EUROPAIRS

Support Action

Co-funded by the European Commission under the
Euratom Research and Training Programme on Nuclear Energy
within the Seventh Framework Programme

Grant Agreement no. 232651
Start date: 01/09/2009 Duration: 21 Months

www.europairs.eu

Safety and licensing evaluation of a (V)HTR coupled to industrial processes

Deliverable D2.1

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End-User Requirements fOr industrial Process heat Applications with Innovative nuclear Reactors for Sustainable energy supply

EUROPAIRS project – Grant Agreement no. 232651

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Document title	Feasibility of the safety assessment of the coupled system
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Number of pages	40
Document type	Deliverable
Work Package	
Document number	D2.1 – revision 0
Date of completion	05/01/2011
Dissemination level	☒ Confidential (EUROPAIRS consortium and European Commission) ☐The above + specific dissemination group: … ☐Public
Document status	

Summary

This repport presents the preliminary review of the feasibility of the safety assessement of a High Temperature Reactor producing electricity and process heat.

Revisions

Rev.	Date	Short description	First author	Task leader	Work package leader	Coordinator
00	8/09/11		*O. Baudrand*	-	O. Baudrand	E. Bogush

Distribution list

Name	Organisation	Comments
EUROPAIRS consortium participants		

EUROPAIRS_D21_V12_final.doc
EURATOM FP7 Grant Agreement no.232651

Demandeur	E. Bogush (coordinateur projet EUROPAIRS)
Référence de la demande	GA 232651
Numéro de la fiche programme	02D/T02-001/06
Processus de rattachement	R4

Feasibility of the safety assessment of the coupled system

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(DSU/SERIC/BEXI)

Service d'évaluation de la sûreté des réacteurs refroidis au gaz,
à neutrons rapides, d'expérimentation

Rapport DSR/SEGRE n° 16

	Réservé à l'unité		Visas pour diffusion		
	Auteur(s)*	Vérificateur	Chef du **DSR/SEGRE**	Directeur **de la sûreté des réacteurs**	Directeur Général de l'**IRSN**
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Dates	05/01/2011	07/02/2011	10/03/2011		
Signatures					

* Excepté pour les les chapitres 6.1, 6.2, 6.2.2 et 6.2.3 rédigés par TUV NORD et FZJ
**Remplacée par C. Deust d'Haultfoeuille

Table of contents

1 INTRODUCTION

Among the Generation IV systems, High Temperature Reactors (HTR) are foreseen to play a role in the reduction of the emission of greenhouse gases because they exhibit great potential for use as heat and electricity co-generation utilities. The HTR concept is also a likely candidate for hydrogen and synthetic fuel production relying on the suitable temperature range of the coolant (from 700°C up to 1000°C in the future). This potential being assumed, licensing uncertainties due to the coupling with an industrial process are arising and may jeopardize the feasibility of the system.

In this frame, this report aims at coming up with the specific assessment issues associated with the coupling of a nuclear plant and an industrial process. The safety assessment of the reactor itself is not in the scope of this report. Results of this study are tentative safety criteria consistent with the safety approach and the risk management strategy applied respectively to nuclear plants and industrial units. This report focuses on the feasibility of the safety assessment rather than on licensing matters which are within the competence of Safety Authorities. Nevertheless, several elements on licensing will be given as interpretation of existing regulations or approaches.

The first part of the report is dedicated to general considerations on the coupling requirements and a short review of the feedback of past safety assessments carried out on HTRs.

In a second part, two examples of coupled systems are proposed to serve as test cases. They were selected taking into account the potential needs expressed by the end users. So electricity and heat (steam) co-generation appears to be the main incentive to develop HTR technology. Moreover, this technology is by definition associated with high temperature (compared with PWR technology); this is why the processes operating at temperatures accessible with PWRs were not retained as potential applications.

The third part of this report is presenting the assessment of the potential impact of the industrial facility on the nuclear plant. In Europe, the Seveso II Directive obliges industrial facilities to take all necessary measures to prevent the accidents and / or to limit their consequences for the environment. As for nuclear plants, a safety report is to be issued by the industrial applicant, which presents a risk analysis and assessment, a risk reduction plan (if possible), and measures taken to prevent accidents. Attention will also be paid to hazards and transients induced by End Users demand variations.

The potential impact of the nuclear plant on the industrial facility is then presented and completed by a review of the approach used for the assessment of external hazards. Another objective is to examine the potential risk for contamination of the products (steam, hydrogen, etc.) by radioactive elements like tritium in relation with the requirements given in the 96/29 EURATOM directive on radiological protection.

Finally, a synthesis will present tentative safety criteria and acceptance criteria which appear to be relevant for the coupled system and key aspects associated with the safety assessment feasibility will be outlined.

2 GENERAL OBJECTIVES

In order to assess the licensing feasibility of the coupled system, the working group has agreed upon two basic objectives:

1. The coupled facility (plant which is dedicated to the production of industrial products) shall not fall under nuclear regulation.
2. The global safety of the nuclear plant shall not be lowered by the coupling.

The first point comes from industrial and nuclear operators. Actually, the nuclear plant shall be seen as a utility from the industrial site and no specific procedures or regulations related to nuclear safety shall be applied to the industrial processes.

Objective n°1 then involves:

- the practical separation between the nuclear and the industrial sites, each one being under the responsibility of its own operator[1];
- the demonstration of the sufficiently low level of radioactive contamination of the coupling fluid that comes out of the nuclear site[2];
- the status of the workers of the industrial facilities is the same as the public, in terms of radiological protection and exposure limits.

These issues will be discussed further in paragraph 6.

Requirement n°2 derives from the fact that the vicinity of the two systems gives rise to specific hazards (toxic release, explosion hazards, etc.). Practically, there would be two specific safety demonstrations addressing each facility of the coupled system. But, the safety assessment of each facility needs to integrate the reciprocal impact of incidental and accidental operation of the plants. As an example, the nuclear plant safety demonstration shall integrate the potentials hazards induced by the coupled process as external events.

In Europe, several nuclear sites (hosting LWRs) are actually located close to industrial areas. The way this is integrated within the nuclear plant safety demonstration could provide a reference approach for a HTR based system. This issue will be detailed later in paragraph 5.

Around LWRs, an evacuation area has usually been established based on effective dose limit in case of accidental release of radioactive elements (50 mSv ref.1). It is obvious that the installation of a process control room within the evacuation area surrounding the nuclear site would require drastic isolation measures to protect the process operators. Indeed, these protection measures would be tailored to allow at least the industrial processes to be safely shut down by the operators. For the HTR concept, the evacuation area could be canceled and the maximum protective measures limited to temporary sheltering in a limited area around the nuclear site. Such an approach, if demonstration is validated by the Safety Authorities, could be convenient for the coupled system because it would remove a constraint on the safety distance between the nuclear plant and the process. As an example, the GT-MHR project does not include provisions for protective measure to be applied out of the nuclear site, which is consistent with the limit criterion as defined in reference 4: "*the effective individual radiation dose for the population at the buffer area during design basis and beyond design basis accidents*[3] *should not exceed 5 mSv for the entire body during the first year after the accident. In this case, special protection measures for the population are not required*". In some countries, the limit criterion may be more stringent (in South-Africa, regulation stipulates a limit criterion of 1 mSv). Similarly, in frame of the MHTGR project, the DOE has proposed a reduced emergency preparedness requirement and a reduction of the radius of the emergency protection zone (ref. 4). It is to be noted that, up to now, the Public Authorities in Europe require the definition of severe accidents that would ground the implementation of the 5th level of the defense in depth. These accidents are then used to establish the emergency planning. Upon these considerations, the impact of the nuclear plant will be further addressed in paragraph 6.

When dealing with nuclear installation, an aspect of the licensing lies in the assessment of the physical protection against the theft of nuclear materials and the sabotage of the facility. This mandatory item aims at ensuring the protection of the public from the consequences of potential acts of malevolence. The regulation of physical protection is a matter of State's legislation, but definitions and recommendations can

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[1] Note that in Europe, the operator of a nuclear facility is responsible for the safety regardless the financial set-up established for the coupled system.

[2] It is to be noted that the so called "nuclear site" is the area enclosed within the protection fence. It comprises the nuclear plant with the reactor, the electricity production plant, the auxiliaries and the fuel storage (fresh and spent fuel).

[3] In Europe these accidents correspond to Design Extension Conditions (DEC).

be found in the IAEA Information Circular 225 (ref. 2). The implementation of the physical protection begins by the definition of sensitive targets and corresponding threats. The threats are defined by the regulator and are usually classified data. Then several zones are established around nuclear materials and sensitive equipments with gradated protections (fence, access control, monitoring, alarm systems, etc.) and procedures which are based on the defense in depth principle. In case of a coupled system, the implementation of the nuclear site physical protection might not impact the system arrangement if the industrial process and the nuclear plant are settled on separated sites. On the contrary, if the industrial process was enclosed in the nuclear site (inside the surrounding protection fence), then the operators of the process should have a specific access clearance, restricted to the process area (see figure 1). The monitoring and control of the access of the process workers to the protected zone would fall under the responsibility of the nuclear operator. This latter option is clearly more stringent than the separated sites option.

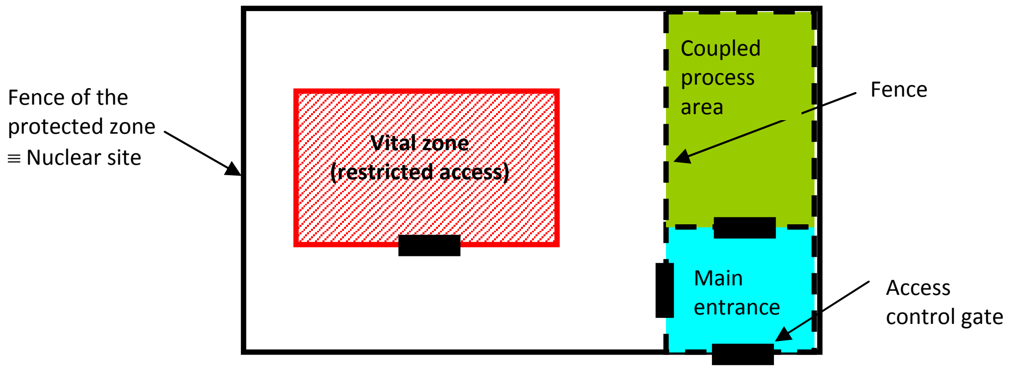


Figure 1: Example of zones of physical protection (with onsite industrial process)

3 ANALYSIS OF THE LICENSING TREND AND THE FEED-BACK

Elements on licensing feedback are developed in details in the WP2 deliverable D2.2 (ref. 5); they relate mainly on power plants design for electricity production. The present paragraph is a reminder of the main conclusions of this document with emphasis on coupling features. Considerations drawn from PWR safety assessment are also presented.

Concerning the basic safety requirements associated with the coupled system, the main issue to be dealt with can be expressed by the following question: is it necessary to strengthen the safety requirements already established for LWRs?

From the reference documents analyzed in the D2.2 document (ref. 5), the safety approach as it has been developed for LWRs is deemed sufficiently complete in itself to cover coupling. As an example, if we consider the approach for design optimization against risks as presented in the European Utility Requirements (EUR), the specific risks associated with the coupling may be taken into account without modification of the design procedure as proposed on the figure below (integration to the risk analysis).

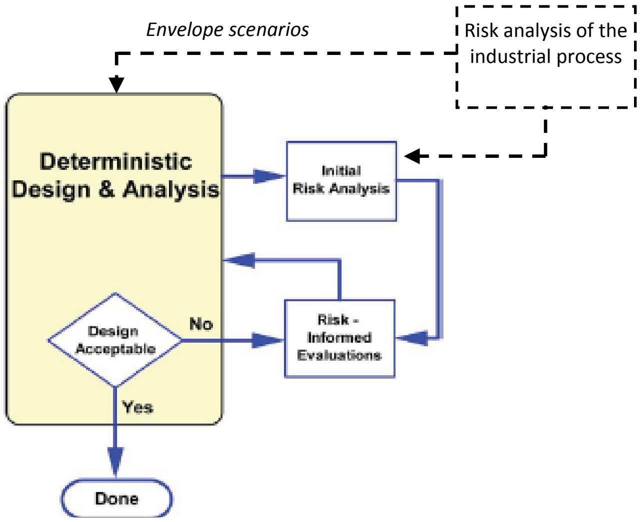


Figure 2: General approach presented in the EUR

The safety objectives for the integral core melt frequency (CMF) are less than 10^{-5}/y for the EPR™, considering all types of events. The integral large radioactive release frequency should be less than 10^{-6}/y (following the IAEA INSAG3 recommendations). So, given the annual frequency of external events[4] induced by the process plant, it should be demonstrated that the impact of the external events on the CMF frequency remains compatible with the above limits. Theoretically, this objective could be expressed as follows: $\sum_f P(ee) \times P(r/ee) + \sum_f P(ie) \times P(r/ie) < 10^{-6}$, where P() is the probability, f is the number of families of events, ee stands for "external event", ie stands for "internal event"[5], r stands for "release" and P (r/…) is the conditional probability of release in case of event occurrence. The risks induced by the coupled process should not exceed the range of radiological risks associated with the internal accidents of the nuclear plant. This means that the two terms at the left of this inequality should be balanced. In other words, the possibility to demonstrate a very low value for the second term of the inequality above (based on the HTR's favorable safety features), should not lead to allow a higher weight to the industrial risks.

In a more detailed approach, P(r/ee) corresponds to the product of two independent terms: P(f/ee), which is the conditional probability that the external event damages at least one safety function, and P(r'/f), which is the conditional probability that the damage results in a radioactive release outside of the nuclear site. First, such an analysis implies a comprehensive review and evaluation of the specific external hazards induced by the industrial processes and of hazards coming from the coupling circuits. After that, a probabilistic analysis would be required to estimate the conditional probabilities described above. On this point, It should be noted that, up to now, French operators have always considered that P(f/h)=P(r'/f)=1. But this conservative approach might not be sufficient in case of a higher number of external hazards. An example of integration of this criterion within the external hazard assessment will be detailed in the paragraph 5.3.

The safety demonstration for PWRs is based on the defense in depth approach. As it has been proposed in reference 4, this approach shall be applied also to the HTR and obviously to the coupled system.

To ensure the protection of the public, the PWR strategy relies on three barriers established between the nuclear fuel and the environment. These barriers are composed and supported by safety classified structures and equipments (liners, walls, filtration systems, emergency power supply, safety related

[4] Initiating events used in the standard safety demonstration are called "External" events in this report when they are induced by sources located outside the nuclear site.

[5] "Internal" events are initiating events occurring inside the nuclear plant.

monitoring systems, etc.). When considering coupled systems, rules to define and list these structures and equipments may be reviewed. Considering HTRs, elements on the proposed methodologies to select the safety related equipments and the boundaries of the safety classified structures are given in deliverable D2.1 (ref. 5). Although the PWR strategy shall be adapted to comply with the specificities of HTRs (see §2.2 and §3.1 in reference 5), it appears that no specific issue may be associated to the coupling. This is obviously because the coupled system may be assimilated to the power conversion system of the PWRs, in terms of separation between safety classified equipments and industrial equipments. The figure 3 illustrates a draft concept of the physical barriers which could be implemented for a HTR based coupled system. The first barrier is the association of the fuel particles, their coatings and the graphite. The second barrier is the primary circuit, including the tubing of the intermediate heat exchanger (IHX). The third barrier is the reactor building which has to contain the radioactive release in case of degradation of the first two barriers. Isolation devices are to be implemented on the intermediate circuit as it crosses the third barrier. In case of damage of the first two barriers, these valves would act as the last confinement device. They can be seen as an extension of the third barrier even if they are activated only in case of leak detection from the primary circuit or in case of accident (IHX tube rupture, etc.). The isolating valves could also prevent the accidental inlet of industrial fluid into the IHX in case of a leak of the heat exchanger (HX). In addition, one or two isolating valves on the industrial circuit (tertiary circuit on the figure 3) may be necessary for the process operation. They are located outside the nuclear site. It is to be noted that, the safety classified valves are subject to the single failure criterion. This could lead to implement redundant valves on the same leg of the circuit (not featured on the figure 3). At least, up to the last confinement barrier, the equipments are safety classified. As an example, the intermediate circuit is safety grade up to the isolating valves. Similarly, if equipments participate in the minimization of the radiological consequences, they might be safety classified (lower level).

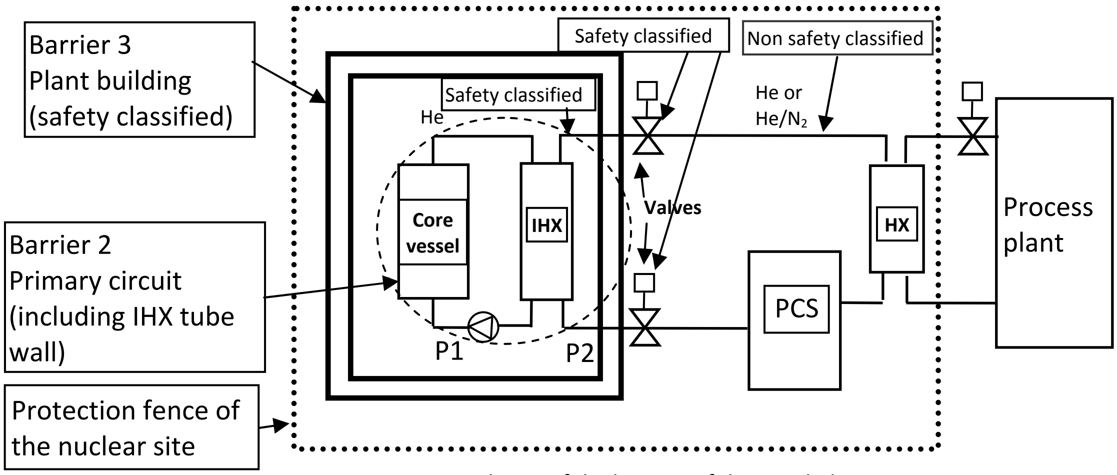


Figure 3: Tentative scheme of the barriers of the coupled system

4 TEST CASES RATIONALE AND DESCRIPTION

4.1 Rationale

To illustrate the considerations exposed in this document, two layouts of coupled systems have been drafted which are called “test cases”. The choice has been driven mainly by the willingness of the manufacturers to address near term applications[6] with also giving an insight into innovative technologies (with medium/long term prospects). The market study performed within EUROPAIRS has also outlined

[6] According to the designers of nuclear reactors, "near term" means a period of 10 to 15 years on average for the development and construction of the HTR (first of a kind reactor). This is clearly longer than the short term outlook for the industry.

several applications of interest. So, criteria have arisen from the discussions held within the EUROPAIRS consortium:

1. The temperature at the outlet of the HTR core should be close to 750°C for near term applications, with a possible increase over 850°C for middle term applications.
2. The nuclear system is producing heat and electricity (cogeneration). For near term development, there is a potential market for steam and electricity production addressing industrial complexes (mainly for the production of chemicals).
3. The nuclear system should be modular to show sufficient flexibility and reliability in terms of energy delivering.
4. The heat is delivered to the End Users through a tertiary circuit.

The first constraint derives from the synthesis delivered by the nuclear plant designers. The main reasons to limit temperatures around 750°C are to use the materials already “available on shelf” for the primary heat exchanger i.e. materials with sufficient operational feedback and compatible with the mechanical and corrosion constraints. So, this core outlet temperature is considered to be achievable without a significant step forward in the field of R&D and nuclear standards. Obviously, the operational feedback, if implemented in construction codes and standards, provides favourable features for the licensing.

The second point is of major importance for the feasibility of the coupled system: besides satisfying manufacturer needs, the cogeneration of electricity and heat provides a flexibility that is essential for the management of the transients imposed by the variable energy demand.

The third point implies that the industrial site should be large enough to be coupled with such an expensive device as a train of nuclear reactor modules and to offer a wide range of potential applications for the heat and electricity. As the nuclear designers have developed reactors with a power range between 200 MW and 600 MW, so given the efficiency of the nuclear system, the End Users’ demand is assumed to be at least a few hundred megawatts.

Although some designers claim that direct coupling would be technically achievable (secondary circuit is “industry grade”), point n°4 is assumed to be necessary to be able to reduce the risk of potential contamination of the process by radio-nuclides. According to the designers, the balance of plant is not significantly worsened by adding an intermediate loop, especially if the core outlet is below 750°C (according to reference 3). This item will be addressed further in the present document. Indirect coupling is also well adapted to cogeneration.

It has to be noted that the present study does not take into account the economics of the coupling. Indeed, industrial sites offer a very wide range of complex coupling schemes (steam turbines, heating of chemical processes, oxygen and hydrogen production, etc.). So the layouts of the systems envisaged in this study do not rely on a technical or economical optimization.

Finally, two test cases have been selected:

- Steam and electricity production for the End Users located on the Chemelot chemical site.
- Hydrogen production based on high temperature electrolysis (HTE).

The Chemelot site is a chemical complex located in the Netherlands. Today, steam and electricity are provided to the processes by several cogeneration units powered by natural gas. Thus, this test case should be considered as a near term application. The second example should be envisaged for medium-term development.

In this chapter, the Chemelot site and the HTE concept will be described to introduce the evaluation of the associated risks which corresponds to external hazards for the nuclear plant. The layouts of the systems and the main parameters related to the coupling are given below. A dedicated paragraph deals with flexibility requirements.

It must be noted that these test cases will be assessed only by looking at the specific issues brought by the coupling. These choices do not stand for validation of the safety of a specific coupled system.

4.2 Test case 1: Electricity and steam production for the Chemelot industrial complex

Chemelot is a huge chemical platform set up on an area of 800 hectares, hosting about 50 chemical facilities. These facilities are operated by more than a dozen companies (DSM, DEXplastomers, SABIC, etc.). The data below were made available through Baaten Energy Consulting with the assistance of USG (Utility Support Group) and are based on documents created by Chemelot.

4.2.1 Utilities for steam and electricity production

The diagram below shows the energy supply system in operation today at the Chemelot site.

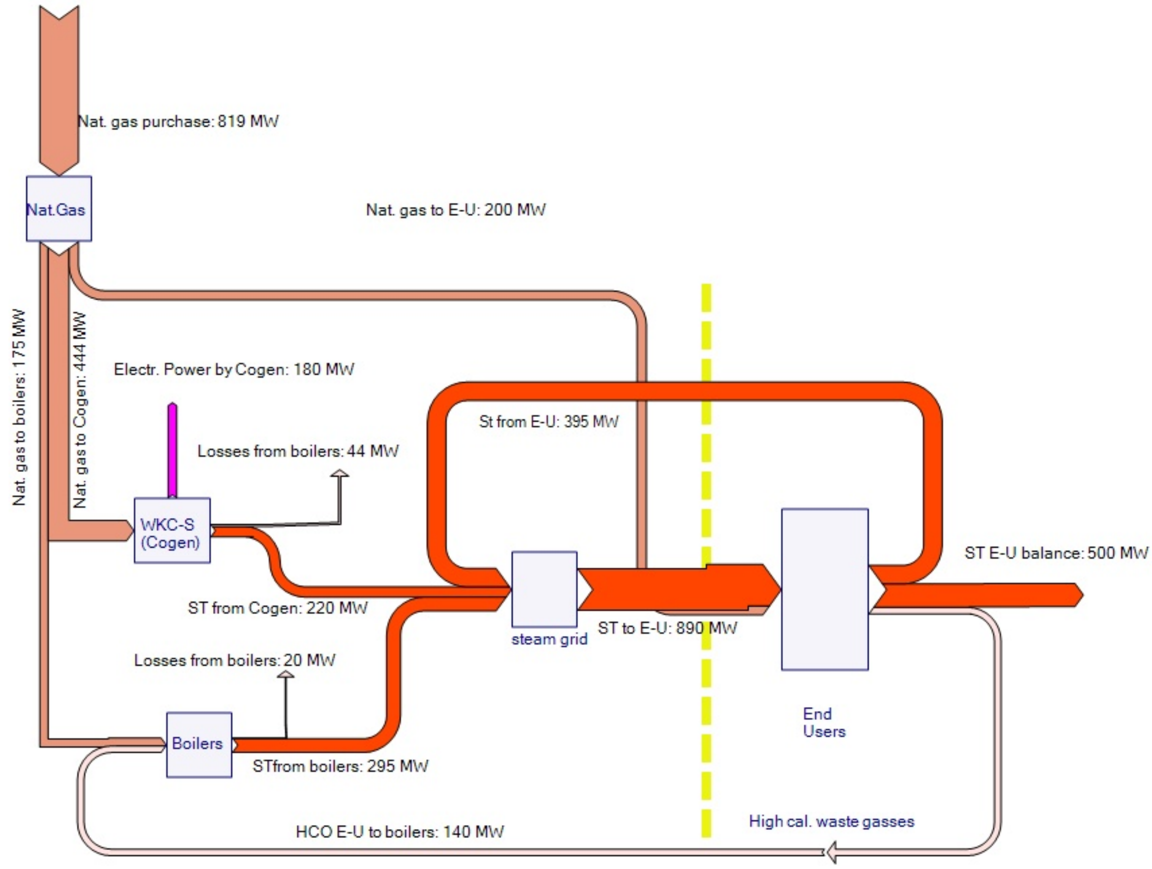


Figure 4: Energy flow diagram for the Chemelot site

The energy production on the Chemelot site relies on natural gas and also on off gases produced onsite. A part of the natural gas purchased is directly supplied to the End Users (ex.: for ammonia production). Electricity and a part of the steam are produced by a cogeneration plant. The electric power output of the cogeneration plant is 180 MW (multiple units). The total site demand in electricity is 270 MW.

The figure 5 below is derived from the description given by the Utility Support unit of the Chemelot site. The cogeneration plant produces steam at two pressure and temperature levels: 14 MPa/525°C and 1.8 MPa/300°C. According to the data given by the USG (2009 data), the average monthly production of steam by the cogeneration plant is around 100 t/h (1.8 MPa) and 120 t/h (14 MPa).

Steam is also produced by boilers fed by natural gas and burnable off gases. The boilers have an average power output of 295 MW but a total capacity around 650 MW. The overcapacity is required to ensure reliable steam supply also in case of maintenance, trips of boilers and sudden increases in demand in case of process start-up by customers.

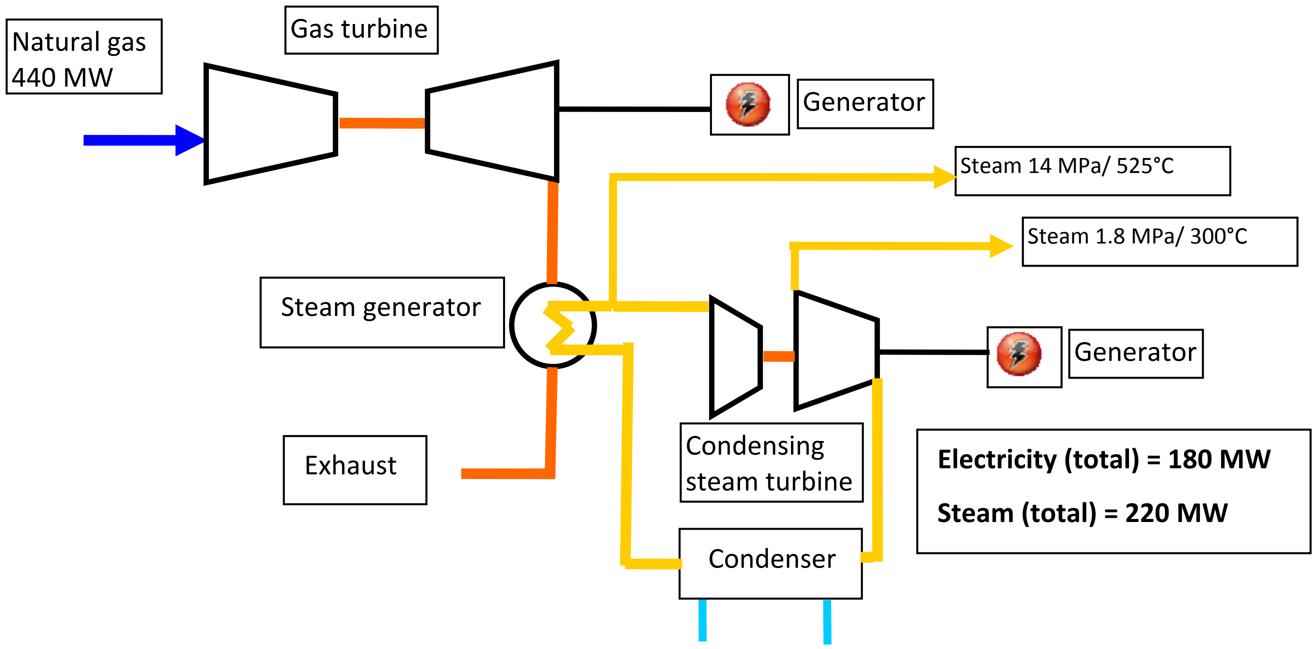


Figure 5: Example of plant layout for cogeneration

4.2.2 Coupled system layout

For the test case, the cogeneration plant is replaced by two HTR modules[7] as shown on the figure 6 derived from the data given in the reference 3 (temperatures, pressures and helium mass flow are assumptions). With a postulated overall efficiency of 40% for the nuclear plant, two 500 MW modules would be needed. The core outlet temperature is assumed to be 750°C, which is considered achievable in a short term objective. The intermediate circuit includes a steam turbine which exhibits a better efficiency than a gas turbine for this range of core outlet temperature. The secondary steam is produced around 570°C to be able to reach the required tertiary steam temperature of 525°C. According to reference 3, a 500 MW HTR module would have two primary loops with steam generators (only one shown in figure 5). The power delivered by the steam produced at present by the cogeneration plant of Chemelot is around 220 MW. The rest is allocated to the electricity production equal to 180 MW. Note that the electric power of the blowers would be close to 10 MW (ref. 3).

The intermediate circuit is isolated from the Chemelot steam grid by the tubing of the reboilers, allowing a dedicated monitoring and conditioning of the water/steam quality, in case the industrial requirements on steam quality would be less severe than the nuclear ones. Finally, this layout is also favourable to the damping of the thermal and mechanical transients on the steam generator using the thermal inertia of the intermediate circuit, thus reducing the potential damage rate on the barrier. Although, it is to be noted that the pressure differential between the primary and the intermediate circuit should be close to 10 MPa.

[7] This report does not deal with the specific safety issues associated with modular layouts.

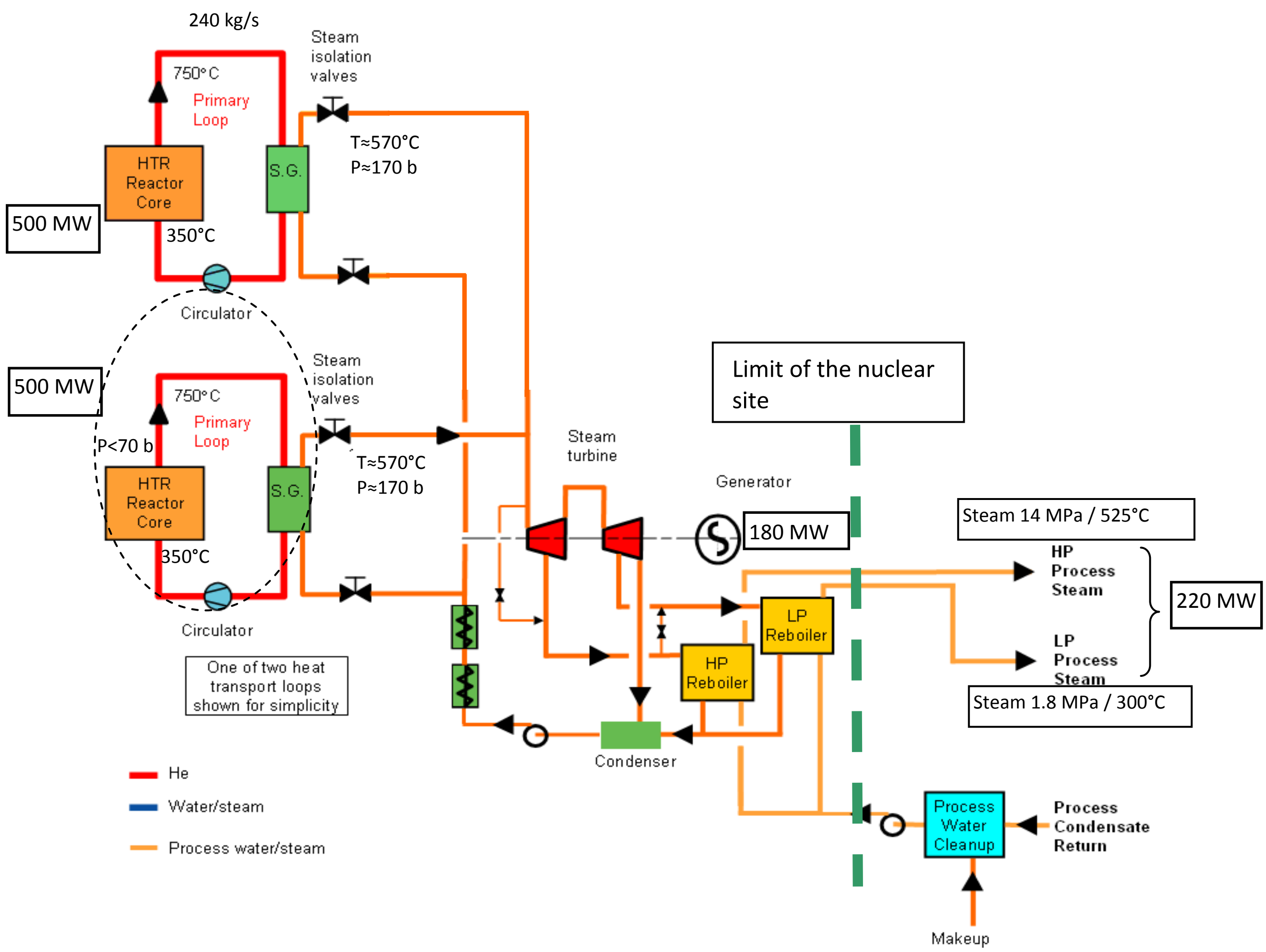


Figure 6: Tentative layout for the cogeneration plant

4.2.3 Building layout

A layout of the reactor building is given on the figure 7 taken from reference 3: it is worth noticing that the reactor vessel and the helium supply and purification systems are located underground. Note that this layout shows only one primary loop with plate type heat exchangers included in a vessel. The core vessel and the intermediate heat exchanger (IHX) vessel are connected by a coaxial hot gas duct.

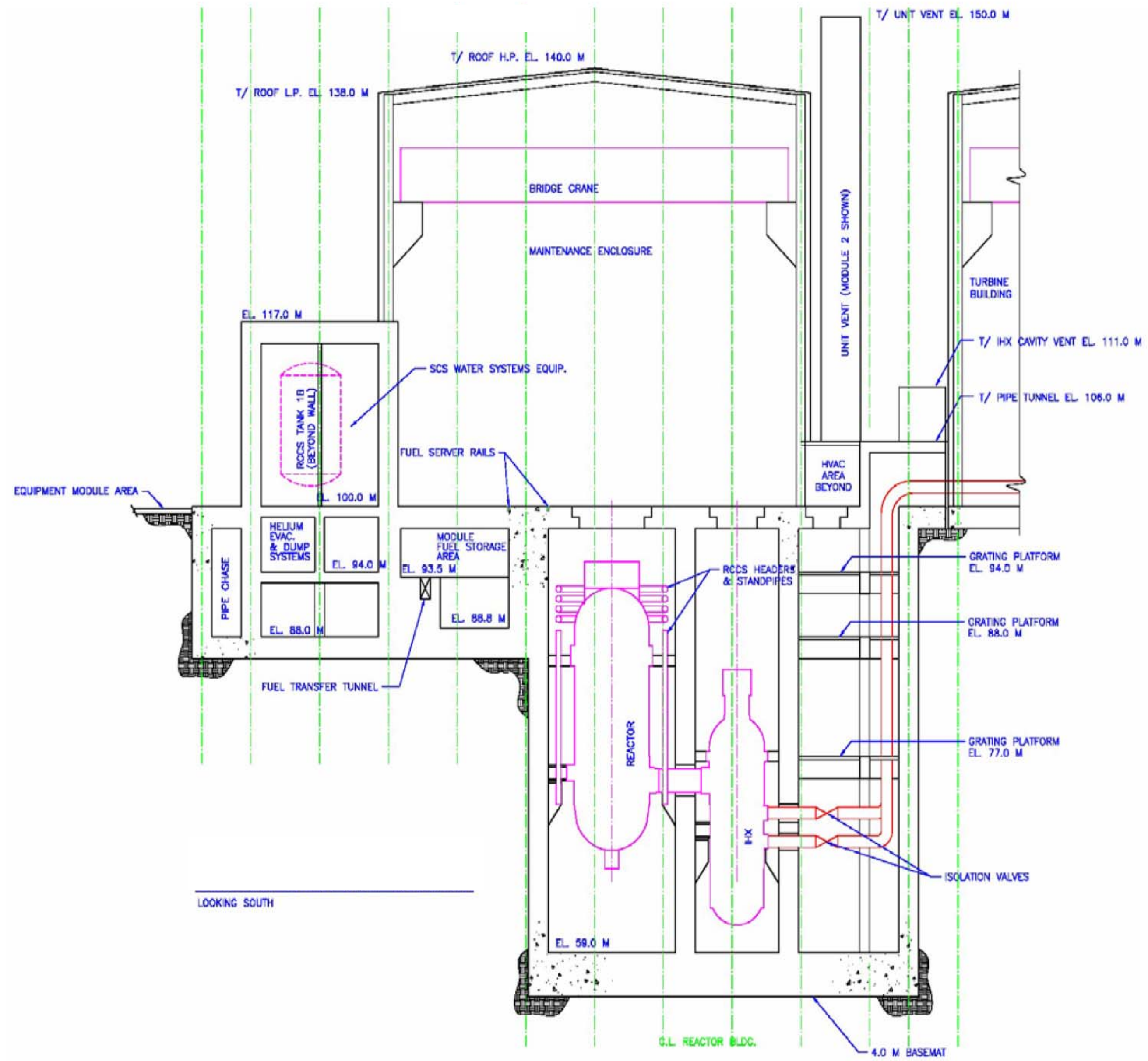


Figure 7: Nuclear buildings section – ANTARES design (data from ref. 1)

4.2.4 Cooling systems

The primary loops are able to remove the heat above a minimum core power level. During shut down states a dedicated system fed by water, which has an exchanger directly in the primary vessel, is started to cool the primary helium (and also the structures) thus reducing the time delay for reaching the shutdown state temperature. In the example provided in the reference 3, this system is not intended to remove the residual heat in case of accident, and therefore is not considered as a safety system. However, this strategy could be modified if it was decided to allow for the shutdown cooling system playing a role in accidental situations. Its use in case of accident would decrease the temperature transient on primary structures and equipment for frequent events. The safety system involved in the decay heat removal is the reactor cavity cooling system (RCCS). This system is implemented around the reactor vessel in the reactor pit (see figure 7). In normal operation, this circuit cools the vessel pit. In case of loss of cooling accident (either by loss of helium flow or pressure), the RCCS should be able to remove the residual heat and this heat is transferred by means of conduction and radiation within the primary circuit and between the reactor vessel and the RCCS, leading first to a temperature increase on primary structures. The RCCS is designed so that the integrity of the vessel and the concrete cavity could be ensured (see § 3.2 of ref. 3 for the description). It is stated by designers that the RCCS should be efficient even if the primary circuit has been depressurized. The RCCS is a redundant circuit. It can be filled by water or air depending on the efficiency needed. It can rely on natural or forced convection. But, in principle a high level of reliability has to be demonstrated both for the equipment and the cold source (see § 5.1).

4.3 Test case n°2: hydrogen production by high temperature electrolysis of steam (HTE)

4.3.1 HTE process

In this test case, a HTR module supplies high temperature steam and electricity to a hydrogen production unit. Practically, such an HTE unit could also be included in a chemical site, but we have considered that it should be connected to a hydrogen pipeline, allowing for a simpler risk analysis. As an example, hydrogen production could be achieved by solid oxide electrolytic cells (SOEC). These cells consume electricity and steam to produce hydrogen and oxygen. HTE technology is derived from the fuel cells which produce electricity from hydrogen (solid oxide fuel cells – SOFC). Therefore the association of reversible fuel cells with a nuclear reactor has been proposed to cope with peak demand in electricity without the help of conventional plants. Until now, this process has been developed up to the laboratory and pilot scale in Germany (in the 1980s), in the US and in Japan. In 2008, the Idaho National Laboratory has operated a laboratory scale facility with 720 cells producing up to 5.65 Nm^3/h of hydrogen (ref. 7). One of the main technical challenges is to increase the lifetime of the fuel cells. The cells and cell stacks are also sensitive to temperature gradient, but temperature is monitored by electric heating rather than steam.

The process diagram of a high temperature electrolysis unit proposed by North West University (EUROPAIRS partner) is presented on the figure 8 below. The production rate corresponds to an industrial scale unit, as compared to medium size methane reforming plants.

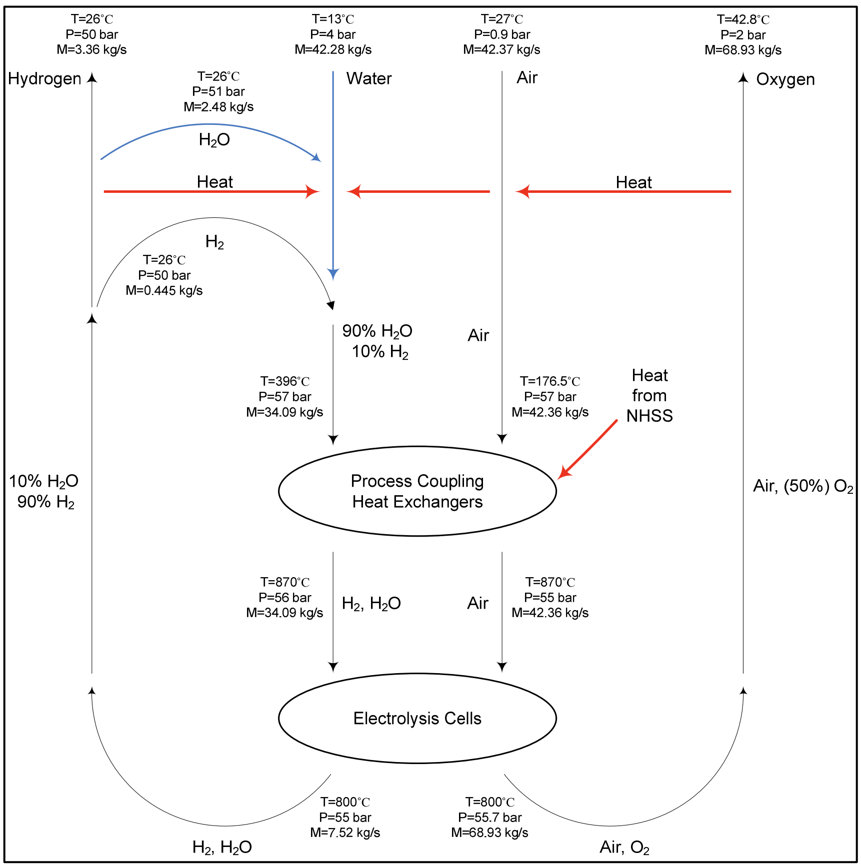


Figure 8: High temperature electrolysis for hydrogen production (NWU data)

This unit is customized for the coupling with a very high temperature reactor (helium outlet temperature around 950°C). Nevertheless, the INL has implemented a process with heat recuperators before the cell modules to allow the use of steam at a temperature lower than 450°C. The recuperator uses the hydrogen and oxygen output streams (820°C) to pre-heat the steam up to 790°C. The rest of the heat is brought by ohmic heating in the cells. The cells can also be operated at thermo-neutral regime: inlet and outlet temperatures are equal. This regime allows reducing the thermo-mechanical stresses in the SOEC modules.

4.3.2 Electrical and thermal power of the nuclear unit

According to the nuclear designers, a mid term application of HTR would allow to raise the helium outlet up to 850°C. The process for hydrogen production can be adapted to available core outlet temperature (850°C), with a minimum steam temperature at the inlet of the recuperator of 500°C (around 750-800°C at the outlet). In this case, an AREVA/EDF study (ref. 8) gives an electricity consumption of 257 MW and a thermal power need of 60 MW (thermo-neutral operating mode) to produce 2 kg/s of hydrogen. The hydrogen production of a standard gas reforming plant is typically 45 000 t per year[8], corresponding to a hydrogen mass flow around 1.6 kg/s (with a process availability of 90%). So, proportionally, the electricity consumption should be close to 205 MW and the thermal power to 48 MW. If the steam temperature is higher than 500°C, the electrical power need is reduced.

4.3.3 Coupled system layout

The heat and electricity could be supplied to the chemical process by two nuclear modules, each with a power around 250 MW. The following layout is extracted from the INL report in reference 9 and refers to NGNP studies.

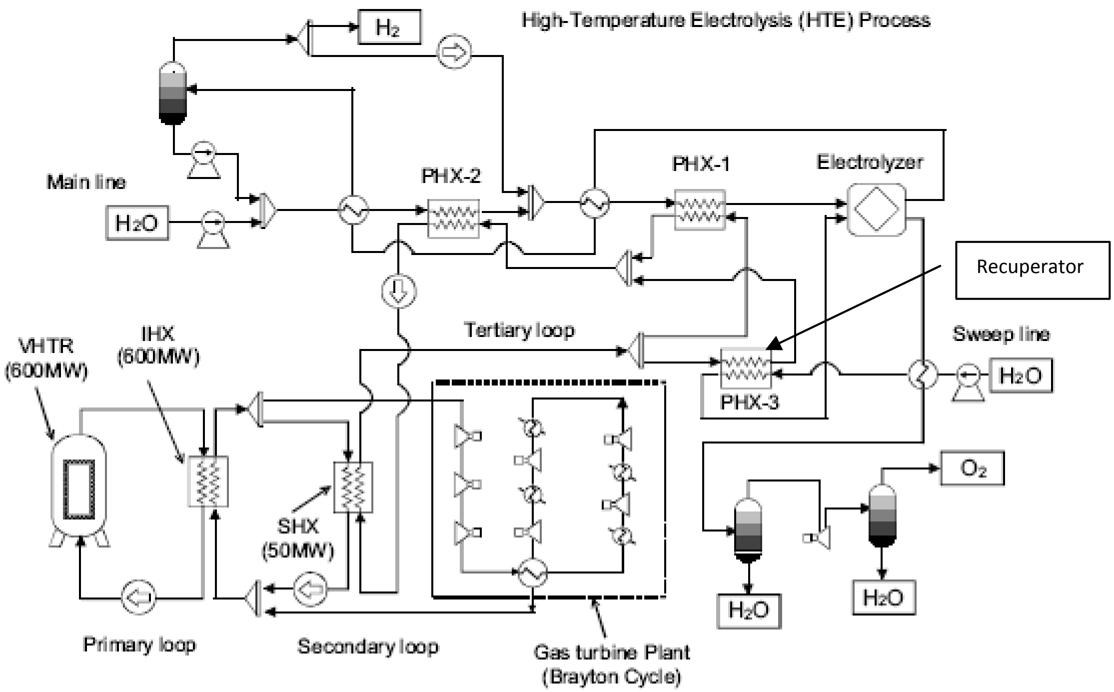


Figure 9: Coupled layout for hydrogen and electricity production (ref.9)

This simplified layout comprises gas circuits only. It has been optimized for a core outlet temperature of 900°C, but the coupling principle is applicable for 850°C. The indirect coupling scheme is postulated in order to be able to limit the tritium contamination of the hydrogen and to have sufficient versatility of the

[8] Data from « Association Française de l'hydrogène »

system. Indeed, it was required that the heat delivering could be switched off without a reactor shut down. This should be practically achievable because the heat power is much smaller than the electricity production.

4.4 Flexibility required by the End Users

A major challenge is to demonstrate the ability of the nuclear plant to follow the variations of the energy demand of the processes without being too much affected by the transients induced. This requirement applies to both test cases defined above.

Practically, different time scales need to be considered to cope with the variations in energy demand:

1. over several years, the availability of the nuclear plant has to fit with the maintenance periods and the seasonal variations of the market[9],
2. over several days, the system has to follow the variation of the demand associated with scheduled ,
3. over several minutes, the system has to cope with peak demand or inadvertent shut down of the processes.

As an example, for Chemelot site, the monthly average mass flow rate of the steam produced by the cogeneration plant is close to 120 t/h (14 MPa / 525°C) and 100 t/h (1.8 MPa / 300°C), corresponding to a power of 200 MW delivered by the secondary circuit to the boilers, whereas the nominal power available is 260 MW. This represents a margin of around 8% of the nominal power of the reactor modules. As a comparison, a PWR is able to follow automatically the electricity grid demand by adjusting its power within a range of ±8% of the nominal power. The increase rate of the demand (for the cogeneration plant) can be estimated between 2 $t.h^{-2}$ and 3 $t.h^{-2}$ of steam[10], which appears to be compatible with the power rise capacity of a HTR, given the PBMR requirement of 1%/min of the nominal power. The total steam demand of the Chemelot site uses to vary with greater amplitude but the conventional boilers may handle such variations because of their important overcapacity.

To achieve a sufficient flexibility, several technical requirements are to be assumed:

- the system is able to operate with only one reactor module,
- the power can be switched to heat production only or to electricity production only (given a reduced operating power of the reactor) i.e. no reactor shut down in case of loss of steam or electricity demand from the industrial site.
- a scheduled power variation can be performed by adjusting the nuclear power (manually or automatically).

For the test case n°1, a by pass of the steam turbine allow to increase rapidly the steam production. In case of brutal decrease in the steam demand, a damping system could be implemented on the intermediate circuit (additional water capacity, etc.).

For the test case n°2, fuels cells are grouped within modular units so a global shut down is of low probability. Moreover, heat represents a relatively small part of the energy production and electricity could be transferred to the grid in case of process shortage.

[9] Typically, operators of refineries use to plan shut downs every five years.

[10] The production of 1 t/h of steam at 14 MPa and 525°C corresponds to a power of 820 kW delivered to the boiler (assuming the boiler inlet temperature of the water on secondary side is 100°C).

5 POTENTIAL IMPACT OF THE INDUSTRIAL FACILITY ON THE NUCLEAR PLANT

The assessment of the potential impact of the industrial facility on the nuclear plant means that industrial risks should be integrated within the demonstration of the safety of the nuclear plant. So, this assessment first relies on the determination of the industrial risks, from which relevant initiators and scenarios may be extracted and used within the safety demonstration.

5.1 Types of events to be considered in the safety assessment of the coupled system

The incidental and accidental events which can occur in the coupled industrial process or in the industrial complex (industries in the surroundings) may have:

- a direct impact on the third safety barrier, its associated equipments (monitoring devices, ventilation systems, etc.) and decay heat removal systems (parts located outside the last confinement barrier),
- an indirect impact
  - on the second safety barrier through the transients imposed to the intermediate heat exchanger or the steam generator (temperature transients, shock waves, chemical challenge),
  - by altering the operational capacity of the workers (in case of toxic releases, heat wave or pressure wave),
  - by damaging auxiliary systems (electrical power supply, command control housing, etc.).

Except for events having a potential impact through the coupling circuit, events induced by the industrial process are called “external hazards” in the safety demonstration of the nuclear plant.

In the paragraph 5.2 below are described the industrial risks associated with the test cases. Paragraph 5.3 will deal with external hazards and paragraph 5.4 will deal with the internal effects due to the coupling circuit.

5.2 Assessment of the industrial risks

After having exposed a methodology for industrial risk assessment, this paragraph deals with the scenarios considered as external hazards for the safety assessment of the coupled nuclear plant, taking the example of the test cases. The associated consequences of the events are quantified below to fix the ideas on the margins.

5.2.1 Methodology for risk assessment

In the European Union, the safety of industrial platforms with significant risks falls under the Seveso II directive on the control of major accident hazards involving dangerous substances (directive 92/82/CE issued in 1982 and amended in 1996 – ref. 10). The member states have transposed this directive in their own regulation.

Following the directive’s approach, the mandatory safety demonstration for the industrial site is based on risks assessment and analysis of the risks reducing measures taken by the applicant.

In a safety report of an industrial site, the study of the major accident hazards relies on a comprehensive risk analysis of the site and its activities. This risk analysis leads to the definition of several major accident scenarios in term of occurrence frequencies and potential impact outside the site limits.

The typical scenario involves a loss of the containment of a dangerous product (examples: catastrophic rupture of tank, leak or full bore rupture of a pipe, etc.). This loss of containment can lead to the following dangerous phenomena, depending on the physical and chemical properties of the product:

- Jet or pool fire: the potential effects on structures and on populations are thermal effects (evaluation of the heat flux).
- Explosion as VCE (Vapor Cloud Explosion) or BLEVE (Boiling Liquid Expanding Vapor Explosion): the potential effects of an explosion on structures and human health are:
  - ✓ thermal effects through a fire ball or a flash fire: the corresponding thermal flux is then evaluated;
  - ✓ pressure effects: the explosion causes a pressure wave;
  - ✓ projectiles.
- Toxic dispersion: the main impact (lethal or irreversible effects) to be assessed is on human health and it depends on the concentration and on of the exposure duration (toxic load).

The quantification of the potential effects of missiles remains an R&D subject. The European HYSAFE Network (www.hysafe.org) has issued a comprehensive report (Biennial Report on Hydrogen Safety) describing all phenomena and modelling situations related to hydrogen dispersion, explosion and consequences including projectiles. Nevertheless, there is no standard method based on a fully validated model for the evaluation of the projectile risk.

For each type of effect (thermal, pressure or toxic) thresholds are defined to evaluate its potential impact on structures or population. These thresholds often depend on domestic regulations. Several European states use international thresholds while others have fixed their own thresholds in their regulation[11].

Examples of several thresholds are presented in annex 1.

For each hazardous phenomenon identified in the risk analysis, maximum effects distances are evaluated corresponding to the thresholds. Those distances are defining areas around the accident spot where any structure or person may be affected. If the distances appear to be unacceptable (inhabitants too close, etc.), the operator must implement mitigation measures to reduce the consequences of the accidents.

Depending on the (European) state risk regulation, mainly two kinds of approaches are developed in the safety reports:

- Probabilistic approach: the probability of dangerous phenomena is evaluated. Criteria are also established to evaluate the risk acceptability;
- Deterministic approach: scenarios are defined by the regulation or guidelines and no probability is taken into account.

"Mixed approaches" are also possible, with a list of scenarios defined by the regulation or guidelines and a probabilistic evaluation of the defined scenarios.

In the case of a probabilistic approach, risk acceptance criteria are defined within domestic regulations. In the paragraphs below are given simplified examples of risk assessment for the two test cases.

5.2.2 Risk assessment for the Chemelot site

It should be noted that the risk assessment of the Chemelot site takes into account some prevention and mitigation measures specifically implemented by the applicants during this assessment.

The Chemelot site hosts a wide range of chemicals. The main processes involving hazardous materials are the following:

- Processes around ammonia (figure 10)

[11] As an example, in France, the thresholds are fixed in the order of the 29th of September 2005 (ref. 11).

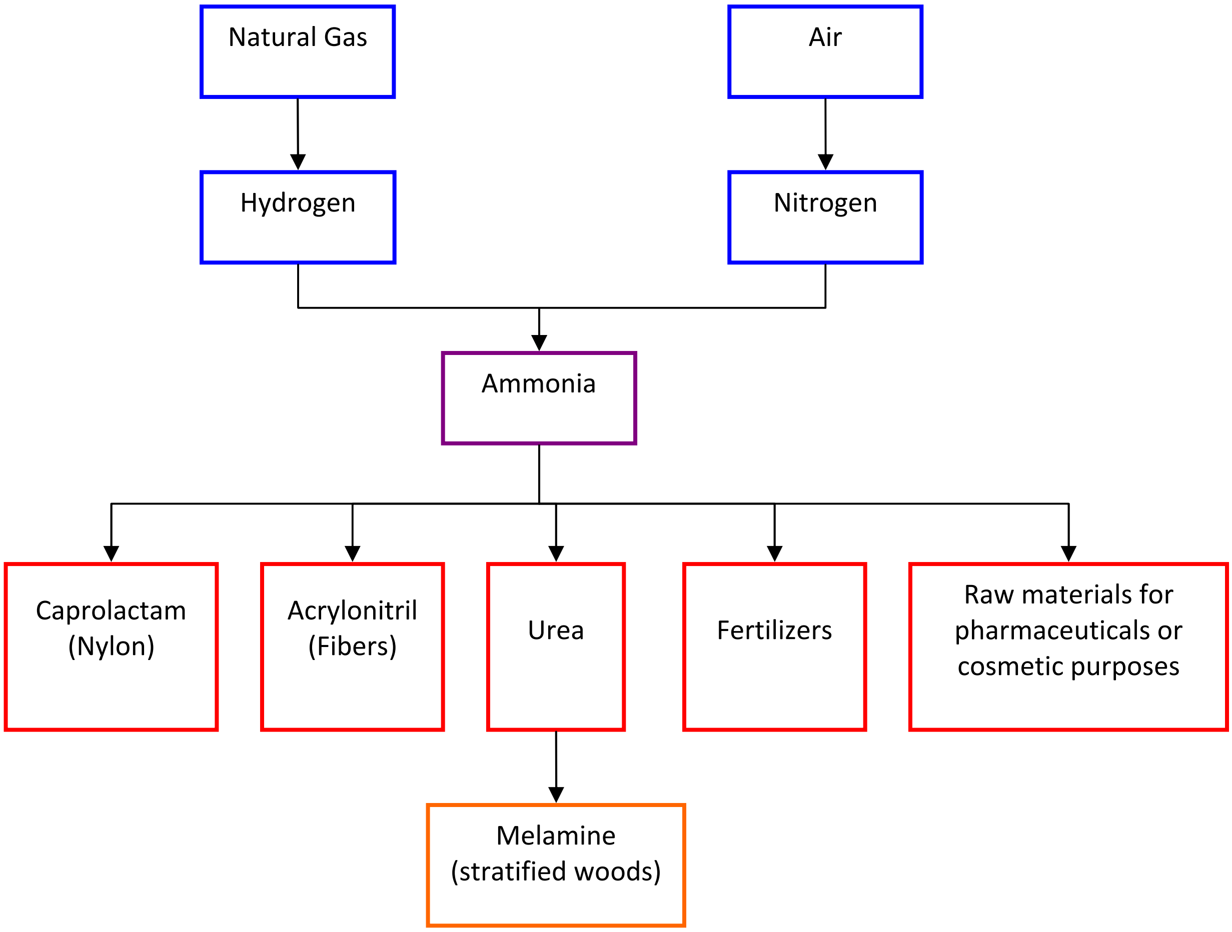
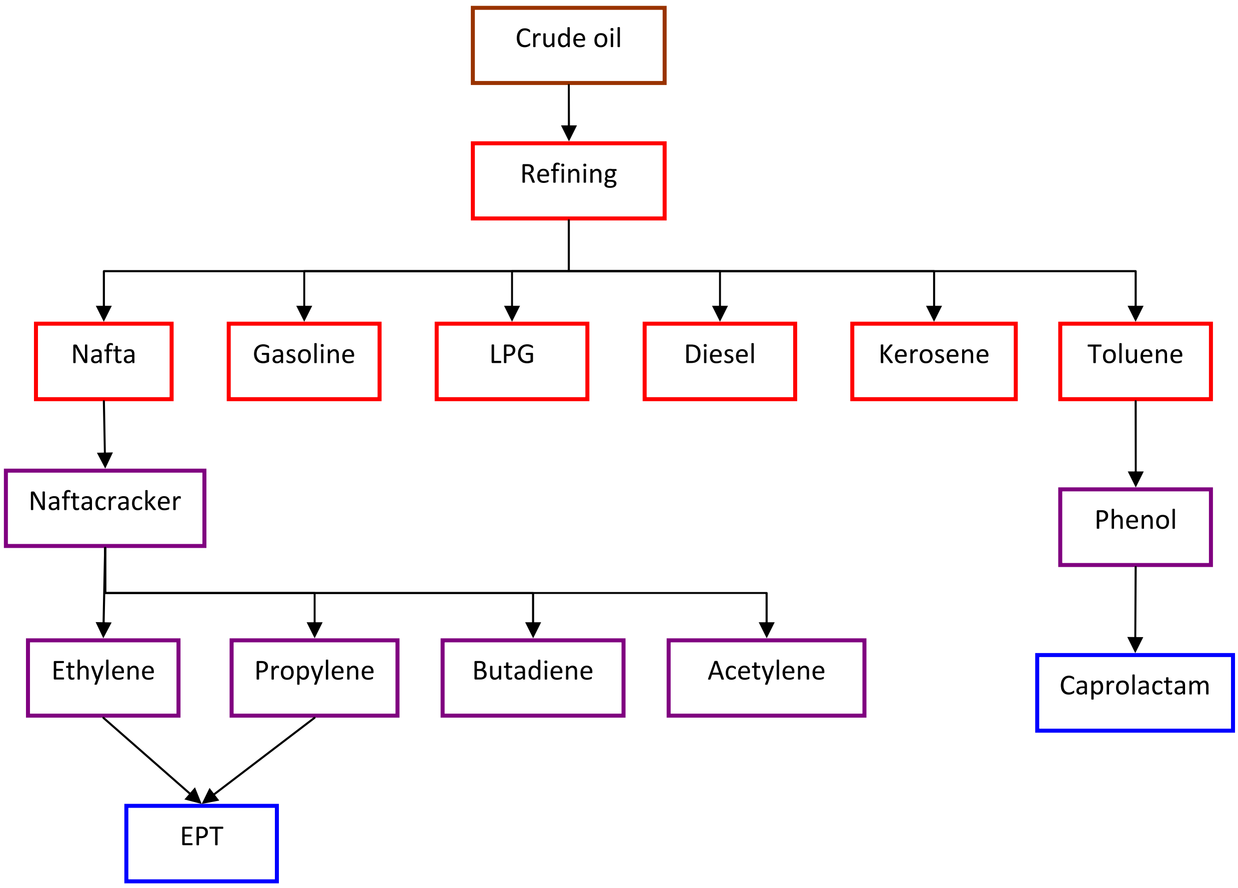


Figure 10: Ammonia and melamine production line

The temperature and the pressure needed vary depending on the different processes used for those chemical syntheses. We have considered that the maximum value for inlet temperature is around 525°C and that the maximal value for the pressure is around 20 MPa.

- Crude oil refining (figure 11)



* LPG: Liquefied Petrol Gas

** EPT: Ethylene-Propylene Terpolymer

Figure 11: Crude oil refining

The processes used in the refining of crude oil are mainly high temperature and high pressure processes. We considered that the maximal range of temperature in the processes is around 500°C and that the maximal range of pressure is around 8 MPa.

Consequently, the following chemicals have been identified as the main dangerous products to be taken into account in the safety analysis of Chemelot facilities:

- Toxics:

  - ✓ Ammonia : although ammonia can be considered in confined situation as a flammable gas , the toxic hazards are the main risks associated with ammonia in major accidents;
  - ✓ Nitrogen oxides (NOx);
  - ✓ Cyanhydric acid (or prussic acid);
  - ✓ Oleum (or fuming sulphuric acid);
  - ✓ Phenol;

- ✓ Other products: acrylonitril.

Considering toxic products, the effects of exposure on man (lethal effects or irreversible effects) depend on the concentration of the product in the atmosphere combined together with the duration of exposure.

- Flammables (explosion or fire risk):
  - ✓ Crude oil:
    - o Pool fire;
    - o Boil over causing thermal effects (fireball).
  - ✓ LPG (butane, propane): butane and propane are highly flammable gases that can lead to the following hazards:
    - o Ignition of a gas cloud in a congested environment or in the open air, causing overpressure effects (VCE) or thermal effects (flash fire);
    - o Jet fire causing thermal effects;
    - o BLEVE of a pressurized capacity, causing overpressure effects and thermal effects (fireball).
  - ✓ Propylene: propylene is also a flammable gas having the same type of hazards as the LPG;
  - ✓ Vinyl chloride monomer (VCM) : VCM is a highly flammable gas that can lead to the following hazards:
    - o Pool fire (the boiling point of VCM is -11°C at the atmospheric pressure);
    - o Ignition of a gas cloud in a congested environment or in the open air, causing overpressure effects (VCE) or thermal effects (flash fire);
    - o Jet fire causing thermal effects;
    - o BLEVE of a pressurized capacity, causing overpressure effects and thermal effects (fireball).
  - ✓ Acrylonitril: acrylonitril is a liquid at the ambient temperature and the atmospheric pressure. It's highly flammable and can lead to the same kind of hazards as VCM (except BLEVE).

A comprehensive study of the loss of containment scenarios of the products listed above and their atmospheric dispersion is presented in the safety report of the Chemelot site (and explosion or fire if existing). Maximal effect distances are then evaluated (thermal effect, toxic effect or overpressure effect).

The maximal potential effect distances evaluated in the safety report of Chemelot are the following:

- For toxic effects:
  - ✓ Scenario identified: catastrophic rupture of an ammonia tank (around 1150 t)
  - ✓ 1% lethality[12]: around 1550 m;
  - ✓ Frequency ≤ 10^{-7} / year;
- For thermal effects:
  - ✓ Scenario identified: instantaneous loss of containment of a propene tank (around 1000 t)
  - ✓ 3 kW/m²: around 2200 m;
  - ✓ 9,5 kW/m²: around 1300 m;

[12] Area where 1% of the population may die due to the effect of the toxic exposure.

- ✓ Frequency ≤ 10^{-7} / year;

- For pressure effects (explosion):
  - ✓ Scenario identified: instantaneous loss of containment of a butane tank (around 1500 t)
  - ✓ 30 hPa: around 2800 m;
  - ✓ 100 hPa: around 1200 m;
  - ✓ 300 hPa: around 700 m;
  - ✓ Frequency ≤ 10^{-7} / year.

5.2.3 Risk assessment for the HTE process

The following dangerous products have been identified for the hydrogen production:

- Hydrogen

Hydrogen gas is an extremely flammable and highly reactive gas. It can form explosive mixtures with air within proportions of 4% to 75%. Its auto-ignition temperature is 570 ° C. The minimum ignition energy is low (0.02 mJ for a stoïchiometric mixture, about 10 times lower than the hydrocarbons one).

- Oxygen

The risk of fire is increased from an oxygen content of 25% in the atmosphere. Moreover, the burning rate increases significantly when the oxygen proportion in the atmosphere exceeds 35%.

In an atmosphere enriched in oxygen, energies between 1 and 10 mJ are sufficient to cause the ignition of various tissues. A discharge of static electricity can easily cause the ignition of tissues.

The following risks are also considered for the hydrogen production:

- fire due to:
  - the enhanced flammability in case of an oxygen leak,
  - jet fire in case of a hydrogen leak (immediate ignition),
- thermal and pressure effects of a hydrogen explosion,
- projectiles generated by an explosion or a turbine failure.

No toxic effect has been identified for the HTE facility.

At the end of the process, hydrogen is a pressurized gas (5 MPa at 26°C). We assume that hydrogen is not liquefied and no liquid hydrogen storage is present on the facility. The hydrogen plant could be connected to other facilities by pipes or directly to a pipeline grid (through a connecting station to regulate the pressure).

The hydrogen production is about 138 t/d, which represents 1.54×10^{6} m^3/d (STP) of gaseous hydrogen. The oxygen production is then 1100 t/d or 7.7×10^{5} m^3/d (STP). In this test case, oxygen is removed from the process by steam used as a sweep gas (see figure 10). It is finally released to the atmosphere.

For the risks described above, we can assume the following effects distances:

- Fire:

- ✓ Enhanced flammability in case of an oxygen leak: the envelop scenario could be a loss of the oxygen containment in the process. Assuming a containment volume of about 1 000 m^3 at 5 MPa. An oxygen concentration greater than or equal to 25% could be observed beyond around 150 m;
- ✓ Jet fire in case of a hydrogen leak (immediate ignition assumed): the envelop scenario could be a loss of containment of hydrogen in the process (loss of containment of hydrogen gas: around 1 000 m^3 at around 5 MPa). The catastrophic rupture of a pipe with internal diameter equal to 6" would lead to a jet fire whose properties are the following (see annex 2):
  - o the flame length of the jet fire would be less than one hundred meters[13];
  - o taking into account the heat radiations, the maximal potential effects distances would be the following:
    - 3 kW/m²: around 180 m;
    - 5 kW/m²: around 155 m;
    - 8 kW/m²: around 135 m.

- Explosion of hydrogen: the envelop scenario could be a loss of containment in the process (loss of containment of hydrogen gas: around 1 000 m^3 at 5 MPa). The maximal effects distances can be estimated as followed:
  - ✓ Distance where the LFL (Lower Flammable Limit) can be reached (thermal effects of the flash fire caused by the explosion : total lethality inside the gas cloud above the LFL concentration): less than 100 m;
  - ✓ Distance at 50 hPa (overpressure effects): less than 400 m.
- Projectiles: calculation of effect distances for missiles may be based only on conservative assumptions. The associated scenarios are often difficult to assess because the definition of the size and mass of the missiles is subject to large uncertainties. As an example for the nuclear plant assessment, there are no method or threshold values proposed by the French Authorities to quantify the potential risk associated with missiles except those coming from turbine failure (turbine blades).

5.3 Potential external impact on the nuclear plant

5.3.1 Methodology for human induced external hazards assessment: state of the art and recommendations

The scope of this chapter is to analyze how the risks identified for the industrial facility could be integrated to the safety assessment of the nuclear plant. As a consequence the study is limited to the risks induced by human activity, which represents only a part of all external hazards.

It must be emphasized that existing nuclear sites were usually settled sufficiently far away from housing and industrial sites, so the external hazards induced by human activity were relatively rare and mainly due to mobile sources (ground transports, airplanes). However, along the nuclear plant life, industries are likely to be settled in its neighborhood. There are also examples of nuclear sites deliberately built near industrial sites, like the Gravelines plant located near the seaport of Dunkerque (6 PWR of 900 MW – thermal power). Moreover, there are in Europe many nuclear research centers gathering several industrial and nuclear installations.

[13] The flame length is estimated by using PHAST 6.54 software and choosing the jet-fire cone model developed by SHELL. In such a model, the flame is modelled as a cone frustum. PHAST is a software developed and marketed by Det Norske Veritas.

According to TSO's feedback, it appears that a current acceptable basis for the nuclear site demonstration can be summarized as follow (ref. 14):

- The site of the nuclear plant is chosen to be as far as possible from any hazardous human activity (especially from stationary sources of hazards).
- A set of standard deterministic requirements are applied for the design of the barriers and the protection of the safety functions (external pressure wave, heat wave, etc.).
- External hazards are listed and grouped into families (explosions, toxics, etc.), for which envelopes are determined.
- The expected frequencies of external hazards (including those induced by mobile sources) are evaluated and compared to acceptance criteria.
- If the acceptance criteria are not fulfilled, the design of the nuclear plant is adapted in order to avoid any significant effect on safety functions, assuming a shut down of the plant. Safety margins are to be demonstrated to avoid cliff effects (i.e. by sensitivity analysis).

Concerning the second point, two examples are the aircraft crash and the explosion. A load-time function associated with a small commercial aircraft and a standard shock wave are usually integrated to the deterministic design of the reactor building.

Concerning the fourth point, the basic acceptance criterion is the occurrence frequency of unacceptable radioactive releases in the environment (with the need of short-time off site response, as defined in the INSAG-3). As an example, this frequency has been set to less than 10^{-6}/y in several countries. This frequency limit encompasses accidents induced both by internal events and external hazards with (as far as possible) a balanced repartition of the frequencies for each of these families (see above paragraph 3).

Indeed, two main questions are arising:

1. Is there a common basis (means "widely accepted by the Safety Authorities") for the nuclear plant safety assessment with regards to the external hazards?
2. Is the methodology used to characterize the industrial risks compatible with the safety assessment of the coupled system?

Question n°1
Although the approaches for the assessment of the external events are yet far from unification, a common basis could be derived from the IAEA documents. In 2003, the IAEA has issued a technical document to gather the experience of plant owners and to serve as a basis for further development of a unified approach (ref. 13). This document deals with "extreme" external conditions (flood, tornados, etc.), including human induced hazards. A safety guide has also been issued, dealing specifically with the human induced hazards (ref. 14).

The following diagram of the figure 12 presents the general procedure (simplified) recommended in the above mentioned IAEA guide. This procedure should be applied for each external event (see footnote n°4) corresponding to a type of external hazard (e.g. BLEVE for explosions). A first criterion is proposed, called "screening distance". For a given external event, this distance is such evaluated that there is no significant effect of the external event on the nuclear plant (based on a deterministic evaluation).

A second criterion is the probability limit of the radioactive release potentially induced by an external event (box n°5). It should be recalled that this probability is usually expressed as a product of three terms: P(r)=P(ee)xP(f/ee)xP(r/f), where P(ee) is the probability of the external event, P(f/ee) the conditional probability of safety function degradation and P(r/f) is the conditional probability of release. A limit value for P(r) is given in the paragraph 3 (10^{-7}/y). As mentioned in the paragraph 3, the current approach is to account for the probability of the initiating event only, without further evaluation of the conditional probability of radioactive release (value arbitrarily set to 1 for P(f/ee) and P(r/ee)).

For external events that have not been eliminated after the screening steps proposed in the IAEA guide, the designer should enter a more detailed evaluation process. In this stage, the designer should determine if the accidental scenario corresponding to the external event is situated within the design basis envelope (box n°6). If it is the case, the afferent loadings are taken into account in the design of the relevant safety items (box n°7). One should note that there is usually one order of magnitude between the upper bound of the design envelope probability range (10^{-6}) and the limit probability of box n°5.

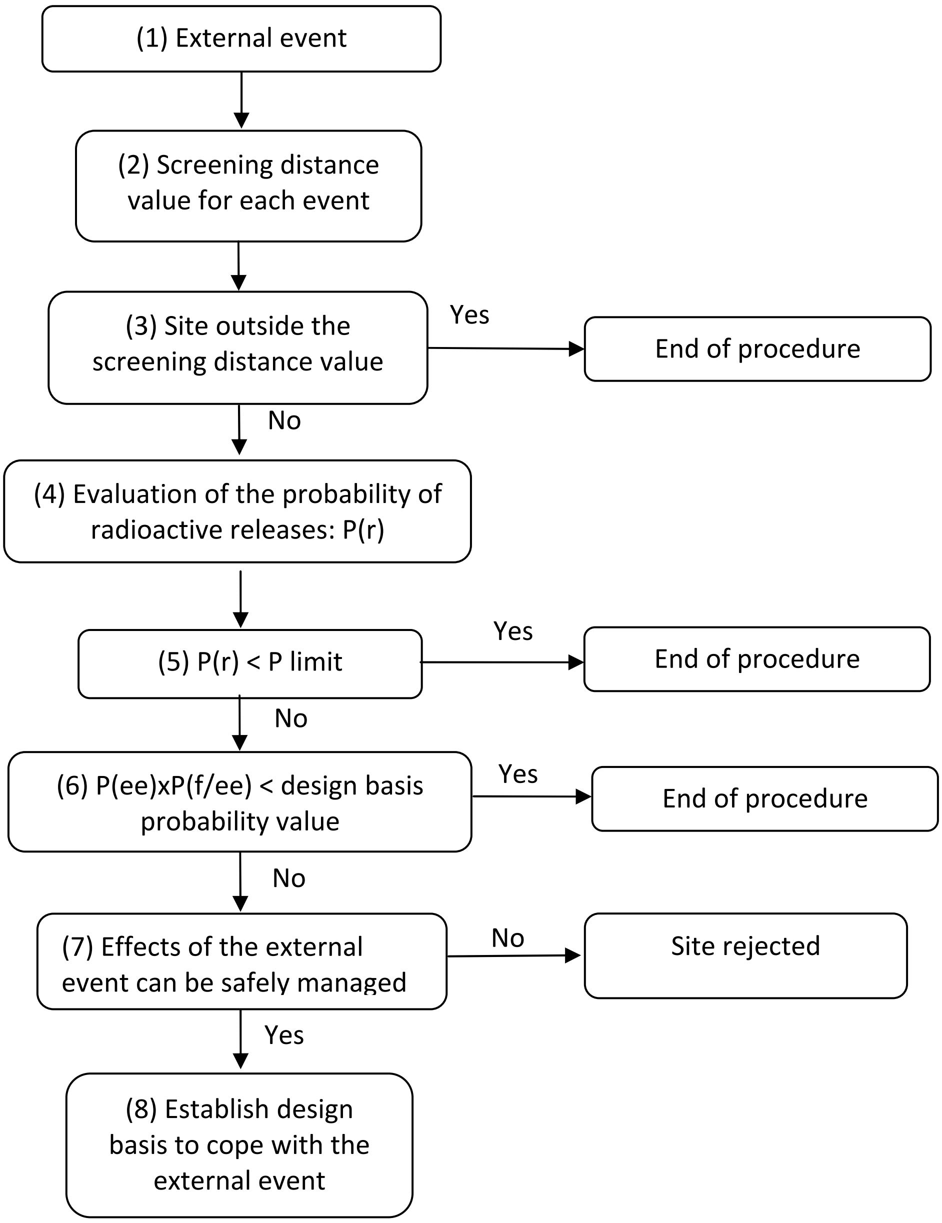


Figure 12: Evaluation procedure for human induced external hazards (ref. 14)

Considering the approach described above, several comments should be made with regards to the coupling of an HTR with an industrial process:

- The sensitive equipments in case of external hazard (also called “targets”) must be determined with regards to the specificity of the HTR design. As an example, the limited need for active safety systems should be a favorable feature.
- In general, the most convenient way to achieve the safety demonstration is to situate the nuclear plant beyond a “safety distance”. Indeed, according to the approach proposed by the IAEA, the safety distance could correspond to the screening distance as defined above in the figure 12. In this case, the

safety distance concept is equivalent to a "zero risk" distance. This means the distance for which no significant loading is applied on the nuclear plant (same type of definition may be applied for hazards induced by toxics). Nevertheless, this concept of "zero risk" distance is quite theoretical. Indeed, the coupling will lead to establish voluntarily the nuclear site near potential sources of hazards. Moreover, the distance between the nuclear heat supply system and the industrial process might be limited by technical and economic constraints. This is why, as stated in the IAEA document (annex I of the reference 13), the concept of safety distance should be simply associated with a distance sufficient for the effects of external events to be mitigated by design provisions and/or operational procedures. This means that, in most of the cases, a detailed analysis as defined in the beginning of this paragraph would probably be required. A prospective example is the approach described in the figure 12 (box n°5 to n°8).

- At least, the inevitable evolutions of the nuclear site environment would lead to the appearance of new risks and the screening distances would become obsolete. This would require to update periodically the safety assessment of the nuclear site.
- Concerning the detailed analysis described in the figure 12, the current practice does not include the evaluation of radioactive releases in case of external events. This is mainly because of a lack of data relevant to the safety equipments behavior and reliability in such accidental conditions. So, we see that the approach which could be based on the IAEA recommendations is still innovative compared with the current practice developed for PWRs.

As a conclusion, the practical determination of the design basis conditions for the safety items based on the external events presented in the safety report of the industrial site is not already standardized. Following the state of the art, as presented in the IAEA safety guide quoted in reference 15, an acceptable basic approach is to "e*stablish the design input parameters by a combination of deterministic and probabilistic methods and to proceed with the design in a deterministic manner*". Nevertheless the following recommendations could be drafted:

- ✓ Especially for the coupled system, some deterministic design loadings may be imposed without limit frequency consideration as regards the potential consequences (e.g. aircraft crash). It also gives margins in case the environment would change during the plant life. In such a case the nuclear plant would be subject to a renewed safety assessment.
- ✓ To give sufficient safety margins, the most penalizing operating conditions have to be selected for each external hazard, and probably combined with transients on the heat supply system like the loss of heat sink or even pressure waves in the secondary circuit (process industrial shut down, explosion effects, etc.).
- ✓ The external hazards may induce common mode failures, so they are more complex than single load cases and at least indirect effects and domino effects. Potential delayed consequences of an external hazard may also be considered. This could be the case for corrosive product or dust release, which may cause delayed degradation of the safety systems.
- ✓ Effort should be made to define a standardized approach to deal with external events which could be based on the IAEA proposals. The detailed analysis described in these proposals relies on probabilistic evaluations and, to be comprehensive, it would also require to categorize the external events and the related design conditions as it is done in the frame of internal events. The objective would be to facilitate the choice of design quality level[14].

Finally, concerning the probabilistic safety assessment taken as a complement to the deterministic approach, it must be reminded that all external hazards have not yet been integrated in the studies of external hazards (level 1 PSA). It is especially the case for human induced hazards. The recommendations to

[14] At present, IRSN does not apply this approach for human induced hazards because, safety requirements associated with external hazards (building stability, cooling capability, etc.) are not always associated with a design quality level as it is the case for the design of safety equipments (rules of ASME or RCC-M criteria in coherence with event category).

be issued by the European project ASAMPSA2 for Gen. IV reactors (2008-2011) will certainly be of interest on this subject.

Question n°2

As far as we can conclude from the example of the Chemelot site, the safety report of the industrial site provides sufficient basis for establishing the input data (envelope scenarios and safety distances) to be used within the approach describe above on the figure 12. Nevertheless, the approach remains different between nuclear industry (in Europe) and other industries, as the latter uses deterministic and probabilistic approach.

5.3.2 Application to the test cases

The hazards identified for the test cases are summarized in the table below and compared with the external hazards induced by human activity listed in the IAEA document (ref.15). They are mainly of the same nature for the two test cases.

Table 1: List of the industrial risks relevant of the test cases

Industrial risks (IAEA ref.14)	Chemelot	Comments	HTE	Comments
Toxic	Yes	Indirect impact on the workers and rescue teams	No	
Corrosive	Yes	Maybe delayed impact on	Yes	Oxygen
Explosion / pressure wave	Yes		Yes	Safety functions and workers
Explosion / Heat wave	Yes		Yes	Safety functions and workers Short effect distance in general
Fire	Yes	Relatively short effect distance	Yes	Risk of internal fire caused by oxygen concentration rise up
Missiles	? (probable)	Scenario not quantified – several turbines on the site – may be enveloped by an aircraft crash	?	Scenario not quantified - may be enveloped by an aircraft crash
Hazards due to mobile sources and pipelines	Yes		Yes	The HTE plant is connected to a pipeline grid for hydrogen delivering to the end users
Electromagnetic interference	?	This risk is partially addressed by the protection against indirect effects of lightning	?	idem
Ground collapse	Yes	In case of excavation and/or explosion. Requirements are imposed in the construction permit procedures of the Chemelot site.	?	Risk of ground collapse in case of explosion needs to be explored but seam relatively low.
Flooding (related to industrial activities)	Yes	Risk of chemical product and water spillage	No	

Chemelot site

For Chemelot, the distance between the nuclear heat supply system and the industrial site would probably not stress the assessment. According to the data given in the paragraph 5.2.2, a separation distance around 2800 m between the nuclear plant and the chemical site would be acceptable as a “safety distance”. This distance corresponds to a storage tank explosion with an estimated frequency around 10^{-7}/y. Such a scenario would lead to pressure loadings compatible with standard design for LWRs. As an example, it is usually required that the containment buildings resist to a pressure wave modeled by a load curve and a peek pressure of 100 hPa. It should be noted also that events which frequencies are estimated below 10^{-7}/y are usually not considered in the safety assessment of the nuclear plant, so the scenarios presented in the paragraph 5.2.2 give limit distances. Nevertheless risks of explosion associated with transports in the vicinity of the nuclear site should be carefully assessed. Given this safety distance, the demonstration should be completed by taking into account standard thermal and mechanical loads (shock waves) for the building design and the safety equipments which may be impacted. Moreover, every modification or construction permits granted on the industrial site should lead to a reassessment of the safety case (commonly done for existing plants).

For assessment completeness, it may be required that the industrial applicant establish scenarios corresponding to the fourth category of the nuclear plant design conditions (frequency above 10^{-6}/y). These scenarios may complete or replace the standard design loads (like the standard shock wave or thermal loads). Moreover, combination of external hazards should also be explored on a probabilistic or deterministic judgment basis. As an example, an explosion could be followed by the release of a toxic cloud. Such a situation could jeopardize the management of the emergency procedures in the nuclear plant.

As regards the toxic clouds, even if this scenario does not lead to the envelope consequences for the Chemelot site, protection of the workers should be ensured in any case. The objective should be to preserve the workers capability to operate a safe shut down of the nuclear plant. At present, this risk is evaluated for PWRs on a probabilistic approach (with some envelope assumptions) according to the French fundamental safety rules. It is to be noted that provisions against toxic hazards is applied into the Chemelot site at least for the control rooms.

It seems obvious that an explosion or a turbine blade failure could generate missiles coming from the industrial site. This scenario has not been studied yet, but would necessarily be assessed for the coupled system. In the French fundamental safety rules, it is stated that the design of the buildings against aircraft crashes is deemed sufficient to cope with missiles hazard.

Hydrogen production plant

For the coupling with an HTE process the safety assessment of the nuclear plant should rely on:

- examination of the acceptability of the transients induced by the HTE process thermal load disturbance,
- checking of the adequate protection of the safety equipments against hydrogen explosion,
- checking of the protection against corrosion and oxidation risks in case of oxygen release.

The first point should be based on the computational fluid dynamics and verification of the creep-fatigue design criteria for the supporting structures and the barriers (the transients will induced temperatures gradients and core power variations). Effects on the fuel shall be investigated too, but considering fuel behaviour, power variations are likely to be enveloped by incidents like inadvertent control rod withdrawal.

The risk of hydrogen explosion will certainly be assessed by verifying that a minimum distance exists between the coupled systems. As we forecast for the test case n°2, this distance should be sufficiently small to ensure technically and economically feasible heat transportation. In this case, the hydrogen is removed continuously from the plant by means of a pipeline (no significant hydrogen storage on site). The hydrogen storage capacity corresponds to the needs during start up of the plant. Nevertheless, the third barrier design will need to include a standard deterministic load function so the protection again other external hazards will be achieved (especially mobile sources).

No specific protection of the workers of the nuclear site shall be required due to the HTE process because, for example, potential heat waves are smoothed by the distance. In fact, HTE process does not involve significant quantities of hazardous substances. As regards the safety classified equipment; tests could be carried out to support the safety demonstration by verifying the sensitivity to oxygen enriched atmospheres.

5.4 Potential internal impact on the nuclear plant

Internal impact on the nuclear plant lies mainly in thermal and mechanical transients induced by the coupled process. The transients induced by the industrial processes may damage the barriers which are the tubing of the secondary and tertiary heat exchanger or steam generator (if possible tertiary component should not be classified as a barrier and protection should be performed by the isolating valves). So the internal impact of the process may affect the confinement function. For the safety assessment, the transients imposed by the process shall be categorized as regards their frequency and intensity and then used as design conditions, according to the procedure described in the standard design codes.

Concerning the thermal effects, a process failure should not trigger a reactor shutdown[15]. To meet this objective, a maximum variation range may be defined for the helium at the inlet of the core. This range will also depend on admissible temperature criteria at the reactor inlet (variation range).

Mechanical loadings on the barriers could also happen in case of leak of the tertiary heat transfer equipment, explosion, etc. In this case, relief valves and surge tanks may ensure the intermediate heat exchanger protection.

Finally, this approach should avoid a reduction of the safety margins. As an example, the rate of temperatures variation in the primary helium must be limited to preserve, among others, the integrity of the graphite structures.

As regards to the test cases, the potential impact on the second barrier should be strongly reduced by the intermediate circuit that will also damp the effects of the transients. Moreover, specific damping systems, like boilers downstream to the heat exchanger, may contribute to cope with the process transients.

Practically, it appears that no major issue would to jeopardize the safety assessment of the coupled system when considering the process transients.

6 POTENTIAL IMPACT OF THE NUCLEAR PLANT ON THE INDUSTRIAL SITE

6.1 General Safety Features of High Temperature Gas-Cooled Reactors

The GenIV reactor concept VHTR is characterized by a helium-cooled, graphite-moderated, graphite-reflected, thermal neutron spectrum reactor core with a reference thermal power production of 400-600 MW. The reactor core type of the VHTR can be a prismatic block core or a pebble-bed core. The low power density and large heat capacity of the graphitic core, the absence of coolant phase changes, and the prompt negative temperature coefficient represent inherent safety advantages. The use of refractory core materials combined with helium coolant allows high coolant temperatures up to 950°C or even higher which are ideally suited for a wide spectrum of high temperature process heat applications. Cogeneration of heat and power makes the VHTR an attractive heat source for large industrial complexes.

The safety basis for all VHTRs is to design the reactor to achieve a high degree of passive safety to avoid release of fission products under all conditions of normal operation and accidents of the design extension conditions. In particular, the tall and slim core geometry allows for the inherent temperature limitation of the fuel temperature due to a self-acting decay heat removal such that the maximum fuel temperature under Anticipated Transients Without Scram (ATWS) and Loss-of-Forced-Convection (LOFC) accident conditions remains below 1600°C, a level that can be sustained by the fuel without further damage (yet to be demonstrated for new generation plants according to the targeted burn-up). Despite still open questions, the VHTR system is a nuclear reactor concept with an easily understood safety basis that permits substantially reduced emergency planning requirements and improved siting flexibility compared to other

[15] Note that a reactor shutdown is considered as a safety related event subject to declaration to the safety Authorities.

nuclear technologies. The passive safety aspect of the design should make the VHTR less vulnerable to the risk of "radiological sabotage" through malevolent acts.

The safety concept for HTRs will be based upon those usually defined for Nuclear Power Plants like the IAEA-Standards. The potential risk induced by a HTR can be less than that of Light Water Reactors because core melt is highly improbable. Furthermore the risk of radioactive releases especially for dust can be reduced by measures like:

- Gas Purification
- Leak before Break and Break Preclusion Concept
- Inertization of the confinement or containment to avoid air ingression into the reactor circuit
- No steam / water in the intermediate circuit to exclude water ingression into the reactor
- Improved leak tightness by a metallic liner
- Controlled gas releases out of the confinement or containment through filters

In comparison to the nuclear plant dedicated only to electricity generation, several modifications are necessary for the process heat variant:

- reduced power density to compensate for the higher core outlet temperature level;
- reduced system pressure as compromise between a high pressure desired for its favorable effect on operating and accident conditions of the nuclear reactor and a low pressure desired for chemical process reasons in the secondary and tertiary circuit;
- two fuel zones in the pebble-bed to minimize the occurrence of hot/cold gas strains in the core to achieve a radial temperature profile as uniform as possible;
- ceramic (graphite) liner to replace the metallic liner because of the higher temperatures.

Helium is under normal circumstances a highly pure coolant with impurities in the ppm range. These impurities, however, typically O_2, H_2O, CO_2, N_2, also H_2 and CH_4, may lead to corrosive interactions with graphitic and metallic materials in the primary circuit, which will enhance at higher temperatures.

Any combination of an HTGR with chemical processes will most probably need an intermediate heat exchanger (IHX) for a decoupling between the primary circuit and the heat utilization system. It represents the pressure boundary of the primary circuit, which sets stringent requirements to the boundary integrity. The clear physical separation allows for the heat application facility to be conventionally designed, and repair works to be conducted under non-nuclear conditions. It also gives flexibility in the design and allows also the use of fluids others than helium in the secondary circuit. The IHX has to work under the harsh conditions of high temperatures and temperature differences, and also of high pressures and pressure differences.

A part of the safety concept is the employment of a non-integrated arrangement with separate vessels for the generation and the conversion of heat, an underground construction of primary circuit and shutdown and decay heat removal systems to ensure protection against atmospheric explosions of gas clouds, fire, or aircraft crash, and inertisation of the containment/ confinement. Coolant leakage from the primary circuit into the secondary circuit is typically prevented by a pressure gradient with the higher pressure on the secondary side to keep radioactivity in the primary system. A further coolant leakage beyond the containment will be limited by closing quick-response shutoff valves (safety graded equipments).

6.2 Potential safety impacts on the chemical plant

According to the general licensing requirements for nuclear installations all negative impacts from the nuclear installation during normal operation, anticipated operational occurrences (AOO) and design basis accidents (including Design Basis Conditions – BDC - and Design Extension Conditions - DEC - as defined in the EUR) must be restricted to the nuclear site. Taking this requirement as a basis there should be no potential hazards for the public and therefore for any industrial plant in the vicinity of the nuclear facility originating from the nuclear power plant. This applies as long as the facilities are completely independent and not coupled in any way.

This is, however, not the case if the industrial plant is supplied with heat in form of steam or any hot fluid from the nuclear plant. Potential hazards for the chemical plant originating from the nuclear plant then are the transfer of tritium into the product and thermal turbulences induced into the chemical system by the inadvertent loss of the heat source.

6.2.1 Tritium contamination

Tritium balancing is an important aspect in planning, operating, and decommissioning of any nuclear plant. Although tritium is deemed to be of less radio-toxicity than other fission products or activated elements, it represents the main source of activity released by nuclear power plants and reprocessing plants during their normal operation. Fortunately, the doses associated to the tritium releases are usually low because of its specific behavior in the environment and the relatively low energy of its beta radioactive emission. Nevertheless, recent reports (ref. 16) support the idea of a higher radiological impact than considered in current standards.

Main sources for tritium in the reactor core are ternary fission (~50% of total tritium production), neutron captures on boron (control rods) and impurities (lithium, boron) in graphite (~35%), and activation of the 3He fraction in the helium coolant (~15%). The principal forms, in which tritium exists in the primary circuit, are HT, CH_3T and HTO. Its extreme mobility requires effective retention mechanisms in a process heat plant.

As most impurities in the helium coolant, tritium is effectively removed from the circuit by the gas purification system. The degree of removal depends on how much of the coolant is bypassed through the purification system. An important sink for tritium in the HTGR are the graphite structures, where the tritium is bound by adsorption, chemisorption, and eventually diffusion into the inside of the graphite. But the tritium might later, with decreasing tritium production, be desorbed again.
For HTGRs, most important is the capability of tritium to pass through metallic walls penetrating the water/steam and/or product gas cycle. The permeation process strongly depends on material and surface effects as well as on temperature and pressure conditions. Gas composition at the heat exchanger walls is such that the metals typically react with the in situ formation of an oxide layer which was found to reduce significantly the tritium permeation rates. Drawback of protective oxide layers is their poor stability against temperature cycling. The controlled addition of water vapor may enhance the formation or repair of oxide layers. In this case, graphite corrosion may be carefully monitored. Pre-oxidation of the metals could also be an option.

For the coupled system, a fraction of the tritium transferred via the tertiary fluid would probably be released from the industrial processes and another part would be diluted in the industrial products. As far as the fluid sent to the process is not directly incorporated in the chemical process but used as a heat carrier (process heating or turbine driving), the tritium contamination of the industrial products may be not significant (i.e. below the sensitivity limit of the measurement). Nevertheless, even at a very low level, consumer products or industrial by-products may be contaminated by a measurable quantity of tritium. As an example, if the steam of the tertiary circuit is directly used to produce hydrogen which is a raw material for resins or synthetic fuel, these products may content tritium. In such a case, the industrial applicant shall perform an impact study which will be mandatory in the authorization procedure.

Possibility to authorize the production of tritium contaminated fluid or by-products

At present, a number of consumer products contain radio-nuclides which are deliberately added in order to profit by their chemical or physical properties (radioluminous products, gas detectors, etc.). According to NRC data, they account for 3% of the annual radiation dose incurred by an individual (ref.17).

As regards the European directive 96/29/EURATOM, practices are not subject to clearance if they involve "*material contaminated with radioactive substances resulting from authorized releases which competent authorities have declared not to be subject to further control*". The exemption criteria are listed in article 3, title III of the directive; for tritium they are expressed as follows:

- the fluid radioactivity, if exclusively due to tritium, must not exceed 1 GBq or,
- the tritium concentration in the fluid must stay below 1 MBq/g.

Obviously, these criteria will have to be met for the coupling fluid and the by-products (steam, hydrogen, chemicals, fuel, etc.). Nevertheless, the criteria specified in the European directive are recommendations. Thus, member states may impose different limits, as it is illustrated in the two examples below.

According to the German Preventive Radiation Protection Ordinance of 1989, neither licensing nor announcement is required for the use of fossil products refined by nuclear process heat, whose tritium content does not exceed 5 Bq/g. This special case is the exception from the rule, where for any fabricated product the specific radioactivity limit is lower by a factor of 10 compared to the above figure, i.e., 500 mBq/g. The background for this special rule resulting from discussions in the context of the PNP project (Prototype Nuclear Process heat) is the fact that, depending on the origin of the feed natural gas, the natural activity content would often have reached already the free limits given by the law.

In the new German Preventive Radiation Protection Ordinance issued in 2001 the free limit for tritium is taken from the European directive. Similarly, French regulation for health protection against ionizing radiations (decree n°2002-460 of April 2002) is in compliance with the Council directive 96/29 EURATOM. In particular the same limits are adopted for exemption. For the test cases, it is obvious that this criterion should be fulfilled at least for the coupling fluid delivered to the processes. Nevertheless, according to the French regulation, any deliberate addition of radioactive elements in goods, raw materials or by-products is forbidden. It is also forbidden to use materials potentially contaminated for the fabrication of any goods or construction materials. Derogations can be granted (IRSN has yet assessed several demands), but if alternative applications which do not involve radioactive materials are feasible, the request may be rejected.

Beside the compliance with mandatory limits (if they exist), the principles for assessment are the followings:

- The industrial practice shall not expose citizens to excessive risks, and potential risks should be favourably balanced by the benefits brought to the society by the use of nuclear energy.
- If the coupling fluid contained in non nuclear grade circuits is contaminated in normal operation, this contamination shall be kept as low as reasonably achievable (ALARA), well below the limits imposed for public protection.

The first principle derives from the IAEA fundamental safety principles (IAEA safety standards SF-1 – 2006). In the case of hydrogen or steam production, the applicant should demonstrate this benefit, for example on the basis of a global environmental impact comparison with production from fossil fuel (reduction of carbon footprint, etc.). It is obviously a difficult point and it will not be further assessed in this report as any proposal in this field should be discussed with State Authorities.

The ALARA objective should be met by a balanced approach based on the limitation of the tritium production in the core, the efficiency of the barriers and the monitoring of the fluid purity. Provisions should also be made to avoid excessive contamination in case of accidents.

As for any nuclear installation, demonstration of viability of the coupled system will require a health impact study. Thus, the way the initial contamination carried by the coupling fluid will disperse in the consumer products will have to be carefully assessed taking into account the whole life cycle of the products. This implies a close collaboration between the reviewer (TSO, State Authorities) and the manufacturer.

Moreover the risk of tritium build up in the industrial process shall be considered. In terms of criteria, the effective individual dose to the public delivered by industrial activities must be kept far below the limit of 1 mSv/y (dose from artificial irradiation except medical exposure - 1990 ICRP recommendations). This criterion is close to the dose from natural radiation exposure (around 2.4 mSv/y in average – UNSCEAR 2008). In the US, the off-site effective dose induced by the liquid effluents from a nuclear site should not exceed 3% (30 µSv total body) of the limit dose for the maximally exposed individual living in the vicinity (ALARA objective). In France, the limit concentration of tritium in the water sampled downstream of a nuclear site has been set to 140 Bq/l (day averaged value – ref 16)[16]. Other examples are given in annex 3 and a synthesis of the limits adopted worldwide can be found reference 18. In particular, for drinking water, it has been recommended by the World Health Organisation to limit the effective dose incurred by the public below 0.1 mSv/y (10% of the total effective dose), taking into account all radioactive nuclides (natural origin or not). Given these values, it could be reasonable to demonstrate that the potential dose delivered by the tertiary fluid and the by-products would not exceed 3% to 5% of the recommended ICRP limit (1 mSv/y), for the maximum exposed individual.

Application to test cases

In the test cases described here, the radioactivity of the coupling fluid and the by-products will not result from a deliberate addition but from the inevitable tritium contamination. Moreover, these products would be commodities for the petrochemical industry rather than consumer products. This involves a dilution of the tritium which cannot be evaluated in this study.

Concerning the test case n°1, the higher pressure in the secondary circuit and the tritium capture in water would efficiently limit the tritium concentration in the steam of the tertiary circuit. Moreover, steam is mainly used for heating process products or to drive turbines. In this case the tritium migration in the industrial products should be negligible (to be demonstrated), but the potential doses associated with the maintenance of the equipments shall be quantified.

Concerning the test case n°2, tritium will permeate through the fuel cells and contaminate the hydrogen and the oxygen. Several studies were recently published concerning tritium contamination in the hydrogen. As an example, calculations performed in the ref.9 (see the layout on figure 10) for the NGNP project showed a tritium concentration of 2.67×10^{-3} Bq/ml (STP) in the hydrogen and 4.85 Bq/ml (STP) in the tertiary helium[17]. This result was obtained with a 600 MW power plant, a primary outlet temperature of 900°C at a pressure of 7 Mpa and a tritium release rate from the particles of 100%. It must be noted that these evaluations are highly dependant on assumptions concerning tritium retention in particles and graphite. Indeed, SiC coated particles effectively retain tritium and a release rate of 1% in normal operation has been considered realistic by experts. Although the above calculations do not allow to draw conclusions about the acceptability of the contamination, they show encouraging results as many technical solutions exist to reduce the tritium production and permeation.

As a conclusion, the European Directive contains recommendations which are not directly applicable to a coupled system insofar as the contaminated products addressed in the Directive are not supposed to be marketed (except for medical applications). The same goes for regulatory limits on radioactive discharge because by-products or consumer products coming from the coupled system can not be considered as effluents. It is reasonable to envisage that the delivering of a low contaminated fluid to the industry will require specific clearance from the Safety Authorities, even if the tritium concentration is below the values given in the European Directive. Thus, it can be assumed that both nuclear and non nuclear operators will have to seek for clearance respectively to deliver and to use the tertiary fluid. The basis for these applications will be the impact study on the workers and the public. Practically, in case of measurable

[16] Such a concentration would yield an effective dose of 2 µSv/year for an individual who drinks 2 l/day of contaminated water.

[17] Note that 1000 m^3 of tertiary helium would contain 5.5 GBq of tritium, which is beyond the exemption limit given in the European Directive (1 GBq).

tritium contamination of by-products and consumer products, concentration limits could be proposed relying on ICRP (International Commission on Radiological Protection) and WHO (World Health Organisation) guidelines.

6.2.2 Loss of heat source

Besides the specific nuclear hazards caused by tritium the unavailability of the nuclear plant must be taken into account. Depending on the nature of the industrial process a breakdown of the energy supply can cause more or less severe hazards for the industrial plant up to a destruction of the process equipment of the plant. Where a continuous process heat supply is necessary the nuclear reactor must be back upped by an adequate system. The back up system can be a conventional boiler as well as an additional nuclear module. The layout proposed for the test case n°1 is consistent with this requirement. Already existing boilers are run together with the nuclear plant.

The main feature of the back up system is to provide energy to the industrial process in such a manner that the process is not disturbed or interrupted when the basic energy supplying system (reactor) trips.

This requirement is however not specific for nuclear systems but applies to all other heat supply systems as well.

6.2.3 Emergency Planning

Another essential issue which has to be kept in mind when discussing the feasibility of a combined nuclear/non nuclear installation is the fact that a common emergency planning for the whole site has to be established.

This combination of the emergency planning for two completely different installations with completely different safety philosophies is a great challenge. On the one hand the chemical plant, operated in accordance with national and international safety standards for chemical plants, and on the other hand nuclear power plant, licensed and operated under nuclear regulatory requirements.

In more detail this means that two regulatory bodies, each of them with its specific responsibility and well established regulatory approach, will have to cooperate in the emergency planning for the combined site in such way that the regulatory requirements for both types of facilities are fulfilled.

This goal can only be reached by adopting the higher safety requirements in all cases where under normal conditions different safety levels apply. In total this should lead to enhanced safety requirements for the non nuclear part.

7 FEASIBILITY OF THE SAFETY ASSESSMENT OF THE COUPLED SYSTEM: CONCLUSIONS AND RECOMMENDATIONS

The tentative safety requirements and criteria sketch out in this document are summarized in the table hereafter. Note that the specific aspects related to modular design are not addressed.

Although it would be possible to find decoupling criteria between the HTR and the industrial processes, some specific requirements and acceptance criteria shall be established for the safety assessment of the coupled system (table 2).

The HTR, the plant is to be designed to have a very low potential impact on the neighborhood in any situation (according to designers). The radiological impact of accidental situations, like the depressurization of the primary circuit, is claimed to be lower than the limits involving emergency evacuation of the public. This would be clearly an advantage because, in case of accident, it is preferable to provide enough delay so that operators can put the industrial installation in a safe condition. As for sheltering procedures, they are usually established for industrial sites with chemical hazards. So, apart from the risk of coupling fluid contamination, the criteria for the nuclear plant impact shall be linked to the plant availability and the consequences of reactor shutdown. The nuclear plant safe shutdown, shall always override the operation of the industrial process. So, this will require the assessment of the provisions taken in order to avoid the degradation of the industrial equipments with a potential back effect on the nuclear plant.

Concerning the potential contamination of the coupling fluid and the by-products of the industrial process, the effective dose brought to workers or general public by the use of the products of the coupled system should not exceed a fraction of the limit dose of 1 mSv/y.

As regards the external hazards, many nuclear sites in Europe are already located in an industrial environment. So approaches and technical solutions have already been implemented to deal with the specific hazards induced by the industrial environment, but rather on a case by case assessment. Moreover, there is no international standard to deal with external hazards and it must be noted that the IAEA proposals imply several innovative modifications as compared to current approaches. In the test case of the Chemelot site, which is located in a densely populated area, provisions have been taken to reduce drastically the frequency of accidents with harmful consequences in the vicinity. These results in relatively short safety distances and low associated consequence frequency which are closed to the exclusion limit usually considered for nuclear plants (10^{-7}/y for a given type of external hazard). Nevertheless, standard external loadings may be applied to the nuclear plant in order to prevent, at least, the evolutions of the industrial environment. Three main areas for improvement of technical and methodological aspects of the assessment of external hazards have been highlighted in this study: the need for a more detailed analysis of the consequences of the external hazards (probabilities of effects on safety functions, categorization of induced accidental loadings, etc.), the treatment of potential common modes failures and the evaluation of the risk induced by the projectiles.

Moreover, it should be noted that all the issues identified are not technical ones. In particular, concerning tritium contamination, a discussion on the free limits with the Safety Authorities should be struck up prior to further design of coupled systems. Similarly, a difficulty lies in the integration of the emergency planning within a document relevant for the whole site.

As a conclusion, this study has exhibited favourable elements for the licensing of a coupled system including an HTR. In particular, the approach based on the safety distance may be compatible with the technical feasibility of medium length connections to carry the coupling fluid between the two sites. Moreover, the risk assessment presented in the safety report of the industrial sites of Seveso class gives quantified scenarios (frequencies and associated loads) adaptable to the assessment of the external hazards of the nuclear plant safety report. However, an important effort has to be made in relation with the safety authorities to establish a relevant regulatory framework and a set of acceptance criteria adapted to coupled systems. Finally, the key aspects for the safety assessment were found to be the followings:

- completeness of the safety documents of the industrial sites and compatibility with the the safety demonstration of the nuclear plant;
- establishment of a standard framework for assessment of the risks induced by the external hazards including missiles;
- possibility to determine safety distances;
- difficulties for emergency planning which shall encompass the industrial and nuclear sites;

establishment of a regulatory framework to address the issue of low level tritium contamination of the coupling fluid, and impact assessment.

Table 2: Proposals for requirements and criteria

Safety functions	Requirements	Criteria	Comments
Physical and radiological protection	Two sites with physical separation	Security	
	Protection measures against toxic release	Operational capacity of the nuclear plant workers	Risk of toxic release from the industrial site
	Limited (or nonexistent) evacuation area around the nuclear site	Protective action limits for the public	At least, compatible with the safety distance resulting from industrial risk evaluations
	Integrated emergency plan for nuclear and industrial facility		
Confinement	Adequate number of barriers between radioactive materials and process product	Criteria for integrity insurance (design, monitoring, inspection)	Nuclear design codes and standards
	Prevention, mitigation and monitoring of the radioactive contamination of primary and secondary fluids (including tritium)	To be defined	
	Contamination monitoring of the secondary/tertiary fluid	Limit concentration or decision threshold	Limits on tritium concentration should be defined based on impact study
	Resistance to external hazards	Frequency limits Design criteria	Standard loads should be applied even in case of limited impact of the industrial facility
	Process transients do not damage the barriers	Design criteria	Coupling does not reduce the safety margins
Residual heat removal	Resistance to external hazards	Frequency limits Design criteria	Indirect effects on safety graded systems
Reactivity control	Resistance to external hazards	Design criteria	Direct and indirect effects on safety equipments
	Low probability of large water ingress or chemical product ingress	Protection measures	Defense in depth approach
	Core not affected by the transients induced by the process	Limitation of temperature transients	Graphite and internal structures integrity
Others (assimilated to safety)	Isolation between secondary and tertiary circuit (valves, etc.)	Reliability of the isolation function (considering safety and quality classifications, maintenance, monitoring)	To prevent process fluids ingress from the tertiary circuit into the secondary circuit

- **Acknowledgments**

Beside the help of all the members of the work package n°2, the authors want to acknowledge a very useful support by Mr Anton Baaten from Baaten Energy Consulting. Helpful cooperation and review were also provided by Mr John Jolicoeur and Mr Sudhamay Basu from the US Nuclear regulatory Commission. We also wish to thank the IRSN's experts for their advice.

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ANNEX 1: Examples of effects thresholds taken from European regulations

As an indication, the following thresholds can be considered in a safety report in France (ref. 11):

- For thermal effects :
  - Effects on human health (prolonged exposure) :
    - ✓ 3 kW/m²: irreversible effects (area of hazards to human life)
    - ✓ 5 kW/m² : lethal effects (area of serious hazards to human life)
    - ✓ 8 kW/m²: significant lethal effects (area of very serious hazards to human life)
  - Effects on structures :
    - ✓ 5 kW/m² : significant destruction of windows
    - ✓ 8 kW/m²: serious damage to the structures
    - ✓ 16 kW/m²: very serious damage to the structures (except concrete structures)
    - ✓ 20 kW/m² : very serious damage to the concrete structures
- For pressure effects:
  - Effects on human health :
    - ✓ 20 hPa : indirect effects by broken windows
    - ✓ 50 hPa: irreversible effects (area of hazards to human life)
    - ✓ 140 hPa: lethal effects (area of serious hazards to human life)
    - ✓ 200 hPa: significant lethal effects (area of very serious hazards to human life)
  - Effects on structures :
    - ✓ 20 hPa: significant destruction of windows
    - ✓ 50 hPa: slight damage on structures
    - ✓ 140 hPa: serious damage on structures
    - ✓ 200 hPa: threshold for the study of domino effects
    - ✓ 300 hPa: very serious damage on structures
- For toxic effects :
  - ✓ Irreversible effects : area of hazards to human life
  - ✓ 1% lethality : lethal effects corresponding to CL* 1%
  - ✓ 5% lethality : lethal effects corresponding to CL* 5%

*CL: lethal concentration

In the Netherlands, a Quantitative Risk Analysis (QRA) is performed (ref. 12): the Societal Risk and the Individual Risk are also evaluated. The probability of death and the fraction of people suffering death should be calculated up to the level of 1% lethality. For a same effect, different thresholds can be applied, whether populations are indoors or outdoors.

For toxic effects, a maximum exposure time of 60 minutes is considered and a level of 1% lethality is retained.

For pressure effects, the following references are assumed:

- ✓ 30 hPa : critical overpressure causing windows to break

- 100 hPa: 10% of the houses severely damaged and a probability of death indoors equal to 0.025 has to be considered
- 300 hPa: very serious damage on structures and total lethality

For thermal effects the following criteria are considered:

- 4 kW/m^2 : glass breaking
- Fire (jet fire…) : 9,5 – 10 kW/m^2 : flame envelop (total lethality)
- 35 kW/m^2: threshold for the building ignition

ANNEX 2: Jet fire in case of a hydrogen leak - Phast model

Process Conditions:

Temperature	25°C
Pressure	5 MPa
Inventory	1000 m^3

Considering a mass flow rate of 1.6 kg/s under the above process conditions, a pipe diameter of 6” corresponds to an average speed of 100 m/s for the hydrogen. This speed is in the common range used in gas pipe design. Higher speed is also investigated, considering a 3” pipe.

Scenario:
Case A: Full bore rupture of a pipe with internal diameter equal to 3”

- case A1 : using the *leak model* ;
- case A2 : using the *line rupture model* with a pipe length equal to 1 meter

Case B: Full bore rupture of a pipe with internal diameter equal to 6”

- case B1 : using the *leak model* ;
- case B2 : using the *line rupture model* with a pipe length equal to 1 meter

Results:

For case A1 and A2 scenario (pipe internal diameter equal to 3”), the flame length is estimated to be 50 meters. The maximal potential effects distances would be the following (standard limits on heat flux – see annex I):

- 3 kW/m²: ≈ 95 m;
- 5 kW/m²: ≈ 85 m;
- 8 kW/m²: ≈ 75 m.

For case B1 and B2 scenario (pipe internal diameter equal to 6”), the length of the flame is estimated as equal to 88 m. The maximal potential effects distances would be the following:

- 3 kW/m²: ≈ 180 m;
- 5 kW/m²: ≈ 155 m;
- 8 kW/m²: ≈ 135 m.

ANNEX 3: Data on tritium concentrations and limits

	Tritium concentration	Comments
Reference measurements		
Sea water	0.1 Bq/l	Sea surface at the Equator
The Channel	10 Bq/l(1)	Sea surface
Rain water	1 Bq/l - 4 Bq/l(1)	
Air	0.01 Bq/l – 0.04 Bq/m^3(1)	1 Bq/l in rain water corresponds to 0.01 Bq/l in air
Clouds	≈ 0.1 to 10 Bq/l(1)	In condensed steam
Drinking water	1.2 Bq/l – 11.16 Bq/l(1)	
Standards		
Drinking Water	740 Bq/l	US EPA standard 1976 (tritium only) Corresponding dose for the Public: 0.04 mSv/y
Water	100 Bq/l	Europe Alert value used as a sign of radioactive contamination
Liquid/gaseous effluents	3,7 Bq/l / 3700 Bq/l	US regulation 10 CFR part 20
Drinking Water	7600 Bq/l	WHO limit for tritium only corresponding to 0.1 mSv/y
Measured or estimated values for intallations		
SFR Phénix	40 Bq/l	Measurement Tertiary circuit (water/steam) – Safety report
NGNP project – Hydrogen product	2.67 Bq/l	Estimated value with “conservative” assumptions Ref.: INL WSRC-STI-2008-0163 - High temperature hydrogen production with intermediate gas circuit

(1): Data from measurements done by the IRSN in Western Europe (ref. 16).