# MO34 Project Lessons Learned - Quality Management System Processes

**Tomáš Naňo**
Slovenské elektrárne, a.s.

## ABSTRACT

The construction and commissioning of Mochovce Nuclear Power Plant Units 3 and 4 (MO34) represented one of the largest nuclear new-build projects in Europe in recent decades. During more than ten years of project execution, extensive experience was gained in the implementation, operation, monitoring, and continual improvement of the Quality Management System (QMS) supporting nuclear safety and project delivery objectives.

This paper presents a structured overview of selected lessons learned collected throughout the MO34 project lifecycle. The lessons were systematically analyzed across key QMS processes, including organizational management, safety culture, design and engineering, procurement, supplier oversight, inspections and testing, non-conformity management, documentation control, audits, training, and continual improvement.

The presented findings identify both risks and opportunities observed during project execution. Particular attention is given to topics such as organizational independence of quality functions, supplier quality management, coordination of detailed design activities, nuclear safety culture promotion, management of non-conformities, root cause analysis, quality oversight of classified equipment, and the application of integrated project databases and performance indicators.

The objective of the lessons learned programme was not only to capture historical project experience but also to transform that experience into practical recommendations applicable to future nuclear new-build projects, major modernization programmes, and the continual improvement of the Slovenské elektrárne Integrated Management System. The paper demonstrates how systematic collection and evaluation of project experience can contribute to enhanced quality performance, improved project execution, strengthened nuclear safety culture, and more effective management of future nuclear investments.

**Keywords:** Quality Management System, Lessons Learned, Nuclear New Build, MO34 Project, ISO 19443,

Nuclear Safety Culture, Supplier Quality Management, Continuous Improvement

# Closing the gaps in deploying new nuclear – Recent EPRI Activities

**Hasan Charkas**
Electric Power Research Institute (EPRI)

## ABSTRACT

This paper presents recent activities by the Electric Power Research Institute (EPRI) aimed at systematically identifying and closing deployment of new nuclear builds gaps, with particular emphasis on the Advanced Reactors Roadmap. The roadmap has highlighted key deficiencies in constructability, standardization, and workforce readiness, as well as limitations in the current supply chain's ability to deliver nuclear-grade components at the required scale and schedule. Long lead items such as large forgings, reactor vessels, and specialized engineered components, which remain critical drivers of project timelines are good candidate for innovative methods to optimize construction schedule.

There has been increasing industry interest in precast and modular concrete technologies to improve construction productivity, reduce site labor intensity, and enhance quality through controlled fabrication environments. Additionally, the ongoing transition from prescriptive, deterministic design approaches to risk-informed, performance-based methodologies is gaining a lot of interest in industry. These methods enable designers and regulators to focus on functional safety outcomes rather than strict adherence to legacy rules developed for earlier reactor technologies. EPRI has been actively supporting the development of frameworks and technical bases that allow for the application of risk insights in areas such as structural design, seismic qualification, and system reliability.

This paper provides an overview of some of the key EPRI projects to provide more certainty in design and construction schedule

# SMR.RETROFIT: A Structured Methodology for Coal-to-SMR Repowering – Application to the Chvaletice Case Study

**Jean Fernandez**
Tractebel

## ABSTRACT

The transition from coal-fired power generation to low-carbon alternatives presents a major opportunity to leverage existing industrial assets while accelerating deployment timelines. This paper introduces SMR.RETROFIT, a proprietary methodology developed by Tractebel to systematically assess the feasibility and value of repurposing coal power plants for Small Modular Reactor (SMR) deployment.

SMR.RETROFIT provides a structured, multi-criteria decision framework that evaluates retrofit strategies across five standardized scenarios, ranging from greenfield development to full integration with existing balance-of-plant systems. The methodology integrates key technical, economic, regulatory, and sustainability criteria, enabling rapid screening of sites and identification of optimal reuse pathways. Particular emphasis is placed on high-value asset reuse (e.g., turbines, cooling systems, grid connection) and its impact on CAPEX reduction, project risk, and time-to-market.

The approach is demonstrated through a case study of the Chvaletice coal power plant in the Czech Republic, a 4 × 205 MWe facility scheduled for retirement. The analysis highlights the site's strong repurposing potential, supported by existing 400 kV grid infrastructure, robust cooling capacity, and large steam turbine units. Among the five scenarios, direct and indirect turbine reuse configurations (Cases 4 and 5) emerge as the most promising, enabling significant infrastructure valorisation.

The study further shows that technology–site compatibility is a key driver for efficiency , particularly regarding steam conditions and turbine integration. While high-temperature SMRs offer pathways for near-direct reuse, intermediate-loop configurations provide greater flexibility and clearer safety boundaries, albeit with moderate efficiency penalties.

---

# From Concept to Reality: Implementation Cases of i-SMR Design

**Yongse Kwon**
Innovative Small Modular Reactor Development Agency

## ABSTRACT

The innovative Small Modular Reactor (i-SMR) is a next-generation integrated pressurized water reactor developed by South Korea, designed to achieve top-tier safety, economic competitiveness, and operational flexibility for the global market. Targeting standard design approval and commercial readiness, this presentation provides a comprehensive overview of the i-SMR project, with a primary focus on how its top-tier design requirements are successfully implemented into engineering solutions.

To enhance safety to an unprecedented level, the i-SMR introduces a fully passive safety system which effectively eliminates the risk of severe accidents sprong from station blackout (SBO). For economic efficiency and carbon-free multi-purpose utilization, the reactor adopts a soluble-boron-free (SBF) operation, significantly reducing liquid radioactive waste and system maintenance demands. Crucially, to mitigate the risk of control rod ejection accidents common in conventional designs, the i-SMR integrates a world-unique In-Vessel Control Rod Drive Mechanism (IV-CRDM) within the reactor pressure vessel. The session will showcase the i-SMR's readiness to lead the future clean energy transition with innovative design features.

# Benefits of dynamic simulation applications to newbuild NPP projects

**Toni Salminen**
Fortum Power and Heat Oy

## ABSTRACT

ABSTRACT

Dynamic process simulation has a number of valuable applications in supporting the design and operation of a newbuild nuclear power plant (NPP) starting from the early design phase. The early applications include engineering simulator for design verification, safety analysis model for deterministic safety analyses and full-scope training simulator. In the commissioning phase, the engineering simulator is used to predict the behaviour of the upcoming commission tests and solve unexpected situations. In the operational phase, these dynamic simulation applications are used to solve unexpected operational occurrences and provide support for plant modification projects.

Fortum has used system code Apros as a dynamic process simulation software to support newbuild NPP projects, such as Olkiluoto 3. The level of detail and the extent of the scope of the simulation models is important for effective use of dynamic process simulation in a wide range of applications. This presentation gives an overview of different plant scale dynamic simulation applications highlighting the obtained benefits.. Three application cases are described in more detail: Fennovoima NPP project, Olkiluoto 3 commission support and Steady Energy's SMR project development support.

Keywords: dynamic process simulation, newbuild NPP, plant operations, simulators, analysis

---

# Developing a Waste Treatment Center for PWR New Builts

**Wolfgang Schuch**
Framatome GmbH

## ABSTRACT

During operation of a nuclear power plant (NPP) radioactive waste (radwaste) is generated in forms of solid waste, liquid waste, waste water and gaseous waste. This radwaste needs to be treated and conditioned to reduce the volumes of radwaste and reach the required limits for waste disposal at an intermediate or final repository.

Consequently, each NPP must be provided with some kind of waste treatment facilities usually incorporated inside a waste treatment center (WTC) or waste treatment building directly at the NPP itself.

The development and design of a WTC requires a holistic approach. Based on the type of the NPP (light or heavy water reactor, boiling water reactor (BWR) or pressurized water reactor (PWR)), the size and the operational regime (duration of a fuel cycle) the generated types and volumes of primary waste can be determined.

On the other hand, requirements and limit values coming from the regulator for clearance as well as for intermediate or final storage of radwaste at a repository must be adhered to regarding

activity levels and concentrations, authorized waste package types, maximum dose rates per package (contact and at defined distances) and authorized conditioned end states of the radwaste inside the waste package.

Consequently, the initial state of the radwaste as well as the end state are known as boundary conditions, from where the processing chain of waste treatment is developed. To reach a certain end state for the conditioned radwaste different process steps like waste segregation and separation, volume reduction, decontamination and solidification in a stable matrix must be applied. Some of these process steps generate secondary waste, which must be considered during the development of a waste treatment center as well.

The holistic approach on the complete waste treatment process chain from the initial waste form to the final conditioned and packaged waste product is in the best case accompanied by a general holistic design approach for the waste treatment center itself regarding layout, size and logistics.

This holistic approach includes the development and design of the waste treatment center through all design phases including the engineering and calculation, manufacturing and construction, commissioning and a close operational support during the first operational years of the NPP as well as continued support throughout the lifetime of the NPP.

Framatome has provided such a holistic approach on the waste treatment center at NPPs in Finland and China. In this paper the Framatome approach to include the different processes, systems and products into the waste management of a NPP is presented, based on the experience from the new-built projects in Finland and China, ranging from the conceptual design until the commissioning and operation of the WTC of the respective NPP.

## Reactor physics code development for SMR analysis

**Csenge Komjáti**
HUN-REN Centre for Energy Research

## ABSTRACT

In recent years, the interest in Small Modular Reactors (SMRs) has grown significantly. These low-power nuclear reactors aim to address some of the main challenges faced by the nuclear industry. Their intended advantages include the possibility of standardized production, transportability and streamlined installation, and the ability to meet a wide range of industrial needs. SMRs have significantly smaller reactor cores containing a large number of geometrical inhomogeneities, in addition, they may employ innovative solutions for reactivity control and cooling. For example, some designs use liquid metal as a coolant and operate with a fast neutron spectrum. Due to their altered reactor physics characteristics, modeling these reactors presents challenges for currently used reactor physics codes, making extensive development and testing of various simulation software necessary.

We have started the assessment of the codes used and developed at the HUN-REN Centre for Energy Research Nuclear Safety Department, by performing calculations for two soluble boron free pressurized water reactor (PWR) benchmark problems and one sodium-cooled reactor physics benchmark problem.

One of the soluble boron free PWRs, NuScale, is a 160 MWth integrated pressurized water reactor (IPWR) MCNP and Serpent 2 Monte Carlo models enabled us to test and investigate the effects of the various approximations employed by the deterministic KIKO3DMG code developed at the laboratory in the case of steady-state conditions.

The other soluble boron free PWR, PRATIC, is a reactor concept based on industrial designs and global trends, and is part of the 2025 Light Water SMR (LW-SMR) reactor physics benchmark. Based on the benchmark specifications, we began performing burnup calculations and generating burnup-dependent group constants using the Serpent Monte Carlo code. We have investigated the effects of the number of radial depletion sub-zones used in the burnup calculations and compared the results with those obtained using the deterministic APOLLO2 code provided with the benchmark.

We are also participating in a benchmark organized by the Nuclear Energy Agency (NEA) Working Party on Scientific Issues and Reactor Systems (WPRS), based on the SEFOR sodium-cooled fast reactor. We have performed several reactor physics analyses and also begun the modeling the reactor's thermal-hydraulics, and planning to perform coupled calculations in the future.

Parallel to these projects, we have started the development of a Python-based framework for generating parameterised multigroup cross-section libraries using deterministic (SCALE/NEWT) and Monte Carlo (Serpent 2) methods. This system will increase our ability to test modelling approximations under different conditions, thereby reducing uncertainties in the final calculations. It would also reduce the time it would take to develop the capability to manage novel and diverse reactor technologies, or to investigate legacy models for traditional reactor types.

---

## Supporting new builds with cutting edge core design and safety analysis methods

**Adam Kecek**
ÚJV Řež

## ABSTRACT

The deployment of new nuclear power plants requires advanced computational methods, validated analytical tools, and comprehensive safety assessments to support design verification, licensing, and regulatory approval. Core design analyses, fuel performance assessments, and deterministic and probabilistic safety evaluations provide an essential technical basis for demonstrating the safety and reliability of new nuclear installations.

Core design and fuel cycle optimization rely on dedicated computational tools, including ANDREA, CYCLEKIT, and LP-OPT, validated for VVER-1200, VVER-1000, and VVER-440 reactors with different fuel types. Automated reload design, operational data evaluation, and comparison of predictions with measurements support the verification of core performance and safety margins. Fuel performance codes, detailed finite element models, and subchannel analysis provide additional assessment of fuel behaviour and thermal limits.

Safety analyses employ established VVER methodologies and advanced computational approaches, including coupled system and three-dimensional core simulations, Best Estimate Plus Uncertainty (BEPU) methods, and computational fluid dynamics. These methods are applied to operational and accident scenarios, including anticipated transients without scram, single-failure conditions, and shutdown states. Severe accident codes, such as MELCOR, ASTEC, and MAAP, support the evaluation of accident progression and mitigation measures, including hydrogen risk management, filtered containment venting, and in-vessel and ex-vessel corium cooling.

Experimental programmes, material qualification, reactor pressure vessel surveillance, and fuel inspection data provide essential evidence for model validation and safety justification. Further activities address passive safety systems, accident-tolerant fuels, and small modular reactors. Licensing support includes the assessment of compliance with Czech and European legislation, international standards, and applicable utility requirements.

The integration of validated analytical methods, experimental evidence, and regulatory

requirements provides a robust technical basis for safety demonstration and supports the successful licensing and deployment of new nuclear power plants.

---

# Design and Thermal-Hydraulic Characteristics of a Passive Auxiliary Feedwater System for Advanced Power Reactors

**Jan Hruškovič, Sang Gyun Nam, Jehee Lee, Youngjae Park, Soon-Joon Hong, and Seong-Su Jeon**

FNC Technology Co., Ltd.

## ABSTRACT

A Passive Auxiliary Feedwater System (PAFS) provides passive heat removal through the steam generator secondary side during abnormal plant conditions. PAFS is typically actuated by a low steam generator water level, and steam is routed to a passive condensation heat exchanger (PCHX) submerged in a large water pool, the passive condensation cooling tank (PCCT), located at an elevated position relative to the steam generator. Steam condenses inside the PCHX while heat is transferred to the water in the PCCT, and the condensate returns to the steam generator by gravity. The density difference within this closed-loop system, together with the elevation difference between the heat source (steam generator) and the heat sink (PCHX), enables natural circulation to be established, allowing continued heat removal without the need for electrical power, active pumps, or operator action. In this way, PAFS is intended to maintain secondary-side heat removal during loss of normal feedwater and other postulated accident scenarios requiring continued steam generator cooling. This paper presents the general design features and major thermal-hydraulic characteristics of PAFS for application in advanced power reactors. System pressure, elevation difference, piping resistance, PCHX geometry and inclination, condensation heat transfer, flow characteristics, and PCCT water inventory are key parameters affecting the natural circulation and heat-removal capability of PAFS. The combination of these

design parameters determines the system response from initial actuation to stable passive cooling operation. System-code analyses supported by results from separate- and integral-effect experiments are used to describe the main thermal-hydraulic phenomena and their implications for PAFS design, performance evaluation, and safety assessment.

KEYWORDS

PAFS, Passive Safety System, Natural Circulation, Advanced Power Reactor