USING PROBABILISTIC SAFETY ASSESSMENT FOR MAKING DECISIONS ON IMPROVING THE SAFETY OF IN-SERVICE AND NEWLY DEVELOPED NUCLEAR POWER STATIONS WITH WWER REACTORS

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Abstract

In this paper the current practice of using a PSA for making the decisions on improving safety of operating and newly designing NPP with WWER reactors is briefly described.

1. INTRODUCTION

At present, more than 30 power-generating nuclear power units with VVER-440 and VVER-1000 reactors are in service in Russia, Ukraine and in Eastern Europe countries. Starting from late 1980s, new designs of nuclear power stations with VVER-1000 reactors have been developed for the second stage of Novovoronezh NPP, Kudankulam NPP in India, Tyanwan NPP in P.R. China, and the project of completion of Bushehr NPP in Iran.

For developing and making decisions on improving the safety of in-service and newly developed NPPs, methods of probabilistic safety assessment (PSA) are widely used.

In this paper, the results of applying PSA for making decisions on improving safety of in-service and newly developed NPPs with WWER reactors are briefly described.

For in-service NPPs, the results of PSA are used for the following purposes:

– To develop and assess the efficiency of measures on upgrading the NPP;
– To develop the routines of carrying out periodic tests on safety systems;
– To develop the procedures for control of beyond-the-design basis accidents (BDBA) management.

For new designs of NPPs, PSA is used as a tool for making decisions on determining the main engineering principles and measures that are required to attain a safety level that is qualitatively new compared with that of the in-service NPPs.

2. THE DEVELOPMENT OF MEASURES ON UPGRADING UNIT 3 AT THE NVNPP

Measures on upgrading Unit 3 at the NVNPP were developed using the results of the PSA that was developed within the framework of Project 1.4 of the TACIS-91 Programme and also the NOVISA Project.

The estimated value of core damage frequency as per the Project 1.4 of the TACIS-91 Programme amounted to 1.8 E-3 1/year. The main contributors to this value were accident sequences (AS) associated with the failure to remove heat through the secondary coolant circuit under conditions of initiating events (IE) with transients. Based on these results, recommendations were made on upgrading the heat removal systems through the secondary coolant circuit, including:

– An additional emergency system for supplying feed water to the steam generators;

– A portable feed water pump driven by a diesel;
– A portable diesel-generator.

According to the results of the PSA made within the framework of the NOVISA Project, the introduction of these measures, will decrease the core damage frequency to 0.97 E-4 1/year. Based on the results of the PSA performed for the NOVISA Project, the additional upgrading measures were determined to further decrease the core damage:

– To carry out an analysis of processes associated with leaks from the primary coolant system to demonstrate that the temperature of water in the B-8/3 tank does not exceed 75^{0}C, which is the maximum permissible temperature for the pumps of this system. The introduction of this measure will reduce the core damage frequency on 3.0 E-5 1/year.
– To have more efficient isolation of leaks from the reactor coolant system, it is recommended to replace manually operated gate valves on the blowdown lines and on the lines for returning the blowdown water of the reactor coolant system (3R-9/1÷6 and 3R-11/1÷6), and to introduce an automatic signal for closing them generated when there are leaks from the reactor coolant system. With the introduction of this measure, the core damage frequency will be decreased on 2.0E-5 1/year.
– To improve the functional reliability of the spray system, an automatic signal should be introduced for opening the gate valves on the line supplying water to the spray heat exchangers 3T-10/1 and 3T-10/2 – the same as the signal for opening the gate valves 3B-20 and 3B-20A on the recirculation line of the spray pumps to the emergency boron storage tank B-8/3.
– To prevent core damage during LOCA because of clogging of the confinement sump with the heat insulation of the reactor coolant pipelines, the design of the sump should be altered.
– The implementation of the above-mentioned measures will allow the core damage frequency to be reduced to 3.0 E-5 1/year.
– A further decrease in the total core damage frequency, which can be attained by implementing the above-mentioned measures, is associated with coping with the effects due to such dominant contributors as beyond-the-design-basic accidents with large breaks of the reactor coolant pipelines or steam generator headers. Because of this, the scope of activities on assessing and substantiating the safety must include a determination of the frequencies of large breaks of pipelines using probabilistic strength models. In developing these models, a leak before break concept should be taken into account.

To improve the reliability of removing heat through the secondary circuit, the following measures are also suggested:

– To make a provision for automatic actuation of the emergency feed water supply system.
– To introduce automatic signals for closing the sectionalizing gate valves of the MSH 3(4)P-123, 3(4)P-624 to isolate the half-sections of the MSH during leaks of steam headers. It is desirable that these gate valves were closed before the gate valves on the connection lines between the MSH and the SG steam lines 3P-10…60.
– To introduce automatic signals for closing the gate valves on the steam lines of SGs 3P-11…61, similar to the signals for gate valves 3P-10…60.
– To introduce an automatic signal to close or to prohibit opening of the control valves on the emergency feed water lines when there are leaks of SG steam lines in the non-isolated part.

The above measures on upgrading will be implemented in the Unit 3 at NVNPP during a period from April to September, 2001.

3. DEVELOPING A SAFETY CONCEPT FOR NEW DESIGNS

The design solutions on safety for nuclear power units with WWER reactors of a new generation are aimed at developing an NPP with an enhanced level of safety so that to have the total risk associated with NPP operation as low as reasonably achievable. Here, of course, the requirements of the Russian regulatory documents on safety as well as the recommendations of IAEA, which are now in force, shall be complied with. In particular, the requirements on the target safety performance indices in the NVNPP-2 Project are based on the requirements of Clause 1.2.17 of OPB-88/97. According to these requirements, the frequency of large emergency release shall not exceed 1.0 E-7 per reactor a year. The emergency release is defined as the release of the amount of radioactive products at which evacuation of population may be required beyond the boundaries determined by the existing regulations for sitting NPPs.

The second group of requirements on limiting the risk level are those outlined in Clause 4.2.2 of OPB-88/97. According to them, the core damage frequency shall not exceed 1.0E-5 per reactor a year. The safety concept for new design of NPP with WWER reactor has been developed to achieve these purposes. The PSA results which have been performed for operating NPP with WWER-1000 (Balakovo NPP) were used for the development of the safety concept for new NPP design.

The results of core damage frequency estimation for unit 4 of Balakovo NPP are presented in Table I. The following insights have been made based on consideration of this results:

1. The estimated value of core damage frequency for unit 4 of Balakovo NPP 4.26E-5 1/year is not sufficient the requirement of OPB-88/97 and recommendation of INSAG-3 for new NPP design. The main contribution in core damage frequency value give the common cause failures (CCF) of safety system components and operator errors. The NPP with WWER-1000 design is based on using the three-train active safety systems in which similar types of components (diesel-generator, pumps, valve, check valves etc.) are employed in individual trains. The operator actions on safety system control are required for post-accident period. The influence of CCF and operator errors does not allow to decrease the failure probability values of active safety system lower than 10^{-3} – 10^{-4} on demand.
2. The design of NPP with WWER is not good sufficient to be balanced. The contributions from initiating event with transients (96%) exceed the contributions from LOCAs (2.6%). This fact is explained the similar structures of safety systems which need to prevent the core damage at transients and LOCAs while the frequency values of transients are fairly high (in 100 or more times) compared the frequency values of LOCAs.

Based on PSA results of NPP with V-320 the following principles and decision were included in safety concept of new NPP with WWER design:

1. To ensure the deep defence from CCF that is based on using the diversity in safety systems. The mutually redundant systems of passive and active operation principles or systems with diverse design of components are used for performing the main safety functions.
   - An upgraded emergency protection system with the number of control rods two times more than that used in the V-320 reactor plant and a quick boron injection system to bring the reactor in a subcritical state and maintain it in this state over a wide range of operating parameters (the emergency protection system is capable to maintain the subcritical state up to a temperature below 100^0C).
   - The active and passive systems for emergency heat removal through the secondary coolant circuit. Both these systems can remove heat during infinite time, whereas the emergency heat removal system for NPPs with a V-320 reactor can operate only for a limited time (about 30-40 hours), which is determined by the inventory of coolant in its tanks
   - Active emergency core cooling system (ECCS) and the 1st and 2nd stage hydro accumulators to maintain the inventory of reactor coolant in the core during leaks from the reactor coolant system. The 2nd stage hydro accumulators together with the 1st stage hydro accumulators provide a redundancy to the active ECCS in terms of the function of maintaining the inventory of coolant in the core during 24 hours after the accident. This time can be used to restore the operability of the active ECCS in case of its failure.

The individual trains of active safety systems (the emergency cooldown system and ECCS) can be used to perform functions of normal operation. Here, most of the components of these systems are in the states that are similar to those when the required emergency functions are performed. Using such operating modes of these systems, it is possible to enhance their availability indices and to provide additional protection against common-cause failures.

2. To ensure the deep defence from operator errors that is based on using the passive safety systems which do not require the operator actions for their operation and on using the high level of automatization in active safety systems.
3. To develop the containment system for the mitigation of radioactive releases at severe accidents with core melt. The double containment system with hydrogen removal system and catch for melting core is used in NV NPP-2 design.

The level 1 and 2 PSA were developed for NV NPP-2 design to estimate the effect of new design decision which are described above. The results of CDF estimation are given in Table I. As can be seen from Table I the CDF values for internal initiating groups during power operation amount to $2.58*10^{-8}$ 1/year for NV NPP-2 and $4.26*10^{-5}$ 1/year for Balakovo NPP. The total CDF value from NV NPP-2 taken in account the contribution from shutdown modes amount to $4.8*10^{-8}$ 1/year.

TABLE I. CONTRIBUTION OF DIFFERENT GROUPS OF INTERNAL INITIATING EVENTS TO CORE DAMAGE FREQUENCY

Initiating event	**Frequency of IE 1/year**	**Contribution of Unit 1 at NVNPP-2 to core damage frequency**		**Contribution of Unit 4 at Balakovo NPP to core damage frequency**	
		Absolute, 1/year	**Relative %**	**Absolute, 1/year**	**Relative %**
Leaks from the RCC to the containment					
1.1. Small break	3.20E-03	1.26E-09	~2.6	3.40E-07	~0.8
1.2. Intermediate break	1.00E-03	3.64E-10	<1	8.30E-08	~0.2
1.3. Large break	3.20E-04	6.79E-10	~1.4	5.40E-08	~0,1
2. Leaks from the primary to secondary circuit	1.00E-03	1.26E-09	~2.6	1.10E-06	~2.6
3. Reactor shutdown	1.00E-00	7.38E-09	~15	1.65E-06	~3.9
4. Loss of normal heat removal through the secondary circuit	1.00E-01	7.38E-09	~15	6.50E-07	~1.5
5. Loss of power supply	1.00E-01	7.91E-09	~16	3.54E-05	~82.9
6. Leak of the steam line in the part isolated from the SG	1.00E-03	2.67E-11	<1	3.40E-06	~8.0
7. Leak of the steam line in the part non-isolated from the SG	4.00E-04	1.29E-10	<1	1.00E-10	~0
8. Loss of heat removal from the depressurized reactor	3.50E-05	1.07E-08	~22		
9. Loss of power supply with the depressurized reactor	3.70E-03	1.12E-08	~23		
All IEs		4.77E-08	100	4.27E-05	100

The estimated on level 2 PSA results value of large release frequency for NV NPP-2 amount to $0.94*10^{-8}$ 1/year. It means that using the new design decisions for NV NPP-2 described above allows to decrease CDF value for NV NPP-2 by factor of about 1700 lower than that for Balakovo NPP and to ensure the requirements of OPB-88/97 and INSAG-3 for new NPP design.

4. CONCLUSION

The PSA is very effective method for making and supporting the decisions on safety improving of operating and newly designing NPP. Using the PSA allow to reach the qualitatively new safety level for newly designing NPP and to upgrade the safety of operating NPP.