructions and manuals to control beyond design basis accidents. 10 special strategies to cope with the most probable accidents are developed and included into the instructions.

There are all needed systems, equipment, devices, tools and materials to support the accident management. Communication systems are redundant and diverse. Additional independent diesel generators provide emergency power supply.

Ignalina NPP has design and process documentation, prefabricated and marked pipe sections, facilities, tools to implement modifications for control of beyond design basis accidents.

Organization and arrangements of the licensee to manage accidents are adequate. Measures to restrict the radioactive releases are adequate.

7.2. Safety issues

7.2.1. Earthquake

Some beyond design basis earthquake scenarios are not analysed, namely turnover of the cask with spent fuel during transportation from Units to the storage site and postulated loss of cask sealing, collapse of walls of cask storage hall or guarding concrete fence and cask blockage by debris, collapse of walls of hot cell when there is spent fuel in the hot cell.

Accident Management Centre and Technical Support Centre are located in the administrative building, which is not qualified seismically.

7.2.2. Flooding

There are no safety issues concerning flooding.

7.2.3. Extreme weather conditions

There are no safety issues concerning extreme weather conditions.

7.2.4. Loss of electrical power and loss of the ultimate heat sink

Supply of fuel to ensure refuelling of diesel generators during operation over a long period of time is not contracted.

The backup power supply of temperature and level instrumentation of spent fuel pools from mobile diesel generator is designed but the design is not implemented.

7.2.5. Severe accident management

There are no safety issues concerning the severe accident management.

7.3. Potential safety improvements and further work forecasted

Ignalina NPP prepared plan of implementation of measures for safety improvement which was presented to State Nuclear Power Safety Inspectorate for approval. The plan foreseen implementation of measures, linked to the issues listed in following clauses.

7.3.1. Earthquake

It is recommended to perform the following beyond design basis earthquake analysis for the New Spent Fuel Interim Storage:

- Turnover of the cask with spent fuel during transportation from Units to the storage site and postulated loss of cask sealing;
- Cracks or collapse of walls of cask storage hall and cask blockage by debris;
- Cracks or collapse of walls of hot cell when there is spent fuel in the hot cell.

It is recommended to perform beyond design basis earthquake analysis of cracks or collapse of guarding concrete fence of the Spent Fuel Storage and cask blockage by debris.

It is recommended to analyse a possibility to use signals of Seismic Alarm and Monitoring System for formalization of Emergency Preparedness criterion and subsequent including of this criterion to the Instruction of Accident Classification at Ignalina NPP.

It is recommended to perform design basis earthquake analysis of the OEP Accident Management Centre and of the Technical Support Centre and to propose and implement appropriate measures.

All the mentioned improvements should be carried out till the end of December, 2012.

7.3.2. Flooding

No additional measures are needed to increase robustness of the plant against flooding.

7.3.3. Extreme weather conditions

No measures required which could be envisaged to increase plant robustness against extreme weather conditions and would enhance plant safety.

7.3.4. Loss of electrical power and loss of the ultimate heat sink

Contract for supply of fuel shall be negotiated till the end of June, 2012 to ensure refuelling of diesel generators during operation over a long period of time.

To ensure power supply of temperature and level instrumentation of spent fuel pools it is necessary to implement new design of backup power supply from mobile diesel generator and to include addenda to corresponding procedures till the end of December, 2011.

No additional measures are required to increase robustness of Ignalina NPP in case of loss of ultimate heat sink.

7.3.5. Severe accident management

No plans for additional strengthening the organisation for accident management are needed.

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Appendix A. 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 and partial defueling of the reactor is carried out. As of 1 July 2011 312 FAs were unloaded and currently there are 1335 SFAs in the reactor. In March 2012 it is planned to complete unloading of 500 FAs.

Figure A-1 represents the distribution of SFAs 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 %.

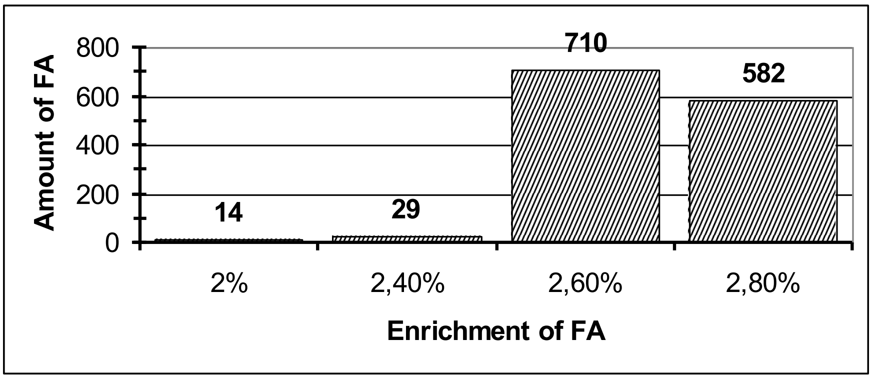


Figure A-1. Distribution of SFAs in the reactor according to the initial enrichment

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 A-2 indicates the distribution of SFAs located in the reactor according to the energy production.

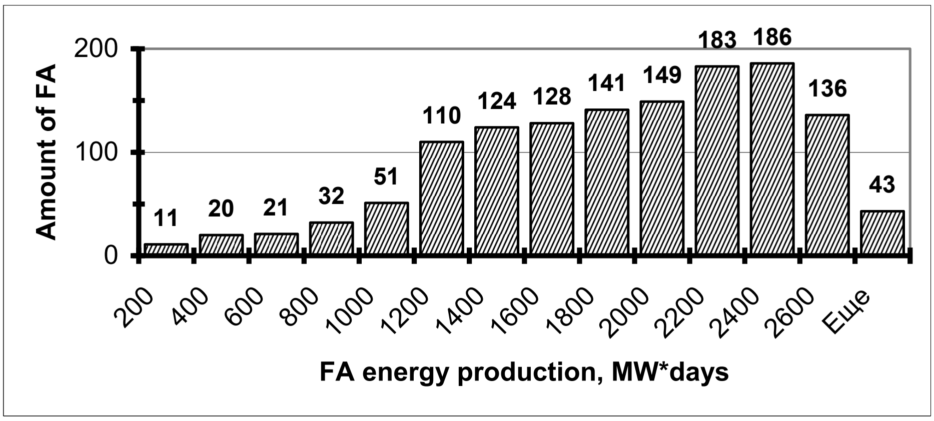


Figure A-2. Distribution of the SFA in the Reactor according to the Energy Production

It is obvious from the distribution that there are the SFAs 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 for protective casks of the SFSF. 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 is:

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

Appendix B. Calculation of the Unit 2 Reactor Heating-up Process with 1335 SFAs in the Reactor Core

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

To provide calculation conservatism 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 circulation circuit is out of operation 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 space 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 fuel elements, 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 fuel elements, water and graphite cladding.

As of July 2011 taking into account that there were 1335 SFAs 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 heating-up time of the graphite cladding are presented in Table B-1.

Table B-1. Data for calculation of heating-up time of the graphite cladding

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

Description	Notation	Value
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:

$$V_{gr} = N_{gr} \cdot H_{r.c.} \cdot (l^2 - \pi \cdot d_{ext}^2 / 4) \quad \text{(B.1.)}$$

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

Mass of graphite:

$$M_{gr} = V_{gr} \cdot \gamma_{gr} \quad \text{(B.2.)}$$

$$M_{gr} = 749 \cdot 1700 = 1.273 \cdot 10^6 \text{ kg};$$

Heat accumulated by the graphite cladding:

$$Q_{gr} = M_{gr} \cdot C_{gr} \cdot (t_{final} - t_{initial}) \quad \text{(B.3.)}$$

$$Q_{gr} = 1.273 \cdot 10^6 \cdot 0.982 \cdot (100 - 45) = 68.75 \cdot 10^6 \text{ kJ}$$

Volume of zirconium in the FC and CPS channels:

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

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

Mass of zirconium in the reactor core:

$$M_{zr} = V_{zr} \cdot \gamma_{zr} \quad \text{(B.5.)}$$

$$M_{zr} = 13.6 \cdot 6500 = 8.84 \cdot 10^4 \text{ kg};$$

Heat accumulated by zirconium in the FC and CPS channels:

$$Q_{zr} = M_{zr} \cdot C_{zr} \cdot (t_{final} - t_{initial}) \quad \text{(B.6.)}$$

$$Q_{zr} = 8.84{\cdot}10^4 \cdot 0.293 \cdot (100 - 45) = 1.42{\cdot}10^6 \text{ kJ}$$

Volume of water in the FC within the reactor core:

$$V_w = H_{r.c.} \cdot \pi \cdot (N_{FC} \cdot d_{int}^2 - N_{FA} \cdot (18 \cdot d_{FE}^2 + d_{ct}^2)) / 4 \quad \text{(B.7.)}$$

$$V_w = 7 \cdot 3.1416 \cdot (1661 \cdot 0.080^2 - 1335 \cdot (18 \cdot 0.0136^2 + 0.0150^2)) / 4 = 32.36 \text{ m}^3;$$

Mass of water within the reactor core:

$$M_w = V_w \cdot \gamma_w \quad \text{(B.8.)}$$

$$M_w = 32.36 \cdot 980 = 3.17{\cdot}10^4 \text{ kg};$$

Heat accumulated by water in the FC:

$$Q_w = M_w \cdot C_w \cdot (t_{final} - t_{initial}) \quad \text{(B.9.)}$$

$$Q_w = 3.17{\cdot}10^4 \cdot 4.186 \cdot (100 - 45) = 7.3{\cdot}10^6 \text{ kJ}$$

Heat accumulated by fuel:

$$Q_f = N_{FA} \cdot M_f \cdot C_f \cdot (t_{final} - t_{initial}) \quad \text{(B.10.)}$$

$$Q_w = 1335 \cdot 125 \cdot 0.287 \cdot (100 - 45) = 2.63{\cdot}10^6 \text{ kJ}$$

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

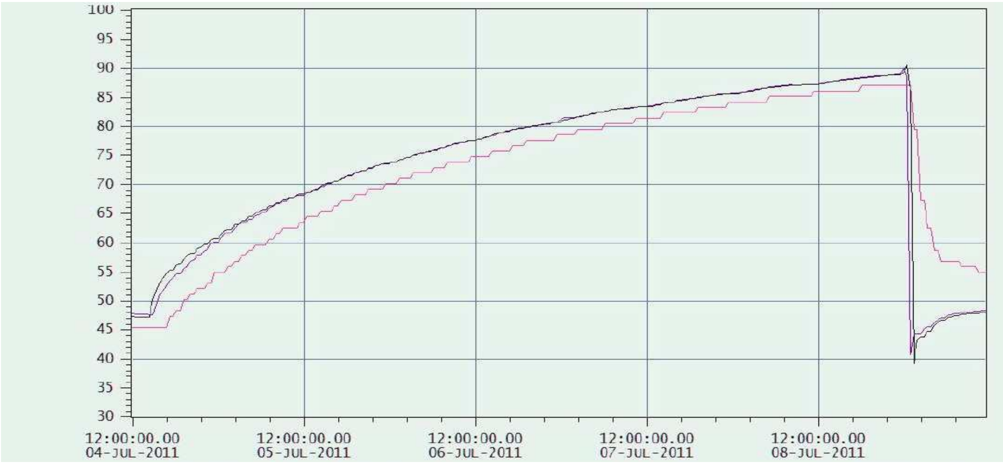
$$t = (Q_{gr} + Q_{zr} + Q_w + Q_f) / W_{dec} \quad \text{(B.11.)}$$

$$t = (68.75{\cdot}10^6 + 1.42{\cdot}10^6 + 7.3{\cdot}10^6 + 2.63{\cdot}10^6) / 443 = 1.80{\cdot}10^5 \text{ s} = 50.2 \text{ hours} = 2 \text{ days.}$$

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 reactor into the environment nor heat removal from the reactor by forced ventilation of the reactor hall and rooms of the DS and SWP were taken into consideration. Restoration of natural circulation of the coolant will allow effectively and quickly reducing 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 broken natural circulation mode for SFA unloading can serve as an indirect confirmation of undue conservatism of the accepted assumptions (see Figure B-1). In this case the drum separators are emptied, the level in the reactor core is controlled according to the MCT level gauges and repair level gauges connected to the ECCS headers.



2YH16T01	MCC TEMPERATURE (LEFT HALF)	GOOD 48.02 DEGREE
2YH26T01	MCC TEMPERATURE (RIGHT HALF)	GOOD 48.27 DEGREE
2ZN00T01.300	MAXIMUM MEASURED TEMPERATURE OF GRAPHITE IN FC	GOOD 54.9 DEGREE

Figure B-1. Graphite Cladding Heating-up in case of the reactor cooling transfer from the Natural Circulation Mode to the Broken Natural Circulation Mode

Over 4.5 days since 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 1332 to 1317 pieces of FA.

Considering that during the defueling of Unit 2 reactor the amount of FAs 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 operational regulations 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.

Appendix C. Calculation of the Temperature Regime of Water in the Unit 1 Spent Fuel Pools

Calculation of the temperature regime of water in Unit 1 SFP was performed after the total 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, Ignalina NPP report [21].

The calculations have shown that as of 1 January 2010 if there is no heat removal by pump-cooling plant, the rate of temperature rise of water in the most heat-stressed compartment (236/2) of Unit 1 spent fuel pools 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 since May till July 2011 the increase of water temperature in the compartment 236/2 was 5°C (increase from 33 up to 38°C).

Appendix D. Calculation of the Temperature Regime of Water in the Unit 2 Spent Fuel Pools

Under normal operation conditions the temperature of water in spent fuel pools is constantly controlled and maintained below 50^{o}C by the pump-cooling plant. The time permissible for out-of-operation condition of pump-cooling plant is determined in view of non-boiling water in the storage pools compartments.

The calculation of the decay heat of SFAs in each compartment of spent fuel pools has been carried out as of 1 July 2011. The calculation of the decay heat of the cut SFAs loaded into 32M baskets was performed using the system of accounting of radioactive materials in protection casks of the SFSF for each 32M basket and then was summarized according to all 32M baskets placed in the compartment. The system of accounting of radioactive materials in the protection casks of the SFSF calculates the decay heat only for the SFA loaded into the 32M basket. Therefore, the decay heat of non-cut SFAs 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 SFSF [22]. The calculation was carried out by the method of interpolation. Interpolation 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:

$$y_i=y_0(z_i)+A_1(z_i)\times\exp(-(x-x_0(z_i))/t_1(z_i))+A_2(z_i)\times\exp(-(x-x_0(z_i))/t_2(z_i)) \qquad \text{(D.1.)}$$

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:

$$\begin{aligned} y = {} & \frac{(z-z_2)\times(z-z_3)\times(z-z_4)\times(z-z_5)}{(z_1-z_2)\times(z_1-z_3)\times(z_1-z_4)\times(z_1-z_5)}\times y_1 + \frac{(z-z_1)\times(z-z_3)\times(z-z_4)\times(z-z_5)}{(z_2-z_1)\times(z_2-z_3)\times(z_2-z_4)\times(z_2-z_5)}\times y_2 + \\ & + \frac{(z-z_1)\times(z-z_2)\times(z-z_4)\times(z-z_5)}{(z_3-z_1)\times(z_3-z_2)\times(z_3-z_4)\times(z_3-z_5)}\times y_3 + \frac{(z-z_1)\times(z-z_2)\times(z-z_3)\times(z-z_5)}{(z_4-z_1)\times(z_4-z_2)\times(z_4-z_3)\times(z_4-z_5)}\times y_4 + \\ & + \frac{(z-z_1)\times(z-z_2)\times(z-z_3)\times(z-z_4)}{(z_5-z_1)\times(z_5-z_2)\times(z_5-z_3)\times(z_5-z_4)}\times y_5 \end{aligned} \qquad \text{(D.2.)}$$

The results of the calculations are presented in Tables D-1 and D-2.

Table D-1. 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 D-2. 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
Subtotal 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
Subtotal in 236/2	**933**	**196.5**
Total	**1638**	**371.6**

It can be seen from the provided results that the most temperature stressed SFP compartments are 236/2 (196.5 kW) and 336 (147.3 kW). The out-of-operation permissible time for pump-cooling plant was calculated for these compartments.

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

In view of relatively low temperature of water in the storage pools, massiveness of building structures, and consequently, long time for heating-up of the building structures, the accumulating capacity of them is not taken into account that results in a calculation margin.

Initial data for calculation are presented in Table D-3. 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 D-3. Initial data for calculation of temperature in Unit 2 SFP

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 bottom plate to the water surface:

$$V_{comprt} = a \times b \times h \tag{D.3.}$$

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

Mass of metal of the walls liner from the bottom plate to the water surface in the compartment:

$$M_{steel} = V_{steel} \times \gamma_{steel} = 2(a+b) \times h \times \delta_P \times \gamma_{steel} \tag{D.4.}$$

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

Mass of metal of the bottom plate of the compartment:

$$M_{tray} = V_{tray} \times \gamma_{steel} = a \times b \times \delta_D \times \gamma_{steel} \tag{D.5.}$$

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

Quantity of SFA in the compartment:

$$n(236/2)=n_1+n_2 \tag{D.6.}$$

$$n(336)=n_{cut} \tag{D.7.}$$

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

Mass of cartridges (compartment 236/2):

$$M_c=m_c \times n_1 \tag{D.8.}$$

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

Mass of 32M baskets (compartment 336):

$$M_{32Mb}=m_b \times n_b \tag{D.9.}$$

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

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

$$M_{FA}=m_2 \times n \tag{D.10.}$$

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

Mass of the SFA without suspensions (compartment 336):

$$M^{Cut}_{FA}=m_1 \times n \tag{D.11.}$$

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

Mass of fuel in the compartment:

$$M_F=m_F \times n \tag{D.12.}$$

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

Mass of the SFA metal in the compartment:

$$M^M_{FA}(236/2)= M_{FA} - M_F \tag{D.13.}$$

$$M^M_{FA}(336)= M^{Cut}{}_{FA} - M_F \tag{D.14.}$$

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

Volume of metal in the compartment:

$$V_M(236/2) = \frac{M_c + M^M_{FA}}{\gamma_{steel}} \tag{D.15.}$$

$$V_M(336) = \frac{M_{32Mb} + M^M_{FA}}{\gamma_{steel}} \tag{D.16.}$$

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

Volume of fuel in the compartment:

$$V_F = \frac{M_F}{\gamma_F} \tag{D.17.}$$

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

Volume of water in the compartment:

$$V_W = V_{comprt} - (V_M + V_F) \tag{D.18.}$$

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

Temperature increment in the compartment:

$$\Delta t = t_{final} - t_{initial} \tag{D.19.}$$

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:

$$Q_W = V_W \times \gamma_W \times C_W \times \Delta t \tag{D.20.}$$

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

Heat accumulated by metal of the compartment:

$$Q_M(236/2) = M_M \times C_M \times \Delta t = \left(M_{steel} + M_{tray} + M_c + M_{FA}^M\right) \times C_M \times \Delta t \tag{D.21.}$$

$$Q_M(336) = M_M \times C_M \times \Delta t = \left(M_{steel} + M_{tray} + M_{32Mb} + M_{FA}^M\right) \times C_M \times \Delta t \tag{D.22.}$$

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

Heat accumulated by fuel of the compartment:

$$Q_F = M_F \cdot C_F \cdot \Delta t \tag{D.23.}$$

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

Time needed for the water temperature rise in the SFP compartment from the initial temperature (50°C) up to the temperature close to boiling (95°C) if there is no heat removal by the pump-cooling plant and there is no heat transfer to the water of the other compartments:

$$T = \frac{Q_W + Q_M + Q_F}{W} \tag{D.24.}$$

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

The results of the calculation show that if there is no heat removal by the pump-cooling plant, the rate of rise of water temperature in the compartment 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 (95°C) is approximately 7 days. This time is quite sufficient to recover the power supply or to eliminate the malfunction of the pump-cooling plant and to recover its functionality.