AES-2006 PSA LEVEL 1.
PRELIMINARY RESULTS AT PSAR STAGE

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Abstract: This report represents PSA level 1 results for AES-2006 project at LAES - 2 first unit configuration on PSAR stage. Report contains short description and base composition of LAES - 2 project, composition and basic requirements of normative documents regulating process of PSA level 1 implementing, list of operating conditions considered in PSA, list of initiating events selected for analysis, brief description of most significant accident sequences leading to fuel damage, characteristics of input data used, results for fuel damage frequency assessment.

Keywords: AES-2006, PSA level 1 results, Leningrad NPP – 2, Baltic NPP.

1. AES-2006 PROJECT SHORT DESCRIPTION

AES-2006 with VVER -1200 reactor is the base project for construction of Leningrad NPP- 2 (LAES -2), Baltic NPP and the Belarusian NPP. Currently, AES - 2006 project is being finalized to meet STUK requirements for NPP «Hanhikivi» construction in Finland. Development of LAES - 2 project was carried out using experience of design, construction and industrial operation for Tianwan NPP. Currently in commercial operation since 2007, are two blocks of Tianwan NPP.

AES-2006 project at Baltic NPP configuration passed Volume 2 EUR [1] requirements conformity assessment by WorleyParsons Nuclear Services JSC. Results of conformity assessment presented at Table 1.

Table 1: AES-2006 project Volume 2 EUR requirements conformity assessment results

Assessment category	**Amount**	**Statistic**
Conformity	3424	87,6%
Conformity with requirement objectives	220	5,6%
Nonconformity	42	1,1%
Not applicable	176	4,5%
Not considered	44	1,2%
Total	3906	100%

LAES -2 project utilize following approaches:
- Maximum use of solutions and studies for already developed projects with VVER type reactors;
- Risk minimizing and system performance improving through usage of proven technical solutions and reference equipment;
- Improving systems and equipment performance by optimizing design margins;
- Providing required level of safety, including beyond design basis accidents, based on the choice of a rational configuration of safety systems with the combination of active and passive elements to implement the principles of diversity and functional redundancy, as well as to reduce the human impact on the basis of the principle of reasonable sufficiency;
- Operating and capital costs reducing;
- NPP`s cost and construction time reducing through usage of existing groundwork for documentation.

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LAES -2 project safety concept based on:
- Defense-in-depth principle;
- Deterministic safety principles;
- Target probabilistic criteria;
- Radiation safety criteria.

Following target probabilistic criteria are established for the project [2, 3]:
- Total frequency of severe fuel damage for all accident sequence must not exceed 10^{-6} 1/reactor per year;
- In order to avoid evacuation of population outside of emergency measures planning zone, determined in accordance with the regulatory requirements for NPP placement, design should strive to ensure, that estimated value for limited accidental release probability, does not exceed 10^{-7} 1/reactor per year.

Following safety barriers are established at project:
- 1st barrier: fuel matrix, preventing release of fission products to fuel cladding;
- 2nd barrier: fuel cladding, preventing release of fission products to main circulation path;
- 3rd barrier: main circulation path, preventing release of fission products to containment;
- 4th barrier: containment, preventing release of fission products to environment.

Communication between plant operational states, goals and levels of defense-in-depth, as well as states of security barriers, associated with failures of defense levels, presented in Table 2.

Table 2: Communication between plant operational states, goals and levels of defense-in-depth, and states of security barriers associated with failure of defense levels

Defense-in-depth levels	**Level 1**	**Level 2**	**Level 3**	**Level 4**	**Level 5**
Defense-in-depth goals	Terms of NPP location and prevention of normal operation violations	Preventing design accidents by normal operation systems	Preventing beyond design basis accidents by safety systems	Beyond design basis accidents management	Preparation and implementation of emergency measures plans at site and beyond
Plant operational states	Normal operation	Normal operation violations	Design basis accident	Beyond design basis accidents	Severe post-accident situation
Strategy	Accident prevention	Accident prevention	Accident mitigating	Accident mitigating	Accident mitigating
Management	Normal operation control system	Normal operation control system, including limiting function	CSS, including RTS & ESFAS	Normal operation control system, CSS, BDBA management system	Emergency management
Procedures	Instructions and guidance for normal operation	Technological regulations for safe operation Normal operation violations management procedures	DBA management procedures	Symptom-based emergency procedures	Emergency measures plans
Reaction	Normal operation systems	Normal operation systems	Engineering protective measures	Special design measures	External emergency preparation

Defense-in-depth levels	Level 1	Level 2	Level 3	Level 4	Level 5
Security barriers state, associated with failure of defense level	Fuel element damage within limits of radiation safety, operability of safety barriers 3, 4	Fuel element damage within limits of radiation safety, impact on safety barriers limitation	Limited 2, 3 safety barriers damage. Impact on containment limitation	1, 2, 3 safety barriers damage. Functionality of containment	1, 2, 3, 4 safety barriers damage

2. COMPARISON WITH REFFERENCE PROJECT

Comparison of main characteristics and parameters for Tianwan NPP with reactor plant V-428 and LAES-2 NPP with reactor plant V - 491 (see Table 3) shows that the - reactor plant V - 491 has some advantages, in particular:

- Tanks of borated water storage system were moved out of the safety building to containment and their functions were combined with those of pit-tanks. This allowed to simplify the water flow scheme to reactor from the ECCS systems, solve problem of pit - tanks valves failure-to-open during LPIS and HPIS pumps transition to recycling, and realize water return scheme from leak or because of a sprinkler system work, back to the pit- tanks.
- Feed and boron regulation system is able to perform the functions of emergency boron injection during ATWS accidents and function of 1 circuit feeding at shutdown modes and "small" leaks of 1 circuit;
- Cooling for responsible consumers process water supply system, using spray ponds;
- Refusal of EDG water cooling in favor of air cooling;
- Presence of passive safety systems;
- Service life for equipment has been increased from 40 to 60 years.

Table 3: Main differences between NPP projects with V-428 and V-491 reactor plants

Design data	Reactor plant V-428	Reactor plant V-491
Service life of main equipment, years	40	60
Pressure in the reactor (nominal) at core outlet, MPa	15,7	16,2±0,3
Coolant temperature at core outlet, °C	321	328,9±5
Coolant temperature at core inlet, °C	291	298,2
The temperature difference (heating) in the reactor, °C	30	30,7
Coolant flow rate through reactor, m^3/hour	86000	86000
Reactor internal diameter (cylindrical portion), mm	4150	4150
Thickness of the reactor wall, mm	192,5	197,5
Thermal power (nominal), MW	3000	3200
Time spent (campaign) fuel in the core, year	3 – 4	4
Average fuel burnup (stationary fuel cycle), MW × day/kg U	43	До 70
Operating time at full capacity during the year (effective), hour	7000	8400
2nd circuit design excessive pressure, MPa	7,84	8,1
Steam generators type	PGV -1000M horizontal	PGV-1000MKP horizontal
Steam capacity, ton/hour	1470	1602
Generated steam pressure at SG steam collector outlet (at rated load) MPa	6,27	7,00
Generated steam temperature (at rated load), °C	278,5	287,0±1,0
Feedwater temperature (at rated load), °C	218	225±5
Coolant flow rate through loop, m^3/hr	21500	21200
Pressure in pressurizer, MPa	15,7	16,2
Turbine type	K-1000-60/3000	K-1200-6,8/50

Design data	Reactor plant V-428	Reactor plant V-491
Turbine power, MW	1060	1170
Containment inner shell height (from inside), m	~63,0	67,85
Containment inner shell free volume, m^3	69169,0	75000
Containment outer shell height (from inside), m	~66.0	77
Containment outer shell vertical concrete wall thickness , mm	600,0	800,0
Residual heat removal scheme	RHRS + LPIS+ Sprinkler system	RHRS+ LPIS
Borated water storage tanks accommodation	Safety building	Under containment
Passive heat removal system through steam generators for beyond design basis accidents management	-	+
Passive heat removal system from containment for beyond design basis accidents management	-	+
Quantity and power of EDG, pcs. × kW	4×5500	4×6300
Quantity and power of normal operation reliable power supply DG, pcs. × kW	2×5000	1×6300

3. PSA IMPLEMENTATION

In accordance with specification requirements for LAES-2 PSA level 1 [2, 3], reactor core, fuel assemblies in the spent fuel pool and assemblies undergoing handling operations, are regarded as a source of radioactivity.

Following regulatory requirements were used during analysis:
- Regulations on main recommendations for development of PSA level 1 for internal initiating events for all NPP operational modes [4];
- Key recommendations for NPP PSA implementation [5];
- Procedures for Conducting PSA of NPP (Level 1) [6].

In the course of PSA following groups of plant operational states were considered:
- Power operation;
- "Hot" state during power shutdown, for planned and unplanned shutdowns;
- Cooldown through 2nd circuit within 1st circuit temperature range of 255 - 135 ° C with disabled ECCS accumulator tanks, for planned and unplanned shutdowns;
- Cooldown through 1st circuit within 1st circuit temperature range of 135 ° C to 60 ° C, for planned and unplanned shutdowns;
- «Cold» state during cooldown, for planned and unplanned shutdowns. Unscheduled repair of equipment;
- Preparation for reactor disassembly, reactor disassembly, for planned and unplanned shutdowns. Scheduled and unscheduled repair of equipment;
- Refueling. Scheduled maintenance of equipment;
- Revision of RP equipment. Scheduled maintenance of equipment;
- Reactor assembly at planned and unplanned shutdowns. Scheduled maintenance of equipment;
- «Cold» state during unit startup, for planned and unplanned shutdowns. Scheduled maintenance of equipment;
- Warming up before RHRS pumps shutdown, for planned and unplanned shutdowns;
- Hydraulic tests for 1 and 2 circuits;
- Warming up within 1 circuit temperature range 135-220 ° C with disabled ECCS accumulator tanks, for planned and unplanned shutdowns;
- "Hot "state during unit startup, for planned and unplanned shutdowns.

For selected groups of plant operational states, a list of events that could disturb normal operation of NPP was compiled. For this study the recommendations of the IAEA, experience of PSA implementation for similar units, and consistent deductive analysis for undesirable processes causes, were used. For further analysis were selected events, directly or indirectly affect NPP normal operation, characterized by estimated frequency of occurrence not less than 10^{-7} 1/year and contribution for total fuel damage frequency less than 1% .

Selected initiating events are grouped on the basis of similarity of accident process percolation paths and final states, caused by these IE, as well as on the basis of similarity success criteria. The aim of such grouping is to limit the number of accident sequences models.
List of IE groups for unit on-power-states shown at Table 4, for shutdown states at Table 5.

Table 4: IE groups for unit-on-power states

Name & content of IE groups
1. IE group, leading to loss of 1 circuit coolant
Compensated 1 circuit leak inside containment
Small 1 circuit leak inside containment
Middle 1 circuit leak inside containment without safety systems dependent failure
Middle 1 circuit leak inside containment with safety systems dependent failure
Large 1 circuit leak inside containment without safety systems dependent failure
Large 1 circuit leak inside containment with safety systems dependent failure
Small 1-to-2 circuit leak
Middle 1-to-2 circuit leak
Large 1-to-2 circuit leak inside containment
Reactor vessel rupture
SG tube rupture, caused by steam line rupture, outside of containment in non-isolated from SG part
Small 1 circuit leak outside of containment
Compensated 1 circuit leak outside of containment
2. IE group, leading to loss of 2 circuit coolant
Steam line/feed water pipes rupture in non-isolated from SG part
Steam line rupture in isolated from SG part
Feed water pipes rupture in isolated from SG part
3. IE group, leading to transient processes
3.1 Transient processes for 1 circuit
Scram
Loss of feed water flow rate caused by control system failure or partial loss of feed water flow rate at one SG
Automatic reactor shutdown with SG isolation by operator
Administrative shutdown (with different configuration of safety system, concerning safe shutdown)
Small feed water/ condensate system leak
3.2 Transient processes for 2 circuit
Loss of normal heat removal
Spontaneous closure of MSIV at one loop
Spontaneous closure of MSIV at all loops
4. IE group leading to failure of support systems
LOOP
Partial loss of own needs power supply
Loss of responsible consumers cooling

Table 5: IE groups for shutdown states

Name & content of IE groups
1. Termination of heat removal from the core because of primary coolant leaks
Compensated 1 circuit leak inside containment
Small 1 circuit leak inside containment
Large & middle 1 circuit leak inside containment without safety systems dependent failure
Middle 1 circuit leak inside containment with safety systems dependent failure
Large 1 circuit leak inside containment without safety systems dependent failure
Large & middle 1 circuit leak inside containment with safety systems dependent failure
Small 1-to-2 circuit leak
Middle 1-to-2 circuit leak
Large 1-to-2 circuit leak inside containment
Reactor vessel rupture
SG tube rupture, caused by steam line rupture, outside of containment in non-isolated from SG part
Small 1 circuit leak outside of containment
Compensated 1 circuit leak outside of containment
2. Termination of heat removal from the core because of 2 circuit leaks
Steam line/feed water pipes rupture in non-isolated from SG part
Steam line rupture in isolated from SG part
Feed water pipes rupture in isolated from SG part
3. Termination of residual heat removal due to support systems failures
LOOP
Termination of residual heat removal due to equipment failures
Termination of residual heat removal through 2 circuit
Termination of residual heat removal through 1 circuit
Spontaneous closure of MSIV at one loop
4. I circuit brittle strength conditions violation due to "cold" overpressure
1 circuit coolant withdrawal lines spontaneous closure
5. Termination of heat removal from SFP
Failure of one SFP heat removal system chanel (including failure of support systems)
6. Fuel damage during handling operations
SFA damage due to heavy objects falling at SFP or reactor pit
SFA damage by refueling machine due to refueling machine faults, personnel errors, violations of normal operating conditions, loss of refueling machine power supply

On the basis of thermal-hydraulic calculations, as well as on results of engineering evaluations performed for PSA, for each initiating events in each operational state group, success criteria were established. Success criteria was supposed as need to find the unit in a controlled state in the final state of accident with absence of threats to leave this state not associated with random equipment failures.

Analysis object modeling and calculation of model characteristics performed, using Risk Spectrum 1.2.1 software. System models takes into account existing equipment interdependence and interconnections between systems that could affect its functions performance. The models also take into account the possibility of common cause failures due to implicit dependencies.

During personnel reliability analysis human errors, which could take place both before and after initiating event (latter subdivided into errors in response to initiating events and errors in the commission of recovery actions), are simulated. Personnel reliability analysis used THERP and SAIC-TRC approaches.

Within data analysis were evaluated frequencies of initiating events, as well as equipment reliability parameters, including the parameters of the common causes for failure.

IE group frequency assessment for operational states groups, in order to quantify contribution of each operational state at total fuel damage frequency, performed using:

- OKB Gydropress data, obtained on the basis of statistic for nuclear power plants with VVER-1000 type reactors operation and results of the probabilistic analysis for destruction of RP equipment and pipelines, performed as part of AES-2006 technical design[7, 8];
- Results of earlier performed PSA [9];
- Operating experience for nuclear power plants with VVER type reactors in Russia and Ukraine [10 - 12];
- Quantitative analysis for nuclear fuel handling operation safety at Unit 1 LAES -2 [13];
- Calculation of reliability for polar crane 360 (205)/32 +10-41, 5 - UHL4 [14].

Equipment reliability parameters estimations were made using:

- Operational data for Novovoronezh, Kalinin and Balakovo NPP equipment within period 1986 – 2010 [15, 16];
- Results of earlier performed PSA [9];
- FRAMATOM reliability data for control safety systems [17];
- Reliability data for Alfa Laval plate heat exchangers [18];
- IAEA-TECDOC- 478 data [19];
- IAEA-TECDOC- 508 data [20];

For common cause failures modeling, α and β - factor models are used. US NRC data used for common cause failures quantitative estimates [21].

4. PSA RESULTS

Quantification results for major accident sequences leading to fuel damage for on-power states are shown in Table 6, for shutdown states in Table 7. Major initiating events contribution at total fuel damage frequency for on-power states are shown in Figure 1, for shutdown states in Figure 2.

Table 6: Quantification results for major accident sequences leading to fuel damage for on-power states

AS code	FDF (1/rpy)	Relative contribution (%)
VS-JND_FM-JNG_GM	$3{,}80\cdot10^{-8}$	29,5
1_2S-P1_2-SPOT_S	$2{,}62\cdot10^{-8}$	20,3
NISP-H-N4	$1{,}32\cdot10^{-8}$	10,2
S-LOCA-JND_FM-JNG_GM	$1{,}16\cdot10^{-8}$	9,0
RR	10^{-8}	7,74

Explanations to Table 6 AS codes:

- VS - compensated 1 circuit leak inside containment;
- JND_FM –1 circuit feed by 1 of 4 HPIS channels;
- JNG_GM - 1 circuit feed by 1 of 4 LPIS channels;
- 1_2S – small 1-to-2 circuit leak inside containment;
- P1-2 - BRU-A works in cooldown mode on 1 of 3 non-emergency SG and water supply in these SG from corresponding channel of emergency feedwater system;
- SPOT_S - work of three channels of SG PHRS at non-emergency SG;
- NISP - steam line/feed water pipes rupture in non-isolated from SG part;
- H – continuous heat removal through 2nd circuit
- N4 – emergency SG feedwater isolation;
- S-LOCA - small 1 circuit leak inside containment.

Table 7: Quantification results for major accident sequences leading to fuel damage for shutdown states

AS code	FDF (1/ rpy)	Relative contribution (%)
LSFPHR-FAK-SFP_FEED-DNU	$1{,}54 \cdot 10^{-7}$	34,45
LOOP_HR1_5-R_HUM-JNG_HUM	$7{,}60 \cdot 10^{-8}$	17,00
LNHR-HUM_EHRS-D_1/4	$4{,}80 \cdot 10^{-8}$	10,74
LHR1-R_HUM-JNG_HUM-KBA_HUM	$2{,}80 \cdot 10^{-8}$	6,26
LOOP_HR1-3,4,8,9,10-RE2-R_HUM-JNG_HUM-KBA_HUM	$2{,}17 \cdot 10^{-8}$	4,85
LOOP_HR1-5-RE2-R_HUM-JNG_HUM	$1{,}74 \cdot 10^{-8}$	3,89

Explanations to Table 7AS codes:
- LSFPHR - failure of one SFP heat removal system chanel;
- FAK – SFPHRS reserve line work;
- SFP_FEED – SFP feed by emergency heat removal tanks &SFP feed pump;
- DNU - SFP feed by diesel-pump unit;
- LOOP_HR1_5 - LOOP under residual heat removal through 1st circuit condition, reactor lid removed;
- LOOP_HR1-3, 4, 8, 9, 10 - LOOP under residual heat removal through 1st circuit condition, reactor lid maintained;
- RE2-R – LOOP duration within 2 -8 hours;
- R_HUM – RHRS startup by operator;
- JNG_HUM – LPIS pumps startup by operator;
- LNHR - termination of residual heat removal through 2nd circuit;
- HUM_EHRS - BRU-A cooldown mode startup by operator;
- D_1/4 - BRU-A works in cooldown mode on 1 of 4 SG and water supply in these SG from corresponding channel of emergency feedwater system;
- LHR1 – termination of residual heat removal through 1st circuit;
- KBA_HUM – 1st circuit feed from feed and boron regulation system, startup by operator.

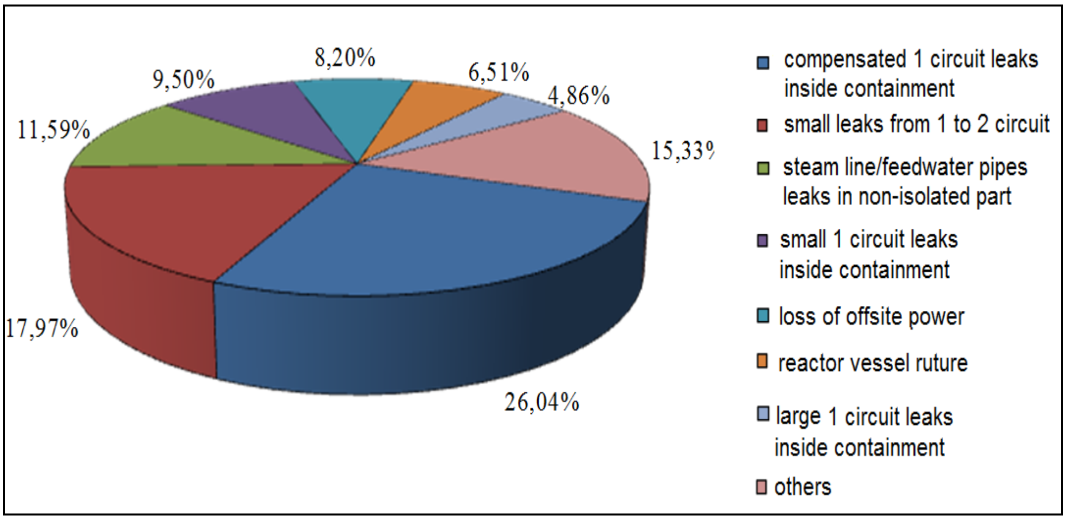

Figure 1 – Major initiating events contribution at total fuel damage frequency for on-power states

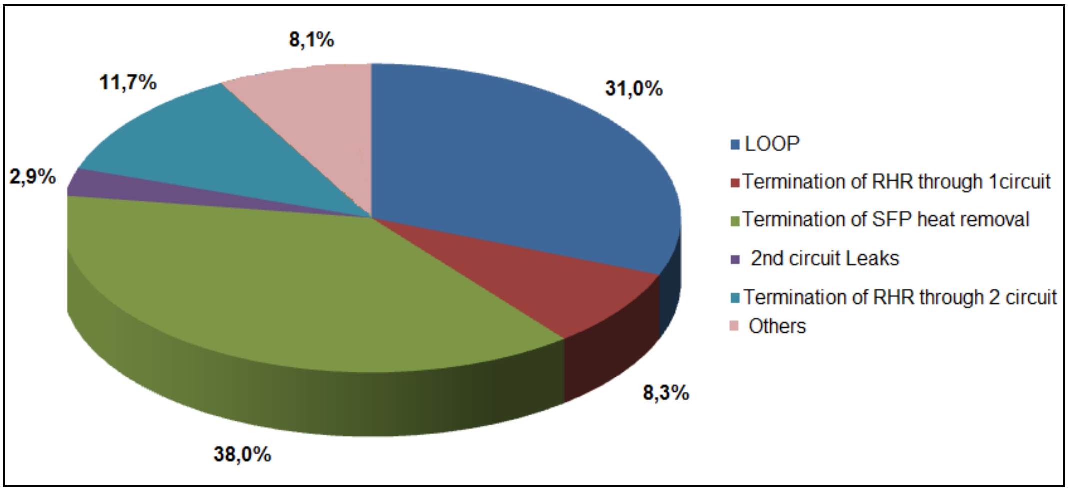

Figure 2 – Major initiating events contribution at total fuel damage frequency for shutdown states

Following results for fuel damage frequency obtained during PSA [22, 23]:

- Average value for total fuel damage frequency at plant-on-power states is 1, $29 \cdot 10^{-7}$ 1/ reactor per year;
- Average value for total fuel damage frequency at plant shutdown states is 4, 47·10-7 1/ reactor per year.

As a result of the uncertainty analysis performed for FDF at on-power states, following 90% confidence interval border obtained:

- Lower (5 %) - 2, $23 \cdot 10^{-8}$ 1/ reactor per year;
- Median (50 %) - 7, $98 \cdot 10^{-8}$ 1/ reactor per year;
- Upper (95 %) - 4, $15 \cdot 10^{-7}$ 1/ reactor per year.

As a result of the uncertainty analysis performed for FDF at shutdown states, following 90% confidence interval border obtained:

- Lower (5 %) - 1, $69 \cdot 10^{-7}$ 1/ reactor per year;
- Median (50 %) - 3, $47 \cdot 10^{-7}$ 1/ reactor per year;
- Upper (95 %) - 9, $52 \cdot 10^{-7}$ 1/ reactor per year.

Below are some recommendations made on the basis of PSA:

- Operational procedures improving: remove feed and boron regulation system outage to beginning of refueling stage;
- Technical solutions: consider possibility of automatic startup for LPIS backup channel, while executing normal operation function; consider possibility of automatic startup for SFPHRS backup channel; envisage backup reactor feed line under reactor dismantling condition to reduce LOOP and termination of residual heat removal through 1 circuit -related IE contribution at total FDF.

4. CONCLUSION

Results of PSA level 1 at PSAR stage for AES-2006 (first unit of LAES-2) shows that project meets its target probabilistic criteria. Further work should be directed to conservatism degree reduction and model detailed elaboration at FSAR stage.

AES-2006 project Volume 2 EUR requirements conformity degree, high level of safety and its possibility to be reconfigured to meet customer specific requirements, allows it worthily compete with other NPP projects at international market.

List of abbreviations

ATWS - Anticipated Transient Without Scrams.
BDBA - Beyond Design Basis Accident.
BRU-A - Fast acting atmosphere steam release installation.
CSS - Control Safety System.
DBA - Design Basis Accident.
DG - Diesel Generator.
ECCS - Emergency Core Cooling System.
EDG - Emergency Diesel Generator.
ESFAS - Engineering Systems Fast Actuation System.
FDF - Fuel Damage Frequency.
FSAR - Final Safety Analysis Report
HPIS - High Pressure Injection System.
IAEA - International Atomic Energy Agency.
IE - Initiating Event.
LOOP - Loss Of Offsite Power.
LPIS - Low Pressure Injection System.
MSIV - Main Steam Isolation Valve.
NPP - Nuclear Power Plant.
PSA - Probability Safety Analysis.
PSAR - Preliminary Safety Analysis Report.
RHR -Residual Heat Removal.
RHRS - Residual Heat Removal System.
RP - Reactor Plant.
rpy – reactor per year.
RTS -Reactor Trip System.
SFA - Spent Fuel Assembly.
SFP - Spent Fuel Pool.
SFPHRS - Spent Fuel Pool Heat Removal System.
SG -Steam Generator.
SG PHRS - Passive Heat Removal System from Steam Generators.

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