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Identification of potentially impacted safety margins and methodology for safetyanalysis of a SMR integrated in a hybrid system

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Summary

Identification of potentially impacted safety margins and methodology forsafety analysis of aSMR integrated in a hybrid system. The study will cover feedback from Germany (GRS), France (IRSN) and eastern Europe (ENERGORISK). Then a methodology for assessment of relevant physical parameters and safety margins of the SMR will be established.

Approval

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D4.2 - Identification of potentially impacted safety margins and methodology for safety analysis of a SMR integrated in a hybrid system

WP4 - Task 4.1

November 2023

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Table of Contents

Funded by the European Union

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List of Figures

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Abbreviations and Acronyms

Acronym	Description
AO	Axial Offset
AOO	Anticipated Operational Occurrences
BoP	Balance of Plant
CCGT	Combined-Cycle Gas Turbines
CHP	Combined Heat and Power
CSG	Compact Steam Generator
DBA	Design Basis Accident
DBC	Design Basic Condition
DEC	Design Extension Condition
DH	District Heating
DiD	Defense in Depth
DNB	Departure from Nucleate Boiling
FOAK	First of a Kind
FoM	figures of merit
GEMINI+	Research and Development in support of the GEMINI Initiative
HES	Hybrid Energy Systems
HOF	Human and Organisational Factors
HTSE	High-Temperature Steam Electrolyser
IAEA	International Atomic Energy Agency
LOCA	Loss Of Coolant Accident
NPV	Net Present Value
NSSS	Nuclear Steam Supply System
OPEX	Operating Experience
PCI	Pellet Clad Interaction
PIE	Postulated Initiating Events

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Acronym	Description
PWR	Pressurised Water Reactor
RES	Renewable Energy Source
SMR	Small Modular Reactor
SSC	Systems Structures Components
WENRA	Western European Nuclear Regulators Association
WP	Work Package

Executive Summary

The integration of SMRs into a HES will involve specific interactions between the SMRs and the rest of the HES, and new potential risks. In the present report the status of European studies on safety analysis of small modular reactors from the operational flexibility and cogeneration viewpoint has been analysed to study the need to define a specific safety methodology. The proposed methodology is based on the IAEA guide SSG2. Its objective is to identify these specific risks and to assess them. In particular, the integration of SMRs in future low carbon and smart grids where, some non-pilotable generators have priority over pilotable means, will necessarily introduce specific interactions with the energy network and constraints to ensure the electricity grid reliability (e.g. make production equal to demand always). Then a crucial step is to establish an exhaustive list of postulated initiating events. Energy network balancing through SMR load following capabilities, energy storage systems or cogeneration flexibility, has been discussed. This report gives elements to address such configurations from a safety assessment point of view by considering the degree of dependence and feedback effects between the various systems.

Keywords

SMRs, Hybrid energy Systems, Safety Methodology, Flexibility, Cogeneration, Europe

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1 Introduction

Although use of nuclear power has achieved high standards of safety, most of the safety experience is related to electricity production. There is limited safety experience feedback in case of use in other purposes with an important part of HES, possible use of cogeneration and aspects of exploiting nuclear energy for non-electric applications. Thus, it is worthwhile to examine the impact of this use on safety approaches, being understood that technology choices shall not challenge safety principles and that the way to implement them may be adapted.

The objective of this report is to provide a preliminary analysis of the transient behaviour of SMRs integrated in an HES in unfavourable situations, specifically related to such uses. This analysis includes to evaluate possible reactions to abnormal (or stressful) conditions created by the two following different possible causes: technical failures and erroneous human actions.

The analysis of the impact of hazardous phenomena that could result from an accident involving an industrial process in the vicinity of a nuclear installation is a standard procedure in the safety demonstration of a nuclear reactor. However, having this industrial process directly linked to the operation of the reactor (nuclear cogeneration) is less common. The reflections carried out in this document are an opportunity to identify the potential impacts of SMRs integrated in a HES, in the light of current knowledge of similar use with NPPs and beyond. They are also a way to identify the possible sources of disturbances, which could weaken the safety of the NPP, to take them into account from the design stage. One of the key objectives is to identify possible normal and abnormal disturbances due to the integration of SMRs in the considered HES.

This report gives elements to address the integration of SMRs in a HES, from a safety assessment point of view by considering the degree of dependence and feedback effects between the various systems.

There is little experience with nuclear cogeneration, some with existing NPP, but none with SMRs. So for such a FOAK assessment, it is important to consider firstly the principles of standard safety assessment methodologies independently of interfaced HES by considering these HES as potential external hazards (Section 2) and secondly some key points from the few safety analysis reports on the operation of NPPs in cogeneration or load following mode (Section 3).

A safety methodology is suggested in Section 6 from the identification of limits and effects on components and systems (Section 4) and reference initiating events and transients are suggested (Section5).

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2 Principles from standard safety assessment methodologies

The first step is a reminder of the key safety principles for new reactors whose application may be potentially impacted by cogeneration operations:

- WENRA -RHWG has performed a review that shows that even though there is a wide variety of SMR designs currently under development, the safety objectives issued by WENRA in 2010 can also be used for SMRs. Thus “WENRA considers these objectives to be the minimum requirement for SMRs. Higher safety requirements should be expected, particularly when considering the potential for such modern technologies to provide significant improvements in safety performance” [WENRA 2021]
- The safety objective stated by the Council Directive 2009/71/Euratom amended by 2014/87/Euratom is universal and as such shall not be challenged by the technology. Moreover, the Directive requires that “*licence holders are to regularly assess, verify, and continuously improve, as far as reasonably practicable, the nuclear safety of their nuclear installations in a systematic and verifiable manner*”.

Then it should be examined if complementary safety objectives shall be considered in case of HES, regarding notably the following issues:

- The potential presence of a large number of people in the vicinity of the plants that requires to drastically reduce the accidental releases.
- Using the reactor coolant to heat another coolant requires safety evaluation concerning this interface and the barriers involved.

To go more in details, among the main recommendations highlighted in ELSMOR which could be linked to the problems addressed by TANDEM, we can note that a particular attention should be paid to the additional missions undertaken by SMR such as producing hydrogen, steam or desalinated water, that might bring about new issues in terms of:

- Operation and maintenance constraints: “In addition to the supervision of several reactors, the management of extra processes, hazards and transients will be required” (See section 4.2 below for hazards and transients).
- HOF issues: “Risk of overload and human errors for the operation team; Should nuclear and other processes elements be separated? This will impact the organization of the supervision team, the design of the control room and management procedures”. The key responsibility for safety concerning cogeneration activities that could impact the increase of radiation risks, should lie with the staff or organisation responsible for the nuclear facility.

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- How to consider safety culture for staff involved in the non-nuclear processes (which will be a major part of the general process) as it implies that safety is an overriding priority (IAEA glossary) and as establishment of a strong safety culture is a major requirement [IAEA GSR].
- How to define an adequate organisation with two important processes whose dependencies are less direct than in case of electricity production (potentially 2 different operators or 2 fully separated structures)
- How to manage accidental situations.
- How to consider both types of risks in case of HES that are also associated to environmental risks (prioritization).
- Next step to address the issues: “*No OPEX available on these issues*” see section 3.1.2 related to existing installations.

Moreover, it is noticed that “*If the SMR is used for cogeneration, the impact on the containment integrity due to hazards introduced by other plants (chemical or hydrogen facility) should be considered as well*”. In addition to the methodology detailed in ELSMOR for taking into account the specific features of SMRs in conventional use, **the fundamental principles of safety remain the basis for investigating the possible disturbances that will be caused by the integration of SMRs into a HES and then to consider firstly such “hybridization transient” with regard to the usual assessment of external hazards** which we discuss in detail in the paragraphs below.

In particular, the methodology should answer the following questions:

- Should the hazards issued from the HES be considered as internal hazards (so, as such, their sources shall be minimized as far as possible considering that the operator is responsible of these sources (see WENRA RL issue SV))?
- Should some indisputable principles be established and fulfilled regarding these sources? (Straightforward illustration: for an electrical NPP, no designer would propose a turbine whose axle would be perpendicular to a reactor building)
- The list of transients to be considered shall duly consider the HES itself, which could complicate the safety case.

In addition, particular attention should be paid to the separation between the nuclear side and the industrial one:

- How to consider the unavailability of the HES which could be not scheduled at all and whose duration could be diverse?
- More generally, how to consider the potential non-scheduled variations of the demands of the HES?

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- When considering the influence of the HES on the NPP, it may lead to state that some equipment of this system is to be classified. This should be done with care to avoid a “banalisation” of classification (moreover, it has an impact on organizational aspects).
- HOF, safety culture… see above.
- Business model: what would occur if the HES had to be dismantled while the NPP operator had not foreseen that?
- Security aspects: large number of activities around the NPP.

If the hazards resulting from hybridisation were to be considered simply as external hazards, the associated safety objective in terms of external reference hazards would be limited to "no core meltdown and no (or minor) radiological impact".

Assuming a clear separation between the activities of the nuclear island and the uses that would be made of the heat produced or its electrical power, a conventional safety analysis would include these cogeneration risks in the list of internal and external risks which are the risks induced by industrial activities and communication routes (mainly explosion and aircraft crash). The other reference risks are earthquakes, lightning and electromagnetic interference, extreme weather or climate conditions, or floods originating from outside the site.

As usual it is important to pay particular attention to the classification of equipment as this is emphasised in the book « Éléments de sûreté nucléaire – Les réacteurs à eau sous pression » [Couturier 2021]: “*Achieving and maintaining an appropriate level of safety requires the implementation of an approach that ensures that equipment is subject to appropriate requirements in terms of design, manufacture, qualification, operation and in-service monitoring, commensurate with its importance for safety. This is the role of safety classification*”.

Identifying the interfaces between the nuclear island and its environment, the potential hazards, their targets at the level of systems and then at the level of individual components, and finally defining the level of damage, therefore is an essential step. The identification process should include also reasonably foreseeable combinations of independently occurring hazards, causally-related hazards and consequential events resulting from a common initiating event.

In a classic approach to analysing hazards, the plant shall be protected against any external attack by protection designed to limit the consequences of the hazard, using SSCs designed or protected to remain operational during these transients to prevent and limit the consequences of a reference external hazard and SSCs to achieve and maintain a safe condition after a reference external hazard.

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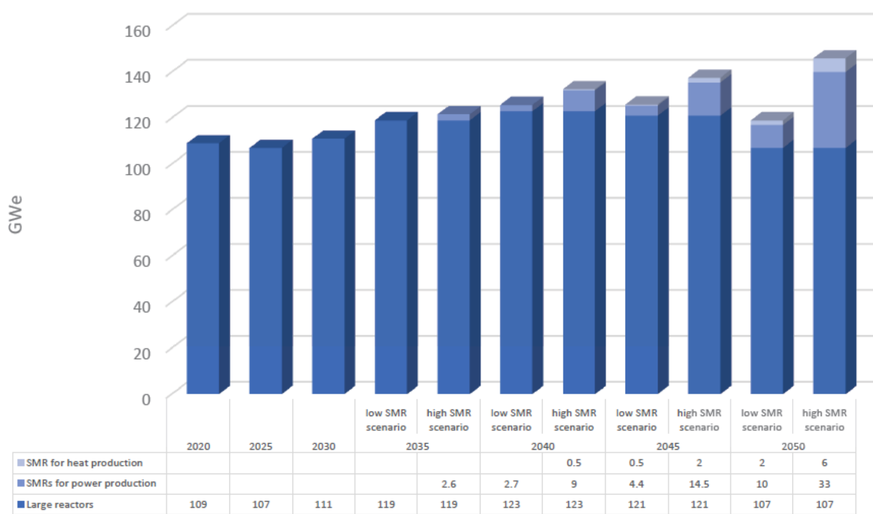
3 Identification of possible issues

3.1 Cogeneration application and energy mix scenarios

3.1.1 Key points from WP1

In the next few years, according to ESTAT [ESTA 2021], a deep electrification process is expected that will make electricity represent more than half of the energy consumed in EU. However, energy transition targets as reduction of CO_2 emission cannot be met without accounting for Heating and Cooling (H&C) which is a sector that constitutes about half of the total final energy needs in Europe. At the same time, the recently introduced REPowerEU Plan [REPowerEU 2022] has increased for 2030 the target to 20 Mt of renewable hydrogen. Low-cost hydrogen may benefit refineries and ammonia industries, that today share the biggest demand of H_2 (90% of which is fossil fuel produced), but also facilitate the growth of emerging "decarbonization" applications, like transport and industrial heat.

When designed not only as a source of electricity but also as source of other energy products such as heat or hydrogen carrier, Small Modular Reactors (SMRs) can assist in achieving the above-mentioned objectives. The TANDEM project aims at defining two HES configurations where SMRs can be integrated in a safe and cost-effective manner. Considering the first SMRs to be operated in Europe within almost ten years, a 2035 and a 2050 temporal scenarios are investigated adopting fixed figures for large reactors installed capacity while varying SMRs deployment as presented in **Figure 3-1**.



	2020	2025	2030	2035 low SMR scenario	2035 high SMR scenario	2040 low SMR scenario	2040 high SMR scenario	2045 low SMR scenario	2045 high SMR scenario	2050 low SMR scenario	2050 high SMR scenario
SMR for heat production							0.5	0.5	2	2	6
SMRs for power production					2.6	2.7	9	4.4	14.5	10	33
Large reactors	109	107	111	119	119	123	123	121	121	107	107

Figure 3-1 Nuclear installed capacity (GWe) by 2050 in EU27 for TANDEM energy scenarios

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Although a distinction between low and high SMR deployment scenarios may be of interest for other considerations (e.g., technical, environmental, etc.),- this is not considered essential for the purposes of safety implications addressed within TANDEM WP4.

The interest of the TANDEM project is to define few main energy scenarios and analyse all the possible technological solutions that can ease a safe and efficient integration of SMRs into hybrid energy systems. These study cases attempt to include market evolution as well as the principal European energy policies. However, different national situations and market volatility prevent to define a unique configuration, suggesting instead introducing figures of merit (FoM) that can help in economically assessing and optimizing composition and operation of HES. A set of techno-economic and environmental criteria is identified like the Net Present Value (NPV), defined as the difference between the present value of cash inflows and outflows, and the Levelized Cost of Electricity (LCOE), i.e. the price at which the generated electricity should be sold for the system to break even at the end of its lifetime. Furthermore, the impact of taxes, tax reliefs and incentives on competitiveness and cost-effectiveness is acknowledged and accounted. At last, in combination with technical FoM as the size of the HES components or the share of load demands/productions of each technology, the only considered environmental criterion is the CO_2 footprint, which includes direct and grey emissions (due to construction and deconstruction of facilities). It is worth noting that although safety assessment is not directly influenced by the above-mentioned criteria, possible measures that safety analysis might end up with could instead impact on components size (e.g., distance between modules), capital and operational costs.

Several studies dedicated to safety of nuclear reactors with cogeneration are analysed in report D4.1 pointing out the most critical issues. Each of them emphasizes different aspects but, at the same time, requires focusing on all the interactions between the SMR and the distinct modules of the HES. Even if the operational and environmental impacts related to the penetration of renewable energy sources and industrial processes will be investigated by WP3, the architectural layout of the two HES configurations already allows for some safety considerations. Therefore, in the following, the components that were found to be the most promising for TANDEM systems as well as their interfaces with the nuclear reactor, are presented.

The first configuration is intended for district heating and power supply and it is envisaged and mainly designed for large urban areas in Northern and Central Europe. The second HES concept is the Energy Hub, meant for complex distribution network of in-out energy carrier fluxes among various end-user segments.

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District Heating HES

The HES configuration for district heating and power supply has the primary goal to decarbonize the production of heat and electricity with emphasis on heating. The choice of DH solutions is adapted to the regional needs and policies of energy systems in Finland and Czech Republic, respectively selected as case studies to represent Northern and Central Europe scenarios. The main difference between the two relies in the incorporation of a large NPP for combined heat and power production (CHP) to the Czech case only.

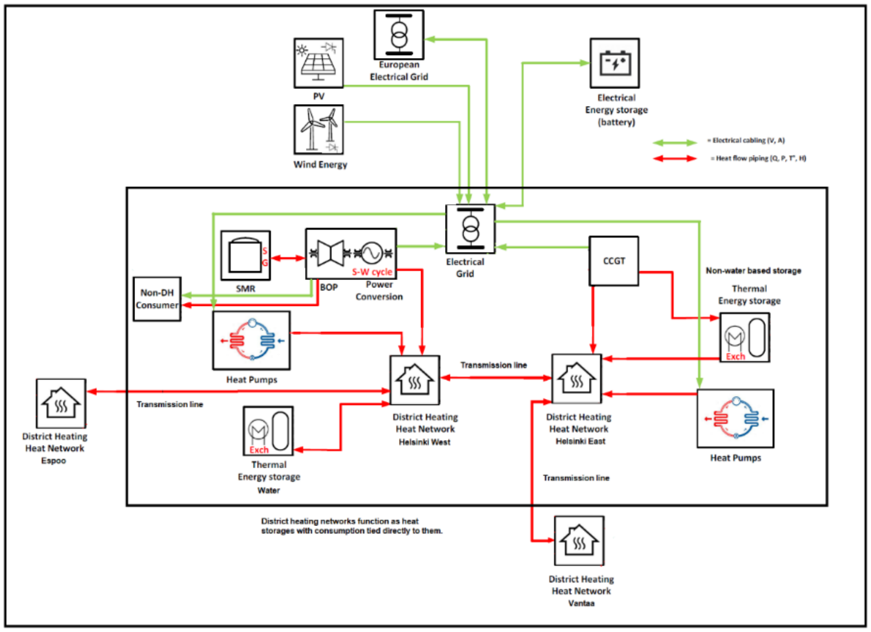

Figure 3-2 Conceptual layout of the hybrid energy system for district heat and power supply

An architecture based on 4 distinct DH network blocks is designed to match real infrastructures constraints (**Figure 3-2**) although the one directly coupled with the BoP is the only module strictly necessary for a safety analysis. Two of them are modelled as simplified and pre-determined heat sinks or outputs, whereas two major blocks can be adapted to the DH energy transition, depending on the local specifications. Power and heat are produced by a mix that includes SMR, RES, heat pumps and CCGT CHPs. The nuclear reactor is supposed to be built as close as possible to cities where the demand for heating is, to better exploit DH potential.

Energy Hub HES

The Energy Hub is a HES architecture where multiple energy carriers issued from various energy sources can be converted, stored, and supplied for end-user needs (**Figure 3-3**). In TANDEM the energy hub for SMR integration demonstration is inspired from a harbour-like infrastructure with only electrical power and heat carrier fluxes. The HES can be integrated with a High-Temperature Steam Electrolyser (HTSE) component for hydrogen production and a thermal storage while downstream applications, industrial needs or batteries can be included through boundary conditions (imposed fixed or adaptive power fluxes). A similar procedure is followed to define photovoltaic and hydroelectric energy supplies which are included as electric fluxes with daily or seasonal variations, defined according to data collected in the past years. Moreover, a generic Heat Usage block is connected to the network in order to include industrial or civil heat applications.

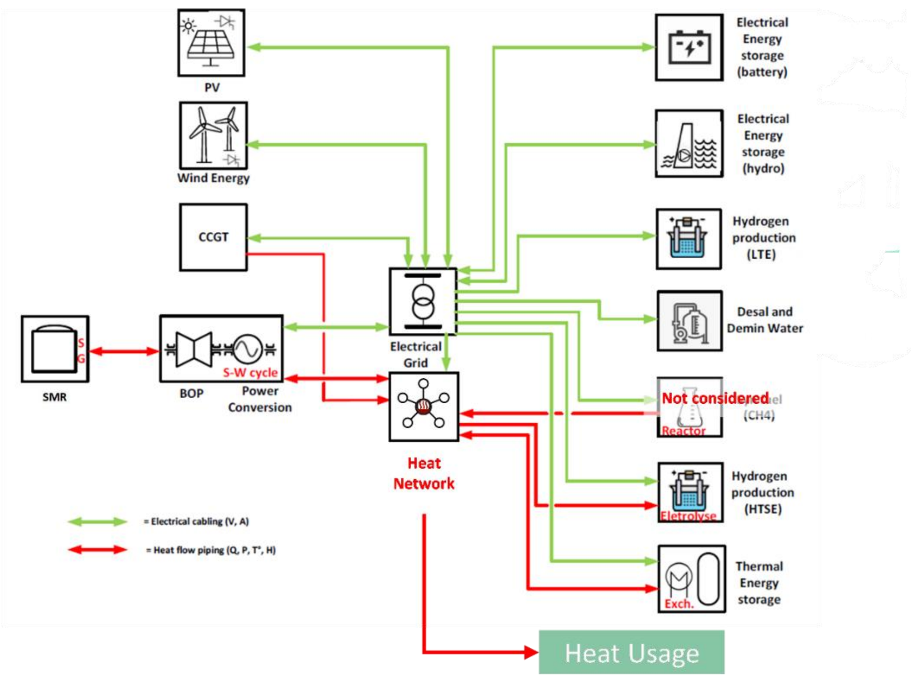

Figure 3-3 Conceptual layout Illustration of the hybrid energy system for the energy hub

The SMR concept chosen as use-case in TANDEM relies on the ESMR academic concept developed by POLIMI, GRS and VTT (the owners of the E-SMR dataset) in ELSMOR. It basically shares the same design philosophy for the reactor system as the NUWARD^{TM} concept with potential for extended applications. The E-SMR is an integral LW-SMR reaching 540 MWth/170 MWe. The core is a truncated classic PWR core with 17x17 assemblies and an active height of 2 m. The vessel contains eight steam generators that are *once-through* compact plate heat exchangers. Six of them are conceived for normal operation and two for accident scenarios. The passive safety system comprises two condensers located at the top of the containment and connected to the safety steam generators to remove the decay heat through natural circulation and thus to cool and depressurize the primary circuit by releasing heat to a water pool surrounding the metallic containment vessel. Broadly speaking the containment of the E-SMR has a vertical cylindrical part and two hemi-spherical parts at the top and the bottom (**Figure 3-**

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4). Connected to the upper plenum a safety system made by four identical accumulators is foreseen. The ultimate heat sink is the water wall (i.e., the cooling pool) around the containment. The metallic containment is immersed in the water wall and is used to evacuate the decay heat in a Loss Of Coolant Accident (LOCA) scenario. To maintain a liquid level in the vessel, two accumulators will inject water. From the reactor control point of view, the E-SMR is designed to operate without boron concentration within the primary fluid. Injection of boron is foreseen only in case of emergency shutdown whereas, in normal conditions, control rods are used to control reactivity.

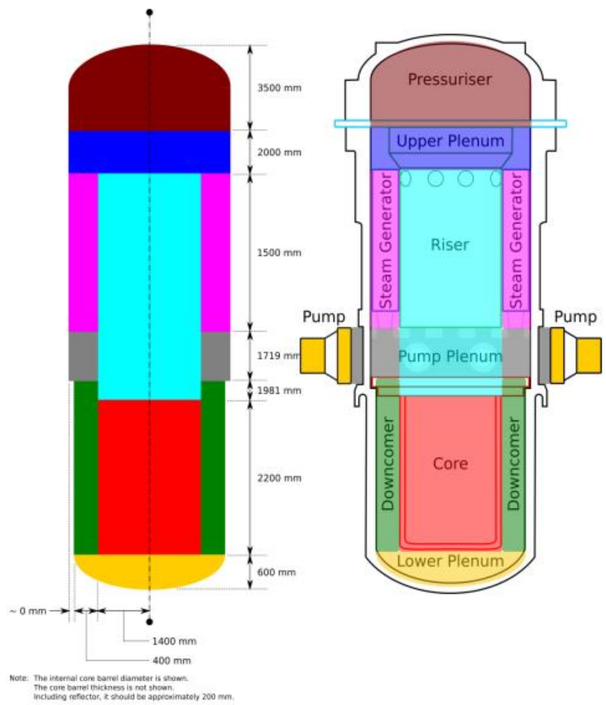

Figure 3-4 E-SMR vessel design and its different heights

The primary fluid (water) is heated in the core. It exists through the riser that also contains the control rod parking position and their internal drive mechanism. The pressurizer is integrated to the vessel, at the top of it and separated from the circulating fluid by a separation plate. The primary fluid reaches the Compact Steam Generator (CSG) at the outskirt of the vessel where it exchanges the heat with the secondary circuit before going through the primary pumps and to the core through the downcomer. It has to be noted that there can be a bypass in the CSG region, through the Safety-CSG that should not be active during normal operation but that still have free flow area allowing the fluid to circulate. The secondary side is at a much lower pressure (around 45 bar) compared to actual large conventional PWR in order to obtain superheated water at the exit of CGS as the latter are once-through exchangers.

Instrumentation and Control systems (I&C) include units designed to constantly control the main physical quantities of the nuclear power plant and actively operate in order to keep them within fixed safety ranges, in a safe and fast way. Based on the well-known I&C systems adopted for other load-following oriented reactor as the French PWRs, a set of primary control systems is proposed to keep temperature, pressure and chemical properties of primary water close to set-point values. A turbine control system is envisaged to regulate the steam mass flow rate through the turbine while the pressure in the secondary circuit is controlled thanks to valves and pump speed control systems. Directly involved in the HES framework, a turbine bypass dump is adopted to manage transient scenarios in which power demand experiences a rapid decrease from the electrical grid and/or the heat users. By delivering additional steam to the condenser, temperature and pressure are limited in both circuits, reducing the risk of automatic shutdown.

The majority of the above mentioned I&C systems must communicate with both the nuclear steam supply system (NSSS) and the BoP. The latter unit encompasses all structures, systems and components that connect the SMR with the HES as well as the power conversion system that is composed by high and low pressure turbines, moisture separator, reheater(s) and condenser. With respect to conventional NPPs, a HES also requires additional safety measures such as intermediate loops and heat exchangers to ensure a physical barrier separating the nuclear modules and external technologies. This measure is foreseen for almost all the applications except for the thermal energy storage that is inherently decoupled from the SMR, through dedicated heat exchangers for the charging and discharging functions. Depending on how each external HES module is connected to the BoP and on the distance from the nuclear reactor, different safety considerations may apply.

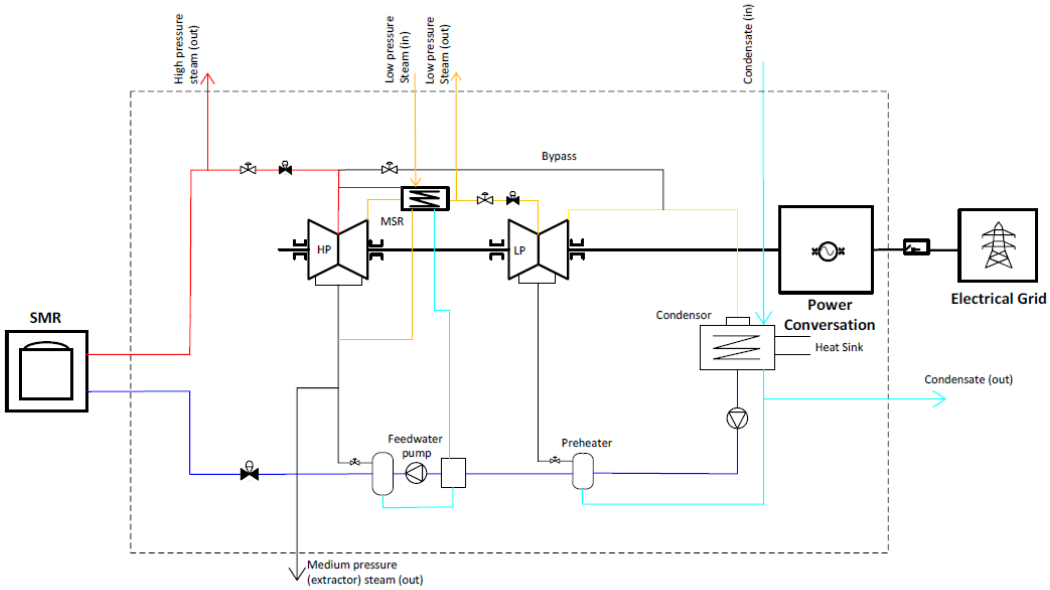

Figure 3-5 Model flow diagram (created by TRACTEBEL in the scope of the TANDEM project)

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Figure 3-5 represents the high-level architecture of the NPP BoP module, while the connections between this and the different external end-users are depicted in **Figure 3-6**. On the other hand, **Figure 3-7** shows how the electrical power is produced by, distributed among and used by the different units.

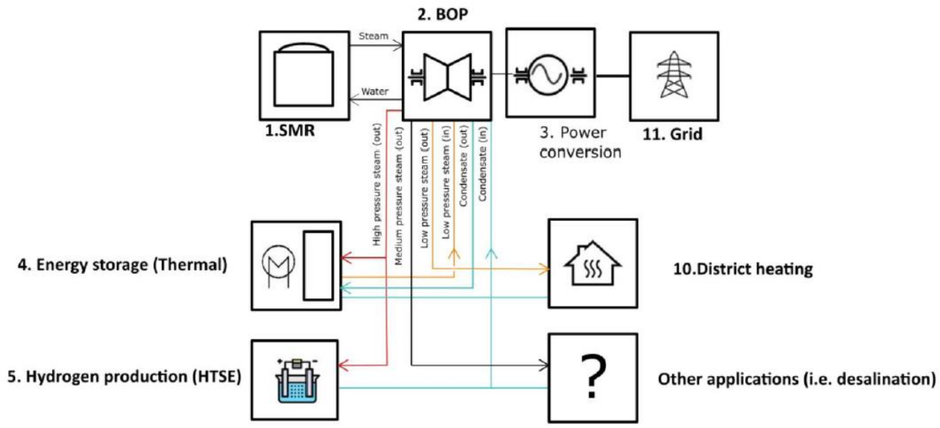

Figure 3-6 Coupling of BoP with other modules of HES (the steam and condensates at different pressures are coloured differently) (created by TRACTBEL in the scope of the TANDEM project)

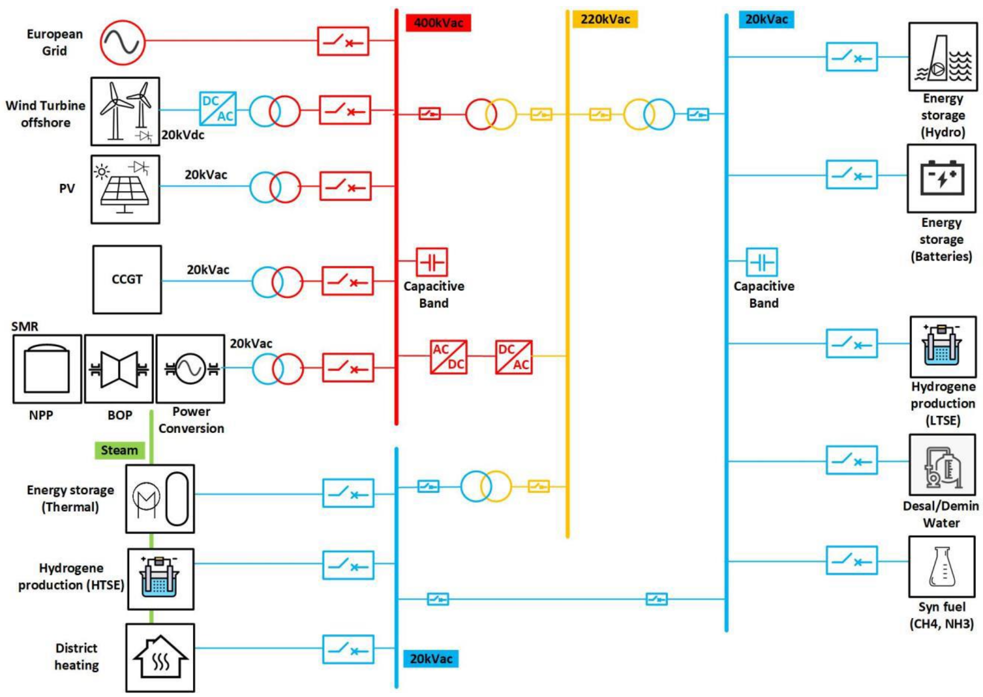

Figure 3-7 Model Flow Diagram (created by TRACTEBEL in the scope of the TANDEM project)

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In the following, the selected technologies for the main external modules are presented and discussed.

Energy storage.

Energy storages represent crucial components to be implemented in a grid with high penetration of intermittent sources thanks to their capability of damping the load variations.

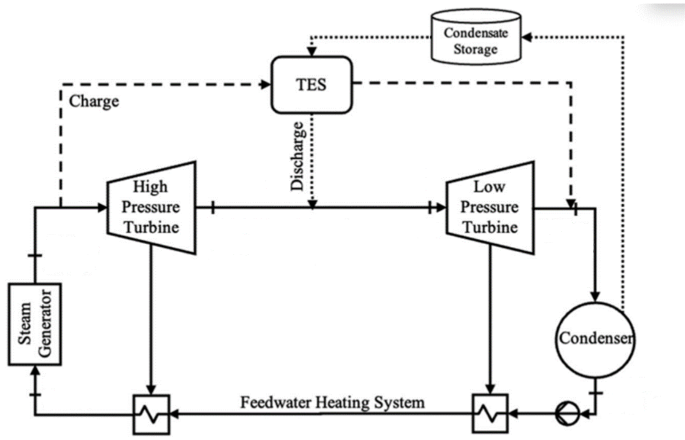

Figure 3-8 TES connection to turbine cycle (created by ENERGORISK in the scope of the TANDEM project)

Electrical energy storages permit to control grid frequency and voltage while thermal energy storages allow to increase the efficiency of the steam cycle of the NPP. As a base case for TANDEM project, taking into consideration E-SMR parameters on steam temperature, a sensible-heat energy storage made of a two-tank system with *Therminol 66* as thermal oil is proposed. As a preliminary option, TES (Thermal Energy Storage) is charged using live steam (high pressure steam at the outlet of the OTSG, before high pressure turbine) while discharge relies on the reintroduction of steam before the low-pressure part of the turbine (**Figure 3-8**). On the other hand, flow-battery systems and regenerative fuel cells were discarded as electrical storage in favour of batteries as lithium-ion, lead-acid, VRB (Vanadium Redox Battery) and NaS systems. The selection of the best BES (Battery Energy Storage) is then submitted to the definition of the specific application requirements in terms of power quality, energy management, emergency back-up power etc.

Hydrogen production.

Hydrogen is a highly versatile energy vector, and this makes it an important contributor to the clean energy transition in particular as fuel, energy storage system and industrial feedstock. Hydrogen can be extracted by different mechanism and among the requested energy source nuclear energy is an interesting choice since it does not imply production of CO_2. For the TANDEM project, both the low-temperature and high-temperature technologies coupled with the E-SMR

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are intended to be investigated. Regarding the HTSE, the Solid Oxide Cell Electrolyser (SOEC) is considered the most promising candidate and a general layout of the possible connections plant is reported in **Figure 3-9**. However, some modification can be applied depending on the outcomes of performance and safety analysis. The recirculation loop that provides hydrogen could be replaced by an appropriate hydrogen storage while the sweep gas inlet is needed only if the electrolyser has to work also as a fuel cell. Low-temperature electrolysis instead can use electricity exclusively to get hydrogen and oxygen from water and the use of Polymer Electrolyte Membrane (PEM) technology is acknowledged as the most mature and efficient for cogenerating purposes.

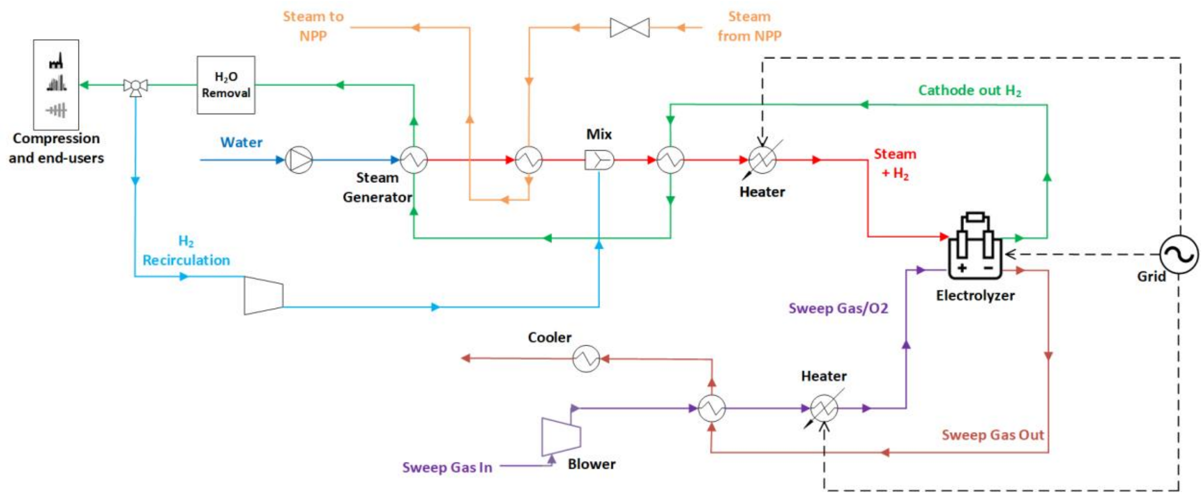

Figure 3-9 General layout of a High Temperature Steam Electrolysis plant coupled with an NPP

Desalination.

The situation of water supply for population is believed to deteriorate in the next years due to global warming. Desalination technologies will therefore become more and more important and given the high amount of energy needed to produce pure water, the nuclear source can play a relevant role in this context. The most commonly adopted desalination technologies involve reverse osmosis (RO) and thermal processes such as multi-stage flash (MSF) and multi-effect distillation (MED). They both represent a possible solution not to waste excess generation capacity during off peak hours, however they differ for the physical mechanisms through which fresh water is obtained. In fact, RO process just needs electrical energy whereas thermal options require also a heat supply which should be provided by the BoP. The two mechanisms can be also implemented in combination to improve the desalination performance. For the sake of this project, comparative analysis yields the reverse osmosis as the most appropriate solution.

Synthetic Fuel.

Another solution that increases the flexibility of the HES, stabilizing the grid, is the usage of electricity/heat to combine hydrogen with CO_2 producing methane by chemical reaction (methanation). This application enables energy storage over long periods, fast energy release in case of electricity shortage but also offers a solution to hydrogen storage. The methanation reactor is not directly connected to the BoP since it receives H_2 and CO_2 from storages or distribution networks and exports the excess heat produced by the process as steam. Whilst promising, this technology has not been considered for the initial HES configuration.

Wind energy.

On/Offshore wind farms can be connected to the local electrical grid of the HES and constitute an important source of intermittent renewable energy. HES, being capital intensive technologies, should operate at full load to maximize profits. This implies to allocate surplus thermal and/or electricity energy to produce more profitably storable product during off peak hours. Thermal and electrical energy from the NPP might instead be used for other processes as water desalination or hydrogen production while the wind farm compensates for the lower power output to the grid. In addition, the installation of batteries (BESS) would ensure a stable and efficient supply, in case of wind absence.

District Heating.

District heating (DH) systems generally exploit water or steam as carrier of thermal energy to heat buildings and warm tap water. However, transporting heat for long distances involves high pumping costs and heat losses which require to locate the power plants only few kilometres apart from the consumer. During colder seasons the heat demand is higher but the same goes for the heat losses which depend on the surrounding environment temperature. Among the different district heating production modes, heat only boiler reactors represent the nuclear version of Heat Only Boilers (HOBs) and they are small size nuclear plants that produce only thermal energy. Whenever the heat needed to warm water is obtained by redirecting part of the steam commonly used for electricity production, the term “Combined heat and power” is adopted to refer to this DH practice. Eventually, HES can be equipped with heat pumps that are able to exploit “low energy sources” (<100°C) to provide heating but also cooling of buildings.

With this overview, the cogenerating layout should appear in a more functional way, easing the following safety analysis and estimation. In particular, the interactions among different units within the hybrid energy system should be investigated when it comes to identify AOOs, DBA and external hazards and assess their frequency of occurrence.

3.1.2 Key points from Safety Analysis Reports

The use of NPPs for cogeneration has been effective for many years in several countries. These configurations have often been considered as the juxtaposition of two industrial processes with a customer-supplier relationship and a small number of interfaces. The risk analyses are then independent, considering the other process as a potential external hazard. First, it is important to recall that installations that could cause significant risks to the public and the environment are subject to national legislation derived from the European SEVESO II Directive. In this context, risk assessments are mandatory.

It should be emphasised that there is a fundamental difference in safety philosophy between a nuclear and an industrial plant. The objective of safety design in a nuclear facility is to confine radioactive materials within the facility. In contrast, confinement of materials in a chemical plant may increase the individual risk to the public and workers due to the potential for confined explosions and hazardous chemical accumulations. For example, in a water-splitting hydrogen production system, H2 and O2 are produced simultaneously. This creates the possibility of an internal explosion if they are accidentally mixed. On the other hand, the two gases are produced in different process steps that are physically separated. To prevent internal explosions, an emergency purging system shall be provided to remove hydrogen from pipes and vessels.

In our case of integrating SMRs into a HES, the interface is more dynamic, and this requires the development of additional specific guidelines, which is the aim of this report. However, as a first step, it is a good approach to use the assessments already carried out in these areas as a basis for our reflections. Cogeneration is also used in other parts of the world, but mainly for industrial purposes. In India, nuclear cogeneration reactors have been built to supply heat to industrial sites. In Japan, they power seawater desalination plants. We then try to get access to PSAR and FSAR documents for existing nuclear cogeneration plants, where freely available, to understand how the problem has been dealt with in previous applications.

From the list of nuclear cogeneration plants in the IAEA document [IAEA 2017], we contacted the nuclear safety authorities of the USA (for the desalination plant at Diablo Canyon), Czech Republic (for the district heating capability of the Temelin and Dukovany NPPs), Japan (for Ohi, Ikata, Genkai, Takahama, Kashiwazaki with desalination), Canada (for the district heating capability of the Bruce A NPP).

The aim of such approach was to obtain answers to the following questions.

- Were there particular initiating events, conducting to specific transients, that must be considered in the assessment?

- Has there been any identification of components or systems (in the nuclear island) that could be sensible at this use or is it considered as totally decorrelated?
- Has it been considered that there could be specific DBC or AOO or DBA?
- Has there been any specification ranges in operating conditions compared to other NPP?

Replies have been received, but unfortunately, PSAR/ FSAR could not be released for sharing often because those documents are controlled nuclear information. Then we have not been able to find any information that could help us with our project. However, the interesting information we received suggests that a standard conventional external industrial risk assessment approach would have been used in these cases.

Usually, the main equipment of district heating system, as heat exchangers and circulating pumps, are in turbine hall, near to source of heating steam, which is main turbine. The safety classification of the system is Safety Class 4 (normal non-safety system).

An example of localization is presented in **Figure 3-10** below.

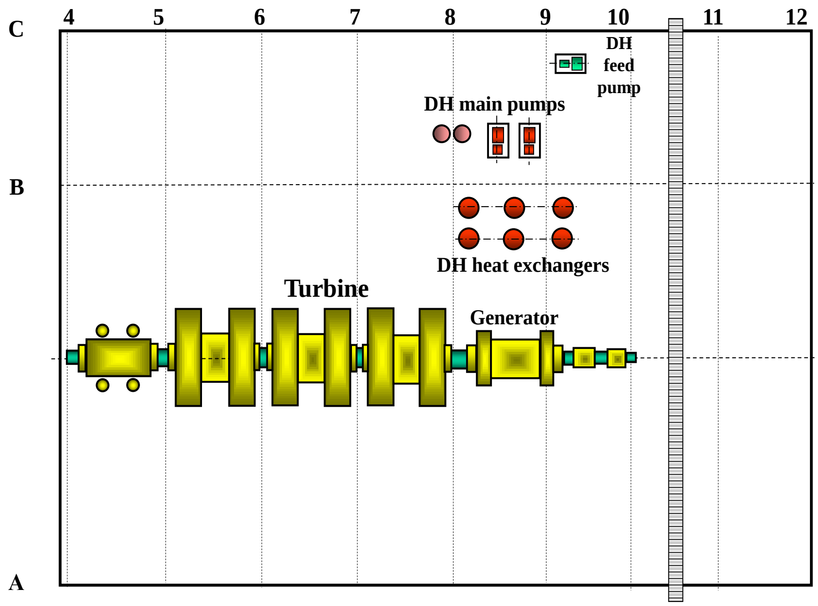

Figure 3-10 Example of location for main equipment of district heating system

Transients /initiating events at district heating system that can have impact on SMR operation include the following:

- Rupture of steam line from turbine to DH system (or rupture of DH heat exchangers) during nominal power operation of reactor. This leads to the release of steam into the turbine hall and the steaming of the area (at turbine hall elevations) of the DH system, to pressure increase in the turbine condensers, and to trip of the condensate pump.

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Thus, some failures at DH system may lead to loss of main turbine and possible failures of another nuclear SSCs due to spatial interactions (pipe whip, high humidity, high temperature, missiles, etc.).

- Rupture of DH pipeline in turbine hall, and consequential flooding of the turbine hall equipment located near to the rupture place and compartment below the rupture.
- Equipment of DH system (pumps, automatic valves, electrical equipment, etc.) as well temporary heat sources can be sources of fires at turbine hall. And thus, ignition sources and fire loads of DH system should be properly accounted for in fire hazard analysis / fire probabilistic safety assessments.

Other malfunctions and events at DH system (e.g., trip of DH pumps, closure of check valves at uncontrolled steam extraction lines, loss of power, clogging of filters) have no influence on operation of nuclear systems.

Also, depending on the design of the nuclear plant, there may be cases when contaminated (radioactive) water enter the DH system. This may be as result of failures in water purification / water treatment systems that are used to clean contaminated water from primary circuit. Therefore, the design of HES should provide measures to prevent ingress of contaminated water to DH system and to consumers of the system, as discussed below.

FORTUM for whom NPP Combined Heat and Power (CHP) is a familiar concept confirmed that Loviisa 3 included CHP option, but it did not receive the decision in principle in 2010. There are some public reports about the topic but without details about safety assessment. In addition to the reports available, FORTUM has identified the following topics of interest (private communication):

- Site location – emergency planning zone. NPP CHP should be located rather near the consumption.
- District heating (DH) exchanger and pipes “are behind the secondary circuit” i.e. kind of tertiary circuit.
- In theory certain transients in DH network could affect turbine island. However, these transients should not propagate back to nuclear island.
- some examples:
    - loss of flow in DH network (pump trip)
    - loss of volume in DH network
- If DH production ends, the NPP should either reduce reactor power or increase electricity production or dump steam into condenser: this depends on design choices.

Moreover, we can refer to international documents related to the safety assessment of CHP plants. A detailed and exhaustive bibliography has been compiled in the Deliverable WP4/D4.1.

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As a basis, we can refer to IAEA SSR-2/1 Requirement 35, which states that NPPs used for combined heat and power, heat generation or desalination NPPs coupled with heat utilisation units (e.g., for district heating) and/or water desalination units, shall be designed to prevent processes that transport radionuclides from the NPP to the desalination or district heating unit under operating and accident conditions. In the same spirit, ELSMOR Deliverable 2.12 [ELSMOR D2.12] recommends these important points:

- "If the SMR is used for cogeneration, the impact on the containment integrity due to hazards introduced by other plants (chemical or hydrogen facility) should be considered as well".
- "the location of SMRs is of considerable importance to identify hazards".
- "Cogeneration could introduce others external hazard related to the industrial activities".

"Embedded cogeneration" with a nuclear reactor requires careful consideration of how the industrial process is coupled to the power source. Safety, economic and operational factors are involved. If we refer to the available PSAR, the main safety concern could be limited to preventing the transfer of radioactivity from the nuclear reactor to the industrial process, even under accident conditions. However, another consideration has to be taken into account, especially if the industrial process using the heat has a relatively strong coupling with the secondary circuit.

Safe operation of the nuclear power plant can be ensured by providing a set of regulatory safety requirements and practical design considerations when coupling the nuclear reactor with a conventionally designed industrial plan (chemical plant for hydrogen production). This notably implies the need to define appropriate separation distances between the reactor building control room and the industrial plant against flammable gas leakage and against toxic gas leakage. As for any other operating NPP, the technical requirements related to the integration of SMRs in a HES will be elaborated with international safety standards and validated by national regulations.

Safety assessment feasibility on coupling nuclear and industrial facilities has also been studied in the framework of the EUropairs project. It dealt with nuclear cogeneration systems connected to conventional industrial processes. Main outcomes of this project have been also detailed in the D4.1 deliverable. It highlights that the evaluation of the consequences of external hazards on the safety functions may consider events combinations and consequences on auxiliary equipment related to safety (venting systems, diesels, filters, etc.). In addition, a more precise definition of the systems is required to cope with family of external hazards (ex.: control and monitoring, ventilation systems, relief valves, level of emergency power supply, etc.). Concerning the risk induced by the coupling fluid, a particular attention must be paid to:

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- The ingress of the coupling fluid inside the nuclear circuits (depends on pressure staging) by considering isolation and fast draining systems.
- The ingress of chemical products in the reactor hall or in the circuits (combined failures) by considering that explosion of chemical products inside the reactor hall should be excluded.
- The shut down process by taken into account consequences of the resulting thermal transients that need to be integrated in the design and by considering mitigation systems possible (damping systems, isolation valves, etc.).

To conclude general safety requirements from conventional cogeneration safety assessment are in particular as follow:

- Process induced hazards shall not represent a major proportion of the accidental initiators of potential radioactive release.
- Transients imposed to the nuclear plant by the coupled process(es) shall not lower the safety margins ensuring the resistance of the barriers (barriers against radioactive release).
- Hazards induced by the industrial process shall not diminish the capability of the workers to operate the plants safely.
- A safely distance (to be defined) between the nuclear island and the industrial process might help mastering the risks.

Moreover, it should be noted that in 2009 Fortum submitted a project to the government for a new reactor at Loviisa (LO3), a combined heat and power plant (combined heat and power is widely developed in Finland but has never yet been applied to a nuclear power plant). However, unlike TVO and Fennovoima, it has not obtained authorisation in principle. An analysis of this situation could provide interesting elements for the construction of our methodology.

3.2 Load Following

3.2.1 Key point from D4.1

The deliverable D4.1 of the TANDEM Project (Pucciarelli et al., 2023) reported the status of European studies on safety analysis of SMRs from the operational flexibility and cogeneration viewpoints. Though the body of the work made in Europe in relation to the safety aspects of LW SMRs when included into HESs in a cogeneration pattern was found really limited, profit was taken from the studies made in the broader field of nuclear cogeneration with other types of nuclear reactors, namely High Temperature or Very High Temperature Reactors ((V)HTRs).

The European Nuclear Cogeneration Industrial Initiative platform (NC2I), in particular, has worked on different projects in the past, to highlight the capabilities and the challenges posed by

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(V)HTRs in a cogenerating environment. In particular, the reports of EUROPAIRS [EUROPAIRS], NC2I-R [NC2I-R] and GEMINI+ [GEMINI] projects offered an interesting background for the work to be performed in TANDEM, with a broad analysis of the two-way interactions that the NPP may have with cogenerating applications. Other documents were also considered, from WENRA, IAEA and the SMR Regulators' Forum, for enriching the analysis. Several suggestions were made for setting up an impact assessment methodology of the cogeneration system on the safety margins of the involved nuclear reactor.

Based on that analysis, "flexibility" was considered as a characteristic a nuclear reactor should bear, to be profitable and safely interfaced with cogeneration system, according to the amount of energy storage devices included in the system. One may have to assess possible additional AOOs in safety analyses, as well as the need of load following capabilities of the nuclear reactor, because of the mentioned two-way interactions the nuclear reactor may have with cogeneration system. In this regard, the cogenerating configuration will suggest if the operating regimes of the nuclear reactor must be reconsidered for part-load operation in more or less extended periods (on yearly or daily basis). In other words, if the reactor can mostly operate at full power by just apportioning the steam generated for electricity production and other uses. In the report, the involved TSOs contributed to load following related issues, based on their experience, in particular in France and in Germany. Finally, first suggestions about the methodology for the assessment of the safety margins impacted by cogeneration were proposed, easing the reporting work for the present deliverable.

Relevant issues highlighted by a literature survey were pointed out in section 2.3 of D4.1, where the following aspects were considered important for the development of the safety margin assessment methodology:

- Definition of the cogeneration application and of the energy mix scenario

Indeed, a first necessary step is the definition of a specific cogenerating scenario to be addressed in the safety analysis of the HES. This is necessary in view of restricting the scope of the analysis to an exemplary HES, to be characterised by the type and number of systems interacting with the nuclear reactor and by their particular needs in terms of operating regimes and possible threats posed to the continuity of reactor operation and to its safety.

- Degree of decoupling of the NPP from the grid and the cogeneration process

In particular, in this context a quite relevant feature of the HES components is related to their degree of coupling (or decoupling) with the nuclear reactor. In principle, if the nuclear reactor can be considered sufficiently far away in a physical and logical sense from the cogenerating applications, its safety assessment can be supposed not to differ in any relevant form from

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the one applied to a generic reactor whose purpose is just electricity generation. On the contrary, two-way interactions can take place between the nuclear reactor and, e.g., electrolysers or a district heating network, and may have to be properly accounted for, in order to assess the safety margins impacted by the presence of the HES. These interactions must be attentively considered for the specific system considered in order to identify their impact on the AOOs and, in case, on the different postulated accident scenarios to be assessed in a NPP Probabilistic Safety Assessment (PSA).

The following aspects were identified in D4.1 as the main issues:

- Differences in regulations applicable to the industrial process and to the nuclear power plants, with the need to manage the related interfaces.
- Differences in standards, e.g., applied for cybersecurity and to component design of the different nuclear and non-nuclear systems.
- Physical separation: a sufficient distance of the NPP from the production process seems to be necessary and sometimes a further barrier is needed (see below) to avoid contamination of the process by the nuclear power plant or, on the contrary, to protect the NPP from external perturbations; one notices that this is a key issue to be considered in TANDEM.
- Ownership separation: whether or not the NPP and the cogenerating application are under the same ownership, responsibilities (e.g., considering nuclear safety as an overriding priority). Moreover, any kind of interface should be attentively addressed.
- Human resource separation: workers from the NPP or the cogenerating application may be subjected to different hazards and may have different authorisations for entering the plants: the degree of separation of the two workforces must be considered.
- Bidirectional barriers: reinforcing what previously stated on physical separation, additional bidirectional barriers should be considered to separate the NPP from the rest of the HES. Level of nuclear safety and interfaced industrial application.

It is recognised in several analyses that the level of safety of the nuclear reactor should not be decreased by the presence of the cogenerating system. For the cogenerating scenario, it is necessary to carefully established, to what extent this objective may be reached, since it is clear that the two-way interactions between the NPP and the HES do affect the safety. It may be also necessary to confirm that, the currently applicable nuclear safety objectives are largely met for operation of the HES. Moreover, it must be considered that the safety requirements normally imposed to a NPP may be much more stringent than those applicable to usual industrial applications. This gives rise to a mix of regulations and standards which must be consistent, while making nuclear safety the top priority.

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- Methodology of safety analysis

The classical safety philosophy principles applied worldwide in NPP design should be applicable also for systems interfaced with the NPP (cogeneration systems, BoP…), all the HES, so that DiD and Multiple Barriers, for instance, should be considered by dealing with the safety assessment of the HES. Anyway, it must be clarified, if the additional risks coming from cogeneration can be treated as common "external" threats, although they are posed to the NPP by the presence of cogenerating systems and traded by usual methodologies applied for events occurring out of the limits of the nuclear reactor or caused by natural phenomena.

- Flexibility in NPP operation

The degree of load-following of the NPP must be discussed considering the specific needs of the rest of the HES. A basic choice may be whether the NPP must be operated on load-following basis to best meet the needs of cogenerating applications or whether it would be more cost-effective to keep the NPP at full power by apportioning its output between the different external applications (electricity and hydrogen production, district heating, etc.). One notices that economy and safety aspects strictly depend on this choice.

- Safety (and security) of the electrical energy supply to the NPP

The availability of a reliable off-site electrical power source for the NPP in case of need, is a crucial aspect for its safe operation. Additional AOO may be related to the intermittency of the electrical grid supplying the plant. This aspect should not be underestimated in future energy networks dominated by renewable energy sources, unless a sufficient degree of energy storage be assured with technologies that, to date, are mostly proposed, tested and made commercially available.

- Impact of the NPP on the industrial process

Possible radioactive contamination of plant effluents routed to the industrial application and also, stability production requirements may be applicable to the NPP, to serve in the cogenerating application. This may affect, as already mentioned, routine procedures as outages for maintenance of the nuclear reactor, thus creating a further interface to be managed, affecting safety of the reactor in an indirect way.

- General safety issues of SMRs in view of licensing

SMRs are presently under conception and start to be considered in licensing applications. Some of their features may suggest a "graded" approach of the safety assessment, considering the claims of improved safety, but may also raise additional concerns for the "advanced" nature of some of their features. Multi-unit and multi-module issues can be considered among these additional concerns brought by the novel features of SMRs.

- Definition of meaningful study cases

As in previous projects organised in the frame of NC2I, specific study cases need to be considered for analysing the safety features of the LW SMR in hybrid cogeneration patterns. Without claiming to be completely exhaustive in assessing study cases at this stage, sufficiently representative cases should be addressed for providing the present work with a reasonable generality.

Considerations on cogeneration aspects are also introduced, since in German NPPs of Stade, Lingen and Krümmel have been variously operated in cogeneration mode and specific experience is available.

In relation to load following, thermo-mechanical consequences are the most important areas to be addressed in the safety assessment of NPPs operating in such regime. The integrity of the pressure-retaining walls and structures, the structural safety of the components and the functional capability of the active or passive components are considered. These components are designed with reference to a load spectrum that must be tailored to the specific conditions envisaged during actual operation. Parameters like the magnitude of power changes and their rate in time are indeed important for the design of components. Adaptation of the control system may be also required to cope with load following scenarios; corrosion issues must be considered as well in case of deviations from the prescribed chemical conditions due to load following transients.

Considerations on nuclear reactors in cogeneration applications are then elaborated, suggesting that NPPs for heat generation need to be specifically designed for combined heat and power production. It is suggested that in such concepts "the aim should be to decouple the nuclear reactors used for energy supply from external power fluctuations as far as possible and to maintain nominal load operation that is as undisturbed as possible". Radiological, chemical and energetic hazards should be considered in the design of coupled systems. In particular, radiological, chemical and fire hazards as well as, missiles generated by pressurised component failure, explosions or deflagrations and toxic substances shall be considered. It is noted that these types of hazards are already within the scope of man-made hazards and internal hazards covered by existing regulatory guidance. Therefore, good practices already in place can be transferred to the sector of nuclear hybrid cogeneration. Possible protective measures are discussed, considering the protection from mutual interaction of the nuclear and the cogenerating plant. Sensors for detecting the accumulation or unwanted dispersion of toxic substances are also

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considered. Postulated aircraft crashes may have a different impact when addressing nuclear and chemo-technical plants and, as such, must be specifically considered in cogeneration plants. Hazardous materials may also leak from the cogenerating plant and the transport of different kinds of goods by heavy vehicles on a site may pose additional threats. Licensing should consider not only technical aspects but also administrative ones, e.g., related to responsibility.

Control strategies in cogeneration need moreover to consider different degrees of freedom: 1) supply of process heat according to demand; 2) supply of the required electrical energy; 3) flexible balancing of power plant capacities and process heat demand by purchasing electricity from or feeding it into the public grid. The reliability of the supply of process heat is a key concern for some cogenerating application and this, needs to be properly considered in designing the nuclear hybrid cogenerating application. Load following and adaptation to rapid variation of regimes are considered, also in view of the use of Gas Cooled Reactors (GCR) and Molten Salt Reactors (MSR) in HESs. Controlling a coupled energy grid is indeed a complex task that needs central management. Moreover, the possible interfaces between safety, security and safeguards (the “three S”) are commented, suggesting that the connection between the NPP and the cogenerating application may create pathways for endangering the safety and the security of the plant. On the other side, the needs for the accounting of materials subjected to safeguard seem not to be changed by the presence of cogenerating plants.

Expanding more closely on load following issues, quick variations, daily variations and weekly/monthly variations are reported in large nuclear fleets as the one of France, owing to the huge share of electricity production covered by the nuclear source. The influence of these practices on the operating plants is considered in terms of their effects on: the reactor core power distribution and even more in general, on core depletion; the complex phenomena of pellet-cladding interaction (PCI); thermal and load fatigue; wear and tear; waste management, owing to the need to vary boron concentration for reactivity control. The main issues anyway concern the core, in relation to the effects on fuel and possible risks of boiling crisis occurrence. The experience accumulated from existing nuclear reactors on load following effects may not be directly applicable to SMRs, which suggest caution in extrapolating the obtained data.

Embryonal suggestions for the safety margin assessment methodology targeted herein, are proposed in D4.1 for further reflection. In particular, different levels of screening of the relevant issues affecting the safety of the nuclear plant in the HES are suggested.

- At a first level, the WENRA safety objectives, as summarised in WENRA [WENRA 2021], are considered, proposing to discuss which features of the cogeneration application may impact the SMR safety margins. The safety objectives are related to: O1. Normal operation, abnormal events and prevention of accidents; O2. Accidents without core melt; O3. Accidents with core melt; O4. Independence between all levels of DiD; O5. Safety and security interfaces; O6. Radiation protection and waste management; O7. Leadership and management for safety.

- At a second level, concerns related to the impact on safety margins can be proposed by considering categories of issues from SMR Regulators' Forum (2021): 1) FOAK issues; 2) Multi-unit/multi-module issues; 3) Passive Safety; 4) Exclusion of Faults from Safety Analysis; 5) Severe Accidents and Design Extension Conditions.
- At a third level, the basic DiD levels can be used for an additional screening, considering coupling between the SMR and the cogeneration system which may affect each DiD level. This is defined in the IAEA Safety Glossary [IAEA 2018]: Level 1: Prevention of abnormal operation and failures; Level 2: Control of abnormal operation and detection of failures; Level 3: Control of accidents within the design basis; Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents; Level 5: Mitigation of radiological consequences of significant releases of radioactive material.

The report finally suggests taking as a meaningful example of the result of the work to be conducted in TANDEM, the material described in a report issued by the EUROPAIRS project [EUROPAIRS], related to (V)HTRs, whose considerations may help in sketching the targeted safety methodologies for LW SMRs in cogeneration systems.

3.2.2 Key points from Safety Analysis Reports of operating NPP

Several NPP, especially in the French fleet, are operated to perform (electrical) load following for several decades. The main concerns of load following from a safety standpoint, compared to base load operation, are related to the Pellet Clad Interaction (PCI) and to hot spots.

The PCI in the load following context is mainly related to the different thermal dilatation of the pellet and the clad. During power increases, the pellet dilates faster compared to the clad potentially leading to gap closure (if not already) and increase in clad mechanical stress. In the worst cases, it may result in rod failure.

To limit the amplitude of the phenomena, the power ramp speed is limited:

- 3% of Nominal Power / hour for operation following shutdown and fuel management.
- 5% of Nominal Power / minute everywhere else.

Moreover, operation at "low" load is constrained in time [MORILHAT] to limit the fuel conditioning and the consequent increased clad stress during the following ramp to nominal power.

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Load following may lead to an increase in of *hot spots* for several reasons. First, power transients *immediately* generate power axial variations, due to the modifications of the axial temperature gradient and to the insertion of control rods (which also generates radial power variations and hot spots). These variations lead to modifications of the equilibrium (production vs destruction/decay) of fission products: some, such the Xenon 135, may be strong neutron absorbers leading to additional and oscillating axial power variation with a *time constant of several hours*; these oscillations, if not controlled by the operators, may diverge in “big” cores. Such axial and radial power variations and heterogeneities, at constant power increases the amplitude of local power, which must be controlled to avoid risk of fuel melting or Departure from Nucleate Boiling (DNB).

The effect of control rods (and consequently of Xenon oscillations) can be reduced by using rod characterized by low absorption, such as the so-called French “grey” rods; the axial flux perturbation due to their insertion is lower than the one due to classical “black” rods (e.g. Ag-In-Cd). Moreover, it has to be known that for “small” core (i.e., reduced core height) such as for the E-SMR design (and other typical SMRs), the problem related to Xenon oscillations is lowered.

To tackle such phenomena during operation, an allowed operating domain in the power / axial-offset plan is defined. The reactor is due to remain with low values of AO, especially for high power values. Excursions to higher AO values are accepted for a limited period (the greater the excursion, the lower the time) to avoid important Xenon perturbations and consequent oscillations. The domain limit is especially constrained for positive AO values, which would have a significant impact on DNB margins.

Finally, the strong local heterogeneities in power due to the insertion of rods during power transients results in significant burnup heterogeneities (for relatively long steady-state operation). The “shaded” area, with a significantly lower burnup, can then generate local hot spots, which can reduce margins to DNB and Fuel Melting. To limit such phenomena, operation with (significantly) inserted rods is limited in time (in such cases, boron injection can be used to substitute the rods, but at the price of a far lower readiness to return to nominal power due to higher characteristic time for boron management compared to rod withdrawal).

3.3 Limits and impacts on components and systems

3.3.1 Introduction

In the previous section of this report, the importance of focusing on the interfaces and connections among the different units of the HES is emphasized. The model flow diagram in ***Figure 3.11*** represents a preliminary architecture of the "energy hub" BoP components and relative connections, with an emphasis on the heat fluxes going in and out of the modules. The

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lines may represent the interfaces with external units such as energy storages, electrolyzers, industrial heating networks, etc., depending on the steam temperature required. As it can be seen, each connection is made possible thanks to one or more Intermediate Heat Exchangers (IHXs), which, together with isolating valves, constitute a safety barrier between the nuclear island and the outer units. The correct functioning of these IHXs is therefore a high-level prerogative for the whole NPP safety approach.

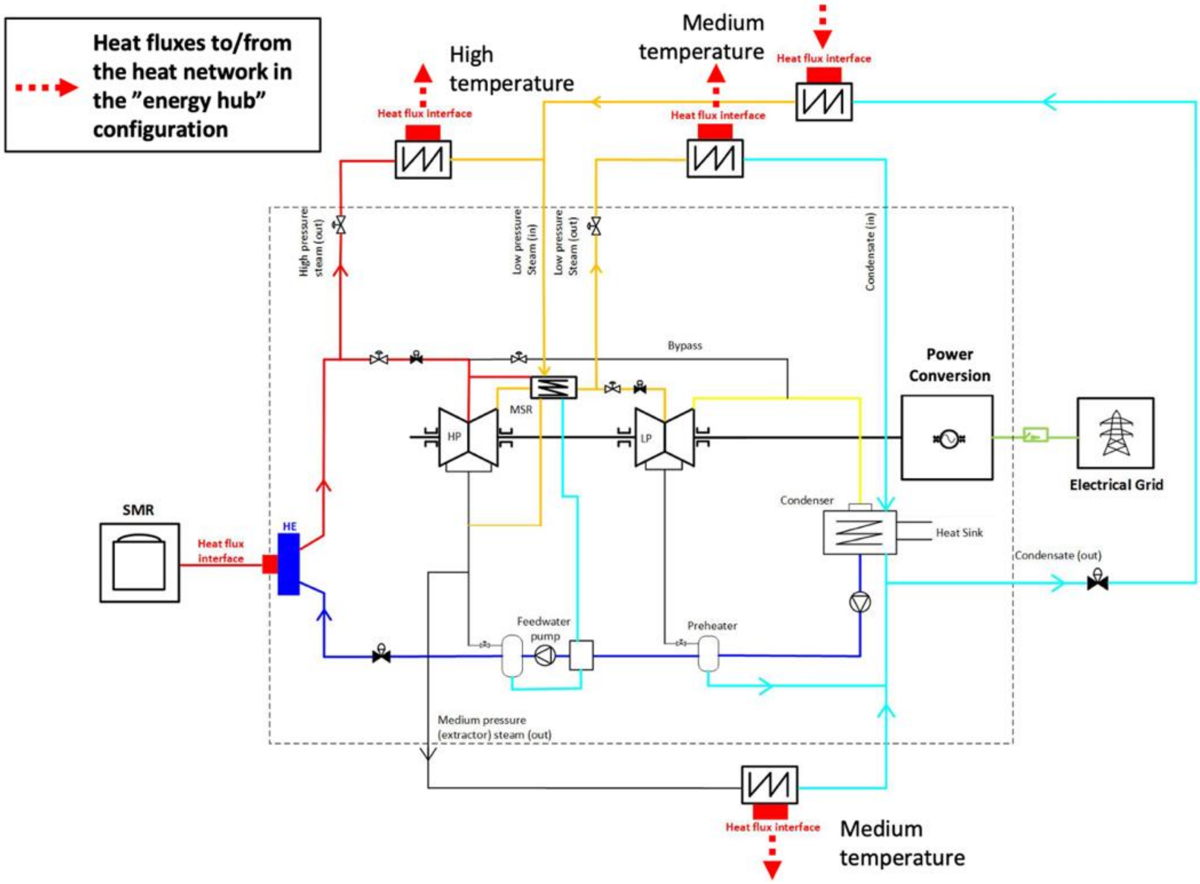

Figure 3-11 Model flow diagram (created by TRACTEBEL in the scope of the TANDEM project)

In general, although every external system has its own components, each end-user is perceived by the BoP systems as nothing but an output or input of power. The opening or closing of a heat supply line is considered as an increase or reduction in the load for the BoP systems. These variations can be part of load-following operations which may also involve adjustments in the reactor power.

As outlined in section 3.2.2, load-following power changes must respect specific rates depending on the extent of the variation, to ensure operation within the nuclear fuel and clad integrity parameters. Moreover, deviations from the nominal power condition influence the balance of plant that in turn is designed to reactively adapt its operations without affecting its functioning. These quite standard control mechanisms generally require the conversion system, which is the only live steam user (for traditional NPPs), to follow and adapt itself to the reactor power behavior.

Instead, NPPs integrated into HES introduce a number of different thermal consumers that inevitably complicate the plant control. These systems are designed to complement intermittent power sources and to follow grid demands by operating the reactor at full power. This can be done by giving to the BoP the role of actively assigning different destinations to the produced steam. Therefore, during the HES lifetime, the balance of plant will be subjected to continuous configuration changes, each one characterized by different connections between the users and the plant. As already defined in D2.2, depending on which line is opened or closed and on the amount of energy that is supplying or receiving, the HES can be interpreted in each moment as assuming a distinct operating regime.

With respect to the above-mentioned conventional NPPs, HESs BoPs must be able to manage scheduled load variations imposed not only by the reactor but also by potential end-users located in the HES. In fact, each regime is characterized by a different subdivision of the energy shares among grid and users and load-following operations usually require more or less frequent transitions from one regime to another. In HESs, when the power variation request comes from the nuclear reactor, procedures similar to those adopted in traditional nuclear plants should be followed, as the external technologies must not interfere with their application. Instead, additional requirements to the BoP systems might be introduced due to power modifications induced by the outer units. For instance, design provisions should be envisaged to handle planned transitions among different operating regimes in a “controlled and safe” way; however, load following implications will be discussed in section 3.2.

Situations still to be investigated are related to load variations imposed by the end-users that are not scheduled and might happen without proper anticipation or scheduling. The entity and frequency of these occurrences are unknown and the capability of the BoP to act as a thermal buffer between the nuclear island and the external units should be validated. In some cases, it could be impossible for the BoP to stabilize the plant and bring it back to a normal operating regime in a controlled way. These situations will be therefore regarded as PIE due to a steam unbalance, triggered by one or more failures of external users.

These considerations have the preliminary objective to define what are the thresholds above which, the BoP system is no more able to withstand and properly adsorb sudden load variations without impacting the nuclear island. The BoP is expected to act as a buffer between the outer units and the reactor, by damping potential harmful effects that could instead require an "islanding" procedure or a reactor shutdown. This possibility depends on the capability of its components to react more or less rapidly to sudden changes in the steam balance.

3.3.2 Unplanned conditions and initiating events

During normal operation, the connection or disconnection of a module from the HES is a delicate procedure that involves unit conditioning, start-up or shut-down of components, etc... These processes generally require specific timing in order not to cause rapid thermal or mechanical gradients, which may result detrimental to the involved pieces of equipment. By exploiting control valves, recirculation lines, robust sizing and similar provisions, scheduled transients can be handled in a controlled way such that all the BoP systems are retained within safe operating ranges, without any impact on the NPP.

However, there might be situations in which malfunctions of external users' components can lead to unplanned conditions. Sudden deviations from the nominal operating conditions might be problematic and difficult to tackle, in particular if the nuclear reactor is impacted. In order to figure out whether the HES is capable of handling these situations, it is necessary to determine the operating limits of each one of the involved components.

This analysis does not aim at defining a comprehensive list of all the PIEs and DBAs that might be caused by external modules. It is rather focused on those unplanned situations that, depending on components and systems capabilities, might be returned to normal operations or, in the worst case, become precursors of accidental conditions. Since the objective is to understand the possibility to keep the plant operating without the need to isolate the nuclear reactor from the HES, at this stage, only unplanned events that might cause "indirect effects" on the systems are considered. These effects include physical or thermal variations in the power conversion system working fluid, such as temperature and pressure changes or sudden increases/decreases of mass flow rate.

In fact, hybrid units might involve different components that could fail in several ways and have distinct influence on the HES, depending, for example, on their physical distance from the nuclear power plant. However, general malfunctions that can be of interest for almost all the applications involve:

- Pumps failure;

- Valves failure;
- Pipelines rupture;
- Heat exchanger leakage or rupture.

Since the listed components are quite standard, well-established design features and provisions including the application of redundancy and diversity principles, must be considered. A redundant system could be designed such that neither a single failure of an active component nor a single failure of a passive one should prevent the system from operating. Redundant components, in fact, may increase the overall reliability of the system if properly designed. However, given the larger number of couplings of the HES with respect to conventional NPPs, including users for which augmented quality standards may not apply, there might be a higher frequency of situations in which two or more components happen to fail within a short period of time from a common cause.

In situations where redundancy and diversification provisions do not offer a significant benefit, the failure of the above-mentioned components would lead to an almost immediate exclusion of the affected module. As already discussed, the opening or closing of a line is perceived by the plant as a load imbalance. More or less rapid thermal load variations are common phenomena that are usually taken into consideration for any reactor design. To monitor and handle these transients, several procedures are developed and adopted in almost all NPPs. What might differ in a HES configuration is the frequency and the scale of these events.

3.3.3 Component limits

As anticipated, unplanned load variations mainly impact the BoP system and therefore the operating limits of its components should be identified a priori. As the TANDEM project does not consider a specific BoP design, the considerations outlined in this report should be intended as suggesting useful boundaries, from a conservative point of view, to start delving into safety issues.

When a variation of steam flow is experienced, the mainly concerned components are the turbines. According to the BoP design reported in D2.1, extraction points have been foreseen before and after the high-pressure turbine. In this regard, two distinct scenarios can be considered: in the first one, the turbo-generator system is operating at partial or full load and is connected to the grid; in the other, all the produced heat is employed by external users and the turbine system is kept non-operating and disconnected from the grid. Although sometimes there could be interest in totally directing the steam to the heat network, it is unlikely that the turbine group could be completely shut off. In fact, part of the electricity generated is supplied to the different internal uses of the nuclear plant itself. Moreover, the turbine start-up would be

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affected by thermal inertia which would make the overall system less flexible than a stand-by hot condition. Therefore, this scenario has not been further discussed.

The events that can lead to an unpredicted reduction in the amount of steam entering the turbine are multiple and various, depending on the HES configuration. If this happens during operating regimes in which the turbine is still working at small flow rates, it would be important to know if the machine has a minimum load below which it cannot be safely operated. This lower limit is usually determined by the capability of the last-stage blades to withstand the ventilation effects. The ventilation phenomenon occurs when the flow of steam through the blades is so small that it starts causing recirculating flows and eddies (**Figure 3-12**) [MAMBRO]. Because of the low flow rate, the pressure rises over the rotor from which the fluid absorb energy increasing the blades temperatures [SIGG]. These effects mainly affect the last turbine stages and cause structure vibrations and overheating that can be partially counterbalanced by the action of bearings and spray hood at turbine exhaust (used to cool down the blades). For a general-purpose turbine, the ventilation limit can be conservatively set to 20% of the nominal load. However, these phenomena can be detrimental for the turbine system if these operating regimes last too long. Therefore, these operations should be limited to a strict minimum time, never stretching the turbine below the motoring load. The latter, that is generally no more than 5% of the nominal load, sets the limit below which the generator begins to work as a synchronous motor requiring energy instead of producing it, in order to keep the turbine blades in rotation.

This should not be a concern in the event of an unplanned load increase. Turbines are designed to be operated with 100% of the nominal load. However, both increases and reductions in load might be dangerous whenever they are associated with time constants that can produce thermal or mechanical shocks to the component structures. For this reason, from a regime of constant load, turbines should not be exposed to variations that exceed rates of 10% nominal load per minute [CANTEACH 1994]. Moreover, if the steam temperature is subject to changes, it is crucial, for the turbine integrity, to keep them to a rate of no more than 3 °C per minute.

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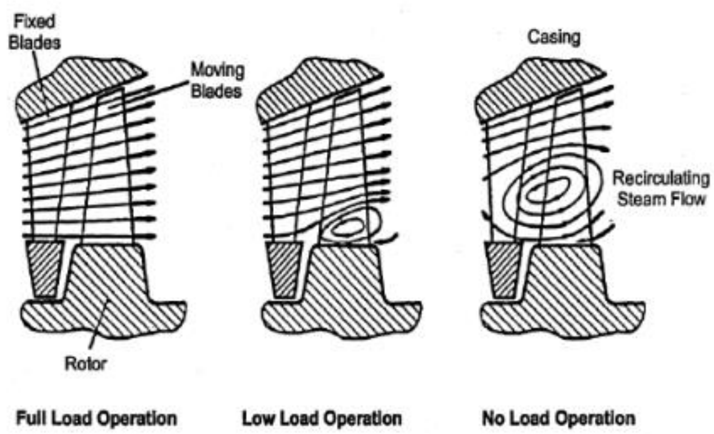

Figure 3-12 Schematic representation of partial load phenomena effects

In particular, to handle unplanned load increases with no impact on the turbine, bypass lines can play a key role. These can be used to discharge the excess heat directly to the condenser, not only in the case of a turbine trip but also whenever an external user happens to fail. By properly regulating the steam flow, sudden variations can be slowly managed by means of valves that can bring the plant back to a stable condition without overcoming any component working limit. In principle, similar procedures could be somehow applied also by relying on external units (if in operation or with no need for start-up), as the thermal energy storage. After the failure of a module, the extra steam can be immediately directed to other end-users and then progressively routed to the turbine. However, external users, whenever available (for instance, the thermal energy storage might be fully loaded and thus not ready to receive further heat input), could be used, although not credited for safety analyses in the licensing space.

As regards to other components, the occurrence of excessive thermal gradients is the major obstacle to rapid load changes within the whole BoP circuit. This is why the Moisture Reheater Separator (MRS) and the various heat exchangers should also be kept safe from temperature variations that go beyond 2-3 °C per minute. In conclusion, all these limits, summarized in *Table 1*, are very approximative and more precise estimations strongly depend on the specific design of each unit.

Turbine		
	Ventilation limit	20 % nominal load
	Anti-motoring limit	5 % nominal load
	Power variation rate	10% nominal load / min
	Temperature variation rate	3 °C / min
MSR, HXs, Condensers		
	Temperature variation rate	2-3 °C / min

Table 1 Operating component limit.

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4 Specific analyses required for the integration of an SMR in HES

4.1 Identification of normal operation (DBC-1) and abnormal operation (DBC-2 or AOO, DBC-3 & DBC-4)

The various operating states of the SMR are presented in Chapter 2 of Deliverable 2.2 "Identification of the modelling/simulation strategy", using the IAEA classification into four categories:

I. Normal operation.

II. Anticipated Operational Occurrences (AOO or DBC-2), which are expected to occur over the operating lifetime of the plant.

III. Design Basis Accidents (DBC-3&4).

IV. Design extension conditions (DEC).

Each operating category covers several situations where the reactor is either at full power, at reduced power, or in a protective situation, with a complete core divergence shutdown due to the fall of the absorber rod groups. We can separate the previous classifications in two main situations for the reactor plant, with the same objective to reach and to maintain a safe state: normal and abnormal operation". The second category covers unplanned incidents and accidents during the life of the reactor and are managed by automatic and operator actions, with the aim of achieving a controlled state of the reactor unit, then a stable state in cold shutdown.

4.1.1 Normal operation in Level 1 (DBC-1)

Definition of Level 1, 2 and 3 corresponds to the escalation from normal operation to accident, in the Defense in Depth Approach. Such definitions are detailed in the WENRA document, specific to the safety of New NPPs [WENRA 2013]. Taking as an example the normal operating situations described in the PCSR for the UK EPR [UK-EPR], and which can naturally be applied to an SMR case, this category includes the following situations:

A. Power operation and normal scheduled operating transients such as increases in load, reductions in load, plant shutdown or start-up.
B. Specific operations due to unplanned events, such as house load operations or loss of power sources.

The first category concerns the different power operations that are presented in chapter 4, both for process heat purposes and variation in (electricity) power demand. There are no specific safety issues associated with this category of operation, as long as reactor conditions are within the expected operating range as specified in the technical specifications of the unit. For example, the preparation of steam generators conditioning during power-up of the reactor unit, and before connection to the turbine on the one hand and to the heat process on the other hand, must comply with the limits set by the safety studies associated with incidental or accidental transients likely to occur during this intermediate phase of operation (typically a steam line break event, DBC-4). In addition, the specific nature of hybrid reactor in normal operation, according to the different operating regimes described above, must not lead to an increase in the frequency of unplanned initiating events classified as abnormal situations.

The second category concerns typical unplanned events (or planned exercises) that can occur in the life of the reactor, corresponding to typical loss of power sources. The islanding procedure, after a reactor power reduction, enables the reactor unit to remain in normal operation, benefiting from the electrical power generated by the turbo generator set to supply all the components required for its operation, albeit uncoupled from the electrical grid. In such a case, typical sequences of operation are (automatically) activated to achieve the islanding situation.

The loss of the external electrical network instantly results in a loss of resistive torque on the alternator, causing the turbine to speed up. The steam flow control system at the turbine inlet adapts instantly, progressively shutting off the inlet steam flow with a specific throttle valve design system. For the first few seconds, the steam generators accumulate the steam produced at full power, before the main control system initiates the islanding procedure by: reducing core power to around 30% full power, opening the MSIV bypass (SDIV) to send the surplus steam produced directly to the condenser. The turbine retains only around 5% of the steam produced for electrical operation in islanding mode. **Figure 4-1** below shows the steam routing from the CSGs, with the main MSIV kept open to supply the 5% of steam to the turbine, and the MSIV bypass further opened by the SDIV valve to supply the SDC with the remaining 25%. After passing through the turbine and condenser, the liquid water returns via the MFWS circuit, with the MFWIV isolation valve remaining open. In this way, a distinction is made between "normal" unit operation, with the MSIV steam isolation valve and the MFWIV feedwater isolation valve kept open. In "abnormal" or incidental operation, these two valves are closed to allow the unit to fell back to classified systems and circuits. The water and steam circuits shown here are simplified, and not all isolating devices or other routing or evacuation functions to ancillary systems are shown. MSSV system is specific to the nuclear island, as a safety device dedicated to overpressure protection.

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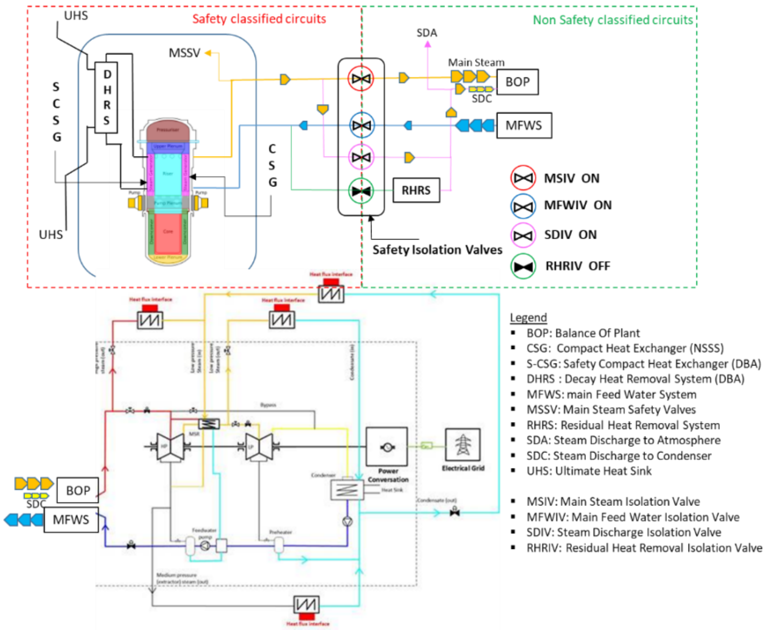

Figure 4-1 Simplified safety classification systems between a SMR nuclear island and BoP House load operation

Normal plant operation will give priority to the electrical islanding function provided by the turbine-generator set, in order to maintain the nuclear unit in normal operating conditions, with priority given first and foremost to the satisfactory supply of steam to the turbine-generator set. The ancillary supply of thermal heat is likely to be interrupted, particularly in the event of loss of the external power supply, when the process requiring thermal heat is likely to be subject to the same problem of loss of power supply. To maintain the supply of thermal heat during an islanding sequence, the nuclear unit operator must solve the following two problems:

- Ensure a satisfactory steam supply regime for the BoP, enabling it to continue to supply both the turbine in islanding mode and the HES process (5% + x %), and to send excess steam to the condenser.
- Manage the additional aggravating transient during the islanding phase, when the supply of thermal heat would be interrupted due to an event specific to the thermal heat process and, in particular, a loss of the external power grid. This would require a rapid return to the previous situation of 25% sent to the condenser, avoiding an overspeed regime for the turbine leading to islanding failure.

During reactor shutdowns in normal operation, a specific cooling system is activated to remove residual power from the core. This system is shown in **Figure 4-2** below, with the steam flow from the normal steam generators CSGs, and the return of condensate in a closed circuit. In this configuration, the SDIV isolation valve is open, as is the specific feed return valve of the RHR system.

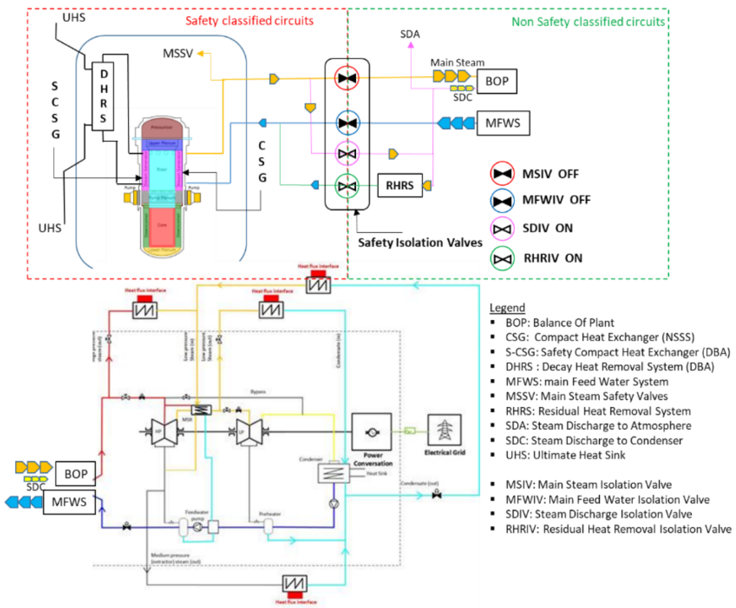

Figure 4-2 Simplified safety classification systems between a SMR nuclear island and BoP Shut down operation

4.1.2 Abnormal operation in Level 2 (DBC-2 or AOO)

There is no specificity of SMRs with respect to large nuclear power plants in terms of safety definition and classification. If we consider the latest WENRA RHWG document [WENRA 2021] dealing with the specificity of SMRs, there are still the three main categories, namely normal operation (DBC-1), control of abnormal operation and failure (DBC-2), control of accident and prevention of escalation to severe accidents (DBC 3/4, DEC). The last category is not considered in this chapter (Level 3-b and Level 4), considering that specific study are already done for the nuclear plant itself, and should be sufficient for the hybrid configuration. Such hypothesis can be done if the three following considerations can be assessed:

- No specific extreme external event has to be initiated from the HES process, with heavy consequences on the respect of the three main safety functions [WENRA 2013] leading to a Level 3-b transient event (DEC-A):
  - Control of reactivity;
  - Removal of heat from the reactor and from the fuel store;
  - Confinement of radioactive material, shielding against radiation, as well as limitation of accidental radioactive releases;
- No specific combination of frequent category 2 and possibly category 3 incident initiators, associated with aggravating situations or multiple incidents, with origins from the HES process, likely to lead to an accidental event falling under level 3b or category DEC-A;

- No specific sequence of initiating event and combination of aggravating situations, with origins from the HES process, likely to lead to a severe accident entering phase (Level 4).

Considering the deterministic approach from the IAEA workgroups [IAEA 2013], AOOs events are expected to occur at least once during the lifetime of the plant. So, frequency of occurrence is higher than 10^{-2} per reactor-year. There are seven families of AOO incidents:

A. Increase in reactor heat removal
B. Decrease in reactor heat removal
C. Decrease in reactor coolant system flow rate
D. Reactivity and power distribution anomalies
E. Increase in reactor coolant inventory
F. Decrease in reactor coolant inventory
G. Release of radioactive material from a subsystem or component

Based on the above 7 incident families, the deterministic safety analysis will therefore consist in studying the incident initiators specific to the hybridization combining the SMR reactor and its BoP, with the additional HES process.

A. Increase in reactor heat removal

This first family consists of an increase in thermal power extraction from the core, either directly from the primary circuit (pressurizer pressure control malfunction, for example), or as a result of a large, unplanned demand for thermal power from the BoP or HES customer via the secondary circuit. This example is typically the safety case to study, depending on the characteristics of both the BoP and the HES specificities. A malfunction in the turbine steam supply control system, on the one hand, or in the thermal supply process, on the other, and corresponding to an event in the category likely to occur once in the life of the reactor or hybrid plant, can then be dealt with in this context. Other incidental initiators may also be considered by the BoP and HES plant designer. Variations in the normal feedwater flow parameters, in term of temperature or mass flow, can induce such an initiating event, if minimum or maximum values are exceeded and reach a threshold to activate automatic reactor shutdown (same consideration for category B).

If we refer to the UK EPR safety study [UK EPR], spurious opening of valves (primary or secondary circuit), or of safety relief or bleed valves, are classified as category 3 events, said to be improbable and therefore outside the scope of the AOO. Similarly, piping ruptures, particularly of the steam type, which could result in a demand for thermal power from the core, are also classified as at least category 3.

As a general rule, therefore, the quality of the components and associated circuitry, particularly for the hybridization and HES process, must be sufficient to avoid including events that would normally be classified as category 3, but whose insufficient reliability would lead to them being classified as category 2.

B. Decrease in reactor heat removal

This second family corresponds to the reverse transient described above. In the event of a sudden drop in the thermal power demand of the BoP or HES customer, the secondary circuit absorbs the difference between the energy evacuated and that produced by the primary circuit, rapidly leading to an increase in average primary enthalpy. The average core temperature rises, leading to a drop in reactivity, but is also accompanied by an increase in primary pressure, which the pressurizer is unable to compensate for. Automatic power regulation begins to lower the control rods, but high primary pressure thresholds can then be activated, even reaching the opening limit of the pressurizer safety valves. This transient therefore corresponds to a thermal power variation that exceeds the load-following limits for both the turbine-generator set and the HES process. Disconnection of the turbine from the grid, or turbine trip, corresponds exactly to this incidental transient, either following electrical disconnection of the alternator and failure of islanding (for example, following a condenser vacuum fault), or for any other incident involving the turbine and leading to rapid closure of the steam inlet valves. A similar event can also be envisaged on the thermal power plant side, when an accidental initiating event leads to a sharp reduction in the thermal heat input.

C. Decrease in reactor coolant system flow rate

This event concerns only initiating events due to the circulation of the primary fluid. It is not related to the hybridization of a BoP and a HES process. However, we can note that the loss of external power supply (LOOP) can also fall into this AOO family, with a rapid shutdown of the primary pumps. For electric power interruption of less than 2 hours, this event is classified as a DBC-2 event (Note that such categorization is not depending on the plant itself, but on the grid reliability). Although the initiating event is uncorrelated with the hybridization function, it will be necessary to ensure that the particular location of a nuclear unit within an industrial area does not significantly increase the risk or probability of losing external power supplies, even though the islanding function in normal operation is supposed to alleviate this problem, albeit with rather reduced reliability.

D. Reactivity and power distribution anomalies

This category of transient event is correlated to the control of reactivity of the core, without any correlation to the hybridization.

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E. Increase in reactor coolant inventory

This category of transient event is correlated to the control of the volume of the primary water inventory, mainly due to a spurious activation of safety injection system, or a CVCS (chemical and volume control system) malfunction. There is no link with the hybridization.

F. Decrease in reactor coolant inventory

This category of transient event is correlated to a minor reactor coolant system leak or a very small break in the primary circuit, as a very small LOCA (loss of a cooling accident). Malfunction of CVCS can be identified too as an initiator. There is no link with the hybridization.

G. Release of radioactive material from a subsystem or component

This category is very specific to the problematic of the main safety function concerning the confinement of the radioactive material and is not correlated to the hybridization.

In addition to the previous family cases, we can add a spurious activation of the reactor shutdown operation (SCRAM mode or reactor trip). Origin of such event is typically a malfunction of I&C, due to combination of bad detection or defaults in instrumentation. We can include this transient event in the general procedure following detection of the various category 2 incidents described above, since the reactor trip action is always taken to stop the core power delivery.

The AOO procedure can be different from the classical safety approach dedicated to DBC3-4 transient event, and depends on the general safety architecture proposal for the DBC2, DBC3, DBC4 deterministic approach. Considering that passive safety systems are mostly dedicated to unplanned events, and to avoid using them both in Level 2 and Level 3 of Defense in Depth, decay heat removal systems that are not classified, or at least not in a class 1 safety rank, can be called upon to manage this type of incidental transient in the first instance. In the event of failure or unavailability, these systems are then covered by the class 1 specific systems, corresponding in particular to the passive DHRS system shown in **Figure 4-3**. Safety strategy of SMR Nuward technology is presented in reference below [ICAAP 2023]. For example, we can notice three different systems and routing circuits associated to the decay heat removal strategy.

1. Auxiliary feed water system and steam discharge to atmosphere

In that case, the auxiliary system from normal feed water system is combined with the normal feed water system line, to ensure the adequate water flowrate feeding in a shutdown situation. In normal operation, such system should be present too to ensure the normal shutdown operation and cooling down from state A (full power operation) to state B (connection to the RHRS system).

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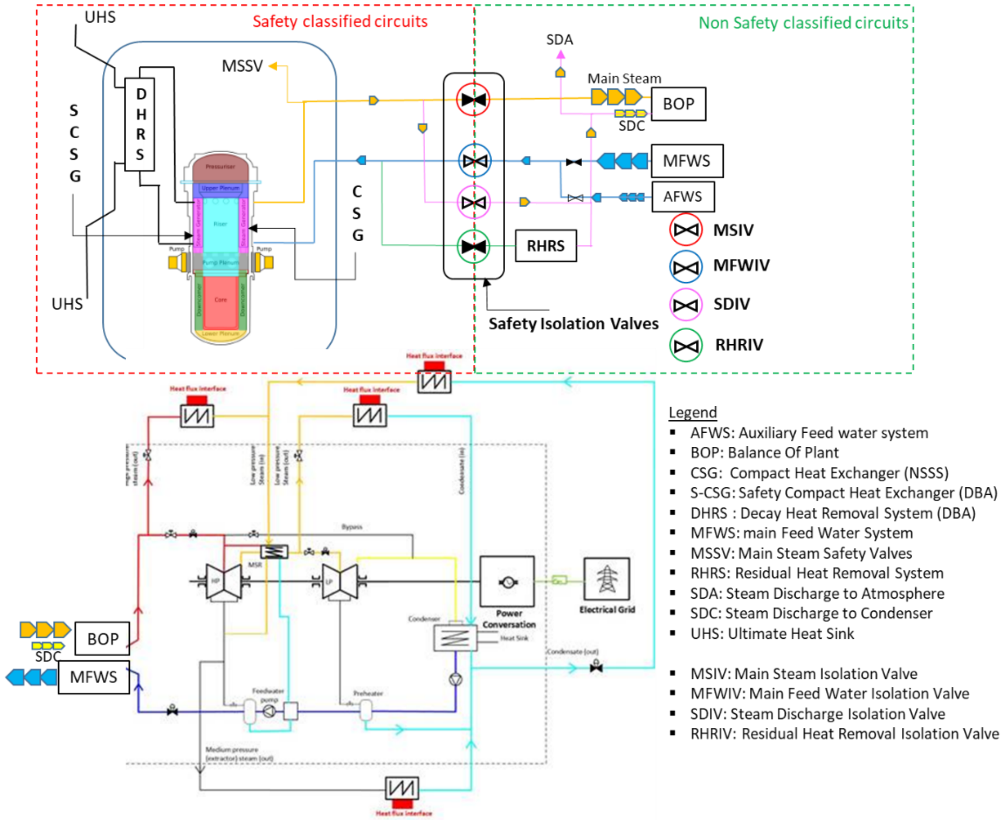

Figure 4-3 Auxiliary feed water system and steam discharge to atmosphere operation mode in a AOO or DBC-2 event

2. RHRS operation mode

This system is normally dedicated to ensure the reactor cooling, in normal operation, from state B of connection to this system, to a state allowing the complete fuel handling and removing from the reactor vessel. Depending on the system connection and activation options, and in particular its maximum temperature operating conditions, this residual heat power removal system can also be used as a first line of defense to manage this type of incidental transient. The routing of such operation is described in the **Figure 4-4** below.

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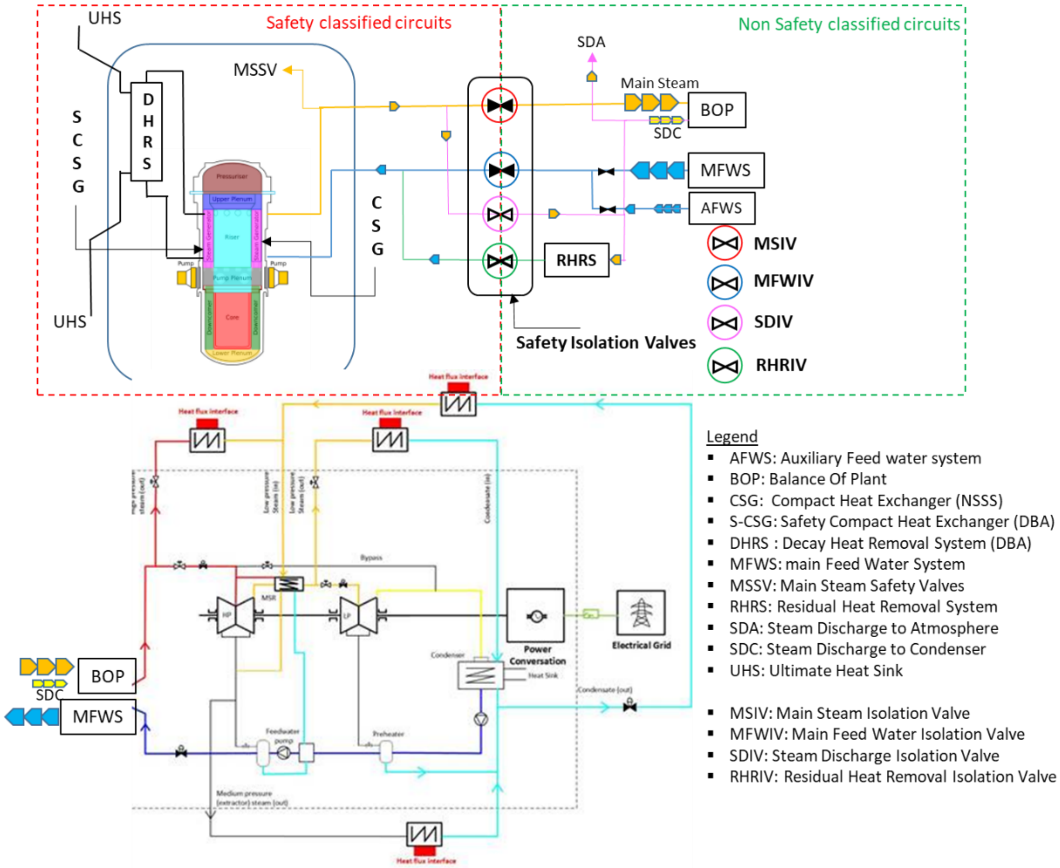

Figure 4-4 RHRs operation mode in AOO or DBC-2 transient event

3. DHRS operation mode (safety class component)

This system is valued as a safety system or important equipment protection system, and is present in the safety demonstration as an essential component of the class 1 safety demonstration. In particular, this system will be used in category 3 and 4 accident situations, with an emergency and reactor protection procedure, including a nuclear unit retreat within its safety-classified architecture (see red box of **Figure 4-1**). Necessarily, all the safety isolation valves must be closed in order to adequately isolate the classified section. **Figure 4-5** shows the following diagram for this safety operating mode, in the same way as for category 3 and 4 accidents.

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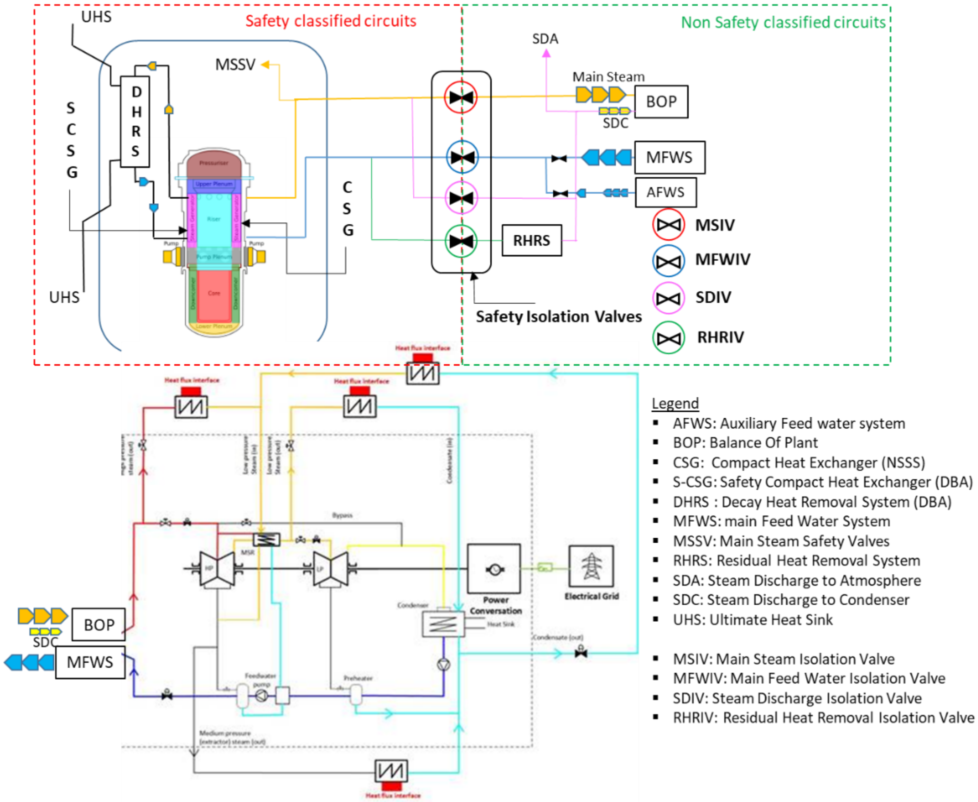

Figure 4-5 DHRS operation mode in AOO or DBC-2 transient event

This document does not deal with all events relating to the safety of fuel assembly storage in a spent fuel pool, which is outside the scope of the issue of hybridization of an SMR reactor and a process combining electricity production and thermal heat supply.

4.1.3 Abnormal operation in Level 3a (DBC3-4)

DBC3 and DBC4 design basis accidents have a lower frequency of occurrence than the AOO and DBC2 incidents described above. These are not expected to occur during the lifetime of the reactor, in accordance with the principles of DiD. The deterministic study consists of examining these accidents, which are likely to occur despite everything, with realistic assumptions about penalties and aggravating situations. The frequency of occurrence of such an event is between 10^{-2} and 10^{-5} per reactor year. The safety demonstration associated to such category of accident is assessed for the reactor plant, in association with a classical BoP.

The safety management of design basis accidents is carried out within the exclusive perimeter of the classified safety systems (perimeter in red box in previous figures). As mentioned above, all the isolation valves on the secondary lines determine the boundary between the classified and unclassified parts of the plant. Hybridization concerns only the downstream part of the isolation valves, and any initiating event likely to give rise to a design basis accident, induced by the specific

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architecture of this hybridization, will then be decoupled from emergency accident management, as soon as all the isolation valves are activated to close.

It is important to note that in a classical deterministic safety study, the conservative design basis accident (with realistic physical penalty parameters) as initiating event is generally accompanied by an aggravating situation, corresponding for example to a single failure condition of a component important for safety (notion of single failure criterion of a family of components important for safety, 10 CFR Appendix A of US NRC code, or Guide 22 IRSN France). Maintaining strict isolation of classified secondary circuits, so as to eliminate the consequences of an initiating event beyond the isolation valves defined above, means ensuring that there are at least two diversified means of ensuring this isolation function, both at the level of the active component itself, and at the level of the I&C system and its power supply. With such consideration, we can conclude that there is no specific additional safety studies to assess, with hybridization of a classical SMR plant and BoP.

4.1.4 Specificity of a thermal storage coupling

The implementation of a thermal storage unit, coupled with a BoP, is summarized in Deliverable D1.4, **Figure 4-6**, with a conceptual illustration of heat energy network distributing several thermal consumer units, including a storage energy system plant. This figure is reproduced below, including the notion of reversibility of thermal power supply between the storage unit and the turbine.

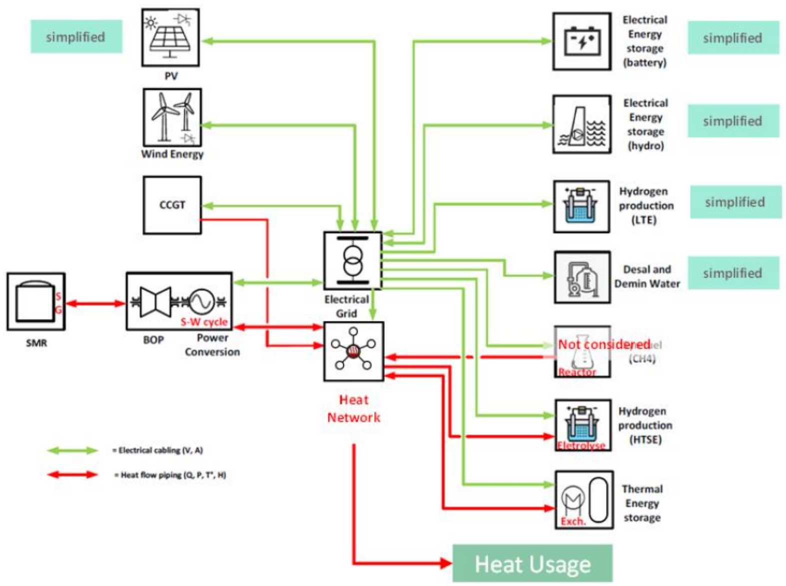

Figure 4-6 Conceptual layout illustration of the hybrid energy system for the energy hub regarding the high SMR deployment scenario in 2035 (D1.4 report Figure 7)

However, a more detailed design is required to understand how these two configurations of thermal energy consumer or producer, and in particular the positioning of the hot and cold legs, as well as the pumping unit ensuring fluid circulation. It is important to note that the secondary circuit of a nuclear facility, SMR type or other, has a limited extension with a volume of water whose chemical quality is controlled and monitored. This is why it is likely that the notion of connecting to an energy network to supply a thermal storage unit is carried out in the form of an intermediate circuit and a heat exchanger present in the BoP, in order to avoid an extension of the conventional secondary circuit piping. If we look in more detail at the direct connection between the thermal storage unit on the one hand and the secondary circuit of the nuclear island and the turbo generator set on the other hand, deliverable D2.1 dealing with the modeling of this thermal storage unit shows an example of a simplified layout in **Figure 4-7**.

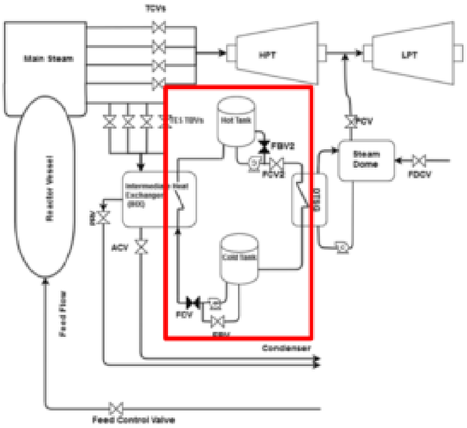

Figure 4-7 Schematic of a two-tank thermal energy storage system (Frick et al.,2019) (D2.1 report Figure 17)

We will therefore distinguish between the two main operating modes of the nuclear plant presented above, i.e. normal operation or DBC1, and abnormal operation including AOO and DBC2 conditions. We do not deal with category 3 or 4 DBA, initiated by the energy storage system, for the same reasons of a complete independent isolation of safety secondary circuit in front of the initiating event. Normal operation mode corresponds to start-up procedures, ramping up to nominal speed, and variations in thermal power supplied to the turbo generator set, as well as to the various thermal supply processes, including thermal storage. Likewise, the return of part of the stored energy to the turbo alternator is also part of this normal operation, within the operating range specified by the manufacturer. The reactor normal progressive power shutdown procedure, decoupling the various customer units (thermal and generator), is also part of the normal operating procedure. Schematically, the layout of the thermal storage unit is part of the classic configuration separating the classified secondary circuit from the extension to a BoP and HES processes, as shown below **Figure 4-8**.

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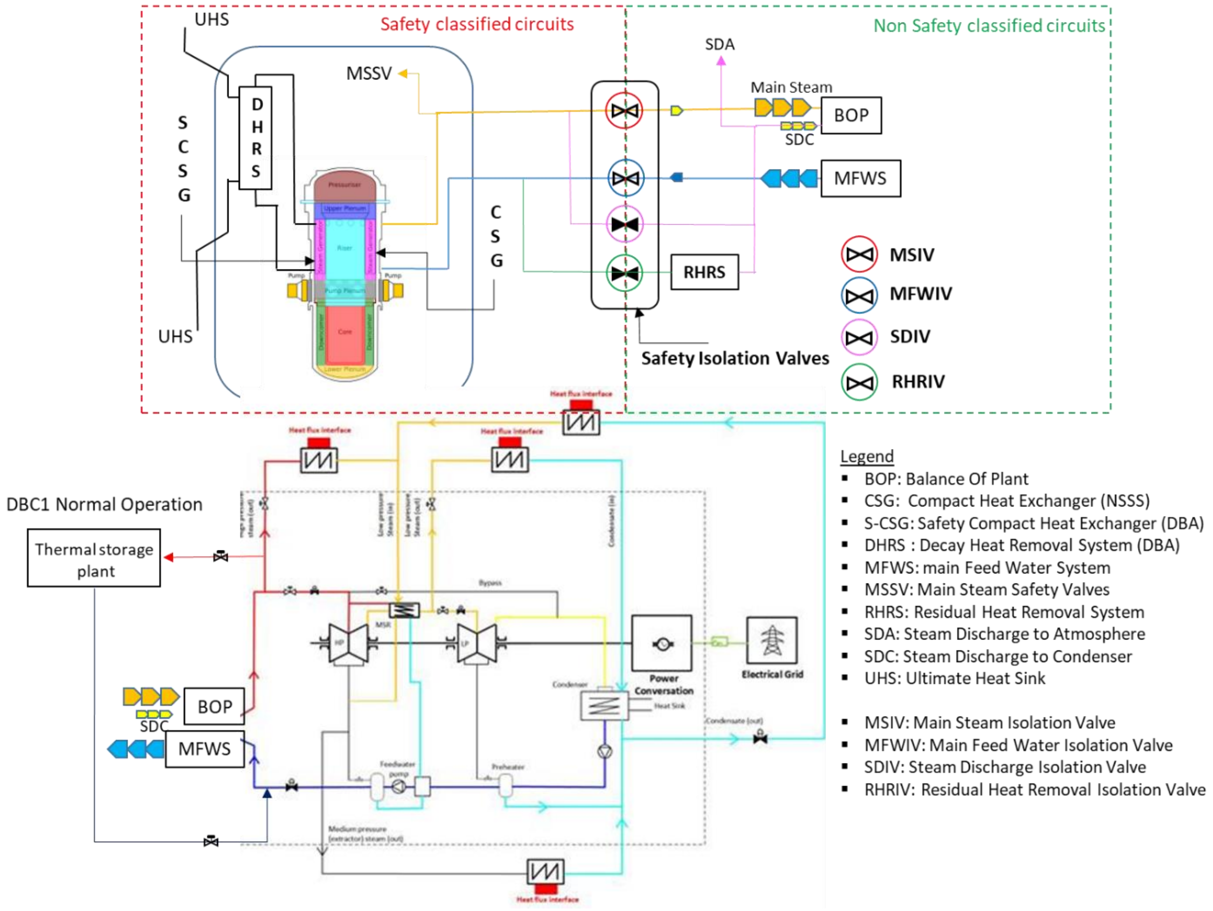

Figure 4-8 Simplified safety classification systems between a SMR nuclear island and BoP with a thermal storage unit in normal operation

A complementary use for the thermal storage unit may be considered, depending on the degree of reliability of the components and piping architecture connecting the secondary frontier (isolating valves group). In parallel with the examples of using unclassified or low-rank classified systems to manage AOO-type events, the thermal storage unit, if available, can also be used as equipment for protecting and managing the incidental transient, in the first instance. Illustration of such AOO management by a thermal storage unit is presented below. Of course, in case of unavailability or failure start for such operation, the safety classified system described in **Figure 4-8** will take over the incidental event management.

4.2 Initial conditions

Though it is presently yet too early to propose the set of meaningful scenarios necessary to evaluate the potentially impacted safety margins of the E-SMR NPP, the following categories of operating conditions can be already envisaged, on the basis of the specificities that characterize a cogenerating NPP integrated in a HES:

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- Operating regimes at prescribed reactor full or part-load power. In these regimes of normal operation, it can be assumed that a share of the energy produced by the reactor is devoted to electricity generation and the remaining part to the applications needing heat. The definition of these regimes in view of answering the users' needs will establish the steady-state conditions for the primary and the secondary side of the reactor during continuous operation.
- Transitions between two different operating regimes. Depending on the number and the characteristics of the considered steady-state regimes, assumptions will be necessarily made about the transitions that will be needed to be simulated between two regimes. These transitions can be the result of a pre-defined power control program (in which case they are classified as normal operation) or can be the result of a perturbing external condition (in which case they are classified as AOOs). Such events may simply imply a transition to a different splitting of the produced power or also a change in nuclear power production.
- DBAs affected by the HES and/or by the cogenerating setup. As said earlier, it must be investigated whether the nuclear island may be affected by the BoP and the HES at least during the early phases of an accident. It must be investigated also whether the cogenerating setup of the power plant may induce some significant variations to some classes of accidents. As an example, a steam line break may take place in a section of the secondary circuit dedicated to heat production. Depending on the configuration of the E-SMR secondary circuit, this could potentially lead to a different scenario than a steam line break happening in a conventional secondary circuit.

These considerations, though still generic, provide an overall sketchy perspective of the possible scenarios that need to be simulated in the safety assessment of the E-SMR, considering its inclusion into a HES.

5 Proposed methodology for transient analysis

As a preliminary step, the recommendations and methodologies documented in the deliverables of the ELSMOR project for the safety assessment of SMRs without considering coupling with a cogeneration process are considered in the following as a basis for the demonstration. This will ensure the compliance with relevant nuclear safety regulations and standards (IAEA, WENRA, national regulatory bodies...).

In addition, the safety of SMRs should be improved by reducing the factors that could cause these disturbances in the NPP. Such an objective should be considered in the new operational management of the integration of SMRs in HES. For this reason, DEC should be avoided. Because the identification of possible remaining DEC situations is not possible at this stage, the approach related to DEC is not covered in this document. The discussions are therefore limited to the identification of constraints that can affect the manoeuvrability of the SMR.

5.1 Description of the different phases

Assessing the safety of a nuclear Small Modular Reactor (SMR) integrated in a HES requires a comprehensive methodology that considers various aspects of nuclear safety. The proposed methodology to assess the safety of such a system is based on the IAEA guide SSG2- rev 1 and can be split in several phases:

5.1.1 Phase 1: Firstly, perform two separate safety analyses

One for the NPP and one for the industrial plant or the electrical grid by considering each installation as potential source of external hazards on the other.

And then for the entire HES in interaction with the nuclear island.

5.1.2 Phase 2: Detail precisely the design and operational characteristics of the SMR

This includes understanding the reactor type, fuel type, cooling system, control mechanisms, and any safety features or redundancies built into the design.

5.1.3 Phase 3: Identify sources of risk and sensitive system components

For the risk of disruption to electrical power, and for the risk of disruption to heat flow, begin by identifying all the components of the hybrid energy system, including the SMR, the energy conversion and storage systems, and any other relevant components and systems.

5.1.4 Phase 4: Define a DBC list

This phase should be guided by the list of the following general guidelines:

- Defining appropriate separation distances between the reactor building and the industrial plant is in favour of minimizing mutual risks.
- There are two possible configurations for coupling a SMR with an industrial process in a cogeneration context. Either the coupling can be characterised as weak if the interface is managed via the BoP, which provide a certain buffer, or the coupling is strong if it interacts directly with the nuclear island.
- An hazard does not necessarily lead to a failure and therefore there is the need to consider an initiating event.

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- All reactor states should be considered (power operation, shutdown, maintenance, etc.)
- In the same way, as design provisions have been put in place for the turboalternators (placed tangentially to the nuclear island), the same kind of consideration must be given to the industrial cogeneration process systems.
- Stopping potential projectiles without any special provisions other than the strength of the building materials.
- It should be noted that the modifications planned as part of the post-Fukushima ECS would make it possible to fulfil some of these safety functions - provided that these modifications withstand the thermal and overpressure effects of an explosion.
- Taking domino effects into account is necessary.
- Local storage (in a form to be defined) as a solution to avoid transients should be considered.

From those requirements, identify PIE associated with AOO & DBA through an analysis of operating experience data for similar configurations and compare with the list of PIEs developed for safety analyses of similar plants. Because at this stage, the evaluation of PIE frequencies is subject to large variations and the interactions with other industrial systems remains insufficiently known, it might be interesting to study certain scenarios in which an external stress event is combined with a DBC operating condition not induced by the stress event.

The DBC list will be subdivided into the four categories (DBC 1-4) based on the frequency of PIEs and on good practice from other projects or from standards. Acceptance criteria are defined for each category. The operator should first and foremost seek technical solutions at the design stage to prevent an external stress situation from leading to a DBC-type event.

5.1.5 Perform safety analysis

Conduct a safety analysis of the nuclear SMR and its integration within the hybrid energy system. This analysis should include deterministic and probabilistic approaches to assess the likelihood and consequences of accidents or incidents. Evaluate transient factors such as loss of coolant, variations in temperature, pressure, disturbances on electricity power, etc. Take a particular attention to human factors in terms of training, staffing levels, operator interfaces between teams of NPP and the industrial processing case of cogeneration interfaces.

5.2 Description of the tools and coupling

Numerical tools must be used to perform the safety studies of the identified reference transients. In general, several codes are used, in coupled or separate simulations, to model different phenomena:

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- Neutrons transport
- Thermal-hydraulics
- Thermo-mechanics

Several models are used for each domain depending on the transient and the applied methodologies. For example, 3D, 2D, 1D or 0D models, or even a combination of them, can be used for neutrons transport. For thermal-hydraulics models, *component* and *system* codes can be used.

In the context of the TANDEM project, one of the main objectives is to focus on the impact on safety of the hybridization of the SMR. In other words, the objective is to analyze how the connection of heat users to the BoP may induce specific effects on key safety parameters. It may be because of initiating events associated to the connection to heat users or by interaction between the BoP and the connection for *traditional* events.

In order to model the impact of the heat users (i.e. the dynamic interaction between connected systems), the whole HES have to be modeled (system modeling approach). The Modelica language have been chosen to build a *modular* simulator of the HES.

Moreover, safety validated codes, such as ATHLET and CATHARE, are used to focus on the primary system and to compute key safety parameters. Coupling between the safety codes, modeling the primary system, and the Modelica simulator, modeling everything else, have to be established.

The following paragraphs give some more details on Modelica, the safety codes and their couplings.

5.2.1 Modelica

5.2.1.1 Modelica Generalities

Modelica is a modelling language dedicated to dynamic systems:

It is Equational and Acausal. It means that the modeler has to define the equations governing the phenomena along with their initial/boundary conditions, without the need to implement the numerical methods for solving them. In other words, the modeler may focus more on the physical phenomena rather than the numerical aspects. Since one has to define the system of equations, one may also define easily the corresponding inverse problem

It is modular (object oriented). The community develops libraries (thermal-hydraulic, mechanical, electrical, I&C…) that can be easily reused. *Modules* are assembled to build more complex models, for example:

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- Pipes + Walls → Heat-Exchanger
- Pumps + Pipes + Heat-Exchangers + Turbines → Rankine cycle

It can deal with discontinuities/events.

The language is supported by some software, such as Dymola or OpenModelica, which allows the user to:

- Pre-process: define equations / assembles available modules
- Compile the test-case and run simulations
- Post-process the results

5.2.1.2 Modelica in TANDEM

The open source TANDEM library, to be developed in the context of this project, is based on several existing libraries. For what concerns the SMR plant modelling (primary and secondary loop), the thermal-hydraulic *ThermoPower* (Polimi) and *ThermoSysPro* (EDF) libraries will be used. They consist in a collection of 0D and 1D thermal-hydraulic elements. Several other libraries are used for specific components (e.g. electrical grid).

5.2.1.3 Modelica for Safety Studies

For what concern thermal-hydraulic, nuclear safety dedicated codes (such as ATHLET, CATHARE, Relap…) have been developed over several decades, so the previously cited libraries have never been used for safety demonstration studies.

However, other Modelica libraries have already been used for safety demonstration submitted (and approved) to a nuclear safety authority. It is the case, for example, for the I&C ventilation of Hinckley-Point C based on TAeZoSysPro calculations (developed by EDF); a validation report for such applications is associated to that library.

5.2.2 Safety Codes

Safety codes are used to assess the safety of nuclear power plants, research reactors and spent fuel pools in different scenarios by vendors, utilities, authorities and TSOs. The two codes used in TANDEM (ATHLET and CATHARE) for safety analyses are described below shortly. More details on their functionalities and for coupling are given in Chapter 4.2 of the deliverable D2.2 of the Tandem project.

ATHLET

ATHLET stands for "Analysis of Thermal Hydraulics of Leaks and Transients". The ATHLET thermal hydraulics code developed analyses the entire spectrum of events and accidents to be assumed in nuclear reactors. The simulation code, which is used worldwide, allows the assessment of the safety of reactors of different construction lines and designs.

ATHLET functions and application

Both GRS and other users in Germany and abroad use ATHLET to assess the safety of NPPs, research reactors and spent fuel pools in different scenarios. ATHLET can be used for the simulation of operating states, DBA and beyond-DBA without core damage in various types of nuclear reactors. Typical accidents are, for example, a break in a reactor coolant line or the failure of the electrical power supply. ATHLET has models for pressurized water reactors, boiling water reactors, RBMK reactors, advanced 4th generation reactor concepts and SMRs.

With the system code ATHLET, all essential components within the plants can be represented. The focus is on the thermal-hydraulic simulation of the fuel assemblies, the cooling circuits and the safety systems. The code simulates both active and passive emergency cooling systems as well as alternative cooling media such as helium, molten salt, and liquid metal (e.g. lead, sodium, lead-bismuth eutectic). In the process, these simulations can also be carried out for complex models on a normal PC within days or a few weeks, so that no high-performance computing centre is required.

Focal points of the simulation code

Current developments and research priorities concern the simulation of:

- passive safety systems in various reactor concepts
- fuel cooling in spent fuel pools
- special fuel assembly designs in research reactors
- cooling media for new reactor concepts
- SMRs

Important fields of application for ATHLET have included analyses regarding the effects of a power increase, the effectiveness of safety devices according to new findings, events with strong, spatially inhomogeneous power change in a reactor core (coupled with 3D neutron kinetics codes) and support for modelling in probabilistic safety analyses.

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Code features and structure

The modular structure of ATHLET allows for a building-block-like system configuration and can thus replicate a wide range of plant designs. ATHLET consists of four fundamental modules:

1. Thermal hydraulics module for calculating the coolant flow and the thermal energy transported with the coolant from the fuel assemblies. With a 3D module, areas with complex, multi-dimensional flows can be resolved in detail. Specific models make it possible to model elements of a reactor such as pumps, valves or steam generators.

2. Heat conduction and heat transfer module for calculating heat conduction in plate-, cylinder- and sphere-shaped structures as well as heat transfer between structural surfaces (e.g. of fuel rods or heat exchangers) and the coolant.

3. Control module for modelling control and governing systems via which active elements - such as pumps and valves - can be controlled.

4. Neutron kinetics module for modelling nuclear heat generation in fuel assemblies.

CATHARE

CATHARE is a system thermal-hydraulics code developed since 1979 by the CEA (French Atomic Energy Commission), EDF (Electricity of France), Framatome and the IRSN (Radiation protection and Nuclear Safety Institute). Originally devoted to the safety analysis of water-cooled reactors (such as PWR, VVER or BWR), it has been subsequently extended to simulate the transient behavior of other reactor families. In the framework of the Generation IV International Forum, the CEA used CATHARE to launch several feasibility studies of advanced reactor concepts including Gas Cooled Reactors (GCR), Sodium-Cooled Fast-Breeder Reactors (SFR), Supercritical Water-Cooled Reactors (SCWR).

The CATHARE code has a modular structure: several modules can be assembled to represent any hydraulic circuit. The three main hydraulic modules are:

- the 1-D module to describe pipes,
- the 0-D module to describe large capacities,
- the 3-D module to describe multidimensional effects in specific components.

To complete the modelling of the circuits, other components can be connected to the main hydraulic modules, such as:

- Thermal walls for radial heat conduction calculation

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- Heat exchangers between two elements of the same circuit or of two circuits
- 1-node turbomachinery (pump, blower, compressor, turbine)
- Valves
- Mass sources / sinks
- Accumulators
- Fuel pin thermo-mechanics to predict cladding deformation, creep, rupture, clad oxidation and thermal exchanges
- Point neutronics

A CATHARE hydraulic circuit simulates a two-phase flow by means of a six equations model (i.e., a model based on the mass, momentum and energy conservation of each phase). The third and most recent major version of the code (known as “CATHARE-3”) also introduced a three-field nine equations model, in which the liquid phase is split into a continuous field (film) and a dispersed field (droplets). This potentially allows improving the simulation of specific flow patterns (like the annular mist flow) in which the liquid film behaves differently from the droplets. Both the six-equation model and the nine-equation model can be extended to take into account non-condensable gas transport.

The numerical method implemented in CATHARE are known for their robustness in a wide range of flow configurations. The spatial discretization consists in a first order upwind scheme with a staggered mesh. The time discretization also consists in a first order scheme, fully implicit for 1-D and 0-D modules and semi-implicit for 3-D modules. The nonlinear system of equations is solved by a Newton-Raphson iterative method following several steps. The solution can be distributed over several processors in order to reduce the CPU time by parallel computing. This allows real time calculation of reactor transients.

Most of the information reported in this section is extracted from the CATHARE official website (cathare.cea.fr), where an extensive bibliography is available.

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6 Conclusion

In practice we note there are no significant open safety issues regarding the integration of SMRs in HES that could require rules adaptations in the way the safety principles are implemented.

As for conventional NPP used for cogeneration, various safety analyses are required to perform the safety demonstration, including thermos-hydraulic deterministic analyses, PSA, hazard analyses, and failure tolerance analyses.

The specificities of SMRs mean that all DBC scenarios can be managed passively, without the need for any action by the operator and without any external source of final heat removal. Consequently, all scenarios involving transients due to the integration of SMRs in HES for industrial heat or electricity cogeneration should be considered in the DBC list and then managed in such a framework. Such transients must not lead to DEC situations.

It should be noted that one of the main difficulties is to identify the PIEs and also the hazards (internal or external) that may be present in such configurations. PIEs are strongly dependent of HES configurations and SMR's design. Therefore, a global methodology can only be general. Moreover, two-way interactions that SMRs may have with other systems shall be considered. . It is important to keep in mind that a "small" malfunction in the SMR can lead to problems in the other systems, which in turn affect the SMR, or vice versa (other systems malfunction affecting the SMR). Are existing models capable of simulating this interdependent response? How can these models be validated if no Operating Experience for SMR is not available? The defined methodology will be tested in the next tasks of the WP 4 through models using Modelica, ATHLET and CATHARE tools in different safety cases. That's why, the main PIEs will be selected considering potential safety issues highlighted in the sections of this deliverable.

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