

## List and description of procedures for Justification of TVS-2M introduction in «Bushehr» NPP Unit 1

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
0	Elaboration of Contract (or Supplement). Making of Contracts		T0		
1	Participation in core NPhC development (transitive loads from UTVS_TV2S-2M with blankets_TV2S-2M without blankets and equilibrium TV2S-2M without blankets). Elaboration of initial data for neutron-physical and thermal-hydraulic calculations.	OKB GP JSC PFC	T0+7 months	877,85	
1.1	Определение и согласование с Determination and coordination with the Customer of TV2S-2M core specifications, nomenclature and work schedule. Elaboration of the initial data to provide NPhC	OKB GP	T0+1 month	39,30	
1.2	Neutron-physical calculations	JSC PFC OKB GP	T0+6 months	560,05	
1.2.1	Calculation of neutron-physical characteristics of transitive fuel cycles starting from the 7-th loading, with outlet to equilibrium fuel cycle of Unit 1 of "Bushehr" NPP	JSC PFC	T0+6 months	486,05	Work purpose — determination of make-up fuel nomenclature for transitive and equilibrium fuel cycles, selection of "Bushehr" NPP reactor core arrangement and calculation of neutron-physical characteristics. The term for delivery of description of make-up fuel nomenclature and basic NPhC calculations to OKB GP is T0+4 months.
1.2.2	Verification of engineering safety factors for transition from UTVS to TV2S-2M	JSC PFC	T0+6 months	74,00	The procedure will be presented considering FA  structure and dimensions (with tolerances) and numerical values of heat flux and coolant heating engineering safety factors, which shall be used in the calculation analysis of transfer to and further operation of TV2S-2M in "Bushehr" NPP. Evaluation of interassembly gap value shall be submitted by OKB GP 1 month before issue of the present report.
1.2.3	Coordination of NPhC calculations provided by JSC PFC	OKB GP	T0+6 months		Coordination of NPhC calculations is required for verification of used design data, either for thermal hydraulic calculation input data sufficiency.
1.3	Elaboration of input data for neutron-physical characteristics (transitive and equilibrium cycles)	JSC PFC OKB GP	T0+8 months	278,50	

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1.3.1	Preparation of input data for thermal-hydraulic and thermal-mechanical calculations. Generation of table frame parameters	JSC PFC OKB GP	T0+7 months		Calculation of developed fuel cycle neutron-physical characteristics is provided using developed library of neutron-physical constants, fuel nomenclature and refuelling patterns considering variations of core main parameters, such as CPS CR group position, core thermal power, coolant temperature at the core inlet, coolant flowrate through the core within the limits of RP normal operation. NPhC initial data and the table with NPhC boundary values used in thermal hydraulic calculations of safety are prepared on the basis of obtained NPhC values considering prior justified fuel cycles for Russian and foreign power units.
1.3.2	For thermal-hydraulic calculations and safety analyses of disconnection of various number of RCP sets	OKB GP	T0+8 months		RP stationary operation and the transient arising as a result of RCP emergency trip (one out of four and two out of four or one out of three) are considered in the calculation. The scenarios are developed for the conditions simulating Unit unloading to a permissible power level respective to the number of RCP in operation as a result of APP group drop and power governor operation by the signal of RCP trip. NPhC boundary values are determined by the results of considered scenarios, including maximum non-uniformity of relative power, realized in the core during RP operation in steady-state conditions with four, three, two RCPs and in the transient related to RCP trip, which are initial data for the analysis of RP heat-engineering reliability for full and partial number of RCP in operation.
1.3.3	For safety analyses of RIA type using the codes of core three-dimensional kinetics	OKB GP	T0+8 months		The calculation includes preparation of the library of neutron-physical constants, calculations of xenon oscillations considering operating group position variation and recommendations for the choice of feedback parameter reference points. The library is prepared using code package SAPFIR_95&RC_VVER in binary format for FA types and is used by program unit KARTA providing three-dimensional simulation of water cooled and water moderated reactor kinetics as a part of complex code KORSAR/GP in the calculations of reactivity conditions. Comparative analysis of neutron-physical characteristics obtained using code packages SAPFIR_95&RC_VVER and KASKAD is additionally provided on the basis of the obtained results
1.3.4	For thermo-mechanical calculations	OKB GP	T0+8 months		The calculation includes fast neutron flux density axial distributions, both maximum burnt out FA, and all FA in the core in the various moments of operating period.
1.3.5	Calculation of decay heat for spent TVS-2M in spent fuel pool (in shutdown reactor and removed top head)	OKB GP	T0+8 months	112,90	Calculation of core and spent fuel decay heat is provided using the procedure of international standard - ISO-10645-92, Calculation of the decay heat power in nuclear fuels in light water reactors. The events of reactor scheduled shutdown for refuelling with FA partial and complete unloading from the core are considered. The data are presented for whole core powers in various states, average and maximum powers of FA unloaded into a spent fuel pool. The results obtained are use in thermal-hydraulic calculations of shutdown reactor and fuel pool cooling.

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1.3.6	Development and justification of the format of Refuelling safety evaluation report (RSER) considering application in the design justification of spatial kinetics codes	OKB GP	T0+16 months	165,60	New format of refuelling safety evaluation report (RSER) will be developed considering application in the design justification of the codes with core spatial kinetics. Limitations allowing to evaluate fuel cycle safety under deviations from the design refuelling pattern on the basis of neutron-physical characteristics calculations will be presented in RSER updated format. Safety analysis of the conditions with core spatial kinetics for various versions of fuel cycle arrangements will be provided as a verification of the report new format.
1.3.7	Input data, necessary for calculation performance of AEP	OKB GP	T0+6 months		
<b>2</b>	<b>Thermal-hydraulic design of the core (transitive cycles from UTVS to TVS-2M with blankets, and then to TVS-2M without blankets and equilibrium TVS-2M without blankets)</b>	<b>OKB GP</b>	<b>To+10 months</b>	<b>1175,00</b>	
2.1	Thermal-hydraulic calculations of RP normal operating conditions with disconnection of various number of RCP sets	OKB GP	T0+10 months	301,10	RP steady-state conditions of operation with four, three and two working RCP set in the first transitive and subsequent cycle, including equilibrium fuel cycles with core arrangement of UTVS and TVS-2M type FA will be considered in the calculations. The following reactor plant primary thermal-hydraulic characteristics will be presented for each of the conditions (four, three and two RCP sets in operation): - loop coolant flows through the reactor and FA; - DNBR in the core on the basis of local parameters in fuel rod bundles for core maximum peaking powers respective to fuel rod limiting design linear load and power values; - coolant temperature at the reactor inlet and outlet; - coolant heating in FA; - maximum temperature of external surface of fuel rod (gadolinium fuel rod) cladding; - maximum steam content in FA hot cell; - pressure differentials in the core and reactor, hydraulic loads to reactor internals, FA and FA individual components (RCCA). Thermal-hydraulic characteristics in RP steady-state operating conditions with four, three and to two RCP in operation will be determined for nominal parameters and considering parameter possible deviations from nominal values within the limits of design tolerances.

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2.2	Thermal-hydraulic calculation of the parameters for reactor vessel brittle failure resistance (BFR) analysis	OKB GP	T0+10 months	301,10	Reactor plant coolant main thermal-hydraulic parameters (pressure, flowrate and temperature values) under the conditions considered in reactor vessel brittle failure resistance analysis according to IAEA recommendations "Guidelines on pressurized thermal shock analysis for VVER nuclear power plants are determined in the calculation. Revision 1. IAEA-EBP-WWER-08 (Rev. 1), IAEA, 2006". Thermal-hydraulic calculation results use used as reference data for calculation of boundary conditions for reactor vessel component heat release, temperature fields and stresses.
2.3	Calculation of boundary conditions, parameters for reactor vessel BFR analysis	OKB GP	T0+10 months	225,80	Heat exchange boundary conditions (time variation of boundary medium temperatures and coefficients of heat transfer from media to surfaces) in nozzles areas and reactor chambers in the areas of coolant mixing and water supplied from ECCS are determined in the calculation for accident conditions. The results are purposed for the calculations of temperature fields and stresses in BFR analysis of reactor vessel components.
2.4	Stop of spent fuel pool cooling	OKB GP	T0+10 months	150,60	Calculation purpose - to determine the time before fuel damage in the SFP for considered initiating event with stop of cooling water supply into the cooling pool. Make-up flowrate required for SFP evaporated water compensation is also determined in the calculation.
2.5	Spent fuel pool leak or pipeline break resulting in spent fuel pool water level decrease	OKB GP	T0+10 months	196,40	The calculation is provided for cooling pool spent FA for the initiating event with water leak through spent fuel pool (SFP) bottom lining under flowrate equal to the capacity of cooling pool standard system make-up pumps for the initiating event with guillotine break of the pipeline for water supply into a spent fuel pool cooling system. Thus SFP water make-up is not provided. The calculation purpose - to determine the time before SFP fuel damage in considered initiating events with SFP water leak.
<b>3</b>	<b>Strength and thermal mechanics of FA, RCCA and reactor core (loadings of transfer from UTVS to TVS-2M with blankets, and then to TVS-2M without blankets and equilibrium TVS-2M without blankets)</b>	<b>OKB GP</b>	<b>T0+16 months</b>	<b>3211,55</b>	
<b>3.1</b>	<b>Elaboration of calculations of reactor internals temperatures and heat exchange boundary conditions for strain determination of reactor internals affecting FA compression</b>	<b>OKB GP</b>	<b>T0+12 months</b>	<b>75,30</b>	Determination of coolant flowrates and heating in core baffle channels and gaps for various versions of fuel loadings. Calculation of temperature fields in core baffle and core barrel under normal operation at different versions of fuel loading. Calculation of core barrel temperature fields in for the version with wall maximum power for the conditions of anticipated operational occurrences and main coolant pipeline breaks.

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<b>3.2</b>	<b>Substantiation of TVS-2M thermo-mechanical stability and strength</b>	<b>OKB GP</b>	<b>T0+14 months</b>	<b>1112,80</b>	
3.2.1	Calculations for determination of the parameters of fuel rods and guiding channels interaction with FA spacing grid	OKB GP	T0+10 months	212,80	The calculations determine local parameters of fuel rod and guiding channel (GCh) interaction with spacing grid (SG) of UTVS and TVS-2M: SG rigidity for turn of fuel rods and GCh depending of current elastic tightness in contact pairs “fuel rod – SG cell” and coefficient of friction in contacting surfaces; radial rigidity of SG cells; parameters describing fuel rod and GCh deformations in the flights between SG considering radiation creeping of structural materials; parameters describing SG deformation considering radiation creeping of structural materials (variation of SG cell internal inscribed diameter, SG deformation under fuel rod and GCh turnings in the grid). Calculation results will be used as input data for simulation of UTVS and TVS-2M thermo-mechanical behavior during operation. Two calculations are scheduled using code package ANSYS
3.2.2	Calculation of FA operation longitudinal strain and rigidity	OKB GP	T0+10 months	94,90	Bend and longitudinal rigidity of UTVS and TVS-2M, both UTVS and TVS-2M GCh in-service length variation will be determined in the calculation. Work results are required for calculation of FA compression forces during operation and determination of permissible range for PTU axial position during preventive maintenance. FA various states will be considered in the calculation. Calculations will be provided using specialized certificated code FAME_N1simulating FA thermo-mechanical behavior during operation
3.2.3	Calculation of FA components shape variation in loss of coolant accidents	OKB GP	T0+14 months	94,90	Analysis of UTVS and TVS-2M SG straining resulted from interactions with fuel rod ballooned claddings, including variation of SG wrench dimension in the course of loss of coolant accident will be provided in the calculation. Evaluation of the conservatism for UTVS and TVS-2M loss of stability under longitudinal compression in the course of loss of coolant accident will be provided in the calculation. Conclusions of FA thermo-mechanical stability in the loss of coolant accident and of the possibility of core unloading after the accident will be made on the basis of the results obtained. Code package ANSYS and code FAME_N1 will be used in the calculation.
3.2.4	Calculation analysis of FA frame strength during operation	OKB GP	T0+10 months	229,10	Two calculations are scheduled. Forces in UTVS and TVS-2M components during operation, considering radiation increase and creeping of zirconium alloys will be determined in the first calculation using code FAME_N1. Calculation analysis of UTVS and TVS-2M component strength considering most loaded structure units will be provided in the second calculation on the basis of the forces obtained using code package ANSYS. Non-elastic analysis methods will be used providing more exact safety margins for composite units using finite-element code packages.

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3.2.5	Calculations of FA and inter-assembly space deflections in NOC. Preparation of reference data generated for calculation of Keng.	OKB GP	T0+14 months	229,10	Two calculations will be provided. Calculation evaluations of UTVS and TVS-2M possible deflections during operation will be provided in the first calculation using specialized code FAME_N1. Single FA will be conservatively assumed not considering supporting effects of adjacent FA. Specified calculation results will be used in the calculation analysis of UTVS and TVS-2M strength. Evaluation of FA compression maximum forces providing FA thermo-mechanical stability during operation will be also made within the framework of specified work. Simulation of thermo-mechanical behavior of all FA in the core using probabilistic methods will be provided in the second calculation. The calculation will be provided using specialized code developed on the basis of certificated code FAME_N1. The gaps between FA depending time will be determined in the calculation for the whole core at different axial levels.
3.2.6	Calculation of FA compression parameters for NOC conditions	OKB GP	T0+16 months	94,90	The ranges of top nozzle displacements for UTVS and TVS-2M, both the ranges of specified FA compression forces during operation will be determined in the calculation for RP in which PTU projection completions were not provided. All main factors defining FA compression during operation, including relaxation of FA spring unit preliminary compression forces and GCh length variation during operation will be considered in the calculation. Solution of the necessity to provide activities on FA compression forces optimisation during operation can be assumed by work results.
3.2.7	Calculation of the permissible range of PTU axial position during scheduled preventive maintenance	OKB GP	T0+12 months	157,10	Permissible range of PTU axial position during scheduled preventive maintenance (PM) will be determined in the calculation for each PM up to reactor brining to equilibrium fuel cycle. The calculation will consider all main factors, which determine PTU axial position in open reactor, including core current cycle, PTU weight in water, variation of FA GCh length, relaxation of the forces of FA spring units preliminary compression. Specified work results are required to control the forces of FA compression during operation, and if required, for the recommendations of PTU projection completion necessity and extent by the results of their axial position measurement during PM.

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3.3	Calculations of CPS CR dynamic characteristics for NOC, AOO and DBA	OKB GP	T0+12 months	204,80	Calculation of CPS CR dynamic characteristics for normal operation and anticipated operational occurrences. The calculation is provided to determine CPS CR dynamic characteristics under the conditions of their displacement in case of reactor scram at NOC and AOO. CPS CR dynamic characteristics (drop time, displacement velocities, overloads during drop and damping) are determined for various number of working RCP sets and various primary coolant temperatures typical for conditions NOC and AOO. Influence of geometrical and operating conditions parameter deviations (within the ranges of design tolerances) onto CPS CR dynamic characteristics is considered in the calculation. Calculation of CPS CR dynamic characteristics for design basis accidents. The calculation is provided to determine CPS CR dynamic characteristics in accident conditions with breaks of MCP and other pipelines, both with CPS drive housing break. CPS CR dynamic characteristics (drop time, displacement speeds, forces and overloads during drop and damping under scram, during emersion from bottom position (accidents with pipeline breaks) and withdrawal from the core (in the accident with drive housing break) are determined considering influence of CPS CR pressure differentials realized in considered accident conditions. Calculation results are used in RP safety analysis and as input data for strength calculations.
3.4	Calculation of core hydro-dynamical conditions for DBA	OKB GP	T0+10 months	168,70	VVER-1000 reactor coolant parameters are determined for the accidents with: - break of ECCS pipeline connected to MCP "cold" or "hot" leg; - break of "hot" and "cold" MCP Dnom 850 (design criteria of the time for CPS CR insertion in the core is checked for specified conditions in the calculation of CPS CR dynamic characteristics). Accident conditions are considered for two initial states: - for reactor operation at nominal; - at operation at zero power, in "hot" state of the reactor. Calculations of accident operating conditions at nominal power are required for determination of CPS CR dynamic characteristics for drop time computation and strength calculation analysis. Calculations of accident operating conditions at zero power are required for determination of RCCA extension shaft dynamic characteristics for emersion from bottom position.
3.5	Calculation of TVS-2M dynamic characteristics for DBA	OKB GP	T0+10 months	506,10	



No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
3.5.1	Dynamic characteristics of TVS-2M under "cold" and "hot" pipeline breaks at nominal power for which LBB concept is not used	OKB GP	T0+10 months	168,70	Dynamic characteristics, axial and lateral loads to TVS-2M are determined for the process of RP reactor coolant system emergency de-pressurization at nominal power resulted from break of the pipeline with maximum diameter connected to MCP for which "leak before break" concept is not considered. Loads to TVS-2M are calculated for minimum and maximum compression of spring units in TVS-2M top nozzle. Calculation results are used as reference data in TVS-2M and reactor internals strength calculations.
3.5.2	Dynamic characteristics of TVS-2M under "cold" and "hot" pipeline breaks at hydrotests for which LBB concept is not used	OKB GP	T0+10 months	168,70	Dynamic characteristics, axial and lateral loads to TVS-2M are determined for the process of RP reactor coolant system emergency de-pressurization at hydraulic tests resulted from break of the pipeline with maximum diameter connected to MCP for which "leak before break" concept is not considered. Loads to TVS-2M are calculated for minimum and maximum compression of spring units in TVS-2M top nozzle. Calculation results are used as reference data in TVS-2M and reactor internals strength calculations.
3.5.3	Dynamic characteristics of TVS-2M under "cold" and "hot" pipeline breaks under PRZ PORV operation pressure for which LBB concept is not used	OKB GP	T0+10 months	168,70	Dynamic characteristics, axial and lateral loads to TVS-2M are determined for the process of RP reactor coolant system emergency de-pressurization under PRZ PORV actuation pressure resulted from break of the pipeline with maximum diameter connected to MCP for which "leak before break" concept is not considered. Loads to TVS-2M are calculated for minimum and maximum compression of spring units in TVS-2M top nozzle. Calculation results are used as reference data in TVS-2M and reactor internals strength calculations.
<b>3.6</b>	<b>Calculation of dynamic loads on reactor internals in DBA</b>	<b>OKB GP</b>	<b>T0+13 months</b>	<b>489,25</b>	
3.6.1	Forces to reactor internals under "cold" and "hot" pipeline breaks at nominal power for which LBB concept is not used	OKB GP	T0+13 months	168,70	Work purpose is to determine pressure differentials and the forces affecting reactor internals at nominal power for the process of RP reactor coolant system emergency de-pressurization resulted from the break of maximum diameter pipeline connected to MCP, for which "leak before break" concept is not considered, for the first time moments characterized by maximum dynamic loads. Calculation results are used as reference data in TVS-2M and reactor internals strength calculations.
3.6.2	Forces to reactor internals under "cold" and "hot" pipeline breaks at hydrotests for which LBB concept is not used	OKB GP	T0+13 months	168,70	Work purpose is to determine pressure differentials and the forces affecting reactor internals at hydraulic tests for the process of RP reactor coolant system emergency de-pressurization resulted from the break of maximum diameter pipeline connected to MCP, for which "leak before break" concept is not considered, for the first time moments characterized by maximum dynamic loads. Calculation results are used as reference data in TVS-2M and reactor internals strength calculations.



No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
3.6.3	Forces to reactor internals under "cold" and "hot" pipeline breaks under PRZ PORV operation pressure for which LBB concept is not used	OKB GP	T0+13 months	151,85	Work purpose is to determine pressure differentials and the forces affecting reactor internals at PRZ PORV actuating pressure for the process of RP reactor coolant system emergency de-pressurization resulted from the break of maximum diameter pipeline connected to MCP, for which "leak before break" concept is not considered, for the first time moments characterized by maximum dynamic loads. Calculation results are used as reference data in TVS-2M and reactor internals strength calculations.
<b>3.7</b>	<b>Strength calculations</b>	<b>OKB GP</b>	<b>T0+16 months</b>	<b>654,60</b>	
3.7.1	TVS-2M strength calculations for NOC, AOO and DBA, external dynamic loads and various combinations of conditions and loads	OKB GP	T0+16 months	271,70	Analysis of FA components static and cyclic strength, both stability during RP operation in the conditions NOC, AOO and combinations of loads is provided in TVS-2M strength calculations. Analysis of FA components safety is provided for the load combinations NOC+DBA and NOC+DBA+OBE: FA components deformation shall not prevent RCCA drop by reactor scram signal and core unloading.
3.7.2	Internals strength calculations for NOC, AOO and DBA, external dynamic loads and various combinations of conditions and loads	OKB GP	T0+16 months	382,90	Analysis of internals static and cyclic strength, both stability during RP operation in the conditions NOC, AOO and combinations of loads is provided in internals strength calculations. Analysis of internals components safety is provided for the load combinations NOC+DBA and NOC+DBA+OBE: internals components deformation shall not prevent RCCA drop by reactor scram signal and core unloading.
<b>4</b>	<b>Safety analyses (loads of transfer from UTVS to TVS-2M with blankets and then to TVS-2M without blankets and equilibrium TVS-2M without blankets)</b>	<b>OKB GP AEP</b>	<b>T0+20 months</b>	<b>8154,20</b>	
<b>4.1</b>	<b>Elaboration of safety analyses for AOO conditions with disconnection of various number of RCP sets</b>	<b>OKB GP</b>	<b>T0+16 months</b>	<b>1481,50</b>	
4.1.1	Malfunction of the systems affecting reactivity. Uncontrolled withdrawal from the core of CPS most efficient control rod group (at nominal power, partial power and from subcritical state during start-up)	OKB GP	T0+16 months	207,20	Thermal-hydraulic calculation of the conditions in the conservative approach considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.
4.1.2	Malfunction of the systems affecting reactivity. Malfunction of the control rod (cluster). Drop of one control rod	OKB GP	T0+16 months	122,70	Thermal-hydraulic calculation of the conditions in the conservative approach considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.

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4.1.3	Malfunction of the systems affecting reactivity. Malfunction of the control rod (cluster). Axial static mismatch of one control rod (cluster) in control group	OKB GP	T0+16 months	90,30	Thermal-hydraulic calculation. The conditions are referred to AOO category and long in time. Different versions of one CPS CR from working group inter-position to other WG CRs resulting in power field considerable distortion are simulated. Meeting of thermal-hydraulic acceptance criteria is proved in safety analyses by the results of neutron-physical calculations.
4.1.4	Malfunction of the systems affecting reactivity. Operator error at suppression of xenon oscillations (displacement of control rod resulting in maximum possible distortion of power field)	OKB GP	T0+16 months	104,70	Thermal-hydraulic calculation. The conditions are referred to AOO category and long in time. Scenarios are simulated in the calculation, when rising or descending phase of axial xenon oscillations is resulted from operator error (inaction) related to RP power variation, resulting in power field distortion to the core top or bottom to (and higher) the design limits for fuel rod local power. Meeting of thermal-hydraulic acceptance criteria is proved in safety analyses by the results of neutron-physical calculations.
4.1.5	Malfunction of the systems affecting reactivity. Malfunction of boric control and volume system or operator error resulting in coolant volume increase or primary boron concentration lowering	OKB GP	T0+16 months	172,20	Thermal-hydraulic calculation of the conditions in the conservative approach considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions. Thermal-hydraulic calculation. The conditions are referred to AOO category and long in time. Minimum time values of reactor critical state reaching after scram considering fuel incomplete overlapping by CPS CR absorber are determined for RP various states. Maintenance of reactor subcriticality is proved by the results of provided calculations within full time specified for the time criterion - 15 min, without considering the time for initiating event identification for the states of start-up, hot state, cold shut-down, power operation and 30 min, without considering the time for initiating event identification, for refuelling conditions.
4.1.6	Malfunction of the systems affecting reactivity. Fuel assemblies improper loading and operation in the wrong position	OKB GP	T0+16 months	90,30	Thermal-hydraulic calculation. The conditions are referred to AOO category and long in time. Various versions are simulated for erroneous rearrangement of two FA in the core with maximum enrichment resulting in significant distortion of power field. States with core maximum relative power are selected in the power range from HZP to nominal for which thermal-hydraulic analysis of core cooling is provided considering deviations of thermal-hydraulic parameters. Meeting of thermal-hydraulic acceptance criteria is proved in safety analyses by the results of neutron-physical calculations. In addition, calculation evaluation of the number of core FA wrong shuffling resulting in compact arrangement of FA with maximum enrichment ensuring reactor subcriticality criterion meeting are provided for the state "Refuelling".
4.1.7	Lowering of coolant flowrate in the reactor coolant system. Disconnection of various number of main coolant pump sets	OKB GP	T0+16 months	52,70	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.

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4.1.8	Lowering of coolant flowrate in the reactor coolant system. Conditions of grid frequency emergency deviations	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.9	Increase of heat removal from the secondary side. Full inadvertent opening of one feedwater control valve	OKB GP	T0+16 months	52,70	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.10	Increase of heat removal from the secondary side. Feedwater system malfunction resulting in feedwater flow increase	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.11	Increase of heat removal from the secondary side. Feedwater system malfunction resulting in feedwater temperature decrease	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.12	Increase of heat removal from the secondary side. Increase of steam flowrate to the turbine (as a result of steam pressure governor malfunction or failure) (turbine load abrupt increase above 10 % of nominal)	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.13	Decrease of heat removal from the secondary side. Loss of a.c. non-emergency auxiliary power supply of auxiliary plant equipment (loss of NPP power)	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.14	Decrease of heat removal from the secondary side. Closing of turbine stop valves or off-site power loss	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.15	Decrease of heat removal from the secondary side. Loss of feedwater normal flowrate (with the exception of FW pipeline break)	OKB GP	T0+16 months	52,70	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.16	Decrease of heat removal from the secondary side. Spurious closing of main isolation valve	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.17	Decrease of heat removal from the secondary side. Loss of vacuum in the condenser or other cases resulting in turbine shut-down	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.

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4.1.18	Increase of primary coolant inventory. Disturbances in chemical and volume control system, resulting in the increase of primary coolant inventory	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.19	Increase of primary coolant inventory. Spurious injection into pressurizer from chemical and volume control system	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.20	Spurious operation of systems. Spurious injection into pressurizer from reactor coolant pump set pressure side	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.1.21	Increase of heat removal from the secondary side. Inadvertent opening of steam generator safety valve, BRU-A or BRU-K	OKB GP	T0+16 months	122,40	Thermal-hydraulic calculation of the conditions in the conservative approach considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.
<b>4.2</b>	<b>Safety analyses in DBA conditions with disconnection of various number of RCP sets</b>	<b>OKB GP</b>	<b>T0+22 months</b>	<b>1396,50</b>	
4.2.1	Malfunction of the systems affecting reactivity. CPS control rod ejection under drive housing break	OKB GP	T0+16 months	99,80	Thermal-hydraulic calculation of the conditions in the conservative approach considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.
4.2.2	Malfunction of the systems affecting reactivity. Operator error at loop connection (connection of RCP set without power preliminary decrease), one for each loop	OKB GP	T0+16 months	84,80	Thermal-hydraulic calculation of the conditions in the conservative approach considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.
4.2.3	Decrease of coolant flowrate in the reactor coolant system. Instant jamming of one RCP set out of various number of working sets	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.2.4	Decrease of coolant inventory in the reactor coolant system. SB LOCA as a result of pipeline (Dnom <100) break	OKB GP	T0+16 months	52,70	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.2.5	Decrease of coolant inventory in the reactor coolant system. LB LOCA as a result of pipeline breaks (Dnom <100 including break of the main coolant pipeline)	OKB GP	T0+16 months	52,70	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
4.2.6	Decrease of coolant inventory in the reactor coolant system. Primary-to-secondary leak within SG ranges (Dnom <100)	OKB GP	T0+16 months	60,20	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.2.7	Decrease of coolant inventory in the reactor coolant system. Breaks of instrumentation lines or other lines from reactor coolant pressure boundary coming through the containment	OKB GP	T0+16 months	52,70	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.2.8	Decrease of coolant inventory in the reactor coolant system. Break of steam generator heat-exchanging tube with subsequent cooldown at the rate of 60 °C/h	OKB GP	T0+16 months	52,70	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.2.9	Decrease of coolant inventory in the reactor coolant system. Main steam header break	OKB GP	T0+16 months	158,10	Thermal-hydraulic calculation of the conditions in the conservative approach considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.
4.2.10	Decrease of coolant inventory in the reactor coolant system. Pressurizer safety valve inadvertent opening with subsequent failure to seat	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.2.11	Decrease of coolant inventory in the reactor coolant system. Primary-to-secondary leak during separation of steam generator collector head	OKB GP	T0+16 months	37,60	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.2.12	Increase of secondary heat removal. Break of steam generator main feedwater pipeline	OKB GP	T0+16 months	52,70	Variant calculations in conservative approach considering single failure and brining to repair of safety system channel to prove meeting of acceptance criteria in transitive and equilibrium fuel cycles.
4.2.13	Failures in systems and equipment for nuclear fuel handling. Drop of fuel assembly during overload into spent and fresh fuel pool	OKB GP	T0+16 months	202,90	Analysis of TVS-2M drop during overload into spent and fresh fuel pool is made to determine FA strain state for subsequent safety analysis. Design basis accident with FA drop during fuel handling procedures shall be considered in safety analysis of the system complex for nuclear fuel storage and handling according to standard requirements. Nuclear safety calculation analysis is provided using the results of strength analysis for the event of FA drop into cooling pool. The criterion of nuclear safety analysis is not exceeding of neutron multiplication factor specified in federal standards and regulations.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
4.2.14	Failures in systems and equipment for nuclear fuel handling. Spent and fresh fuel container drop	OKB GP	T0+16 months	122,70	Analysis of spent and fresh fuel container drop is provided to determine pitch variation of FA arrangement in the container in the drop event for subsequent safety analysis. Design basis accident with drop of FA container during handling procedures shall be considered in the safety analysis of system complex for nuclear fuel storage and handling according to standard requirements. Results of strength analysis including container structure possible variation (as a result of drop) and FA structure component possible shape variation are used as initial data for calculation analysis of nuclear safety. The criterion of nuclear safety analysis is not exceeding of neutron multiplication factor specified in federal standards and regulations.
4.2.15	Increase in secondary heat removal. Spectrum of steam line breaks	OKB GP	T0+16 months	160,10	Thermal-hydraulic calculation of the conditions in the conservative approach considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.
4.2.16	Calculations of distribution and accumulation of hydrogen in containment rooms during DBA. Check of meeting of hydrogen safety criteria	AEP	T0+22 months	131,60	
<b>4.3</b>	<b>Elaboration of safety analyses for BDBA conditions</b>	<b>OKB GP AEP</b>	<b>T0+22 months</b>	<b>2052,20</b>	
4.3.1	Complete loss of all a.c. power supplies (complete loss of NPP power)	OKB GP	T0+16 months	68,70	BDBA analysis in realistic approach for the scenarios with and without accident control (to the moment of reactor vessel damage). Evaluation of RP and containment parameter variations for the typical events, including the time of fuel severe damage, reactor vessel damage (through melting). Evaluation of coolant mass and energy, hydrogen and corium release from RP into containment during an accident.
4.3.2	Complete loss of all feedwater supply into steam generators	OKB GP	T0+16 months	68,70	BDBA analysis in realistic approach for the scenarios with and without accident control (to the moment of reactor vessel damage). Evaluation of RP and containment parameter variations for the typical events, including the time of fuel severe damage, reactor vessel damage (through melting). Evaluation of coolant mass and energy, hydrogen and corium release from RP into containment during an accident.
4.3.3	Anticipated transients without scram. Uncontrolled withdrawal of CPS most powered control rod group from the core (at nominal power, partial power and from subcritical state during start-up)	OKB GP	T0+16 months	75,30	Thermal-hydraulic calculation of the conditions in the realistic approach not considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.
4.3.4	Anticipated transients without scram. Loss of auxiliary emergency a.c. power supply (loss of NPP power supply)	OKB GP	T0+16 months	75,30	Thermal-hydraulic calculation of the conditions in the realistic approach not considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
4.3.5	Anticipated transients without scram. Condenser loss of vacuum	OKB GP	T0+16 months	75,30	Thermal-hydraulic calculation of the conditions in the realistic approach not considering equipment single failure or operator single error. Calculation is provided to prove acceptance criteria meeting for considered conditions.
4.3.6	Loss of coolant accident, LB LOCA without emergency core cooling	OKB GP AEP	T0+16 months	102,10	Analysis of BDBA in a realistic approach for the scenarios with or without the accident management (before the moment of the reactor vessel failure). An assessment of change in the RP and containment parameters, times of the specific events including the times of severe fuel failure, reactor vessel failure (penetration). An assessment of yield of coolant mass and energy, hydrogen and corium from the RP into the containment during the accident.
4.3.7	Loss of coolant accident, SB LOCA without emergency core cooling	OKB GP AEP	T0+16 months	114,90	Analysis of BDBA in a realistic approach for the scenarios with or without the accident management (before the moment of the reactor vessel failure). An assessment of change in the RP and containment parameters, times of the specific events including the times of severe fuel failure, reactor vessel failure (penetration). An assessment of yield of coolant mass and energy, hydrogen and corium from the RP into the containment during the accident.
4.3.8	Loss of coolant accident, LB LOCA and pump recirculation interlocking	OKB GP AEP	T0+16 months	102,10	Analysis of BDBA in a realistic approach for the scenarios with or without the accident management (before the moment of the reactor vessel failure). An assessment of change in the RP and containment parameters, times of the specific events including the times of severe fuel failure, reactor vessel failure (penetration). An assessment of yield of coolant mass and energy, hydrogen and corium from the RP into the containment during the accident.
4.3.9	Loss of residual heat removal for 24 h in the shut-down mode	OKB GP	T0+16 months	60,20	Analysis of BDBA in a realistic approach before the moment of the reactor vessel failure. An assessment of change in the RP parameters, times of the specific events including the times of severe fuel failure, reactor vessel failure (penetration). An assessment of yield of coolant mass and energy, hydrogen and corium from the RP into the containment during the accident.
4.3.10	Making a calculated analysis of the changed parameters of hydrogen distribution and accumulation in the containment rooms	OKB GP AEP	T0+22 months	225,50	OKB GP will give the input data
4.3.11	Analysis of core criticality and corium during the severe beyond-design basis accidents	OKB GP	T0+16 months	282,50	To make the calculation analysis of criticality we make use of the results of simulation of four key modes using the thermalphysic computer codes. In each of the modes it is simulated a certain stage of the beyond design-basis accident scenario characterized by a degree of the core component degradation and a set of thermalphysic parameters. The fact of reaching the criticality state is a criterion when making the analysis.
4.3.12	Failure of feedwater supply into the steam generators (BDBA with reactor scram failure)	OKB GP	T0+16 months	75,30	Thermohydraulic calculation of the mode in a realistic approach without consideration of the equipment failures or an operator's isolated error. Calculation is performed to confirm the meeting of the acceptance criteria for the mode under consideration.



No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
4.3.13	Inadvertent opening of steam dump valve to the atmosphere (BRU-A) or steam dump valve to turbine condenser (BRU-K) (BDBA with reactor scram failure)	OKB GP	T0+16 months	75,30	Thermohydraulic calculation of the mode in a realistic approach without consideration of the equipment failures or an operator's isolated error. Calculation is performed to confirm the meeting of the acceptance criteria for the mode under consideration.
4.3.14	Complete loss of NPP power supply (BDBA with reactor scram failure)	OKB GP	T0+16 months	75,30	Thermohydraulic calculation of the mode in a realistic approach without consideration of the equipment failures or an operator's isolated error. Calculation is performed to confirm the meeting of the acceptance criteria for the mode under consideration.
4.3.15	Closing of stop valves (BDBA with reactor scram failure)	OKB GP	T0+16 months	75,30	Thermohydraulic calculation of the mode in a realistic approach without consideration of the equipment failures or an operator's isolated error. Calculation is performed to confirm the meeting of the acceptance criteria for the mode under consideration.
4.3.16	Calculation analysis of the severe beyond-design basis accidents (in-vessel stage)	AEP	T0+22 months	500,40	
4.3.16.1	Calculation of corium interaction with concrete and evaluation of radiologic consequences during the severe BDBA at "Bushehr" NPP	AEP	T0+22 months	316,80	
4.3.16.1.1	Calculation analysis of interaction of corium from the reactor vessel with concrete of the reactor concrete cavity	AEP	T0+22 months		
4.3.16.1.2	Analysis of radioactive fission products outlet to the environment during the severe BDBA	AEP	T0+22 months		
4.3.16.2	Analysis of performance of localization functions by the containment of "Bushehr" NPP during in- and out-vessel stages of BDBA	AEP	T0+22 months	183,60	
4.3.16.2.1	Calculation of change in parameters of steam-gas medium and hydrogen concentration in the containment rooms during in- and out-vessel stages of the severe BDBA	AEP	T0+22 months		
4.3.16.2.2	Calculation of distribution of radioactive fission products in the containment volume and outlet to the environment during the severe BDBA	AEP	T0+22 months		
<b>4.4</b>	<b>PSA development</b>	<b>AEP OKB GP</b>	<b>T0+24 months</b>	<b>3224,00</b>	

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
4.4.1	PSA, level I, during power operation (HZP mode included)		T0+24 months	1407,74	
4.4.1.1	Thermohydraulic calculation for probabilistic safety analysis under the power operation conditions. LB LOCA in the primary circuit	OKB GP	T0+24 months	188,20	The thermohydraulic calculations for IE group “LB LOCA in the primary circuit” are carried out. The TH calculations are carried out to determine the criteria of success for safety functions. The results of these calculations will be used to group the IE, generate the trees of events and failures when simulating the accident sequences, determine a sequence of functioning the frontal systems for each IE group
4.4.1.2	Thermohydraulic calculation for probabilistic safety analysis under the power operation conditions. MB LOCA in the primary circuit	OKB GP	T0+24 months	188,20	The thermohydraulic calculations for IE group “MB LOCA in the primary circuit” are carried out. The TH calculations are carried out to determine the criteria of success for safety functions. The results of these calculations will be used to group the IE, generate the trees of events and the trees of failures when simulating the accident sequences, determine a sequence of functioning the frontal systems for each IE group
4.4.1.3	Thermohydraulic calculation for probabilistic safety analysis under the power operation conditions. SB LOCA in the primary circuit	OKB GP	T0+24 months	188,20	The thermohydraulic calculations for IE group “SB LOCA in the primary circuit” are carried out. The TH calculations are carried out to determine the criteria of success for safety functions. The results of these calculations will be used to group the IE, generate the trees of events and the trees of failures when simulating the accident sequences, determine a sequence of functioning the frontal systems for each IE group
4.4.1.4	Thermohydraulic calculation for probabilistic safety analysis under the power operation conditions. The SB LOCA in the primary circuit compensated by the normal make-up system	OKB GP	T0+24 months	112,92	The thermohydraulic calculations for IE group “SB LOCA in the primary circuit compensated by the normal make-up system”. The TH calculations are carried out to determine the criteria of success for safety functions. The results of these calculations will be used to group the IE, generate the trees of events and failures when simulating the accident sequences, determine a sequence of functioning the frontal systems for each IE group
4.4.1.5	Thermohydraulic calculation for probabilistic safety analysis under the power operation conditions. Primary-to-secondary coolant leak	OKB GP	T0+24 months	150,56	The thermohydraulic calculations for IE group “Primary-to-secondary coolant leak”. The TH calculations are carried out to determine the criteria of success for safety functions. The results of these calculations will be used to group the IE, generate the trees of events and failures when simulating the accident sequences, determine a sequence of functioning the frontal systems for each IE group
4.4.1.6	Thermohydraulic calculation for probabilistic safety analysis under the power operation conditions. Secondary leak.	OKB GP	T0+24 months	150,56	The thermohydraulic calculations for IE group “Secondary leak”. The TH calculations are carried out to determine the criteria of success for safety functions. The results of these calculations will be used to group the IE, generate the trees of events and failures when simulating the accident sequences, determine a sequence of functioning the frontal systems for each IE group

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
4.4.1.7	Thermohydraulic calculation for probabilistic safety analysis under the power operation conditions. Transients	OKB GP	T0+24 months	150,56	The thermohydraulic calculations for IE group “Transients”. The TH calculations are carried out to determine the criteria of success for safety functions. The results of these calculations will be used to group the IE, generate the trees of events and failures when simulating the accident sequences, determine a sequence of functioning the frontal systems for each IE group
4.4.1.8	Elaboration of the initial data for PSA-1 concerning the RP. A choice and grouping of the initiating events concerning the RP for the power operation conditions	OKB GP	T0+24 months	22,60	It is made the analysis of the IE that can occur during the RP power operation. We consider the internal IE being caused by the failures of the process equipment or RP systems including the spurious actuations or personnel’s errors. Based on the analysis the considered IE are distributed into the groups included into the PSA-1 logical-and-probabilistic model. The boundaries of the IE groups related to the loss of primary coolant are determined according to the results of the thermohydraulic calculations in support of PSA.
4.4.1.9	Elaboration of the initial data for PSA-1 concerning the RP. Evaluation of frequencies of the initiating events due to the primary leaks for the power operation conditions	OKB GP	T0+24 months		It is performed an evaluation of frequencies of the IE groups considered in PSA that can occur owing to the primary coolant leaks through the cracks or untightness of the RP equipment or pipelines. Evaluation of frequencies of the IE groups is carried out considering the data on the operational experience of the VVER-1000 RP equipment and the results of the probabilistic analysis of failure of the RP equipment and pipelines.
4.4.1.10	Elaboration of the initial data for PSA-1 concerning the RP. Neutron physics calculation for determination of realistic criteria of success of the reactor emergency protection functions	OKB GP	T0+24 months		In the calculation for the various moments of burn-up of loadings we determine the values of emergency protection worth and repeated criticality temperature with the different number of the CPS CR being stuck at the various height from the core bottom. The results are used in the PSA to determine the probability of the reliable performance of the emergency protection function by the EP system.
4.4.1.11	Elaboration of the initial data for PSA-1 concerning the RP. Analysis of reliability of reactor emergency protection functions	OKB GP	T0+24 months	22,58	It is performed an evaluation of the reliability indices of the CPS equipment fulfilling the reactor emergency function considering the differences in each of the IE groups to be discussed in PSA. In this case the mechanical part reliability indices are evaluated considering the realistic criteria of success to be determined as a result of the neutron physics calculation. In the present analysis the frequency of the false reactor scram is not evaluated, but it is determined in PSA based on the operational experience.
4.4.1.12	Elaboration of the initial data for PSA-1 concerning the RP. Simulation of emergency sequences for initiating events with the primary leaks	OKB GP	T0+24 months	45,17	It is developed a logical-and-probabilistic model of development of the emergency sequences (ES) for the IE due to the loss of primary coolant during the RP power operation. Thus, based on the results of thermal-hydraulic calculations in support of PSA we determine the safety functions and the systems that ensure their fulfilment, and also the criteria of success of these safety functions.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
4.4.1.13	Elaboration of the initial data for PSA-1 concerning the RP. Simulation of emergency sequences for initiating events with reduction in heat removal	OKB GP	T0+24 months	22,58	It is developed a logical-and-probabilistic model of ES development for the IE related to loss of heat removal from the fuel being in the reactor during the RP power operation. Thus, based on the results of thermal-hydraulic calculations in support of PSA we determine the safety functions and the systems that ensure their fulfilment, and also the criteria of success of these safety functions.
4.4.2	PSA, level 1 for conditions with shutdown reactor		T0+24 months	865,70	
4.4.2.1	Thermohydraulic calculation for probabilistic safety analysis under the standby conditions. Determination of criteria of success for the safety functions with the sealed primary circuit	OKB GP	T0+24 months	376,40	The TG calculations are performed for PSA under the standby conditions with the sealed primary circuit. The TG calculations are carried out to determine the criteria of success for safety functions. The results of these calculations will be used to group the IE, generate the trees of events and trees of failures during simulation of the emergency sequences, determine a sequence of functioning the frontal systems for each of the IE groups
4.4.2.2	Thermohydraulic analyses for justification of probabilistic models under standby conditions with the unsealed primary circuit	OKB GP	T0+24 months	188,20	The calculation is aimed to: - determine the time the personnel have at their disposal to implement the emergency provisions: from the initiating event till there occurs the failure (the beginning of uncovering) of the reactor core fuel and in the spent fuel pool; - determine the minimum required water flowrate to be supplied into the core to prevent the core degradation.
4.4.2.3	Elaboration of the initial data for PSA-1 for the RP. A choice and grouping of the initiating events concerning the RP for standby conditions	OKB GP	T0+24 months	30,10	It is made the analysis of the IE that can occur during the RP operation under the conditions with the shutdown reactor (the reactor is subcritical and all the CPS CR are in the LLS) for the operational states determined for PSA, level 1. We consider the internal IE the causes of which are related to the failures of the process equipment or the RP systems including the spurious actuations or the personnel' errors. Based on the analysis the IE considered are distributed into the groups included in the logical-and-probabilistic model of PSA, level 1.
4.4.2.4	Elaboration of the initial data for PSA-1 for the RP. Evaluation of frequencies of the initiating events related to the primary leaks for the standby conditions	OKB GP	T0+24 months	45,17	It is performed an evaluation of frequencies of the IE groups considered in PSA which can occur owing to the primary coolant leaks through the cracks or untightness of the RP equipment or pipelines under the standby conditions. The frequencies of the IE groups are evaluated for the operational states (OS) determined for PSA, level 1 to be made considering both the data on the operational experience of the VVER-1000 RP equipment and the results of the probabilistic analysis of the RP pipelines and equipment failure.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
4.4.2.5	Elaboration of the initial data for PSA-1 for the RP. Neutron physics calculation for the initiating events with dilution of primary coolant under the standby conditions	OKB GP	T0+24 months	45,17	In the calculation for the various moments of burn-up of loading we determine the time periods the personnel have at their disposal to perform the activities on termination of supplying the pure condensate into the primary circuit from the moment of initiating event occurrence to that of reaching the criticality in reactor core for the RP various operational states characterized by the cold shutdown boric acid concentration value in the primary coolant. The results are used in PSA when determining the probability of bringing the reactor into the critical state.
4.4.2.6	Elaboration of the initial data for PSA-1 for the RP. Simulation of emergency sequences for the initiating events with the primary coolant leaks with the sealed primary circuit	OKB GP	T0+24 months	60,22	It is developed a logical-and-probabilistic model of development of the ES for the IE related to loss of primary coolant in the OS with the sealed primary circuit. Thus, based on the results of thermal-hydraulic calculations in support of PSA we determine the safety functions and the systems that ensure their fulfilment, and also the criteria of success of these safety functions.
4.4.2.7	Elaboration of the initial data for PSA-1 for the RP. Simulation of emergency sequences for the initiating events with loss of heat removal with the sealed primary circuit	OKB GP	T0+24 months	30,11	It is developed a logical-and-probabilistic model of development of the ES for the IE related to loss of heat removal from the fuel being in the reactor for the ES with the sealed primary circuit. Thus, based on the results of thermal-hydraulic calculations in support of PSA we determine the safety functions and the systems that ensure their fulfilment, and also the criteria of success of these safety functions.
4.4.2.8	Elaboration of the initial data for PSA-1 for the RP. Simulation of emergency sequences for the initiating events with coolant leaks in the primary circuit with the unsealed primary circuit	OKB GP	T0+24 months	60,22	It is developed a logical-and-probabilistic model of development of the ES for the IE related to loss of primary coolant in the OS with the unsealed primary circuit. Thus, based on the results of thermal-hydraulic calculations in support of PSA we determine the safety functions and the systems that ensure their fulfilment, and also the criteria of success of these safety functions.
4.4.2.9	Elaboration of the initial data for PSA-1 for the RP. Simulation of emergency sequences for the initiating events with loss of heat removal with the unsealed primary circuit	OKB GP	T0+24 months	30,11	It is developed a logical-and-probabilistic model of development of the ES for the IE related to loss of heat removal from the fuel being in the reactor for the ES with the unsealed primary circuit. Thus, based on the results of thermal-hydraulic calculations in support of PSA we determine the safety functions and the systems that ensure their fulfilment, and also the criteria of success of these safety functions.
5	<b>Justification of fuel rod and gadolinium fuel rod (transient loadings from UTVS to TVS-2M with blankets, and then to TVS-2M without blankets and equilibrium TVS-2M without blankets)</b>	VNIINM OKB GP JSC PFC	T0+23 months	997,95	
5.1	<b>Development of initial data to justify the operability of TVS-2M fuel rod and gadolinium fuel rod</b>	OKB GP JSC PFC	T0+16 months		

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
5.1.1	Preparation of initial data for justification of operability of fuel rods and gadolinium fuel rods under both the steady-state operation conditions and the NOC transients	JSC PFC	T0+8 months		The rod-to-rod power distributions under the considered unsteady-state conditions will be obtained considering the changes in thermal core power, displacements of CPS CR groups, change in coolant temperature, boric acid concentration in coolant and xenon transients. By the results of the calculational analysis of the conditions in the format agreed upon with the Consumer, the binary files of the initial data will be prepared for justification of operability of fuel rods and gadolinium fuel rods.
5.1.2	by NPhC under the AOO transients and high distribution of fuel rod cladding temperature	OKB GP	T0+10 months		In the calculation the conditions that refer to AOO category are simulated: uncontrolled withdrawal of control group, inadvertent decrease in boron concentration in the primary coolant, operator's error in case of xenon oscillation suppression, improper loading of the FA. Based on the performed calculations we prepare the binary files of the arrays of the rod-to-rod power distributions, burn-up and fast neutron flux on the fuel rod cladding and the temperatures values of the external surface of the fuel rod claddings for their use in future in the analyses of the fuel element operability.
5.1.3	by the thermohydraulic characteristics under the steady-state conditions	OKB GP	T0+10 months		The input data of the core NPhC (arrangement of fuel rods/ gadolinium fuel rods considering burn-up) for RP steady-state operation conditions. On the basis of the obtained information and thermal-hydraulic characteristics of the circuit a thermal-hydraulic calculation is performed for each FA of the core. Work for preparation of data on t/h characteristics results in high distributions of fuel rod cladding temperature for each fuel rod of the core.
5.1.4	by LOCA	OKB GP	T0+16 months		Analysis of the temperature state of the claddings of fuel rods and Gd-fuel rods with different power under large break loss-of-coolant accident (LB LOCA). Generation of the power and boundary condition data for thermomechanical calculations of fuel rods and Gd-fuel rods with different power and burnup: - variation of coolant pressure during the accident; - variation of fuel power linear rate during the accident – for groups of fuel rods with different initial power; - variation of coolant temperature and heat transfer coefficient during the accident. Submission of the initial data by VNIINM for safety justification of fuel rods and Gd-fuel rods under LB LOCA.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
5.1.5	by RIA	OKB GP	T0+16 months		Analysis of the temperature state of the claddings of fuel rods and Gd-fuel rods with different power under reactivity-initiated accident (RIA). Generation of the power and boundary condition data for thermomechanical calculations of fuel rods and Gd-fuel rods with different power and burnup: - variation of coolant pressure during the accident; - variation of fuel linear power rate during the accident – for groups of fuel rods with different initial power; - variation of coolant temperature and heat transfer coefficient during the accident. Submission of the initial data by VNIINM for safety justification of fuel rods and Gd-fuel rods under RIA.
<b>5.2</b>	<b>Development of justification of operability of fuel rod and gadolinium fuel rod under the steady-state operation conditions, NOC and AOO transients, analyses of behaviour during DBA</b>	<b>VNIINM OKB GP</b>	<b>T0+23 months</b>	<b>997,95</b>	
5.2.1	Development of justification for the TVS-2M fuel rod and gadolinium fuel rod	VNIINM	T0+23 months	<b>958,68</b>	
5.2.1.1	The technical reference information “Elaboration of the initial data on thermalphysic characteristics of fuel rods and gadolinium fuel rods of various power and burn-up to perform the thermalhydraulic calculations of design basis accidents, data transfer to JSC PFC and OKB GP”	VNIINM	T0+11 months		
5.2.1.2	Justification of operability of fuel rods and gadolinium fuel rods under the steady-state operation condition at "Bushehr" NPP, Unit 1 during TVS-2M implementation	VNIINM	T0+16 months	283,42	The paper analyzes the initial data with the results of neutronics calculations of BNPP-1 fuel loadings and selects a representative array of FAs and fuel rods, Gd-fuel rods in them. Thermophysical and thermomechanical characteristics of fuel rods and Gd-fuel rods are calculated using the code START-3A and meeting of the design criteria is checked to justify operability of fuel rods and Gd-fuel rods under steady-state operating conditions at BNPP-1.
5.2.1.3	Justification of operability of fuel rods and gadolinium fuel rods during the transients with NOC, AOO at "Bushehr" NPP, Unit 1 during TVS-2M implementation	VNIINM	T0+16 months	268,66	The paper analyzes the results of neutronics calculations of governing transients of NOC, AOO at the beginning and the end of the transient and the steady-state fuel loading of BNPP-1. Thermophysical and thermomechanical characteristics of fuel rods and Gd-fuel rods are calculated using the code START-3A and meeting of the design criteria is checked to justify operability of fuel rods and Gd-fuel rods under transients of NOC, AOO at BNPP-1.



No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
5.2.1.4	Justification of safety behavior of fuel rods and gadolinium fuel rods during the design basis accidents LOCA and RIA at "Bushehr" NPP, Unit 1 during TVS-2M implementation	VNIINM	T0+21 months	203,44	The paper analyzes the initial data for justification of a safe behavior of fuel rods and Gd-fuel rods under design LOCA and RIA for BNPP-1. Thermomechanical and corrosion characteristics of fuel rods and Gd-fuel rods are calculated using the code RAPTA-5.2 and meeting of the criteria is checked to justify a safe behavior of fuel rods and Gd-fuel rods under design LOCA and RIA at BNPP-1.
5.2.1.5	Determination and justification of design criteria of VVER-1000 fuel rods under NOC and AOO	VNIINM	T0+12 months	93,67	The objective of the paper is: to determine the design criteria applied in development of engineering material to justify the operability of fuel rods and Gd-fuel rods under NOC, AOO for licensing the uranium-gadolinium fuel for WVER-1000; to describe the reasons (causes) for introduction of an appropriate design criterion of fuel rod; to perform calculation-experimental justification of the magnitudes of the limiting values and dependencies for the criteria of operability of WVER-1000 fuel rods under NOC, AOO.
5.2.1.6	Determination and justification of criteria of safe behaviour of VVER-1000 fuel rods during the design basis accidents	VNIINM	T0+19 months	109,50	The objective of the paper is: to perform experimental justification of the criteria of a safe behavior of WVER fuel rods under design basis accidents with coolant loss (LOCA) and reactivity-initiated accidents (RIA); to evaluate the properties of alloy Э110 (E110) as material for WVER fuel rod claddings within the context of fuel system coolability criteria and fuel rod damage criteria whose meeting is necessary for safety assurance of the reactor plant with WVER.
5.2.2	Coordination of justification on fuel rod and gadolinium fuel rod of TVS-2M	OKB GP	T0+23 months	39,27	
<b>6</b>	<b>Justification of RCCA operability and service life characteristics</b>	<b>OKB GP JSC PFC VNIINM JSC MSZ JSC NCCP</b>	<b>T0+21 months</b>	<b>1411,80</b>	
6.1	Calculation of the limiting deviations of relative position of fuel stack and absorber in the absorbing rod of the control and protection system	OKB GP	T0+4 months	55,00	In the calculation it is determined the high-altitude position of the absorber with respect to the fuel stack with the various RCCA positions in the core. We consider the RCCA position on upper limit switch, lower rigid stop, lower limit switch, and also several different positions along the core height. The results of the calculation are required for elaboration and justification of the core reactivity control algorithms and for the calculated analysis of RCCA strength and serviceability.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
6.2	Justification of RCCA service life. Determination of the radiation loads acting on the RCCA structural and absorbing materials during NOC	OKB GP	T0+10 months	75,30	To justify the radiation stability of the AE structural materials it is checked the meeting of the acceptance criteria of CPS CR to be spent during the automatic control and EP conditions. As a result of the analysis made considering the fuel cycle characteristics we determine the burn-ups of boron-10, dysprosium isotopes and neutron fluences onto the absorbing materials and AR end piece during CPS CR operation over an adjusting range with the various high-altitude positions from the core bottom followed by the operation in the EP to confirm the RCCA operability during the assigned service life. The calculation results are the initial data for calculations of change in the worth of the separate CPS CR groups and emergency protection considering the absorber burn-up during the fuel cycle operation.
6.3	Calculation of RCCA worth considering the absorber burn-up during operation	OKB GP	T0+10 months	75,30	The worth of the CPS CR emergency protection and working group is calculated considering a change in the isotope composition of boron carbide and dysprosium titanate during an irradiation for the AE assigned service life. The worth values are calculated for the chosen reactor fuel cycles. To confirm the frame values of the emergency protection worth it is chosen a conservative version of positioning the burnt CPS CRs considering the sticking of one CPS CR of the maximum worth. It is evaluated a decrease in the emergency protection worth during operation caused by a change in isotope composition of boron and dysprosium in the AE of CPS CR.
6.4	Calculation of power distribution in the RCCA absorbing and structural materials	OKB GP	T0+10 months	75,30	The calculation is carried out to obtain the initial data on power distribution in the AE structural materials for the analyses of AE cooling under the NOC and AOO including DBA. The conditions are broken down into the groups in which the CPS CR motion is initiated in addition to the EP signal, and the CPS CR motion occurs only by the EP signal. The relative local power distributions in the structural materials of AE and GCh, and also the fuel rods, surrounding the GCh, corresponding to the different CPS CR group positions are calculated for two groups of conditions.
6.5	Analysis of RCCA thermal state during NOC, AOO and DBA	OKB GP	T0+13 months	52,70	The analysis is performed to determine the data on the temperature state of AE of CPS CR under the steady-state conditions of NOC. The document deals with the calculation results of velocities, coolant flowrates in GCh, coolant temperatures in GCh, temperatures of GCh wall and temperatures of AE cladding for the various positions of absorbing element in the reactor core and with the various power profiles along the core height. The concentric and eccentric positions of AE in the GCh are considered. The analysis results are used for RP safety analysis.
6.6	Strength analysis of RCCA under NOC, AOO and DBA	OKB GP	T0+17 months	180,70	In the calculation analysis the strength analysis of RCCA under NOC and AOO and during DBA as well will be performed.
6.7	Modification of RCIS of CPS drives in connection with using TVS-2M	OKB GP VNIIEM	T0+14 months	408,80	Modification of RCIS (including the software) is related to adjustment of the high-altitude position and the CPS CR adjusting range

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
6.7.1	Development of technical requirements for modification of RCIS in connection with using TVS-2M	OKB GP	T0+8 months	255,30	
6.7.2	Modification of RCIS (including the software)	VNIIEM	T0+14 months	153,50	
6.8	Analysis of RP dynamic stability	OKB GP	T0+12 months	77,60	Thermohydraulic calculation of normal operation conditions and several conditions with the anticipated operational occurrences in a realistic approach with simulation of operation of the main controllers, interlockings, EP-PP. The calculation is performed to confirm an absence of automatic unscheduled decrease in power, automatic unscheduled reactor shutdowns or actuation of control safety systems under the considered conditions without the failures of systems, equipment and personnel's erroneous actions in addition to the initiating event or transient condition.
6.9	Elaboration of the documentation of the detailed project reports of AE and RCCA (drawings, explanatory reports, project of technical specifications, specification for structural materials, patent forms)	OKB GP	T0+21 months	392,80	The drawings together with specifications and specification for structural materials contain the sufficient information for elaboration of WDD (working design documentation). Explanatory reports contain the complete information on description of the design, operating conditions, handling requirements, justification of safe operation under the design conditions.
6.10	Agreement on the documentation of detailed project reports of AE and RCCA with the foreign organizations	JSC PFC JSC MSZ JSC NCCP	T0+21 months	18,30	The agreement on the documentation will allow us to confirm the underlying characteristics in the design on conformity to the neutronic characteristics, characteristics of structural materials, manufacturing processes
<b>7</b>	<b>Materials of detailed project report of FA and core</b>	<b>JSC PFC OKB GP VNIINM JSC NCCP</b>	<b>T0+32 months</b>	<b>1358,80</b>	
7.1	Elaboration of the documentation of detailed project reports of TVS-2M and core (drawings, explanatory reports, project of specifications, specification for structural materials, patent forms)	OKB GP	T0+27 months	1154,70	Drawings together with specifications and specification for structural materials contain the sufficient information for elaboration of WDD. The explanatory reports contain the complete information on description of the design, operating conditions, handling requirements, safe operation justification under the design conditions.
7.2	Agreement on the documentation of detailed project reports of core and TVS-2M with the foreign organizations	JSC PFC VNIINM JSC NCCP	T0+27 months	23,40	The agreement on the documentation will allow us to confirm the underlying characteristics in the design on conformity to the neutronic characteristics, characteristics of structural materials, manufacturing processes
7.3	Putting into production at the Manufacturer	JSC TVEL JSC NCCP OKB GP	T0+32 months	180,70	Making a trial batch, carrying out the acceptance tests to confirm the readiness of manufacture

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
8	Nuclear safety, releases, doses	JSC PFC OKB GP AEP	T0+21 months	1617,70	
8.1	Calculation of radiation characteristics of spent fuel during long-term storage (residual heat, activity and nuclide composition of spent fuel)	JSC PFC	T0+21 months	42,90	
8.2	Justification of nuclear safety when handling TVS-2M at the NPP	JSC PFC OKB GP	T0+21 months	102,00	
8.2.1	Development of justification of nuclear safety when handling with TVS-2M at the NPP	JSC PFC	T0+21 months	102,00	The nuclear safety analysis will be carried out on the basis of the results of neutron-and-physical calculations for the normal operation conditions and accident situations through the whole path of transportation and fuel storage at the NPP (before and after the reactor).
8.2.2	Agreement of nuclear safety analysis when handling with TVS-2M at the NPP	OKB GP	T0+21 months		For the agreement of the results of the nuclear safety analysis to be made by JSC PFC it is proposed not only an analysis of the information obtained, but also carrying out the selective independent calculations. It is possible that simulation of the separate states characterized by the probability of reaching the safety criterion will be required.
8.3	Activation of the corrosion products on the primary circuit equipment surfaces and in the primary coolant and calculations of fission product accumulation within the claddings and in the primary coolant	JSC PFC OKB GP	T0+13 months	147,10	
8.3.1	Development of the calculations of the corrosion product activation on the surfaces of the primary circuit equipment and in the primary coolant considering transition to TVS-2M	JSC PFC	T0+12 months	85,90	In these calculations we shall determine the values of corrosion product activity on the primary circuit surfaces and in the primary coolant, and also the values of activity of the main fission products under the claddings of the leak-tight fuel rods and in the primary coolant with the assigned number of leaky fuel rods in the core. The results of these calculations are required as the initial data to calculate the radiation protection (item 8.4), and also to compile chapters 11,12 FSAR.
8.3.2	Development of the calculations of fission product accumulation under the fuel rod claddings and in the primary coolant	JSC PFC	T0+12 months	61,20	
8.3.3	Agreement of the calculations of the corrosion product activation on the surfaces of the primary circuit equipment and in the primary coolant and the calculations of fission product accumulation under the claddings and in the primary coolant	OKB GP	T0+13 months		Agreement of calculation data is necessary, as OKB GP uses their results during justification radiation protection (i. 8.4) and during development of input data for chapters 11 and 12 of FSAR

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
<b>8.4</b>	<b>Development of calculations of the radiation load on the personnel and the RP equipment</b>	<b>OKB GP</b>	<b>T0+21 months</b>	<b>572,10</b>	This item of the technical and commercial offer (TCO) provides for elaboration of a set of the calculations intended to justify the reactor plant radiation protection (chapters 11,12 FSAR), and also to generate the initial data for calculations of temperature fields and strength of reactor internals and brittle fracture resistance of the reactor vessel. The calculations by the given item were performed within the framework of detailed project report of the V-446 reactor plant and made the basis for the valid FSAR.
8.4.1	Calculation of radiation protection. Part 1. Radiation levels at the nominal power	OKB GP	T0+18 months	75,30	In the calculation it will be determined the intensity of the main neutron and gamma sources due to the RP, and also a dose rate due to the mentioned kinds of radiation in the different points of near-reactor space during the RP operation at the rated power. The calculation results are necessary to compile chapters 11,12 FSAR and can be used for assessment of the possibility of a short-term access of the maintenance personnel under the containment (if required) during the RP operation at any power up to the nominal one.
8.4.2	Calculation of radiation protection. Part 2. Radiation power distributions	OKB GP	T0+10 months	90,30	In this calculation we shall determine the radiation power in the reactor internals and other structural components of the reactor. The calculation results will be used as the initial data in the calculation of temperature fields in the reactor internals (item 3.1).
8.4.3	Calculation of radiation protection. Part 3. Readiness of the stopped equipment	OKB GP	T0+19 months	75,30	In this calculation we shall determine the radiation sources and levels at the personnel's workstations during the RP routine maintenance in the course of PM. The calculation results will be used to evaluate the dose commitments of the personnel involved in the work on the routine maintenance of the RP during PM related to disassembly (assembly) and inspection of the equipment. The calculation results are necessary to compile chapters 11, 12 FSAR.
8.4.4	Calculation of radiation protection. Part 4. Radiation situation during transport-and-technological procedures	OKB GP	T0+19 months	75,30	In this calculation we shall determine the radiation sources and levels in the places where the personnel is located during the transport-and-technological procedures with the reactor plant equipment and spent TVS-2M. The results of the present calculation will be used for evaluation of the personnel dose commitments during PM and for compiling chapters 11,12 FSAR.
8.4.5	Calculation of radiation protection. Part 5. Dose commitments during the equipment maintenance	OKB GP	T0+20 months	90,30	In the calculation we shall determine the annual average collective dose of the personnel involved in the RP routine maintenance activities related to the reactor sealing and unsealing and the equipment revision during the scheduled refuelling. Meeting the criteria on this parameter will be shown. The calculation results are required to compile chapter 12 FSAR.
8.4.6	Calculation of neutron fluxes	OKB GP	T0+10 months	90,30	In the calculation we determine the basic functionals of fast neutron flux onto the reactor vessel and reactor internals (fluence, damaging dose, neutron spectrum). The results of the calculation will be used as the initial data for strength calculations of reactor internals (item 3.7.2) and brittle failure resistance of the reactor vessel (item 8.5).

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
8.4.7	Topical report for submittal to the Iranian Customer "Radiation load onto the personnel and the reactor plant equipment"	OKB GP	T0+21 months	75,30	In the topical report the basic results obtained in calculations as per items 8.4.1-8.4.6 are proposed to be combined for submittal to the Iranian Customer.
<b>8.5</b>	<b>Development of the reactor vessel calculation for brittle failure resistance considering a changed neutron fluence</b>	<b>OKB GP</b>	<b>T0+14 months</b>	<b>431,60</b>	We shall perform the calculations of temperature fields, stressed-strained state and fracture mechanics parameters in the reactor vessel for the design operation conditions including the accident conditions with a thermal shock. By the results of the calculations the reactor vessel service life will be evaluated by the brittle failure resistance criterion considering the new values of the neutron fluence. The "temperature - permissible pressure" (P-T curve) dependences will be developed for the normal operation conditions and the minimum permissible temperatures of hydrotests will be determined.
<b>8.6</b>	<b>Delivery of initial data for AEP</b>	<b>OKB GP</b>	<b>T0+19 months</b>		
8.6.1	Delivery of the initial data on the core (specifications, NPhC, thermal-hydraulic characteristics, radiation loads onto the RP equipment, results of BDBA analyses, decay heat in TVS-2M, etc.) for AEP	OKB GP JSC PFC VNIINM	T0+19 months		Initial data preparation for evaluation of radiological consequences on the structural characteristics of the core components and materials in an agreed format.
8.6.2	Initial data for evaluation of radiological consequences during AOO	OKB GP	T0+19 months		Preparation of the initial data to evaluate the radiological consequences (coolant mass to be discharged through the steam dump valves of the secondary circuit, data on fuel temperature and fuel rod cladding, assumptions on the loss of integrity of fuel rod cladding) for each initiating event considered.
8.6.3	Initial data for evaluation of radiological consequences during DBA	OKB GP	T0+19 months		Preparation of the initial data to evaluate the radiological consequences (coolant mass to be discharged through the steam dump valves of the secondary circuit, data on fuel temperature and fuel rod cladding, assumptions on the loss of integrity of fuel rod cladding) for each initiating event considered.
8.6.4	Initial data for evaluation of radiological consequences during BDBA	OKB GP	T0+19 months		Preparation of the initial data to evaluate the radiological consequences (coolant mass to be discharged through the steam dump valves of the secondary circuit, data on fuel temperature and fuel rod cladding, assumptions on the loss of integrity of fuel rod cladding) for each initiating event considered.
<b>8.7</b>	<b>Elaboration of justification of NPP radiation safety, including radiologic safety under NOC, AOO and accident conditions</b>	<b>AEP JSC PFC OKB GP</b>	<b>T0+24 months</b>	<b>322,00</b>	
<b>8.7.1</b>	<b>Radiation consequences during AOO</b>		<b>T0+24 months</b>		
8.7.1.1	Radiation consequences of the conditions with malfunction of the systems affecting the reactivity	AEP OKB GP	T0+24 months		

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
8.7.1.2	Radiation consequences due to decrease in coolant flowrate in the reactor coolant system	AEP OKB GP	T0+24 months		
8.7.1.3	Radiation consequences due to increase in the secondary side heat removal	AEP OKB GP	T0+24 months		
8.7.1.4	Radiation consequences due to decrease in the secondary side heat removal	AEP OKB GP	T0+24 months		
8.7.1.5	Release of radioactive substances from the systems and equipment	JSC PFC AEP	T0+24 months		
8.7.1.5.1	Damage to the liquid radioactive waste tanks	AEP OKB GP	T0+23 months		
8.7.1.5.2	Radiation consequences of the accidents of the liquid radioactive waste treatment systems	AEP OKB GP	T0+24 months		
<b>8.7.2</b>	<b>Radiation consequences under DBA conditions</b>		<b>T0+24 mec</b>		
8.7.2.1	Radiation consequences due to malfunction of the systems affecting the reactivity	AEP OKB GP	T0+24 months		
8.7.2.2	Radiation consequences due to decrease in coolant flowrate in the reactor coolant system	AEP OKB GP	T0+24 months		
8.7.2.3	Radiation consequences due to small break accident Dnom<100	AEP OKB GP	T0+24 months		
8.7.2.4	Radiation consequences due to large break accident Dnom 850	AEP OKB GP	T0+24 months		
8.7.2.5	Radiation consequences due to the primary-to-secondary leak within the steam generator (Dnom<100)	AEP OKB GP	T0+24 months		
8.7.2.6	Radiation consequences due to break of instrumentation lines or other lines from the reactor coolant pressure boundary passing through the containment	AEP OKB GP	T0+24 months		
8.7.2.7	Radiation consequences due to the accident with rupture of steam generator heat-exchanging tube followed by cooldown at the rate of 60 °C/h	AEP OKB GP	T0+24 months		
8.7.2.8	Radiation consequences due to the accident with break of the main steam header	AEP OKB GP	T0+24 months		



No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
8.7.2.9	Radiation consequences due to increase in the secondary side heat removal	AEP OKB GP	T0+24 months		
8.7.2.10	Release of radioactive substances from the systems and equipment	AEP OKB GP	T0+24 months		
8.7.2.10.1	Damage to the gaseous radioactive waste treatment system	AEP OKB GP	T0+24 months		
8.7.2.10.2	Radiation consequences due to the gaseous radioactive waste treatment systems	AEP OKB GP	T0+24 months		
8.7.2.11	Radiation consequences due to the fuel assembly drop into the spent and fresh fuel pool during refueling	AEP OKB GP	T0+24 months		
8.7.2.12	Radiation consequences due to drop of the cask for the spent and fresh fuel	AEP OKB GP	T0+24 months		
<b>9</b>	<b>Development of justification of operability of the process equipment and systems of the Unit</b>	<b>OKB GP JSC PFC JSC IZ JSC AM AEP</b>	<b>T0+14 months</b>	<b>2223,90</b>	
<b>9.1</b>	<b>Development of justification of operability of the process equipment of the Unit (DADS, fuel handling procedures, systems of storage of fresh and spent fuel)</b>	<b>OKB GP JSC PFC JSC IZ JSC AM</b>	<b>T0+36 months</b>	<b>1512,70</b>	
<b>9.1.1</b>	<b>Justification of DADS operability</b>	<b>OKB GP</b>	<b>T0+18 months</b>	<b>182,30</b>	
9.1.1.1	Thermohydraulic calculation of DADS	OKB GP	T0+18 months	42,60	Determination of sufficiency of DADS heat exchanger surface for heat removal and non-exceeding of temperature in the DADS circuit above 90°C when carrying out the LCC (for the most-powered TVS-2M)
9.1.1.2	Calculation for determination of the permissible time of water circulation termination in the DADS circuit	OKB GP	T0+18 months	37,60	Determination of the permissible time of termination of water circulation in the DADS circuit (pump trip in the infusion mode) when performing the LCC of TVS-2M based on the condition of non-exceeding of the water saturation temperature in the circuit at the pressure in the infusion mode (for the most-powered TVS-2M)
9.1.1.3	Calculation for strength of the DADS equipment (revision)	OKB GP	T0+18 months	42,60	Calculation revision in connection with an increase in TVS-2M weight
9.1.1.4	Loading pattern for the embedded components (revision)	OKB GP	T0+18 months	25,50	Loading pattern revision in connection with an increase in TVS-2M weight

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
9.1.1.5	DADS. Explanatory report (revision)	OKB GP	T0+18 months	34,00	Revision in connection with a change in the calculation results by items 9.1.1.1; 9.1.1.2 and 9.1.1.
<b>9.1.2</b>	<b>Development of the Manual on LCC of the fuel rods at the RP being in operation and shutdown for "Bushehr" NPP during transition to TVS-2M new fuel (considering the LCC up-to-date requirements and procedures)</b>	<b>JSC PFC OKB GP</b>	<b>T0+19 months</b>	<b>78,00</b>	
9.1.2.1	Development of the materials for including in the Manual on LCC and agreement of the Manual	JSC PFC	T0+19 months	35,40	Development of the documents (criteria, procedures and requirements) on LCC of the fuel rods at the RP being in operation and shutdown for new fuel (TVS-2M) and considering the up-to-date approaches and requirements
9.1.2.2	Development of the Manual on LCC	OKB GP	T0+19 months	42,60	Development of manual on LCC of the fuel rods at the RP being in operation and shutdown for "Bushehr" NPP when using TVS-2M (on the basis of JSC PFC documents)
<b>9.1.3</b>	<b>Justification of operability of FHE with fuel, system of storage of fresh and spent fuel</b>	<b>OKB GP JSC IZ JSC AM</b>	<b>T0+36 months</b>	<b>1252,40</b>	
9.1.3.1	Output of the initial data on justification of FHE with fuel, systems of storage of fresh and spent fuel	OKB GP	T0+14 months		Preparation of the initial data for evaluation of FHE strength (packed fuel storage rack, rack for FA, etc.) and nuclear safety analysis during the FA storage and transportation
9.1.3.2	Calculations for strength of the packed fuel storage racks (PFSR) during normal operation and external dynamic impacts	JSC IZ	T0+17 months	43,20	Evaluation of strength of PFSR during the normal operation and seismic impacts considering an increase in the loaded fuel mass
9.1.3.3	Analysis of drop of heavy objects into spent fuel pool	JSC IZ	T0+17 months	39,60	Evaluation of PFSR distortion during the accidents connected to the heavy object drop
9.1.3.4	Thermohydraulic calculation. Cooling of SFA during storage in spent fuel pool	OKB GP	T0+18 months	75,30	The calculation is aimed at determining the cooling conditions of the spent FA located in the racks of the spent fuel pool during NOC; AOO; DBA; BDBA.
9.1.3.5	Nuclear safety calculation during accidents with drop of heavy objects into spent fuel pool	OKB GP	T0+20 months	117,90	For the calculation it is proposed to make the nuclear safety analysis during storage and transportation of nuclear fuel at "Bushehr" NPP. The system of handling with the "fresh" and spent nuclear fuel is considered. The normal operation conditions, and also the design and beyond design-basis accidents, including accidents with drop of heavy objects into spent fuel pool, are simulated.
9.1.3.6	Strength calculation of the rack in FFS	OKB GP	T0+22 months	85,10	In the calculation it will be considered the strength of the structural components of the rack for TVS-2M fresh fuel assemblies under the normal operation conditions, and also due to the external dynamic impacts: safe shutdown earthquake and crash of the middle class airplane onto the reactor building.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
9.1.3.7	Pattern of loading the SFP embedded components due to PFSR	JSC "IZ"	T0+17 months	39,60	The loads onto the SFP civil structures are given on the basis of the strength calculations.
9.1.3.8	Pattern of loading the FFS embedded components due to PFSR	OKB GP	T0+18 months	42,60	The loads onto the FFS civil structures are given on the basis of the strength calculations
9.1.3.9	Modification of FHM	JSC AM OKB GP	T0+36 months	<b>809,10</b>	
9.1.3.9.1	FHM control system caused by a change in the FA shuffling velocities	JSC AM OKB GP	T0+36 months	260,50	Change in FHMCS (fuel handling machine control system) due to increase in the FA motion velocities and mass.
9.1.3.9.2	Implementation of LCC in FHM working shaft	JSC AM OKB GP	T0+36 months	548,60	The fuel handling machine mast is additionally equipped with the leak detection system in fuel rod claddings.
<b>9.2</b>	<b>Analysis of NPP systems and buildings during utilization of TVS-2M in 12-month cycle</b>	<b>AEP</b>	<b>T0+24 months</b>	<b>711,20</b>	The following procedures will be performed: - capability analysis of SFP constructions; - capability analysis of fresh fuel storage constructions of the reactor building; - calculations of coolant system under SFP cooling conditions (at annual fuel unloading, total fuel unloading, accidental fuel unloading)
<b>10</b>	<b>ICIS modification</b>	<b>OKB GP JSC PFC</b>	<b>T0+23 months</b>	<b>1026,40</b>	
10.1	Justification of the ICIS design metrological characteristics that fulfil the core local parameter protection function for the fuel loadings with the use of TVS-2M for "Bushehr" NPP, Unit 1	OKB GP	T0+23 months	382,90	It is made the calculation analysis by the necessary number and optimum arrangement of the ICI indicators in the core considering the requirements for accuracy and reliability of determination of the in-core local parameters, and also for updating the ICIS codes and devices.
10.2	ICID modification (NTMC extension, change in the core axial arrangement of the indicators)	JSC PFC OKB GP		31,10	
10.3	Justification of representativeness of temperature control at the core outlet and determination of operational temperature restrictions	OKB GP JSC PFC	T0+23 months	312,90	To assure thermal-technical reliability of fuel rods in the core (DNBR > 1,0 under NOC and AOO), the coolant thermal inspection representativeness is performed. NTMC is applied to assure accuracy of heat rate inspection and to improve reliability of thermal control..
10.4	Procedure of determination of operational safety factors for local power distribution and permissible volume power peaking factors	OKB GP	T0+19 months	22,60	The report deals with the procedure of determination of the permissible volume power peaking factors considering the operational safety factors, and also the setpoint values of the preventive and emergency protection by the linear power rating in the cores of "Bushehr" NPP, Unit 1 when changing over to TVS-2M operation are determined
10.5	Cross-verification of the ICIS software as regards realization of protection by DNBR	OKB GP	T0+23 months	276,90	Cross-verification of the ICIS software as regards realization of protection by DNBR is performed by comparing the calculation results obtained in the ICIS SW with the results of the safety analyses. The aim of the ICIS cross-verification is to verify that the functional requirements imposed on the function of DNBR determination are met.

No	Procedure	Provided by	Term of procedure performance	Amount, thousands \$	Brief description of provided procedures
<b>11</b>	<b>Licensing</b>	<b>OKB GP JSC PFC VNIINM</b>	<b>T0+36 months</b>	<b>2944,85</b>	
11.1	Participation in development of section 4.3 "Nuclear- physical design" and chapter 15 FSAR (working materials)	JSC PFC	T0+25 months	24,70	
11.2	Development of technical materials on fuel rod and gadolinium fuel rod in chapter 4, 15 FSAR "Bushehr" NPP, Unit 1	VNIINM	T0+25 months	55,80	
11.3	Revision of FSAR (Sections 4, 5, 6, 9, 11, 12, 15) by the results of the justification performed and the agreement with the Executors	OKB GP JSC PFC VNIINM AEP	T0+27 months	773,30	Results of justification are entered into the corresponding FSAR sections. Agreement with the Executors.
11.4	Revision of FSAR by the results of the justification performed and the agreement with NPPD (Customer)	JSC TVEL OKB GP AEP	T0+29 months	250,95	Results of justification are entered into the corresponding FSAR sections. Agreement with the Customer.
11.5	Revision of FSAR by the results of agreement with NPPD and cooperation with NNSD	OKB GP AEP	T0+32 months	671,00	Rendering the consulting services during cooperation with the Iranian Regulatory Authorities.
11.6	Revision of FSAR by the results of agreement with NNSD and cooperation with the IAEA experts	OKB GP AEP	T0+36 months	722,70	Rendering the consulting services during cooperation with IAEA.
11.7	Revision of FSAR by the results of experimental operation of TVS-2M at "Bushehr" NPP	OKB GP AEP	T0+60 months	446,40	The corresponding FSAR sections are corrected by the results of experimental operation. Agreement with the Iranian Regulatory Authorities and IAEA.
<b>THE TOTAL COST FOR THE WORKS IS, THOUSANDS \$</b>				<b>25 000,00</b>	