Transactions of the 15th International Conference on
Structural Mechanics in Reactor Technology (SMiRT-15)
Seoul, Korea, August 15-20, 1999

An Assessment of Selected Safety Issues of the New Russian VVER-640

R. Kirmse, J. Rohde and H. Schaefer

Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH, Germany

ABSTRACT

As part of the activities initiated by the Bavarian-Russian Commission co-operating on peaceful use of nuclear and solar energy, GRS performed two phases of an assessment of the new Russian reactor concept VVER-640 of type V-407. The main topics of the first phase were: safety systems analyses, accident analysis, I & C and electrical systems. The second phase included the assessment of (a) function and reliability of the passive safety systems, (b) consideration of severe accident conditions in the design, and (c) digital instrumentation and control. Besides a general overview on the main results of the two phases, the paper concentrates on the function of passive safety systems and summarises the results of the assessment of the reliability of passive safety systems and of mitigation of severe accidents.

INTRODUCTION

As part of activities initiated by the Bavarian-Russian Commission co-operating on peaceful use of nuclear and solar energy, GRS received orders from the Bavarian State Ministry of State Development and Environment (BStMLU) for two phases of an assessment of selected safety issues of the new Russian reactor concept VVER-640 of type V-407.

This concept of a new-generation pressurised water reactor of medium capacity (640 MW_{el}) and advanced safety is being developed in Russia under the leadership of the Russian project organisation Atomenergoprojekt St. Petersburg and the design organisation OKB Gidropress at Podolsk. This development is not completed yet but has already reached an advanced state. The license for the beginning of construction at two sites in Russia has been granted by the Russian safety authority Gosatomnadsor. Tenders were submitted also to foreign prospective customers. The two leading organisations are actively supported in the verification and validation of novel design solutions by several Russian scientific centres, e.g. by the Alexandrov Research Institute of Technology (NITI) Sosnovy Bor, the Kurchatov Research Institute (RRC KI) in Moscow, the Leypunsky Institute of Physics and Power Engineering (FEI) at Obninsk, and the Institute ZKTI in St. Petersburg. Siemens as the major foreign partner in planning and implementation of the VVER-640 is sharing its know-how in major areas associated with plant safety as well as in overall plant design and project execution. The VVER-640 will be equipped with systems for control of normal operation and with an I&C safety system, both based on Siemens technology. The electrical systems of the VVER-640 will likewise feature Siemens-made equipment and products. For selected functions, such as secondary-system overpressure protection, mechanical components built to Siemens standards will be employed. The development of the VVER-640 is a challenge of a co-operation between Russian and German companies (1).

Since it is co-ordinating this international co-operation, the Bavarian-Russian Commission was interested in an independent safety assessment of the VVER-640. GRS received in 1993 a first order to perform an orientating safety related assessment of this reactor concept. This work included a limited number of selected topics with priority on the assessment of design basis accidents, of safety systems, of electrical engineering, and of instrumentation and control. To a lesser extent the topics core design and pressure boundary components were assessed in addition. The results of the assessment were discussed with the Russian designers before preparing the report on the first phase completed in 1994 (2).

The work was continued in performing a second phase of assessment beginning in late 1996. The working plan mutually agreed with the Russian Ministry of Atomic Energy, with Atomenergoprojekt St. Petersburg and with Gosatomnadsor, consisted of investigations and evaluations of three subjects. The first subject comprises the function and reliability of essential front line passive systems designed for the VVER-640, including the performed and planned separate effects tests and integral tests for the demonstration of the function of those systems and for the verification of the computer codes applied in the process of safety demonstration. The second subject is dealing with the consideration of beyond design accidents in the design of the concept VVER-640, in particular the procedures and systems in the preventive and mitigative area including their analytical and experimental proofs. As the third subject the presented concept of digital instrumentation and control (I&C) had to be evaluated. Taking into account the stage of the design of I&C during the period of the present assessment project, the aim of this work was changed into an support for an improved understanding of the basic requirements of international and German rules for the implementation of digital systems in the safety system of nuclear power plants. The second phase was completed in 1998 (3).

OVERVIEW ON THE MAIN RESULTS OF THE FIRST PHASE OF ASSESSMENT

The orientating safety-related assessment of the new Russian reactor concept VVER-640 was performed by GRS and its subsidiary ISTec on the basis of rules and guidelines which are in force in the Federal Republic of Germany considering also the requirements for future reactors as they were discussed in several countries. In particular, the general safety approach for new pressurised water reactors elaborated and recommended jointly by the French and German safety commissions and supported by the responsible French and German ministries wer taken into account (4). An overall assessment of the VVER-640 concept on the basis of the performed safety assessment could only be made with reservations according to the limited scope and depth of the assessment.

Most of the information on the new reactor concept VVER-640, as it was presented to GRS, was contained mainly in the third draft of the Preliminary Safety Analysis Report (PSAR) on the VVER-640, supplemented by additional explanations given in the course of several discussions with Russian experts. The degree of documentation quality of the PSAR was remarkable with regard to a clear and detailed description, an in-depth proof and consistency of the statements.

The VVER-640 belongs to the category of evolutionary-passive reactor concepts. Some details of the reactor concept can be found in another presentation of this conference (5). The well-proved features, which were adopted from the VVER-1000, are for example the materials of the safety-relevant components and pipes. The construction of the primary system with the reactor pressure vessel (RPV), the cold and hot legs connected to it at different elevations, the horizontal steam generators, the combined hot- and cold-leg injecting emergency core cooling system with own injection nozzles at the upper plenum and the downcomer of the RPV were also adopted from the VVER-1000. Operational experience, especially in the field of materials used for the secondary system, were evaluated by the Russian side and detected deficiencies were largely eliminated.

Numerous safety-related improvements as compared to the VVER-1000 were noticed, e.g. the power density of the reactor core is reduced to about 60%, the neutron fluence onto the RPV-wall is reduced by the lower power density and by the planned low-leakage loading of the core using burnable poisons, the shut-down reactivity is strongly increased by doubling the number of control assemblies, the water volumes of the primary and secondary system are enlarged, the volume of the pressuriser is enlarged, loop seals are avoided, copper-free material is exclusively applied for the secondary system, a double shell containment is foreseen, external events are considered in the design of all safety-relevant buildings, improved mitigative measures are foreseen against the consequences of primary to secondary (PRISE) LOCA by double isolation valves in the main steam lines and the arrangement of the main steam safety valves within the containment.

These improvements belonging to the evolutionary development are supplemented by an innovative approach, which entails partly replacing the active safety systems by passive safety features. The driving force behind this development is to enhance the safety by simplification and to allow for increased grace periods (not less than 24 h) after the onset of accident conditions before active intervention by operating personal is required. Hence the effect of human error on reactor safety is reduced considerably. In addition to the passive safety systems which usually are designed in 4x50% redundancy, active emergency-power supplied safety-relevant systems are provided in usually 2x100% redundancy. These active systems fulfil occasionally operational functions.

The second innovative approach concerns the implementation of precautionary measures to mitigate the consequences of severe core melt accidents, taken into consideration already in the design, e.g. the transition of core melt at high to low pressure, the retention of core melt inside the reactor pressure vessel (RPV) by ex-vessel cooling.

The results of the overall assessment performed during the first phase were:

- By VVER-640, a plant concept is presented for the first time on the basis of the new Russian safety principles OPB-88 (6) and PBJa RU AS-89 (7), even exceeding clearly these requirements in some cases. A considerably higher safety level is striven for in comparison to existing plants, particularly with regard to avoiding the evacuation of inhabitants after severe core-melt accidents. Here, OPB-88 requires a probability of $< 10^{-7}$/reactor year.
- The experimental testing programme with regard to the function of the new passive safety systems must be continued. The planned experiments at the large scale integral test facility KMS are required before commissioning of the first reactor plant in addition to the separate effects tests and small scale system tests which partly have already been performed.
- The assessment of the reliability of the passive systems needs to be performed. Several possibilities exist, e.g. probabilistic safety analyses using a well proven data base from available experience and from specific long-term experimental investigations, on-line control or functional test of the passive safety systems during power operation. Influences due to corrosion which reduces the reliability of certain components or which can cause flow obstructions or even clogging by deposits of chemicals in systems which are not circulating have to be considered.

- The safety relevance of incidents which are initiated by inadvertent actuation of passive safety systems as well as by leaks and breaks in these systems needs to be investigated. The consequences of an inadvertent opening of the JNB valves, which are provided for the pressure relief of the primary circuit into the fuel pool in case of LOCA, are considered as extremely serious in particular if this would occur at high system pressure. The Russian experts assured that such an inadvertent opening is practically excluded by means of the construction. The proof has not been shown as the construction is considered confidential. For this reason, an assessment on this point has not be performed.
- The diagnosis of the plant state during an incident is generally more difficult when using passive systems instead of active systems. The concept of the reactor protection system must be adapted to this characteristics. It was found that the list of the safety-relevant instrumentation did not contain the parameters required for identifying the function of the passive safety systems.

Several further detailed conclusions were drawn for each of the selected safety issues at the end of phase 1. They are described in (1). These recommendations concerned essentially the completion of safety demonstration, the extension of safety analyses, e.g. into the area of low power states, included proposals for additional systems in order to obtain an improved balance of the safety system functions, e.g. an active emergency feed water system.

RESULTS OF THE SECOND PHASE OF ASSESSMENT

Three selected issues, the function of passive safety systems, the reliability of passive safety systems, and the mitigation of severe accidents are dealt with in this chapter.

Demonstration of the Function of Passive Safety Systems

The assessment of system functions is based on three reports about the verification of computer codes PARNAS (7), SPAS (8), and SPOT (9), These codes are developed by different Russian institutes for the simulation of the system function of

- depressurisation of the primary system and natural circulation modes in the connected system of pools (fuel pool and emergency pool) and reactor pressure vessel (system JNB),
- containment heat removal under single- and two-phase natural circulation conditions to the external tanks BAOT located at high elevation (system JMA), and
- heat removal from the secondary coolant system via the steam generators and heat exchanger/condensers into the tanks BAOT (system LBT).

Two main reasons were decisive for the development of new computer codes for the simulation of passive systems instead of adapting proven codes or modifying suitable ones for the purpose of simulating the involved processes. At first the new codes can be better equipped with appropriate interfaces which allow the formation of an integrated code package for the simulation of the interrelated physical processes during an accident. This concerns also further codes besides the three codes mentioned above, e.g. the code KUPOL which simulates the heat and mass transport processes in the containment. The conventional part of the simulation of the accident sequences are analysed by modified versions of existing codes which are provided also with the necessary interfaces for the integration into the code package. An example is the simulation of the thermal hydraulic behaviour inside the primary coolant circuit. Modified versions of the existing codes DINAMIKA and TETSCH are applied for this purpose. The second reason is the opportunity to individually tailor the code with specific models for the processes involved in different parts of the circuit. These processes are characterised by low flow parameters (pressure, mass flux, heat flux, steam quality) in natural circulation conditions and by mixing processes in large tanks. An additional advantage of developing new codes is the unlimited availability of the code by the designer including industrial application without restrictions, i.e. there is no dependence on foreign licenses. Nevertheless calculations with existing and extended foreign codes, e.g. RELAP5/Mod2 (SCDAP), were also performed for reasons of comparison.

The simulation of the physics of the involved processes by means of the codes PARNAS, SPAS, and SPOT follows in general a realistic approach. A major part of the experimental work for the purpose of code validation and for the demonstration of the function of the passive safety systems has been conducted already. This work is continued at present and in the near future. At the end of this process a comprehensive and well validated code package will exist which will serve for the purpose of simulating the entire chain of heat removal from the core to the ultimate heat sink during different accident scenarios. Complex thermal hydraulic processes are involved in these chain. In case of a LOCA or after forced depressurisation: heat removal from core by natural circulation to the water pools surrounding the reactor, mixing and evaporation of water in these pools into the containment atmosphere, heat and mass transport within the containment, passive heat removal from the containment to the external water tanks, located outside the containment at high elevation, and mixing/evaporation of water in these tanks into the atmosphere as the ultimate heat sink. A parallel and partly diverse chain of heat removal from the core exists with heat transport at any primary system pressure by natural circulation to the steam generators, passive heat removal from the steam generators to the heat exchangers located in the same external water tanks and from there again to the ultimate heat sink. These complex and partly interacted processes are to be simulated and validated by means of suitable experiments. The validation concept corresponds with the state of the art. The progress obtained so far promises that the final goal will be achieved.

At first the codes were exposed to a systematic process of testing. An contrast to code validation by using experimental results for comparison, code testing comprises numerical tests, variation and optimisation of local resolution and time integration steps, comparison of the numerical solution with known analytical solutions, and comparison the results with calculations obtained with other codes. The testing of the codes PARNAS and SPAS has been completed successfully. For SPOT this process is still in progress. The codes PARNAS and SPAS obviously fulfil the requirements on robustness, stability and convergence. The simplifications which contributed to this features seem to be adequate, e.g. mainly a 1-D description, neglecting the dissipation compared to the transferred thermal energy and neglecting the diffusion, stationary values of phase slip, and an always saturated steam phase.

GRS proposed extensions of the scope of simulation for codes, e.g. for PARNAS to include the local variation of boric acid concentration, and to consider the influence of non-condensable gases on the process dynamics.

The validation process for the three codes is in progress. A higher progress has been already achieved for the codes PARNAS and SPAS.

This process is illustrated now taking the validation of PARNAS as an example. Experiments in three of four test facilities listed in the verification matrix have been performed up to now.

The analyses with code PARNAS of tests from facility A show general good agreement between calculations and experiments. Low power tests typical for decay heat, are missing and should be added. An interesting result from facility BASIN was the confirmation of density stratification with a moving horizontal temperature front when injecting hot water in vertical bottom-up direction. It is recommended to extend the experimental programme by investigating also pool mixing with inclined direction of injection and with two-phase flow entering the pool from the primary circulation circuit. This would cover also condensation processes in the pool. The processes investigated so far were calculated by PARNAS and acceptable agreement with the measured results was achieved. Limitations of the integral effects test facility IST(EZ) are the relative small size and the reduced length of the core, about half of the real core. The investigated sequences did not include the initial phase of the accident. The inclusion of smaller break/leak sizes than the double ended guillotine break into the experimental programme is recommended. A remarkably good agreement between the calculations by code PARNAS with the experiments was obtained. Also the rather complex oscillatory behaviour during the period of two phase flow in the core and sub-cooled fluid in the pool was calculated well.

Name	Type	Scale		State	Task of verification experiments
		H	V/P		
"A"	SETF	-	-	c	thermal hydraulics of two-phase flow in vertical heated channel at low coolant parameters (0.1 MPa, up to 500 kg/m²/s)
"BASIN"	SETF	1:10	1:1000	c	non stationary heat transfer, mixing and stratification in pools, coolant entrainment of non isothermal submerged jet at low coolant parameters (0.1 MPa, up to 0.2 kg/m²/s in emergency pool)
"IST" (EZ)	ITF	1:1	1:1050	c	non stationary processes of residual heat removal from reactor core by means of pool cooldown during the final stage of LOCAs
"KMS"	ITF	1:1	1:27	p	comprehensive safety demonstration of the function of all safety systems including their interaction during accidents, reducing the impact of scaling

SETF = Separate Effects Test Facility
ITF = Integral Tests Facility
H = Height (Elevation)

V/P = Volume / Power
c = conducted
p = planned

The experiments at the KMS test facility, the fourth facility of the validation matrix, have not yet been conducted. The construction of this test facility is in progress. GRS considers the completion of the envisaged validation programme by including the tests at KMS as meaningful and necessary. The reasons for the necessity of such experiments are five-fold and equally applicable for the validation of the other codes. The experiments at KMS will serve

- to reduce essentially the gap between the scaled down facilities with factors of 1:1000 (in case of PARNAS) respectively 1:110 (in case of SPAS and SPOT) and the real reactor,
- to include the simulation of interactions between the reactor coolant circuits , the containment and the different passive systems,
- to include the simulation of interactions between passive and active systems,
- to provide data on the integral accident behaviour of the entire chain of heat removal from core to the ultimate heat sink, and
- to model the complete time period of the accident from accident initiation up to safe shut-down state; these data are required for the validation of the integrated code package .

Further recommendations were raised by GRS in particular with regard to the continuation of the still ongoing process of validation of the code SPOT.

A final proposal is aimed at the evaluation of code uncertainties.

Reliability of Major Passive Safety Systems

The assessment was concentrated on distinct system functions, namely the pressure driven coolant injection from accumulators into the reactor (system JNG), the gravity driven injection from ECCS pools (system JNK), the passive containment heat removal system (system JMA), the depressurisation of the primary circuit (system JNB), and the secondary passive heat removal system from steam generators (system LBT).

These passive safety systems as they are designed up to now for the NPP concept with VVER-640 can be grouped into three types:

- passive safety functions isolated by double check valves (in series) from the primary cooling circuit, namely the injection systems JNG and JNK,
- passive safety functions aligned by opening /closing of hydraulic or pneumatic valves, namely the systems for depressurisation of the primary circuit JNB and the steam generator heat removal system LBT,
- passive safety functions aligned permanently without the need to open any valves, namely the containment heat removal system JMA.

The reliability of these systems depends to a large extent on the type and performance of such valves, see also (11).

The designer has applied generic data for the estimation of the common cause failure (CCF) probability. GRS proposes a further plant-specific validation of these data. It is recommended that the Russian side creates and establishes an own empirical basis for CCF assessment.

The reliability assessment of the containment heat removal system (JMA) is proposed to be improved or expanded. Valuable steps towards demonstrating the order of magnitude of the failure probability at demand could be the check of the data base (12) and related references, as well as the assessment of hydraulic robustness and model uncertainties of SPAS. It is recommended to improve by minor design changes the fault tolerance of the redundant trains of system JMA. This would allow not to lose an entire train by one local leakage.

Up to now the reliability evaluation was related to the failure probability of separated passive system functions. It is proposed to continue the work by developing an integrated view on the systems for core melt prevention. This can be performed by the extension to event tree analysis. For selected postulated initiating events the active and passive system functions should be combined and the dominant event sequences regarding core melt frequency deduced and quantified with regard to their frequency. Finally, a sensitivity analysis should be performed with regard to uncertain parameters of the reliability estimates of the passive safety systems.

Mitigation of Severe Accidents

Essential improvements have been made to extend the „defence-in-depth“ by design. They imply a significant reduction of the probability for those type of accidents, which could lead to core degradation up to core melting. Furthermore, provisions have been made to exclude or to drastically limit potential sources for the release of radioactive materials to the environment. In this context selected topics concerning accidental conditions which could lead to early containment failure and severe accident mitigation measures like

- prevention of core melt at high pressure in PCS
- coolability of molten core materials in the lower RPV head by ex-vessel cavity flooding
- RPV failure mode
- melt coolability in a core catcher device below the RPV
- interaction of molten core materials with water (→ steam explosion)
- formation of combustible gas mixtures in the containment atmosphere and
- containment heat removal

have been discussed and assessed. The technical solutions, chosen by the designers are supported by an extensive experimental research programme and related computer code development, validation and application. To assure the function of the technical solutions, still existing uncertainties should be reduced further by the ongoing research activities. For this some proposals and recommendations were made by GRS to further investigate solutions for the VVER-640 conception.

REFERENCES

1. Donath, R., Kirmse, R. and Kuchtevich, I., "WWER-640, Ein neuer Druckwasserreaktor mittlerer Leistung mit einem fortgeschrittenen Sicherheitskonzept (VVER-640, A new Pressurised Water Reactor of Medium Power with an Advanced Safety Concept)," *Proc. of Jahrestagung Kerntechnik '97, Berichtsheft der Fachsitzung "Verbesserung der Sicherheit von WWER-Reaktoren"*, pp 19-50, Bonn, July 1997
2. Kirmse, R., Langenbuch S., Mulka, B., Wenk, W. and Zabka, H., *Orientierende sicherheitstechnische Bewertung des Konzepts des neuen russischen Kernkraftwerkstyps W-407 (Orientating Safety-Related Assessment of the Concept of the New Russian Nuclear Power Plant Type V-407)*, Report GRS-A-2130, March 1994
3. Kirmse, R., Mulka, B., Rohde, J., Schäfer, H., Schnürer, G., *Safety–Related Assessment of Selected Features of New Russian NPP with VVER-640/V-407, Continuation of Previous Assessment*, Report GRS-A-2642, December 1998
4. BMU (German Federal Ministry of Environment and Natural Protection), *Gemeinsame Empfehlungen von RSK und GPR für Sicherheitsanforderungen an zukünftige Kraftwerke mit Druckwasserreaktoren (Joint Recommendation of RSK and GPR for Safety Requirements for Future Nuclear Power Plants with Pressurised Water Reactors)*, Bundesanzeiger (Federal Gazette) 20.11.1993
5. Avrov, A., Donath, R., Ermolaev, V., Gorbaev, V., "Advanced VVER-640 Development and Construction Programme", *Proc. of the 15th International Conference on Structural Mechanics in Reactor Technology*, ID No. S1-A2-RU, Seoul, Korea, August 1999
6. Gosatomnadsor of USSR, *General Directions to Ensure the Safety of Nuclear Power Plants*, OPB-88, PNAE G-1—011-89, Moscow, 1989
7. Gospromatomnadsor of USSR, *Rules of Nuclear Safety of Reactor Plants of NPPs*, PBJa RU AS-89, Moscow, 1990
8. Chabenski, V. et al., *"Substantiation of Computational Techniques and Verification of Code "PARNAS" for Calculation of Non-stationary Processes During Residual Heat Removal from the Reactor through the Emergency Pool with Depressurisation of Primary Circuit of NPP of New Generation with VVER-640/V-407"*, NITI Report in Russian, Sosnovy Bor, 1994
9. Beslepkin, V. et al., *"Code for the Calculation of Non-stationary Processes in the System of Heat Removal from the Steel Shell of the Containment of NPP with VVER-640/V-407, Verification Report"*, SPbAEP Report in Russian, St. Petersburg, 1996
10. Atomenergoprojekt: "Information about Code and Analytical-Experimental Substantiation of the System of Passive Heat Removal from Steam Generators in NPP with VVER-640/V-407", SPbAEP Report in Russian, St. Petersburg, 1997
11. Makihara, Y. et. al., "Passive safety systems for PWR using simplified PSA methodology", Proc. of ARS '94 (Int. Topical Meeting on Advanced Reactors Safety), Vol. 1, pp 95, Pittsburgh U.S.A., April 1994
12. Nyman, R. et. al., *Reliability of Piping System Components, Vol.1: Piping Reliability - a resource document for PSA application*, SKI Report 95:58, December 1995