Post-Fukushima PSA Development for New Reactors in Russia

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Abstract: The Atomenergoproekt engineering company, general designer of nuclear power plants (NPP), has been performing probabilistic safety assessments (PSA) for six NPPs in design. They belong to generations of advanced plants constructed or planned to be constructed in Russia, India, Turkey and Bulgaria including Generation 3+ plants.

The main advantage of the NPP with a new generation reactor compared with Russian designs of previous generations is the use of advanced equipment and introduction of additional passive safety systems in a combination with conventional active systems. Implementation of diversity increases likelihood of the safety function fulfilment. The new VVER plants having inherent safety features are addressed in terms of their anti-Fukushima properties.

The Fukushima accident has the impact on the PSA development itself. This is a challenge to PSA developers. Lessons learnt from the Fukushima event are to direct additional efforts to some points within the PSA development which are discussed.

The paper is aimed at sharing some issues and experience gained from design modifications and PSA development for new advanced plants.

Keywords: new reactors, design, PSA, Fukushima accident.

1. INTRODUCTION

JSC «Atomenergoproekt» is an engineering company, general designer of nuclear power plants (NPP). For years of its existence the company has developed the designs of the majority of the NPPs on the territory of Russia, Eastern Europe and CIS countries. The company performs the complete scope of works and services in the area of the construction of the VVER type plant family. More than 1, 350 reactor-years of the VVER NPPs operation confirm theirs high safety level.

Several different advanced VVER plants including Generation 3+ ones are now in design for some new sites. Provision of a required level of inherent reactor safety and effectiveness of protective barriers on the way of radioactive substances propagation are justified by deterministic and probabilistic analyses. The Atomenergoproekt has been performing probabilistic safety assessment (PSA) for six NPPs in design. They are constructed or planned to be constructed in Russia, India, Turkey and Bulgaria. Plants are located at sites having various geological and environmental conditions; therefore it is essential to carry out plant specific safety evaluation including PSA for any unit.

The Fukushima disaster made the future of the nuclear energy questionable. However the situation varies from country to country. Fortunately the Russian advanced VVERs have anti-Fukushima design features to cope with such a type of accidents. These design decisions were made before the Fukushima event that confirms correctness of the chosen path. Nevertheless, the accident highlighted the line of further development in the area of both NPP design and PSA. The paper is aimed at sharing some issues and experience gained from design modifications and PSA development for new advanced VVER plants.

2. FEATURES OF NEW RUSSIAN PLANTS

The main advantage of the NPP with a new generation reactor compared with Russian designs of previous generations is the use of advanced equipment and introduction of additional passive safety systems in a combination with conventional active systems. Execution of safety functions can be performed by either active or passive safety systems independently of each other. Application of a complex of active and passive safety systems to cope with design basis accidents (DBA) and beyond design basis accidents (BDBA) is beneficial because implementation of diversity increases the likelihood of the safety function fulfilment. It is essential that modern safety requirements are considered, including those issued by EUR, IAEA, etc. On the other hand, the advanced VVER design is evolutional and has a high reference.

The main distinguishing features of the advanced VVER design in terms of safety are as follows:

- Use of different functional and/or structural principles in systems involved in each separate safety function. Used are mutually redundant safety systems of «active» and «passive» principles of action that ensure protection against common cause failures and allow for increase of safety system reliability factors by several orders. To reserve all the active safety systems the passive systems performing the safety function in the full scope are provided. For example, innovative technical solutions provide emergency heat removal via secondary circuit both in active and passive modes without limitation in time.

- Use of «active» safety systems trains for normal operation functions that allows for increase of safety system availability level and provides for additional protection against common cause failures, excluded are latent failures being a main reason of system non-availability in a standby mode.

- Providing low sensitivity to human errors that is assured by means of increase of an automatic control level of systems (exclusion of human actions) in case of occurrence of some DBA and, in particular, in the case of primary-to-secondary leaks; and of the use of passive systems, which activation does not require participation of operating personnel.

- Double-shell containment with controlled annulus. The inner shell, internal prestressed reinforced concrete shell (designed for 0.5 MPa abs) with steel lining is designed to withstand forces and conditions arising from DBA and BDBA. The core catcher is mounted inside the inner shell. The annular space is maintained at underpressure at all normal operation modes and accidents including station blackout. The outer shell is designed to protect against natural and man-made loads and together with the inner shell forms the annular space for collection, monitoring and cleaning of radioactive substances under accidents. Some plants are equipped with filters to reduce pressure in the containment in case of a long-term accident.

- Corium catcher is designed for reception and retention of the corium liquid and solid fragments, core fragments and reactor structural materials, heat transfer from the corium to cooling water, and minimization of hydrogen egress.

Three main innovative passive safety systems designed to enhance safety are the following [1]:

- Second stage hydroaccumulators are used for passive flooding of the reactor core in response to a primary pressure decrease below 1.5 MPa. The system is comprised of four trains containing 1000 tons of boron acid solution with concentration 16 g/kg.

- Passive heat removal system is used for long-term removal of residual heat from the reactor plant in case of the loss of all alternative current power sources (station blackout). System operation time is unlimited. There are four independent circuits of secondary coolant natural circulation - one for each circulation loop. The system is operable when the primary circuit is both sealed and leaky.

- Annulus passive filtration system performs controlled removal of a steam-gas mixture from the annulus in case of an accident associated with the loss of all alternative current power sources. There is no need in power supply for its operation because heat energy of the passive heat removal system is used as a driving force. Purification efficiency of specially designed filters for Caesium -137 aerosols and Iodide-131 in both aerosol and gas substances is 99.9 %. As a result, any uncontrolled release from the annulus to atmosphere (via the outer containment) is excluded.

The operability of all the innovative passive safety systems was justified by experiments at large-scale experimental facilities and results of passive safety systems tests during commissioning of the Kudankulam NPP constructed in India.

3. ANTI-FUKUSHIMA PROPERTIES OF NEW RUSSIAN PLANTS

Russia keeps building nuclear plants despite the impact of the Fukushima accident on the nuclear power sector. However nuclear community has come under public pressure to review the nuclear program and improve safety. As the first response, two months after the Fukushima disaster the Russian authorities ordered stress tests of all national NPPs. The stress tests focus on the areas highlighted by events in Japan. The state of the plants was thoroughly analyzed taking into consideration the Fukushima factors to see how operating and constructed Russian plants would perform in the face of external threats from earthquake and flooding that result in a long-term loss of the grid, ultimate heat sink, or both. Issues connected with the management of severe accidents were also under consideration. In performing the stress test engineering evaluation, deterministic analyses, and the results of PSA performed were used.

Although Fukushima like consequences are mainly analyzed based on the deterministic stress tests a PSA is also used for mapping the accident to the Russian plants. Post-Fukushima measures being implemented in Russia include the provision of autonomous mobile diesel generators and pumps, extension of battery capacity, tracing of new cable lines, involvement of fire facilities in the safe shutdown process, etc. The PSA is used to support the cost benefit evaluation of severe accident management options and support the ranking, justification, and licensing of plant upgrades based on the level of risk reduction associated with each alternative.

The safety analyses for accident conditions like Fukushima showed that advanced VVER plants can withstand such conditions [2]. Initiating events and failures that took place at Fukushima are as follows:

- Seismic impact. The site-specific VVER designs are developed basing on the value of maximum earthquake intensity approximately corresponding to 7-8 points on the MSK-64 scale once every 10, 000 years. In order to provide the possibility of construction of the plant at the sites characterized by higher parameters of seismic impact (8-9 points on the MSK-64 scale), the possibility of design updating without sizeable changes of buildings and pipelines layouts is envisaged.

  Some seismic PSAs had been performed before the Fukushima disaster and gave valuable insights used for safety evaluation. One of them was carried out as a part the PSA for Belene NPP in Bulgaria at the design basic stage [3]. The design basis earthquake ground motion for the Belene site is defined in terms of the peak ground horizontal acceleration as 0.24 g. The total core damage frequency is calculated for internal initiating events, internal and external hazards as 5.1E-07 per reactor a year. The seismic-induced core damage frequency is quantified to be 7.1E-9 per reactor a year. The main contributor (58%) is a loss of off-site power accompanied by dependent ruptures of main steam lines due to the destruction of the turbine hall and seismic-induced failures of reliable power supply preventing steam generator isolation and «feed and bleed» actions. The second contributor (40%) is related to seismic-induced failures of the passive heat removal system.

  A specific point that should be addressed for any plant in design is lacks of some design/operating information, especially for a PSA performed during the conceptual or basis design stage. Such a PSA may contain substantial uncertainties. The lack of design information affecting PSA development may be associated with incompleteness of the design development that is typical for interim design stages. Confirmatory walk-downs aimed at supporting the screening analysis results are also impossible to conduct at pre-construction stages. To dea1 with this issue bounding technologies are used. However this approach sometimes contradicts to the «best estimate» one recommended by regulatory guides. For instance, general criteria for the assessment of the seismic margin capacity permitted in plant design are quite conservative. On the other hand, the seismic margin capacity of the Belene plant estimated in more realistic assumptions corresponds to a higher maximum earthquake intensity at least increased by a factor of 1.4.

- <u>Loss of normal and emergency (diesels) electric power supply (also referred to as station black-out).</u> As mentioned above the Russian concept of safety provision is based on application of safety systems utilizing different principles of operation: «active» and «passive». All safety functions are performed by active systems, and if they are unavailable – by passive ones. In case of the station blackout the function of residual heat removal from the core is carried out by the passive heat removal system during an unlimited time period. As a matter of fact, originally this system was designed to cope with blackout accidents. As a result, contribution to the core damage frequency from such accident sequences is negligible.

  Heat removal from the spent fuel storage pool performed in the boiling mode is also important. The water reserve in the pool is sufficient for 10-20 days depending on the heat generation by spent fuel. Subsequent feeding of the pool for next 10-20 days can be done from the second stage hydroaccumulators installed inside the containment or hydroaccumulators located outside the containment. The pressure rise rate inside the containment due to pool boiling is such that in the worst case after 10 days the pressure will reach the design value for the containment. As a management measure for limitation of pressure rise one can use available dumping lines provided for carrying out containment tests or containment filters.

  The PSA results show that event timing should be analyzed, i.e. a transient followed by a blackout can lead to greatly different scenario than accident sequences directly started with the blackout.

- <u>Failure of ultimate heat sink</u> (sea water). It was caused by plugging of channels and pipes by debris brought by tsunami at the Fukushima plant. Independently of the cause of the ultimate heat sink loss at Russian advanced plants residual heat from the core is removed by the passive heat removal system to outside atmosphere air. Therefore, the performance of the system is not affected by the absence of other heat sinks such as service water, sea water, cooling ponds, etc. The heat-exchangers are mounted at the height around 40 m and are protected by civil structures. These design decisions exclude or minimize their damage by flooding and other natural or man-made events.

  The heat removal from the spent fuel storage pool is a separate problem and discussed above.

- <u>Hydrogen generation</u> by zirconium-steam reaction, release into the containment building, hydrogen explosion and damage of the containment. Inside the inner shell of the containment of Russian plants passive hydrogen recombiners are installed that exclude rising of hydrogen concentration to hazardous values in all accident modes including the beyond design basis ones. The calculations of the ex-vessel phase accident performed to support severe accident analyses and PSA Level 2 show that the maximum value of hydrogen concentration is about 6% given a considerable concentration of water vapour. Therefore, the possibility of both explosions and damage of the containment is as low as possible. As a result, the release of radioactive products to environment is also negligible. The additional protection against radioactive releases is provided by maintaining underpressure in the annulus space by active and passive annulus filtering systems.

- <u>Damage of the basement part of containment and activity escape to environment.</u> In the bottom part of the containment the special device for localization and cooling of the corium in case of a hypothetical accident that may result in core damage is placed called the core catcher. The core catcher allows keeping the containment integrity and excluding release of radioactive products to environment even under the hypothetical severe accidents except for high pressure melt ejection cases. The frequency of accident sequences leading to molten core material being ejected from the reactor pressure vessel under high pressure is estimated to be 2.1E-08 per year.

Following the Fukushima disaster more burdened emergency conditions compared to Fukushima are also investigated. The example is the loss of all alternative current power sources (blackout) combined with break of the maximum diameter primary side pipeline (large LOCA). According to PSA results the large LOCA is the dominant contributor among internal initiating events. Depending on grid properties the large LOCA followed by the plant trip may cause some power supply troubles. That is why the above mentioned BDBA was investigated in detail. Under the stated accident conditions the removal of residual heat from the core is carried out by joint operation of the passive heat removal system and the second stage hydroaccumulator system. It is found that the NPP autonomy (absence of the core damage) in this mode is determined by the water reserve in the second stage hydroaccumulators (from 26 up to 280 hours depending on a break size). Using of water reserve in the spent fuel storage pool makes it possible to provide autonomy not less than 72 hours for breaks of any size. After exhausting of water reserves in the second stage hydroaccumulators and spent fuel pool, if normal or emergency power supply is not recovered, feeding of the reactor and spent fuel pool can be provided from additional hydroaccumulators located outside the containment (for some designs) or pumps powered by mobile air-cooled diesel-generators. The use of the mobile air-cooled diesel-generators is an example of a post-Fukushima modification. The reason for air cooling is the assumption that water as a coolant source for diesels may be unavailable.

4. PSA TRENDS

The Fukushima accident in Japan has the impact on the PSA development itself. This is a challenge to PSA developers. The following lessons learnt from the Fukushima accident are to direct additional efforts to the following points within the PSA development [4]:

- <u>Analysis of correlated internal/external events.</u> Following the Fukushima accident in Japan the main attention has to be paid to external hazards especially for the plants having inherent safety features. In Russia this work was done before the accident and is extended after that. Consequences of the Fukushima accident show that combinations of hazards may be significant for risk. As a matter of fact, a multiple hazards analysis should involve a systematic check of the dependencies between all internal and external hazards. It is evident that combinations of hazards may have a significantly higher impact on plant safety than each individual hazard considered separately. On the other hand, the frequency of combined events may be comparable to that of the individual hazards. Regarding experience from the Japan accidents occurred on March 11, 2011, at least, three types of hazard dependencies can be found. First, a seismic hazard induced another external event (tsunami). Secondly, a fire (internal hazard) occurred in the turbine section of the Onagawa NPP following the earthquake (external hazard). Thirdly, a flooding caused by recovery actions discharging a large amount of water kept safety system pumps disable at the Fukushima plant. Definitely the analysis of combined internal/external events is supposed to be extremely time consuming.

- <u>Seismic PSA.</u> Following the Fukushima accident special attention is paid to the development of seismic PSA. A full scope seismic PSA for the operating Balakovo Unit 1 had been finished by the end of 2011. Many important findings, especially found during plant walkdowns, are now under regulatory review. The problem is the seismic PSA is a very time consuming task. Our main task now is to elaborate the seismic PSA methodology already applied to analyze the design of new plants constructed in seismic regions like India, Turkey and Bulgaria. For instance, some delayed consequences such as a seismically induced loss of diesel fuel pumps may become important when considering a long-term loss of off-site power. Severe aftershocks and their impact on weakened facilities or equipment could also be important.

- <u>Delayed consequences.</u> For new designs that provide the features to delay spent fuel damage, consideration of extended fuel storage and delayed consequences is necessary. Some delayed consequences may become important when considering a long-term loss of off-site power. It is clear that following the loss of off-site power spent fuel cooling pumps need to be powered by essential diesel generators even if water inventory is sufficient to remove residual heat for several days by evaporation. Other important items may be resources shared between the spent fuel pool and reactor core or among several units in case of a long-term or/and multi-unit accident.

  Regarding a standard PSA to be performed to quantify the core damage frequency much longer mission times for components of, at least, three days need to be considered in the PSA for a new plant design in comparison with the usual time of 24 hours. Accidents with durations greater than 24 hours or procedures for events lasting longer than 24 hours are of utmost importance. For instance, as mentioned above, Russian advanced VVERs have low pressure passive hydroaccumulators, called the second stage, with a long-term capability. During this time the active emergency core cooling is unnecessary. Therefore, in this case a 24-hour mission time is inadequate to quantify actual contribution to the core damage frequency from loss-of-coolant-accidents (LOCA). In general, the calculations for accident sequences should be extended beyond the time point when the reactor has been tripped and other safety systems actuated, until a long term stable state has been reached. On the other hand, a greater mission time can be used for recovery actions and repair usually ignored in the PSA for existing plants. As an example, the core damage frequency at the Belene NPP was estimated to be of 1.93E-07 per year for internal initiators and 24-hour mission time without considering any post-accident recovery actions. As an alternative 720-hour mission time was considered taking into account possible recovery actions after 24 hours. The core damage frequency increased by 1.2E-08 per year and was quantified to be 2.05E-07 per year [5].

- Investigation of multi-unit accidents. Historically some multi-unit accidents were considered in Russian PSAs such as a loss of off-site power at several units. Some dependencies like shared diesels, switchyards, transformers, heat exchangers, etc. are evident and usually analyzed while performing a PSA. Particularly important are subtle interactions that have the potential to result in the simultaneous unavailability of safety systems at adjacent units following a long-term accident. Common cooling water and diesel fuel inventory are of utmost importance. Other important points are manager reliability analysis in case of the multi-unit accidents as well as availability of spare parts and repair staff for several units simultaneously. Allocation of the available resources may be a very useful PSA application. For a multi-unit site, the potential spreading of a hazard like seismically induced fire to other units should also be considered in the analysis.

- Availability of an extended list of the procedures for severe accident management. It appears that Japanese operators were not trained to cope with the accident occurred. Accident management and emergency recovery human factors should be under consideration in the PSA. Main, alternative or additional systems and measures should be evaluated by PSA tools and implemented in the accident management procedures with the purpose of restoring the function of safety related systems and preventing degradation of events into more severe accidents.

5. CONCLUSION

We believe that in the design of Russian advanced NPPs suggested for construction the full scope of technical solutions providing NPP safety reduces the release of radioactive products to the environment to the «as low as possible» level in case of external natural and man-made events combined with internal initiating events and additional failures. The main goal is our belief must be supported by the comprehensive, consistent and transparent PSA of a high quality providing insight into contributions to safety of different aspects of the plant that can be understandable for everybody.

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