Probabilistic Safety Assessment of Novovoronezh Unit 3 within NOVISA project.

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As part of the International Nuclear Safety Program (INSP), Novovoronezh Nuclear Power Plant (NOVISA) project is funded by US DOE and Novovoronezh NPP /1/. The general objective of the NOVISA project is to assess the current safety level of Novovoronezh NPP through the performance of the in-depth safety analysis. The management and technical work are carried out by the Novovoronezh NPP and Russian subcontractors. Management and technical assistance are provided by US Argonne National Laboratory and DS&S. The overall NOVISA project was divided into several specific activities. This paper presents results of the Level 1 internal events probabilistic safety assessment (PSA) performed within the NOVISA project for the power operation.

The NOVISA PSA study was started in October, 1998, although some preliminary work associated with the development of the system description documents and the plant specific raw data collection was performed in advance.

In order to ensure that PSA was performed according to acceptable standards and accepted international methods, a set of project guidelines to define the methodologies to be employed, and to describe the documentation requirements was also developed prior the main activity. The Novovoronezh project guidelines were developed by the Novovoronezh NPP and their support organizations and reviewed by the U.S. technical oversight team (DS&S) and by the U.S. assisted Russian quality assurance and peer review team (IBRAE institute).

To support success criteria analysis, thermohydraulic calculations were also performed under a separate task order.

At the moment, the NOVISA PSA is about to finish: all tasks within PSA activity have been already competed and are reviewed by the IBRAE responsible for external review of the overall PSA.

The Level 1 internal events PSA for Novovoronezh was divided into seven major subtasks as follows:

- **Initiating Event Identification and Grouping.** A comprehensive list of initiating events for the power operation was developed and grouped into 27 initiating event groups (see Table 1).
- **Success Criteria Analysis.** Based on the results of thermohydraulic calculations performed, the success criteria were developed for each particular safety function considered in the event/fault tree analysis.
- **Accident Sequences Analysis.** Event trees were used to represent the accident sequence modeling for each initiating event group. Each accident sequence that could lead to core damage was identified, described and documented as part of this task. The development of the event trees was based on the analysis for success criteria.
- **Systems Analysis.** This task involved preparing the system fault trees, failure modes and effects analysis, and finalizing the system design descriptions. Twenty three systems

were analyzed including new ones incorporated in the Novovoronezh design under Safety Upgrading Program such as Additional and Autonomic emergency feedwater systems as well as Mobile diesel generator.

- **Data Analysis.** This task involved:
    - the analysis of the component reliability data collected for the last six years from operating experience of the Novovoronezh Units 3 and 4, including component reliability parameters, and component unavailabilities due to test and maintenance, and
    - the collection and analysis of the initiating event frequency data required to quantify the Novovoronezh PSA models. To calculate initiating event frequencies, Novovoronezh Units 3 and 4 plant operating history data for plant trips was gathered from 1986 through 1998. The plant trips was categorized according to the initiating event groups defined. Frequencies of the large and medium LOCAs were derived using results of Leak-before-Break study.
- **Human Reliability Analysis.** This task involved the identification, modeling, screening and quantification of human failure events in the Novovoronezh PSA. As part of this task, existing plant operating procedures and training materials were reviewed. Interviews were also conducted with plant operators to better understand the anticipated plant response under specific accident sequence conditions postulated in the PSA.
- **Accident Sequence Quantification and Sensitivity Analysis.** This task involved preliminary and final quantification as well as uncertainty and sensitivity analysis. SAPHIRE 7.0 computer code was used for the development of the event/fault trees and quantification.

Table 1

Initiating event groups analyzed in NOVISA Project

IE Group	Description
RPVR	Reactor Pressure Vessel Rupture
LL	Large LOCA (>100 mm)
ML	Medium LOCA (32-100 mm)
SL	Small LOCA (7-<32 mm)
IOPSV	Inadvertent Opening of Pressurizer Relief Valve/Safety Valve
CRTR	Control Rod Tube Rupture (rupture of control rod driver jacket)
ILOCA1	Interfacing System LOCA through Normal Makeup and Purification System
ILOCA2	Interfacing System LOCA through Intermediate Cooling Circuit of Main Coolant Pumps
ILOCA3	Interfacing System LOCA through Intermediate Cooling Circuit of Control Rod Drives
SGTR	SG Tube Rupture
SGCR1	SG Collector Rupture ≤100mm
SGCR2	SG Collector Rupture > 100mm
GT	General Unit Transient
LMFW	Loss of Main Feedwater System
FWDLB	Feedwater Discharge Line Break Isolated Outside Confinement
FWSLB	Feedwater Suction Line Rupture
NILSG	Non-Isolated Steam Line Break
NFWT	Non-Isolated Feedwater Tube Rupture
IOSGSV	Inadvertent Opening of SG Relief Valve
MSH1	Rupture of the 1st Semisection of Main Steam Collector
MSH2	Rupture of the 2nd Semisection of Main Steam Collector
IOSDA1	Inadvertent Opening of Steam Dump Valve to Atmosphere BRU-A1
IOSDA2	Inadvertent Opening of Steam Dump Valve to Atmosphere BRU-A2
LCW	Loss of Circulating Water System
LSW	Loss of Service Water System
LOOP	Loss of Offsite Power
LOOP3&4	Simultaneous Loss of Off-site Powers at both Units lasted more than 2 hours

The overall core damage frequency was estimated to be 1.43E-4 per year. The significant contributors to the core damage frequency are as follows:

- Small LOCA – 34%
- Medium LOCA – 20%
- Inadvertent opening of pressurizer relief valve/safety valve – 12%

- SG Tube Rupture – 8%
- SG Collector Rupture > 100mm – 7%
- SG Collector Rupture ≤100mm – 4 %
- Control rod tube rupture – 3%

It was found that all significant contributors are associated with different kind of LOCAs or primary-to-secondary leaks. Following implementation of the safety upgrading measures, transients left off being dominant contributors.

There are issues still open to be resolved in the future like analysis of turbine hall effect or reducing conservatism of some success criteria:

- Rupture of the feed water header or steam lines may result in effects of steaming or flooding the turbine building rooms that, in its turn, may cause dependent on these events failures of equipment and components of the system for heat removal through the secondary circuit. System interactions like pipe whip or jet impingement may be also important factors causing dependent failures of piping and components. The full analysis of such events shall be performed as a part of analyses of on-site ("external") impacts.
- To reduce conservatism of some success criteria an additional set of thermohydraulic calculations should be performed.
- To assess the overall level of Novovoronezh Unit 3 safety, the shutdown PSA and analysis of external events such as fires, floods, etc. should be also performed.

The plant upgrading to be implemented based on PSA results was identified. The core damage frequency is supposed to be reduced by the following measures:

- Automatic opening of motor-operated valves installed on the lines of service water supply to spray heat exchangers following LOCAs
- Increasing capacity of the spray heat exchangers
- Replacing manual valves by motor-operated valves on primary blowdown and blowdown return and automatic closing them following LOCAs
- Upgrading design of the confinement sump to avoid clogging in case of LOCAs

REFERENCE

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