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D3.21: Orientations on a licensing process of a nuclear co-generation system – elements for the roadmap set-up

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Summary

In the framework of the FP7 NC2I-R project the Working Package 3 (WP3 – "Safety and Licensing") covers safety and licensing issues for existing and future nuclear plants with co-generation capability. The main objective of WP3 is to advise and support the definition of the general specifications of the demonstrator program in the field of safety. While the licensing of a demonstration nuclear reactor will have to be carried out based on the national regulations in force in the host country, WP3 aims at seeking Authorities' acceptance at a wider geographic scale under the premise that a national licensing process should be as consistent as possible with a European consensus in terms of safety objectives and criteria. For this purpose, this report D3.21 firstly provides some licensing guidelines based on the partners' feedback to the licensing of past projects. This task has also been performed in collaboration with the Polish, Lithuanian and Hungarian representatives which are exploring by now the potential development of co-generation systems. With the independend contributions by these partners, key licensing issues, regarding their own state regulations, have been identified. Specifically to the co-generation systems, the risks induced on the nuclear plant by the industrial processes and those induced on the processes by the nuclear plant have been considered. Finally, as the result of this work, the partners have elaborated a tentative roadmap and content of the licensing process applicable to a demonstrator.

Approval

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1 Introduction

The task 3.2 of WP3 aims at establishing preliminary guidelines to support the licensing of a prototype co-generation plant. Building on experience gained in past projects, the up to date recommendations for the safety assessment and licensing of new NPPs will be adapted to the specific needs of a co-generation installation coupled to an industrial process. Finally, as the result of this work, the WP3 partners have elaborated a tentative roadmap and content of the licensing process applicable to a prototype reactor. The organization of this task intended to be highly collaborative as far as each partner should bring in its domestic vision of the licensing requirements and procedures suitable for a co-generation plant. In particular, this work took into account the licensing approaches foreseen in countries which are currently exploring the potential development of co-generation systems like Lithuania, Hungary and Poland.

General note: The following report considers licensing issues associated with the energy co-generation capability of nuclear power plants and in particular for HTRs. Licensing aspects of the specific reactor will only be considered as far as they are correlated to the co-generation applications.

The results of this task as well as of task 3.3 will be used in tasks 4.4 and 4.5 to define the general specifications of the demonstrator programme and a detailed roadmap to deliver a demonstrator specified in this way.

2 Application for NPP co-generation in general

Nuclear heat sources for public services or industrial applications have been considered worthwhile already in the past. Typical fields of application for providing heat or steam at different temperature levels and flow rates for end users are:

- District heating, steam for local electricity production by domestic turbines or to actuate compressors
- Bulk chemical industry: steam at various pressures and temperatures supporting chemical processes at fixed temperatures, in many cases in addition to electricity production, “plug in” market feasible (exchange of fossil based energy infrastructure without much impact on the existing process infrastructure)
- Desalination
- Oil field exploration
- Refining industry

Considering the fossil fuel related environmental consequences (increasing level of greenhouse gas CO_2) the hydrogen/syngas production (feasible at 700°C by applying a sophisticated membrane reforming process) e.g. for transport fuel, advanced steel making processes or Coal-to-liquid (CTL) processes (see “Nuclear Energy for Hydrogen Production” /4/) may provide a low carbon alternative.

These applications are met very well by a HTR co-generation or process heat application because it offers unique features as:

- high temperature output up to 750 °C (higher temperature levels feasible with VHTR technology to meet sophisticated hydrogen production methods as SI (sulfur/iodine) at >800 °C, HyS (hydrated sulfur electrolyze) at >900 °C or HTSE (high temperature steam electrolysis at 950 °C)
- high level of nuclear safety (applying modular HTR concepts)
- high availability of process energy supply by combination of several modular HTRs with one industrial end user
- electricity co-generation feature for process heat application (if needed to use additional capacities of the parallel running HTRs)

Beside the option to support an existing industrial infrastructure by an HTR based nuclear heat source, an HTR could also be an attractive advantage for integrated new industrial installations.

For a detailed discussion of HTR specific industrial co-generation and process heat applications in general see report /1/ and /3/ of the EUROPAIRS project.

3 Experiences from past and recent projects

This chapter summarizes the experiences from nuclear co-generation projects in the past as given in report D3_11 /20/ of this work package 3. We like to point out that the experiences described in this chapter are general experiences with nuclear co-generation dominated by the light water reactor co-generation applications. The applicability of the experience for HTR projects will be discussed in chapter 4.

- Based on the results of the questionnaires no specific questions on co-generation installations are raised by the respective regulators. But additional considerations may be needed for the additional heat exchanger and for back up boilers located close to the NPP (see Stade NPP).
- No licensing specific procedures identified for district heating.
- Co-generation applications provide a decoupling of the co-generation system from the nuclear system by technical provisions to prevent any significant impact on the plant operation.
- No differences observed in the licensing procedure/the upgrade procedure for the NPP depending on the type of co-generation because the authority by the regulator follows the same regulations and the safety of the plant is always in the responsibility of the operators side.
- The project of district heating always involves a strong partnership of NPP plant operator and owner, district heating company and heat grid operator. Usually the nuclear heat source is integrated in a production mix of other sources for energy. For Beznau NPP, the back-up sources are directly connected to the grid and operated as usual for projects build on an existing heat grid. No influences of the licensing procedure are identified correlated to the initiator or consortium of such a project.
- NPP co-generation installations are generally driven by the objective of independence/reduced dependency from fossil fuel supply (e.g. oil crisis in the seventies). But environmental advantages (CO_2 reduction, reduced truck traffic/fuel supply) are welcome and helpful.
- It seems that the heat losses from connection lines of low temperature co-generation systems are not the most important economic issue and can be compensated by pumping power of substations. More important for the economy is the investment for the connecting pipelines. This comprises not only the construction costs but also the investment for the construction sites. In France this would be an issue according to CEA. In conclusion, the economy of a co-generation application is strongly dominated by the distance to the end user.
- Usually the temperature requirements for hot water are in the range of 60 °C to 200 °C which correlates very well to the steam conditions of a PWR turbine. Process heat application usually uses the latent heat of steam condensation (at constant temperature). HTR process heat seems to be adaptable for Coal-to-Liquid (CTL) applications. So NPP based heat/steam meet very well end user requirements and additional installations are not needed.
- Co-generation applications consuming < 6% to 10 % of the total NPP power are usually not considered in the licensing procedure because of the complete decoupling from the nuclear plant operation. Nevertheless specific studies may be requested to identify the risks and demonstrate their correct management, even if no safety case is finally resulting from these studies. For process heat applications consuming a significant fraction of the generated power (e.g. Stade NPP), it is considered in the licensing procedure (then heat grid is considered as a heat sink in normal operation).

The question of significant impact on the business case of the NPP co-generation installation by the back-up systems for heat/steam will be addressed in WP4.

3.1 Lithuania

Like many other countries in Europe, Lithuania is facing challenges in the energy sector on three main dimensions: security of energy supply, competitiveness and sustainability of the energy sector. Such a situation in Lithuania is due to both the historical and political context, and the limited availability of local energy resources. Most energy resources used in Lithuania are imported. Lithuania has a well-developed electricity network, but it is not connected to the European electricity system and therefore electricity can be imported only from a very limited number of countries (Latvia and Belarus).

The biggest electricity generator in Lithuania, i.e. Ignalina NPP, was closed at the end of 2009. Before closure of the Ignalina NPP the electricity production from nuclear fuel was dominant in Lithuania. Share of electricity produced at Ignalina NPP in 2005-2009 made 60-70% of total generated electricity while before

2005 its production was even higher. The nuclear power was replaced by the electricity generated using natural gas and electricity import from neighbouring countries. Starting from 2010 the variety of primary energy used for electricity generation decreased and the energy sector dependence on natural gas from Russia has increased even more. In addition, as was mentioned, there are still no electricity grid connections with the Swedish and Polish electricity systems which will be ready in 2016 (NordBalt) and 2015/20 (LitPol Link). Such situation in Lithuanian electricity sector is causing a complex of serious economic, ecological, and social problems.

In order for Lithuania to become a fully-fledged Member State of the European Union, the Lithuanian energy sector should be entirely integrated into the European energy system. The country must have sufficient local capacity to satisfy the internal energy demand and, with regard to energy related questions, should be able to participate and compete in common EU energy markets. Implementation of these strategic tasks could be facilitated only by close co-operation with other Baltic countries – Estonia, Latvia and Poland. The National Energy Strategy approved in 2012 /25/ sets a number of tasks and major solutions in the fields of electricity, heating, gas, oil, renewable energy sources and improvement of energy efficiency, environmental protection and reduction of greenhouse gas emissions.

The Lithuanian power system was constructed during several decades in the time of the Former Soviet Union and was oriented towards large-scale electricity generation and district heat consumption. In 2014, the installed capacity of the power plants in Lithuania was over 4300 MW. The greatest share of the installed capacity belonged to: the Lithuanian Power Plant (1955 MW), co-generation power plants (900 MW), the Kruonis pumped Storage Plant (900 MW). Lithuania produced 4.4 TWh of electricity, exported 0.7 TWh and imported 7.6 TWh of electricity in 2013. The total demand of the country for electric power was 11 TWh /26/. The total demand of the country for heat was 12 TWh.

3.1.1 Combined production of heat and power

43% of the total generated heat and ~24% of electricity gross consumption were produced in combined heat and power plants in 2013. Heat/electricity production (TWh) amount: 5.2/2.4 in public CHP plants and 0.02/0.38 in auto producer CHP plants /27/.

The first combined heat and power (CHP) plants were constructed in Lithuania in the 50’s of the last century. They have been generating electricity and heat to this day. These CHPs are working on fossil fuel. The biggest CHPs are located near (<10 km) the biggest cities (Vilnius, Kaunas). All district heating supply companies (not only CHPs) supply around 8–9 TWh of heat to heat consumers per year. In urban areas 75% of the residential area, as well as public and commercial buildings, and a part of the industry, is heated by the district heating systems.

Lithuania has excellent conditions for the development of biofuel-fired CHP plants which fired with biofuels could become "green" electricity, and the residual heat can be used for heating of buildings. In Lithuania, unlike in many other countries, district heating sector is developed very well. The country is quite large for biofuels, which can be used without harming the environment and resources. Lithuania is a relatively cold climate country, service sector is dominant, there is almost no heavy industry and the industry sector is likely to decline in the future. These considerations provide a perfect opportunity to implement biofuel co-generation. Such actions are foreseen in National Energy Independence Strategy; new biofuel-fired CHPs are designed and constructed now.

The Lithuanian Power Plant (LPP, Condensation power plant) – a structural department of the company Lietuvos Energijos Gamyba – produces electricity and heat, and provides ancillary services. After the shutdown of Ignalina NPP, LPP became the largest source of electricity generation in Lithuania (fossil fuel). LPP has an installed capacity of 1955 MW, and it generated 1.114 TWh of electricity in 2013. The plant also supplies district heating to the town of Elektrėnai, established in 1962 to support its workers. Additionally a new Combined Cycle Gas Turbine unit 455 MW started its operation in 2012.

Lithuanian small-scale industrial CHP operators usually use the power and heat generated on-site and do not export electricity to the grid.

The stock company Achema is a leading manufacturer of nitrogen fertilizers (2 million tons per year) and chemical products in Lithuania and the Baltics. Achema has two gas-fired co-generation plants No.1 (21 MW) and No. 2 (47 MW). The last new is the only plant of such capacity in Lithuania that fully complies with the EU CHP (co-generation) Directive. Power efficiency of this plant is very high - over 90 percent. Used German Siemens turbine SGT-800 is considered the most effective among the world's industrial turbines.

The basic product manufactured at Joint-stock company Lifosa (located near town Kedainiai) is the nitrogen-phosphorus fertilizer Diammonium Phosphate, the process of which requires phosphoric acid and sulphuric acid, which are also produced at the Company. Lifosa produces heat and electricity on-site (35 MW). Most of the energy produced is consumed in the company's needs, the rest of the energy, if any, is sold. In 2007

Lifosa completed the installation of the Heat Recovery System HRS. This system enables to produce more heat and electricity by more efficiently utilizing the sulphuric acid process heat. The excess heat of technological process is used for district heating. In 2013 about 100 GWh of heat was provided to district heating system (DHS) network. 238 GWh of electricity power was produced at the plant, 50 GWh of which was provided to the country's electricity grid.

ORLEN Lietuva operates the most advanced refinery built in the former Soviet Union and the only one in the Baltic States. The design capacity of the refinery is 15 million tonnes of crude oil per year. Given the existing technologies and current marketing conditions, the efficient refining volume is 8 million tonnes a year. The interlocked Mažeikiai CHP (210 MW) is a natural gas-fired power plant; its primary use is to serve oil refinery ORLEN Lietuva with electricity, steam and heat.

3.1.2 Nuclear co-generation

AtIgnalina NPP (INPP) the first reactor started operating in 1984 and the second one in 1987. The plant was built in Lithuania, near the border with Belorussia. The first unit of the plant is under decommissioning since 1 January 2005 and the second one since 31 December 2009. The reactors of this plant (RBMK type) were among the most powerful in the world: each of them had 1500 MW of installed capacity. In 2008, before the closure, the plant produced about 70% of the electricity necessary for Lithuania. The main part (~85%) of energy used in Ignalina NPP region goes to heating needs, ~13% to electricity and about 3% is used for transportation.

Construction of apartment buildings in Visaginas, the youngest city in the most Eastern corner of the country, was started in 1975 and completed in 1990. Visaginas is a purpose-built satellite town serving the power plant. There live now about 20 000 inhabitants in the territory of almost 9 sq. km now. The maximum population of 30 000 was reached in 2001. There are no other big industrial enterprises in the INPP region. There are around 254 blocks of flats of poor condition in Visaginas. The average heat consumption for heating reaches 183 kWh/m^2 annually and is one of the highest rates in Lithuania /28/. Uncertainties over the future of the town have militated against investment by the inhabitants and the condition of buildings has further deteriorated.

After Ignalina NPP commissioning the main heat source for INPP and Visaginas was the power plant boiler house. While district heating was produced by the power plant, it was supplied at very low cost. The INPP boiler house (including intermediate circuits with heat exchangers, radi. monitoring etc.) was designed to produce hot water (70-120^{o}C) supply for the Visaginas district heating network by tapping steam from either Unit 1 or Unit 2 turbines when in operation. Boiler house was connected to the main building of Unit 1 from the west (**Fig. 1**). Using a special technology, the INPP gathered a part of vapour for the turbine and heated water in a thermal manner. Nominal power output of INPP district heating facilities from one power unit made a total of 230 MWh of thermal energy; total volume was 460 MWh of thermal energy /29/.

Fig. 1: Location of INPP Boiler House (red square)

Fig. 2: INPP Boiler House Heating System

When INPP district heating facility was under repair, there was a standby boiler station for the summer time. This heating facility was built in 1979 for steam, heat and hot water provision of the town construction, INPP and building industry area. 3 hot water boilers were commissioned at the heating plant with the total power of 136 t/h of steam. Steam part was in operation day-and-night for the house loads of the heating plant, hot-water part was put into operation either when the district heating facility of INPP was under repair or in case of outage of both units.

After final shutdown Ignalina NPP itself became the consumer of heat for heating. In these circumstances the standby boiler station was the only heat source for the town and INPP needs. It was specified during INPP Decommissioning Plan development that the existing standby boiler station cannot be the continuous heat source due to insufficient power and outdated equipment. In this connection during INPP Decommissioning Plan development the construction of new heat-only boiler station was foreseen. The contract with the contractor‘s company Elektrim-Megadex S.A. (Poland) concerning the construction of heat-only boiler station was signed in 2003 (project completed in 2005). The capacity of the new heat-only boiler station is 160 MW (7 boilers, each with capacity 26.6 MW). Main fuel of the boiler is gas, standby – liquid fuel (dieseline) /30/. Now Visaginas town uses heat supplied from the gas fired boiler house located 8 km away from the town, thermal energy demand 260 GWh per year.

3.1.3 New NPP

On 18 January 2007 the Seimas of the Republic of Lithuania (Parliament) approved the National Energy Independence Strategy which reaffirmed the long-term commitment of the Republic of Lithuania to develop nuclear energy. The same commitment was repeated in the National Energy Independence Strategy approved in 26 June 2012. One of the strategic goals of the Republic of Lithuania is a new Visaginas nuclear power plant construction and commercial operation start. It is intended, that Visaginas NPP (**Fig. 2**) will supply electricity to the entire region of the Baltic States and Poland. Hitachi Company was selected as the strategic investor in the Visaginas NPP project, offering to provide an ABWR.

Fig. 2: Visaginas NPP sites

The project is being developed by state-controlled energy group Lietuvos energija, a special project company. The Visaginas NPP project is the first regional nuclear energy development project in the Baltic region involving three national energy companies of the Baltic States and the strategic investor Hitachi Ltd. as well as the technology provider via Hitachi-GE Nuclear Energy. The preliminary envisaged project development company’s shareholding composition is as follows: Hitachi (Japan) – 20%, EestiEnergia (Estonia) – 22%, Latvenergo (Latvia) – 20% and Lietuvos energija (Lithuania) taking the remainder at 38%. The Lithuanian share of at least 34% is stated in the national Law on Nuclear Power Plant /31/. The Visaginas NPP project also remains open to Poland’s participation. Visaginas NPP technology is based on the ABWR having a net capacity of 1340 MW. Commercial operation is expected to start between 2020 and 2022 according to preliminary plans.

Up to now many of the Visaginas NPP preparatory works have already been completed: environmental impact assessment (in 2009), full-scale site evaluation in accordance with IAEA requirements and geological, seismic, logistic and other pre-development works completed in 2010. In June 2012, the European Commission, in accordance with the EURATOM Treaty, issued its opinion on the Visaginas NPP project. After careful assessing the Visaginas NPP project, the European Commission has taken the view that the Visaginas NPP project fulfils the objectives of the EURATOM Treaty. The European Commission states that Visaginas NPP contributes to the security of the energy supply in the Baltic region and to the full integration of the Baltic States into the internal European energy market.

In October 2012, however, a nation-wide consultative referendum on a construction of new NPP in Lithuania was organised. 62.6% of Lithuanian citizens voted "No" on a statement “I am in favour of constructing a new nuclear power plant in the Republic of Lithuania”. After that, Lithuanian politicians questioned the legal consequences of the referendum. At first the Lithuanian Government stated that safe nuclear energy development is a necessary integral part of the Lithuanian energy supply mix and that the project might be continued if it is being developed together with Regional Partners (Estonian and Latvian states and their utility companies) and in case it is economically viable. The Strategic Investor submitted proposals on improvement of the financial conditions for the project, which were reassessed by the project’s potential investors, and certain outstanding issues were identified. Currently all interested parties are in the process of discussing possible ways for resolving these outstanding issues. On March 29th, 2014, Lithuanian political parties signed an agreement which commits to complete Visaginas NPP "as soon as possible". In the presence of Ukraine-Russia crisis, a new nuclear power plant is being promoted more strongly as solution for energy independence from Russia.

Co-generation capability for Visaginas town or special studies, were not investigated and considered for new Visaginas NPP.

3.2 Hungary

According to the Energy policy thesis of Hungary 2006-2030 /14/Hungary is poor in primary energy resources. The sometime remarkable and economically exploitable fossil resources just like the natural gas, oil, black and brown coal deposits will be drained in some decades. The only remaining remarkable, in the long run economically exploitable fossil resource is the lignite located in the Mátra Mountain in North-East Hungary (**Fig. 3**). The natural oil and gas fields are under exploitation while the coal mines have been closed in the last two decades due to economic reasons (not competitive exploitation price).

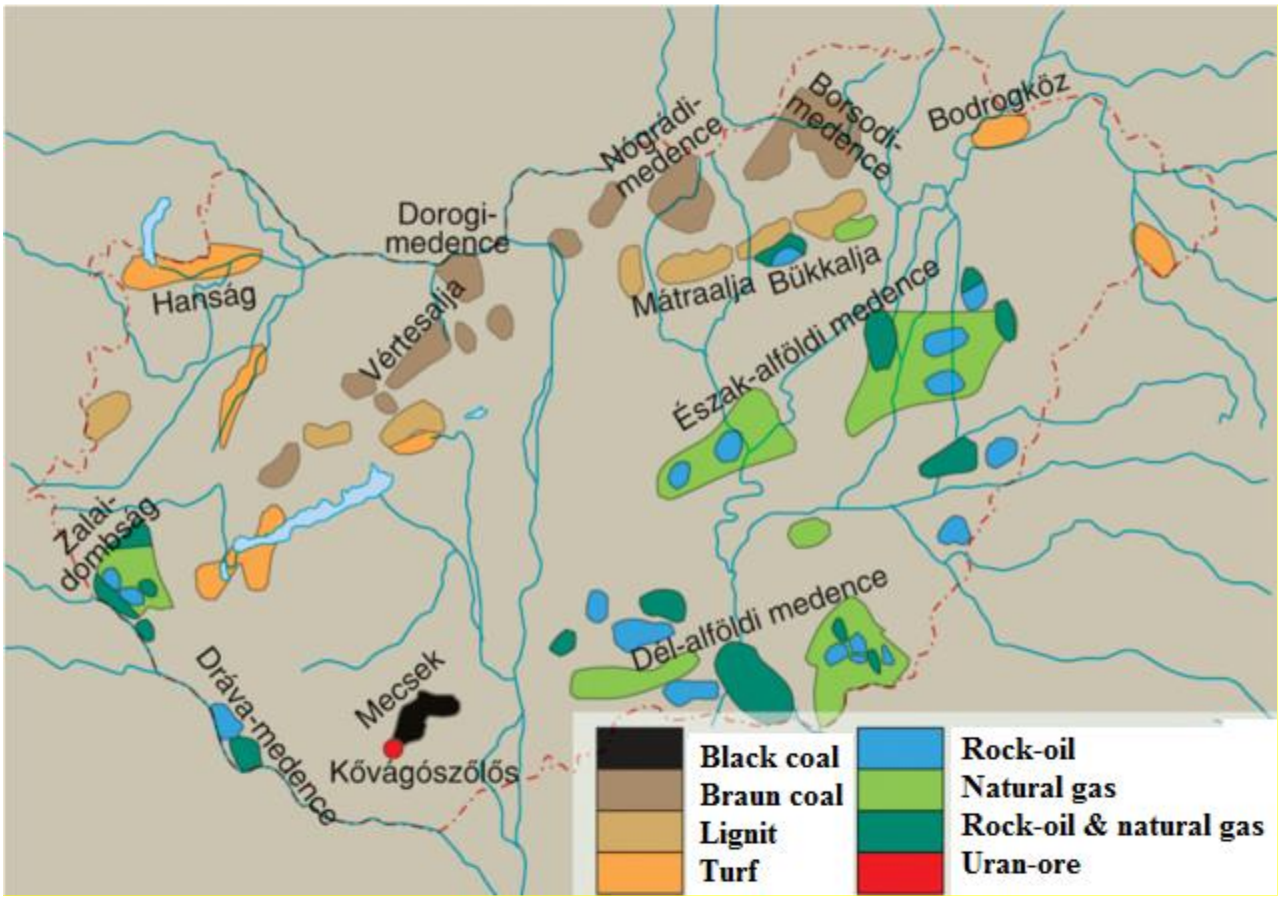

Fig. 3: The deposits of energy sources in Hungary

Hungary had significant uranium deposits in the Mecsek Mountain in South-West, but its mining has been stopped in 1997 due to economic reasons. Some uranium still remained there but its exploitation is economically not reasonable.

According to /14/ the situation is similar in the dimension of renewable energy potential: Hungary is rather poor in renewable energy resources (in the following called renewables), too. But the utilized biomass and wind power capacity seems to reach significant levels /15/: 365.11 MWe and 328.93 MWe respectively (compare these numbers to the peak consumption of the Hungarian electricity grid which is around 6500 MWe). The utilized water and solar power capacity is very limited /15/: 57.73 MWe and 35.02 MWe respectively. There seems to be some potential for further increase in the utilized capacity of the above mentioned four renewables. All other renewables represent only 49.7 MWe utilized capacity. It is worth to mention that Hungary has a 1000 MWe water energy potential on the Hungarian section of the Danube, but a utilization seems unlikely due to reservations of some parts of the society. Furthermore, Hungary has a very good capability in geothermal energy, but this could be utilized for district, greenhouse heating and balneology purpose not for electricity production or process heat supply to industry.

Hungary has to import the majority of its demand of primary energy resources due to the lack of domestic deposits. Furthermore, numerous newly built gas fired power plants are temporarily shut down due to competitiveness reasons in comparison to lower price of imported electricity. This fact raises energy security issues. For example 75% of natural gas import (the total gas consumption represents 44% of the total energy consumption in Hungary) came from the Russian Federation and mainly through only one pipe.

Considering the above mentioned circumstances, it is understandable why Hungary relays on nuclear reactors as the country’s main base load electricity producers. The nuclear fuel can be stored in a relatively small hall in advance for years. Furthermore, once the nuclear fuel has been stored it can be handled as domestic energy source which increases the country’s energy security. Last but not least, the necessary nuclear fuel can be bought from many countries (for e.g.: from France, USA, Canada) not only from the original builder of the reactors, the Russian Federation (formerly the Soviet Union).

The requirement system of Hungarian energy policy contains the energetic supply security, the assurance of necessary conditions for sustainable development, support the domestic economy and economic competitiveness and finally the social responsibility to the citizens in difficulty /14/.

Due to the previously mentioned circumstances, it is stated by the Hungarian policy makers that the future energy industry strategic objectives of Hungary can only be achieved, if besides the energy efficiency improvement, the increasing usage of renewable energy sources and the application of other measures, the nuclear energy further plays a significant role not only for today, but for the future energy mix of Hungary /14/.

3.2.1 The past of nuclear co-generation in Hungary

Hungary has a nuclear power plant near the city of Paks (Paks I NPP) which has four Russian type light water units /16/. These reactors operate in co-generation run. It is a multi-unit site with four VVER-440/213 reactors. All of them capable to provide heat for district heating purpose, but due to the relative low heat demand normally one reactor (Unit-3 or Unit-4) provides heat for the district heating system. The installed thermal capacity was 1375 MWth per unit, and after power uprate it is 1485 MWth. The units of the NPP have gone to operation between 1983 and 1986. These units have recently undergone a power upgrade process and their nominal electrical capacities are 500 MWe each instead of the original 440 /16/. The total nominal electrical capacity of the Paks I NPP is 2000 MWe. Paks I NPP provides district heating to Paks on a relatively low price since 1984-1985. It is the only nuclear district heating application in Hungary since then.

3.2.1.1 Initiator and key players

The Hungarian Government has taken the initiative for Paks NPP in 1966. The NPP has been built between 1969 and 1987. The main parameters and milestones of the installation of the NPP can be seen on **Tab. 1**.

Unit #:	Type:	Net Power [MW]	Gross Power [MW]	Start date of the installation	Date of system synchronization	Start date of operation	Plant decommissioning date
Paks-1	VVER-440/213	470	500	1 August 1974	28 December 1982	10 Aug.1983	31 December 2032
Paks-2	VVER-440/213	443	500	1 August 1974	6 September 1984	14 Nov. 1984	31 December 2034
Paks-3	VVER-440/213	443	500	1 October 1979	28 September 1986	1 Dec. 1986	2017 (life ext. up to end of 2036)
Paks-4	VVER-440/213	473	500	1 October 1979	16 August 1987	1 Nov. 1987	2017 (life ext. up to end of 2037)

Tab. 1: The main parameters of the Paks NPP /16/

Generally, the People's Republic of Hungary and Soviet Union were the key players in this project. Most of the reactor main components have been manufactured in the Soviet Union with some exception. For example: the heat exchangers were made in the Hungarian "Április 4 Gépgyár" (April 4 Machine Factory) but the other main components just like the circulation pumps were manufactured in the Soviet Union.

3.2.1.2 Opposition and public acceptance

Before the governmental decision or during the installation of the NPP, there was no opposition against it, because the NPP was built in the last two decades of Communist Era in Hungary. In that time public opposition (thus, the public acceptance) was out of question. Today, in a democratic Hungary, this issue is on the contrary: civil and the green organizations can form oppositions against NPP; the public acceptance is an important issue for a NPP extension or for a potential plan of new NPP.

There wasn't any specific public inquiry held on nuclear co-generation before, during or after the installation, because the district heating is very cheap compared to the cost in other Hungarian cities.

3.2.1.3 Technical aspects

The installed VVER-440 units in Paks have the district heating capability by default due to a modification in the original VVER-440 design. This capability was necessary and planned (in advance) to exploit the district heating opportunity from the beginning of the construction. Its cause was that the new residential district (built for the employees of the NPP) in Paks needed district heating. The building housing the district heating system was a part of the construction of Paks I NPP. There was not a standalone project dedicated for the district heating. The experience on district heating gathered in the Soviet Union had to be used during the soviet design of the units. Further details on experiences and interactions can be found in /93/. It is worth to mention that the planned inclusion of the district heating system into the design of the NPP didn't influence the site selection process.

As it was mentioned above, the delivered co-generation products of Paks NPP are mainly electricity and in a minor share district heat. The carrier fluid of the district heating system is hot water. The throughput is 42 MWth while the temperature is 130°C/70°C on 8-9 bar pressure. The provided amount of district heat per year is about 160,000 GJ. Its minimum heat demand is during summer (5-6 MWth), while the maximum is during winter (20 MWth). **Fig. 4** shows the annual profile of provided district heat amount of Paks NPP. As it can be seen higher values occur during winter and lower values during spring and summer.

During the construction of the NPP, it was necessary to build infrastructure to interface with the district heating customers. For example a departure station (the heat goes to the departure station from the NPP and it gets distributed here) and local heat exchangers which provide district heating to residential blocks. The operators of the district heating subsystem have encountered some specific unexpected technical and organizational difficulties during the operation. It was identified that there is no ability in the district heating system to control the circulation of coolant. So constantly, during the whole year the coolant in the district heating pipelines is circulated by pumps. Furthermore, pumps with frequency changing ability are needed instead of the current simple pumps. The measurement instrumentation of the district heating system is not up to date and suitable either. Regarding the organizational aspect, there is a problem identified connected with administration because the district heating system is operated by 3 different departments of the NPP which causes inconveniences and sometimes certain problems.

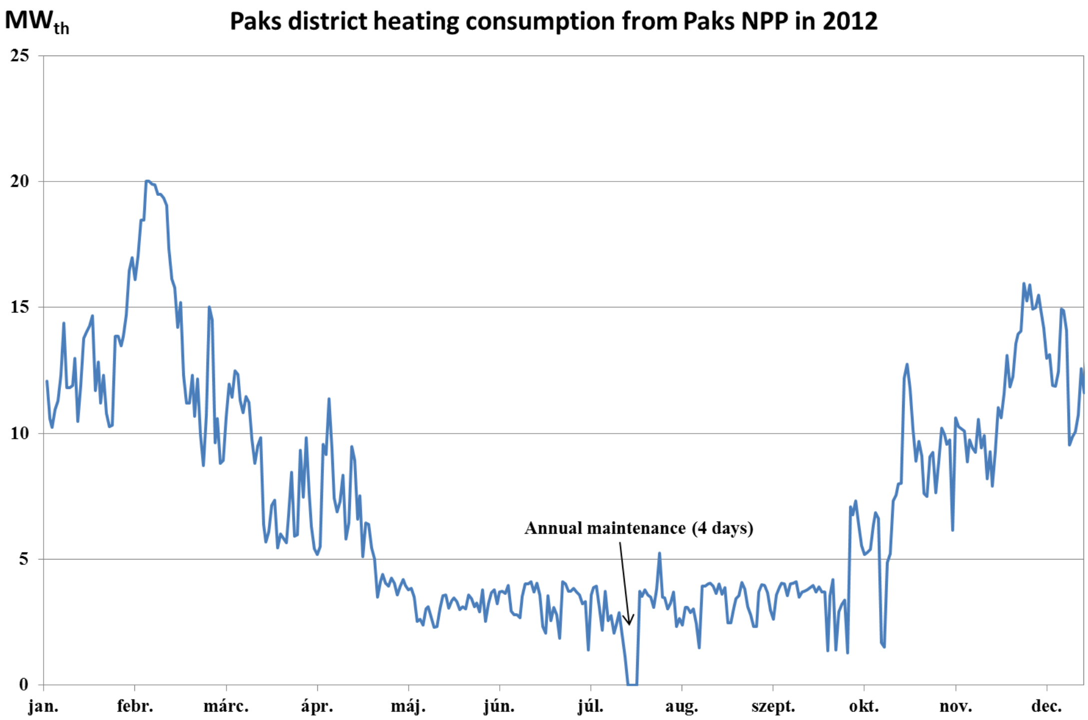

Fig. 4: The consumption of Paks district heating system provided by the Paks NPP in 2012

The presence of the district heating system at Paks NPP slightly decreases its electric efficiency, but strongly increases the overall efficiency without any impact on the availability of plant and other operation and maintenance aspects. There was no impact on the cooling provisions due to low heat output for district heating (see **Fig. 4**). Constructing of a back-up heat/power supply in case of NPP outage was not necessary. The boiler (which provided the necessary heat during the construction of NPP) has been removed.

3.2.1.4 Safety and licensing aspects

The licensing of the co-generation was included in the licensing of Paks I NPP. The co-generation was not handled separately in the safety aspect either. The heat production capability was not taken into account in the safety report of the nuclear installation. There were no specific initiating events to be considered for the safety assessment, in association with the heat production capability. There are no specific criteria and radiological controls required on the heat transfer fluid of district heating system. Radioactivity in the secondary circuit of the reactors will initiate a SCRAM. The co-generation influences the environmental footprint of the nuclear plant, but its contribution to the environmental footprint of the NPP is very limited.

3.2.1.5 Timing aspects

The design of Paks I NPP has been performed from 1968 to 1977. The design was started in 1968 and shortly after that there was a pause (4-5 years) before continuation. Licensing process for the NPP or especially for the district heating system was not yet performed. The construction has been done between 1977 and 1987. The original operation period spans the period of 1982-2017 (unit one has started in 1982, Unit-4 would be shut down in 2017 whithout the ongoing lifetime extension). The lifetime of Paks I NPP was 30 years originally which is and will be extended up to 50 years. The units 1 and 2 have undergone the lifetime extension process while Unit 3 and 4 are under the lifetime extension process. That is the reason why the foreseen decommissioning period for Paks I NPP will start after 2032.

3.2.1.6 Conclusions

It can be pointed out that a similar government controlled project without licensing can't be performed today in Hungary. The regulations have been changed a lot and the potential construction initiator of a new NPP would face a serious licensing process in Hungary.

The district heating system hasn't changed for 30 years. It is successful but technically out-of-date. It is under revision from the technical point of view nowadays and ceveral technical changes aer under discussion (e.g upgrade of control and measurement systems, extension to a higehr number of consumers). This also means that conclusions on economic efficiency may hardly be drawn from present installations.

The district heating is financially successful. It took approximately 15 years (first half of original lifetime of Paks I NPP) to write off the initial investment. It is successful in both social and environmental aspects. First of all, the society has accepted the nuclear district heating. On the other hand, fossil energy source has been substituted for 30 years by the nuclear district heating in Paks resulting in a huge saving in CO_2 emission.

The customers in Paks are very satisfied with the nuclear district heating. The project is more accepted among them today than at the start-up.

The co-generation project is linked to the lifetime of the NPP. According to the current plans, when the units of Paks I NPP will be stopped then the units of Paks II NPP (which is under licensing) will take over this commitment. If support provided in the field of licensing, Hungary would be a good place to develop nuclear co-generation.

3.2.1.7 Next plans

Regarding the currently existing system, the next plans are about extension of the current district heating system.

Furthermore at the beginning of 2014, Hungary has signed a contract with The Russian Federation about the construction of 2 new VVER type units (1200 MWe class) at Paks site. So, the extension of Paks site can be expected in the next 5-10 years.

3.3 Poland

Poland is a country that does not have a nuclear power plant. However, in the 60s of the last century the then government decided to develop nuclear power and began the process of design and preparation of the construction of the first Polish nuclear power plant. Poland, as a country of the communist bloc, developed intensively the heavy industry and shipbuilding (development of Gdynia and Gdańsk shipyards and ship repair companies). The dynamic electrification of towns and villages caused increasing electricity consumption, which was connected with problems in its supply to end users. In 1964 the evaluations and researches of the areas designated for the location of the first nuclear power plant were started. The conducted analysis indicated four locations: Lubiatowo, Żarnowiec, Przegalina, Biała Góra. Its geographical location did not come about by chance : all these sites are located on the north of the country. This part of Poland does not have natural resources - coal, lignite, oil. That is why the electricity had to be "imported" from rich in coal south of the country, through relatively poorly developed grid. The construction of a nuclear power plant in the northern part of Poland was a very good solution to eliminate the energy deficit and losses of energy during its transmission. Regardless of the choice of location, the Presidium of the Council of Ministers approved finally the construction of the NPP on August 12th, 1971. One year later, on December 19th, 1972, the government made decision to locate the NPP on the terrain of the liquidated village Kartoszyno, at the lake Żarnowiec. The then Polish nuclear power program predicted the total installed power in NPPs will be 8.5 GW by 1990 and 30 GW by 2000. The start of the first unit was expected in 1983.

3.3.1 Żarnowiec Nuclear Power Plant

For the Żarnowiec Nuclear Power Plant an operation with co-generation capability was proposed. The first concepts appeared in "Technical and economic study of the optimal choice of supply of heat energy for the Gdańsk Agglomeration" /40/. Six options for a district heating concepts in this area were prepared. The analysis showed that one of the best solutions for the base load heat source would be the proposed Żarnowiec NPP.

Żarnowiec NPP was originally projected for electricity production. As a result of agreements between Energoprojekt, Investor and the Zamech company the NPP design was upgraded with a pass-out and condensing turbine providing the option of heat removal - 460 MJ/s from each turbine. The NPP was designed to supply half of the district heat energy needed. It required the NPP should run in co-operation with peak heat sources, i.e. with two new heating plants localized in the Gdynia and Gdańsk area.

According to the then schedule the NPP should be constructed in two stages:

- Phase I - 2 units 465 MW - the expected start date - 1991 and 1992,
- Phase II - 2 units 465 MW - the expected start date - 1994 and 1995.

In this way, the technical conditions have been created to achieve 1840 MJ/s from the 4 turbines of the Żarnowiec NPP. In this scenario, the supply of heat from the NPP to Gdansk and Gdynia, and to nearby Reda and Wejherowo, was proposed. The construction of one pair of pipelines with a nominal diameter 1200 mm, the maximum which was available in Poland, was foreseen.

The delivery of large amount of heat to the district heating grid required a well-designed transmission system. With two 1200 mm diameter pipelines, approximately 1200 MJ/s can be sent in an optimal way. This power could be obtained from three 440 MWe blocks, but in order to increase the reliability of heat delivery, the utilization of all four units was proposed.

The Żarnowiec NPP thermal system consisted of three parts:

- nuclear power plant as the heat source,
- transit lines and
- distribution centre of a district heating system, connected to the local heating grid in Gdansk and Gdynia.

Two options of the non-nuclear part of the NPP were developed and proposed for the supply of the district heating distribution centre subsequently connected to local heating systems:

- water from transfer line is directly heated by steam from the turbine with exchangers,
- with indirect short water circuit and an additional heat exchangers station "water-water" in the area of the NPP; water from the line would be heated in the additional heat exchangers.

The type of interface between the non-nuclear part of the NPP and the district heating distribution centre has a significant impact, among others, on:

- investment costs,
- temperature of water in the transit line and
- loss of electrical power production of the NPP because of the separated thermal energy.

These issues strongly influenced the efficiency of heat supply from the Żarnowiec NPP to the Gdansk Agglomeration.

As another option for the heat source the nuclear heating plant with two low-pressure water reactors AST500 was predicted. According to the plans it should be started in 2005. The planned thermal power of this nuclear heating plant was 1000 MJ/s and the temperature of water in the heating grid 153/77°C. The location of this nuclear facility was in the southern part of the Gdansk Agglomeration, in Jagatowo. It was to be a source of heat for Gdansk and Gdynia. It was predicted that nuclear heating plant will be the main source of heat for the Gdansk Agglomeration.

Apart from Żarnowiec Nuclear Power Plant, the construction of Nuclear Power Plant "Warta" was planned, which also had to be equipped with pass-out and condensing turbine and a connection to the district heating grid of Poznan.

The accident in Chernobyl changed all plans and caused violent protests against the development of nuclear energy in Poland, which arose after the political transformations of 1989. On 27th May, 1990, the local referendum concerning ŻNPP was carried out. In the then Pomerania region 86% of votes were against the construction of ŻNPP, but this voting was not valid – only 44% of residents took part in it. Nevertheless on 4th September, 1990 the then government decided to stop this investment. The construction progress of the NPP was 36% and the progress of construction facilities was 85% (Fig. 1).

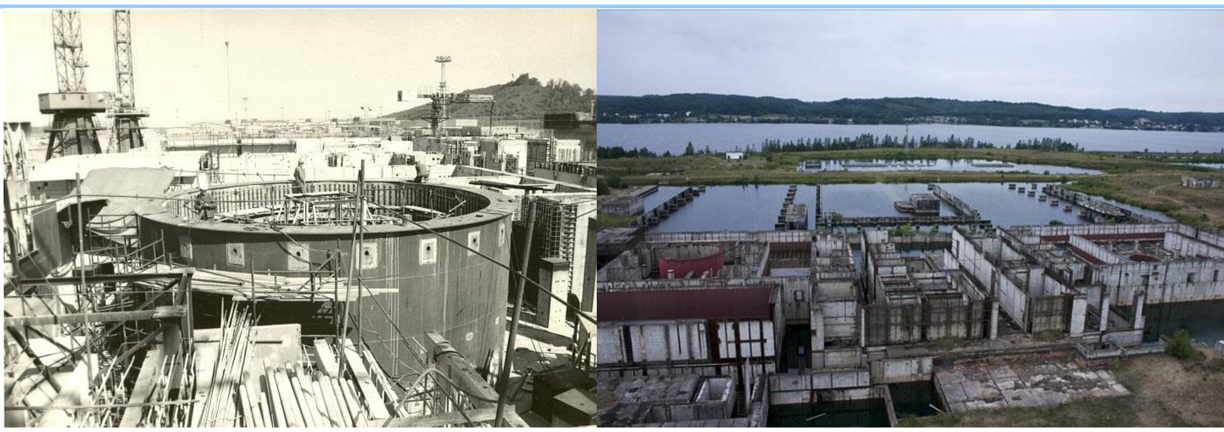

Fig. 5: Left: the construction of Żarnowiec Nuclear Power Plant (1st June, 1989), photo: Stanisław Wiesiołowski, right: Żarnowiec Nuclear Power Plant – current status, photo: Wikimedia Commons

3.3.2 Renaissance of nuclear power in Poland

The idea of the atomic energy development reawakened in Poland, which was the result of an analysis of the needs and potential directions of development of the national energy sector, trends in the European energy sector as well as environmental demands associated with them. A milestone for these activities was the decision made on 13th January 2009 by the Polish Council of Ministers to develop the Polish Nuclear Power Program. The task of constructing and operating the first Polish nuclear power plant was assigned to PGE Polska Grupa Energetyczna S.A. According to the document: „The Polish Programme of Nuclear Energy", which was endorsed by the government in 2014, the first NPP will be open between 2015 and 2030. Unfortunately the work of the NPPs in co-generation is not predicted.

3.3.3 The Polish Prototype Nuclear Heating Station

Polish Prototype Nuclear Heating Station (PPNHS) /41/ was oriented on two projects of reactors which could deliver hot water for heating purposes in densely populated urban areas, namely on:

- Finnish-Swedish LWR reactor SECURE, (developed in 1975-77), and
- French LWR reactor THERMOS.

The underlying assumptions for both projects were practically identical; the main difference is the design: a closed reactor pool for SECURE and open one for THERMOS.

The PPNHS was to be used as source of hot water (~100°C) for heating Pruszków near Warsaw. It was planned to deliver 200 MWth. Polish project started in 1976 and ended in 1978.

From the public point of view, the "forgiving" district heating reactor should guarantee that:

- In the case of design basis accidents, due to internal events or external hazards, personnel errors or possible hostile actions, the amounts of radioactivity released outside the reactor plant shall not require any intervention measures outside the exclusion area, with the radius of 800 meters.
- The protection against significant releases shall be based on simple natural physical phenomena and independent of active safety devices and of human intervention.
- This protection will last for a time long enough, after the initiating incident, that local authorities have enough time to take adequate, but relatively simple and easy to apply, countermeasures.

At early stage of PPNHS project it was decided, that fuel types used for its core should be readily available in Polish conditions. As a consequence, the unified core was proposed both for SECURE and THERMOS type systems, with extensive use of typical USSR made fuel pins. This ruled out French type plate fuel.

The high requirements concerning safety of the nuclear facility located close to densely inhabited areas were decisive in choosing low parameter core submerged in large volume pool of water, which can assure its long time cooling in case both operational and accident related shut-off of the reactor.

The characteristics concerning heat distribution to public were optimized, taking into account cost of PPNHS, cost of peak heat generating station, and heating grid. In conclusion a core inlet temperature of 92°C and for outlet of 118°C was selected. This would result in grid inlet temperature equal 101°C, which is not sufficient for winter conditions in Poland. Therefore operation in coupled mode with co-generation (heat + electricity) district heating power station was envisaged. It was estimated, that PPNHS could satisfy ~75% of heat demand during a year, assuming standard parameters of water in the district heating grid equal 92°C in summer and 118°C in winter.

Because of safety and complexity related comparisons the idea of open pool was abandoned and the project with closed pool was chosen for further development.

PPNHS was envisaged to be a pilot plant for introduction of nuclear energy in Poland. In parallel, preparations for construction of the Żarnowiec NPP were in progress. Finally, the construction of Żarnowiec started in January 1982, stopping other nuclear projects in Poland. In Scandinavia the SECURE project lasted till 1985.

3.4 Norway

3.4.1 Halden Test reactor (boiling water)

The Halden facility is a Heavy Boiling Water Reactor (HBWR) located close to the sea shore in the south-east of Norway. The reactor vessel and primary circuit are enclosed in an artificial rock cavern which plays the role of a containment barrier and a physical protection in case of external hazards.

The reactor has been run from the beginning by the Norwegian Institute for Energy Technology (IFE), founded in 1948. It was foreseen primarily as a demonstrator of nuclear heat production with the purpose to produce steam for a paper mill. Then, in 1958, it turned to an experimental device as part of the international OECD Halden reactor project process for research on nuclear materials and fuel. The Halden reactor has been operated on a regular basis since 1959. It is always delivering steam, but R&D activities are prevailing on heat production.

The nuclear plant comprises a reactor pressurized vessel and three water/steam circuits. The cooling medium in the reactor vessel is heavy water, which circulates in a closed circuit within the containment. The energy carried by the primary coolant is transferred by steam transformers to a secondary closed circuit containing light water. The pipes transporting the secondary circuit pass out of the containment, transferring the heat to a tertiary light water circuit connected to the neighbouring paper factory.

The quality of the primary heavy water and secondary light water is monitored by detection of gamma emitters, leading to further water analysis if an alarm value is reached (no automatic isolation of the tertiary circuit on abnormal activity detection). It was noted that the steam generator tubes could be considered as a third barrier as far as like leakage testing (periodical testing) and in service inspection are established. At least, although tritium is continuously produced by nuclear reaction on the heavy water, the operator indicated that the delivered steam is not contaminated.

Halden reactor is an exception among the co-generation systems examined here, because all the produced heat (around 20 MW) is delivered to the paper mill, although this represents a maximum of 10% of the paper mill's need. The majority of the process heat is produced by conventional boilers which serve also as back-up. So the industrial process does not request the reactor availability, which is imposed by the experimental programs.

Since the commissioning, the installation underwent several safety reassessments (e.g. IAEA INSARR mission in 2007) but the recommendations and comments made by the IAEA reviewers were not related specifically to the heat generation for the paper mill.

As a conclusion of this review, the process heat production achieved in Halden does not appear to have posed specific safety problems. For further details see contribution D3.11 /92/

3.5 Switzerland

3.5.1 NPP Beznau

The Beznau nuclear site hosts two PWR units of 380 MWe, which were commissioned in 1969 and 1971. The units are operated in a base load mode. Heat production has been implemented 30 years ago and support the district heating grid connecting around 2500 customers, The maximum potential heat delivery is 80 MW, but usually it varies seasonally between 60 MW and a few megawatts (in summer). Both reactors are connected to the district heating grid providing a certain redundancy in case of reactor shut down, maintenance of refuelling procedures.

Beside the general steam production (at 122°C/128°C, pressure range 0.22 MPa to 0.28 MPa) it delivers water is delivered to the district heating grid by the central pumping station between 85°C to 126°C (max.) and 1.6 MPa. The closed heating grid makes a pipeline “backbone” of 31 km connected to a distribution grid of more than 100 km. Several back-up boilers installed on the grid may take over the heat supply during simultaneous outages of the reactors. With the closed heating “backbone” pipeline 2 barriers exist to the primary circuit, an additional barrier is usually applied by the customer to connect to the main district heating grid. Contamination of the “backbone” pipeline is technically eliminated by a higher pressure level (minimum 0.6 MPa) than in the interface of the secondary circuit and safety isolation valve in the “backbone” pipeline. Radiation monitoring of the delivered water is not applied. No transients on the reactor operation were generated from the district heating grid by technical decoupling measures.

Despite the relative long distance to the customer an efficient district heating source can be provided by the co-generation facility. Additionally significant positive environmental contributions are provided: save of around 46,000t/a CO2, truck traffic reduced by 8500 trips, reduced input to the heat sink (Aare river).

3.6 Japan

3.6.1 HTTR

The High Temperature Engineering Test Reactor (HTTR, 850 °C outlet coolant at 30 MWth and 4 MPa) built in 1998 at Oarai, Japan, is proposed for a demonstration plant for hydrogen production by steam reforming (SR) process at a rate of approx. 4000 Nm³/h. The coolant temperature level is significant higher that for a standard HTR and represents a first step forward to VHTR conditions. In 2004 even successful operation at 950 °C outlet coolant temperature and hydrogen production by sulfur-iodine (SI) process in lab-scale were demonstrated. Originally focused on the methane steam reforming process for the hydrogen production the higher temperature levels are in particular important for more efficient ways of hydrogen production as the water-splitting process based on the sulfur-iodine (SI) thermo-chemical cycle. This plan to proceed with pilot-scale demonstrations of the SI process and the coupling to the HTTR is also mentioned in the NGNP report “Hydrogen production down selection” (*p 40) /5/,/12/.*The interface between the nuclear and the chemical system is realized by a secondary helium cooling circuit.

Because of the close distance between nuclear and chemical systems a new category of hazards was identified covering potential effects from one system on the other. The hydrogen production facility is located outside the reactor building nevertheless for severe accident analysis an accumulation of flammable compositions inside the reactor building via leakage/rupture of secondary pipelines is considered.

The HTTR plant is located in the Oarai Research Establishment, 5 km away from the centre of Oarai (approx. 20 000 inhabitants). Beside the HTTR also a material testing reactor (JMTR) (LWR, 50 MWth, 400m away) and an experimental liquid fast breeder (JOYO, 130 MWth, 650 m away) are located in the area.

The HTTR is equipped with a steel containment vessel to retain fission products and minimize air ingress for the respective severe accident scenarios. Because of thermal shields and internal structures the reactor pressure vessel (RPV) is only designed for 440 °C.

The information of this chapter are based on the report to hydrogen product with HTTR by K. Verfondern and T. Nishijara /6/.

3.7 Germany

3.7.1 PNP (Prototype Nuclear Process heat) project and Tritium Permeation

In Germany besides the development of LWR for energy production, an industrial application of an HTR heat source for the coal gasification process was also investigated. This HTR process heat concept was called PNP (Prototype facility for Nuclear Process heat) and was established in 1975. Based on the operational experience from AVR an HTR with a coolant gas output temperature of 950 °C was proposed and different options for coal gasification were investigated, namely hydrogenating gasification of brown coal and steam gasification of hard coal. Synthetic Natural Gas (SNG, i.e. methane), methyl alcohol (methanol) and hydrogen (produced from SNG) were investigated as end products of the gasification process. Besides analytical work on pebble bed HTR reactor designs with a hot gas temperature of 950 °C, the PNP project concentrated on the development of two key components to couple nuclear heat into the processes: for the hydrogenating gasification a steam reformer heated by primary helium, catalytically producing hydrogen and carbon monoxide from methane and steam, and a He-He IHX for the steam gasification process. This IHX was developed as a tube type HX in two variations, one with helically wound tubes and one with U-shaped tubes in a compact arrangement. Test components were built in a 10 MW scale and successfully tested for 10,000 hours in a dedicated test loop.

Since the PNP project started in 1975, the reactor concepts investigated first were based on the German HTR design of the time with a large cylindrical pebble bed core, i.e. the THTR. PNP-500 with 500 MW thermal was intended as a first-of-a-kind demonstration plant and PNP-3000 as a large scale commercial plant. With the development of the modular HTR (HTR-MODULE in Germany) in the early '80s, this concept was also applied to PNP conditions with modular pebble bed reactors of 170 MW thermal power each.

While stopped after designing a large-scale nuclear plant for process heat production the investigations, tests and experiments of the PNP project resulted in valuable findings on the feasibility of continuous coal gasification (steam reformer), manufacture and successful operation of high temperature heat exchanger components (He – He, He – H_2O) and the demonstration of licensing capability of nuclear heat HTR by respective safety research.

In contrast to actual designs of process heat applications the PNP concept considered a steam reformer closely coupled (directly by He – steam generator or via a He-IHX) to the primary He coolant circuit to achieve high thermal efficiency. The reactor itself and the subsystems (steam reformer, steam generator) are located inside the reactor building. For the close coupling of the primary He coolant to the conventional systems the tritium permeation to the product side was identified as one of the most important critical aspects. Therefore extensive investigations were performed on the Tritium permeation performance through HTR powered heat exchanger.

Several aspects of Tritium and hydrogen permeation were investigated within the PNP project in particular for steam reformer application. It was found that on the secondary side (intermediate and product) the tritium limit of 5 Bq/g according the German StrSchV of that time /23/ would not be exceeded in any operation state of the PNP-500 /9/. This value could be reduced significantly by application of an intermediate heat exchanger which was necessary for the steam gasification of hard coal and might also be applied to hydrogenating gasification. In combination with a gas purification system in the intermediate circuit a contamination level of about 0.37 Bq Tritium per 1g H_2 product was estimated /10/.

Also from other high temperature reactor test facilities valuable information on the hydrogen and tritium permeation can be derived /7/.

Estimations on the tritium mass balance of the operating Peach Bottom reactor showed that the helium purification system nearly retained all of the tritium because of the highly efficient titanium based getter effect and the effectiveness for capturing other gases (H_2O, O_2, N_2), too. Additional advantages are the possibility of regeneration of the Ti bed and the avoidance of expensive heat and cooling sections. Main tritium sources again are ternary fission in the fuel (55 %), ^{10}B reactions (40 %), ^{6}Li graphite impurities (3 %) and ^{3}He reactions (2 %). In contrast the observed tritium level in the helium coolant was close to the ^{3}He production level indicating that the majority of the generated tritium is bound (for the considered operation conditions) and will not contribute to the permeation fraction to secondary or intermediate coolant systems. The values given above are estimations specific for the operation of Peach Bottom.

Measurements during the THTR-300 initial operation phase showed a tritium concentration in the primary coolant of about 10^4 – 10^5 Bq/kg.

Performance limits of the helium purification system can be derived from operation experiences of Fort St. Vrain. After several water ingress incidents into the core and the subsequent oxidation of the graphite structure, a partial deactivation of the titanium sponge getter bed and the charcoal bed was observed. Nevertheless the tritium concentration in the primary coolant (mostly in form of HTO) was relatively low (about 0.4 MBq/m^3) demonstrating a still effective purification system.

Quantitative values on the different tritium sources can also be derived from estimations to the German HTR-MODULE concept (see chapter 3.7.4).

Tritium behaviour at even higher temperatures of about 950 °C was investigated in the Japanese HTTR during a 50-day test in 2010. Results are not yet available.

In addition to experimental investigation also computational models were developed (THYTAN by JAEA, TPAC by INL) with promising results regarding measured primary coolant tritium concentrations (Peach Bottom 2nd core data). Comparisons to secondary coolant tritium fractions are not yet presented also because of the lack of experimental data.

Tritium permeation

One of the major concerns for co-generation or process heat applications is the contamination risk by tritium generated in or close to the nuclear reactor. The issue of potential tritium contamination of end user systems, components or products by an HTGR system is addressed in the following section in more detail. The information is based on a report by FZ Jülich and AREVA GmbH /7/.

In a HTGR system the total tritium content is dominated by the fraction from the ternary fission products (≈ 50 % of the total tritium production) and (of lesser extent) from ^{6}He (which immediately decays to ^{6}Li and by neutron capture subsequently to tritium). This tritium fraction is mostly contained inside the TRISO coated fuel particles and contributes to the coolant tritium level only in case of failed particle coating. Contribution from uranium impurity of the graphite components or from tritium permeation through the intact coating is comparably small (1 % to 10 % of the coated particle tritium content). Significant contribution to the coolant tritium level (≈ 35 % of the total tritium production) is created by activation reactions on lithium and boron impurities of the graphite components, control rods (in particular for block type design with CR slightly inserted in the core) or from secondary circuit leakage (see DRAGON reactor). Finally the coolant tritium level is contributed by the decay of ^{3}He (about 2 % to 15 % of the total tritium production, depending on the helium make-up compensation the helium leakage). In the primary coolant tritium exists as HT, CH_3T and HTO due to isotopes exchange reactions. The specific tritium production rate in total can be estimated to 370 – 810 GBq (10-22 Ci) per year and MWth with higher rates in the initial operation phase and an equilibrium stage after 4 - 5 years.

Significant tritium removal by decay is usually not considered because the resulting ^{3}He will be reactivated to T by (n,p) reactions. As an intermediate tritium sink the tritium bound by adsorption inside the graphite structures and fuel elements should be mentioned. A certain fraction of the bound tritium will be removed by discharging finally burned fuel elements. But the graphite bound tritium fraction in general should be considered only as intermediate sink and will contribute to the overall release dose level for severe accident scenarios with graphite corrosion effects (water ingress, air ingress). In addition the tritium concentration of the graphite structures needs to be considered for maintenance, repair and final disposal.

To minimize the tritium contamination of the primary coolant and finally the end user products several options are available. The first opportunity is a well-designed helium purification system. Usually designed as a bypass system of the coolant flow it continuously and effectively removes impurities from the He coolant (e.g. 99.6 % for a 1/h helium purification system). For the tritium impurity a purification performance of about 100 % can be achieved by a two-step procedure: first oxidation of HT on CuO-based catalysts and subsequent HTO capture by a molecular sieve at 180-240 °C (higher temperatures up to 800 °C needed to effectively burning methane). And secondly low temperature (20 °C to 30 °C) adsorption on charcoal beds (Note: A liquid nitrogen cooled charcoal bed will also be effective for Xenon or Krypton capture, see experiences from the PHENIX project).

This purification effect may be improvable by an intermediate sweep gas flow or controlled addition of oxygen or steam to improve permeation inhibiting oxide layers inside the primary circuit. But these options are not yet proven technology and subject of further investigations.

For a He - steam interface (HTR MODULE, primary: 7 MPa, secondary: 19 MPa), a pressure grading prevents from a contamination of the secondary circuit. To ensure a pressure grading for He-He interface the pressure at the secondary side has to be maintained by a separate pressure preserving system and the pressure difference is significantly smaller (usually a few 0.1 MPa).

Because of the primary tritium level the contamination effect on the secondary circuit has to be evaluated. Contamination of the secondary circuit is created by tritium permeation though the heat exchanger wall driven by the partial pressure difference between the two gas spaces. Because of the high mobility of tritium this contamination path is hard to limit / minimise. In case of a missing IHX also the hydrogen permeation in the opposite direction from the product side to the primary side should be considered in the design of the primary purification system to minimise corrosion reactions with the graphite structure.

Options for minimising the permeation of tritium and hydrogen through the heat exchanger wall material are optimisation of the heater material, establishment, maintenance and optimisation of oxide layers as Cr_2O_3 or Fe_3O_4 (effective for a temperature range > 600°C, permeation rate reduction by two orders of magnitude feasible), protective coating e.g. by SiC or Y-stabilized ZrO_2 and last but not least optimised heat exchanger materials (e.g. Incoloy 800H, Incoloy 802, Hastelloy H). While several experiments were conducted in the past indicating promising results (see AUWARM experiment /8/, PNP project /9/) these options are not yet proven technology and need further investigations. A HTR demonstrator therefore should start in a first step with a tertiary circuit to the end user facility. Experimental investigations on the permeation of hydrogen and tritium accompanied by the development of computational models with increasing prediction performance were conducted. (German computer model by Cordewiner 1979) in analogy to the ORNL code TRITCO).

In conclusion past experiments and investigations and ongoing R&D indicate promising options to significantly minimise the tritium contamination level of the secondary coolant circuit by optimizing the heat exchanger and the purification system regarding tritium retention. These options seem to be not yet ready to challenge the need of a tertiary coolant circuit to the end user facility/product but may be successfully demonstrated by corresponding tests and investigations with the prototype HTR. Recent investigations in the ARCHER project indicate that, for an HTR generating process steam in the secondary circuit, a viable tritium control strategy may not be problematic to achieve. However it is proposed that plant designers build in some flexibility in operating conditions to allow compensating for the various uncertainties. Furthermore it was seen as a precondition for the elaboration of a more refined tritium control strategy to take into account the national licensing conditions for new-build HTR, in particular the allowed opeartional and accidential tritium release. (see ARCHER report D23.71 /95/).

Past experiments and nuclear applications demonstrate that tritium concentration levels significantly below the licensing limits could probably be achieved (in the range of background noise, for river water in the range of 10 Bg/l). In case there is no domestic regulation yet applicable, the safety analysis should be based on the principle that any exposure of the public (including personal of the non-nuclear part of the co-generation/process heat facility) to artificial radiation doses must be below the 1 mSv/a limit.

Past experiences from LWR show that the tritium contamination level of the primary coolant is driven by neutron activation reactions on boron and lithium and decay reaction of ^{3}He. The contribution from ^{3}He will be unavoidable (in particular in a HTR) and remains rather constant because of the equilibrium reaction between ^{3}He and T. In addition other isotopes may be considered for the burnup and shout down control and/or respective components may encapsulated by a tritium retaining barrier.

The tritium fraction from the ternary fission reaction of heavy metals is retained very effectively by the cladding (LWR) or coating (HTR) of the fuel elements and is only of concern for severe accident analysis. In case of severe accidents the connection to the end user will be interrupted therefore this accidental tritium concentration levels is of no concern for the design of the co-generation facility.

3.7.2 Stade NPP

The process heat supply project at Stade was initiated in the 80’s, around ten years after the plant commissioning. Stade PWR has been producing electricity and steam for about 20 years over a total operation lifetime of 30 years. The steam was used mainly in salt works located at a distance of 1.5 km from the NPP. The steam was delivered through a closed tertiary circuit. Considering the total heat produced in the reactor, only a small portion of the power was diverted to the industrial grid (around 2.5%).This coupling scheme appears quite comparable with some of the district heating systems described above (the Polish projects focus on higher factions of co-generated heat).

According to the operator, the implementation of the heat production system “was a major change of the overall plant concept” because important technical modifications have had to be made on secondary circuits and a new tertiary circuit established. So, a safety review has been performed.

The specific risks originated by the new boilers installation have been investigated resulting in a demonstration of a pressure blast resistance of the concerned building. Fast transients of the coupled system have been assessed to ensure that there were no harmful side effects caused by the added heat production system. Nevertheless, the scram and turbine trip procedures did not need to be modified.

Potential for unmonitored contamination of the salt work heat grid was identified as a main issue to be dealt with. In support of the licensing of the heat production system, the operator has assumed and evaluated single and combined initiating events (leaks of the steam generator tubes, blow-down system failures, operation failures, malfunction of the monitoring system, simultaneous leakages between the three circuits) which could lead to contamination of the secondary and tertiary circuits in order to define the adequate counter-measures. As a consequence, the secondary and tertiary circuits have been equipped with radioactivity monitoring systems connected with the automatic isolation and blow down systems. Moreover, the radiological impact of hypothetic contamination accidents of the tertiary circuit has been assessed and rules were established for monitoring reporting and mitigating the consequences of these accidents. Anyway, the new heat production facility was deemed as “non-safety system” and the regulator has not required an update of the safety report.

The information related to Stade NPP gives a quite complete overview of the issues to be assessed when implementing a heat production system. In particular, the study of external hazards has to take into account the events potentially induced by the non-nuclear facility.

For further details see contribution D3.11 /92/

3.7.3 The PWR process heat project for BASF

The company BASF proposed in Mai 1976 a PWR for process heat production (385 MWe, process steam: 500 kg/s, 18 bar, 265°C) /11/. The intended location of the NPP on the BASF premises was close to the Rheine River in the federal state of Rheinland-Pfalz.

The closest infrastructure were the BASF sewage plant (150 m), a highway (1400 m) and the oil transfer port (1800 m) all located in the south of the BASF area.

No flammable, toxic or narcotizing deposits of the BASF company were located within a radius of about 2.5 km. The BASF petrol depot was located in the south more than 2.3 km away. The distance to the river line was about 470 m. To the west of the reactor building an ethylene pipeline was planned. In conclusion the closest distance of the source of an external event which could affect the reactor building would be ships (470 m), the ethylene pipeline (600 m) and traffic on the road 640 m away from the reactor building. The ground conditions were checked to be reliably stable down to 600 m.

Several additional investigations performed (e.g. environmental impact analyse, protection from external hazards, severe accident analysis) but none of them are specific for the process heat application but originally derived from the licensing requirements of the NPP. While for this project the official licensing procedure was not initiated, the drafted documentation nevertheless gives an indication on the expected issues to be considered from the operating companies view.

It can be concluded that in this case satisfying the licensing requirements of the NPP would also have covered the process heat applications.

3.7.4 The HTR-MODULE-200 project

In the late 1980s a modular pebble bed reactor, the HTR-MODULE, was proposed in Germany. The HTR-MODULE is a high temperature pebble bed modular gas-cooled reactor with a cylindrical core and passive decay heat removal features as required by the German regulatory authorities. The HTR-MODULE design was reviewed by the German authorised inspection agency (TUEV Hannover) and approved by the German reactor safety commission (RSK-expert commission of the German federal government). At the time the design was considered ready for final design activities in Germany /21//22/.

The HTR-MODULE plant is configured with two 200 MWth reactors and steam generator sets, each generating super-heated steam for independent turbine-generator sets for electricity production and reboilers to provide high temperature steam for industrial application (Fig. 6).

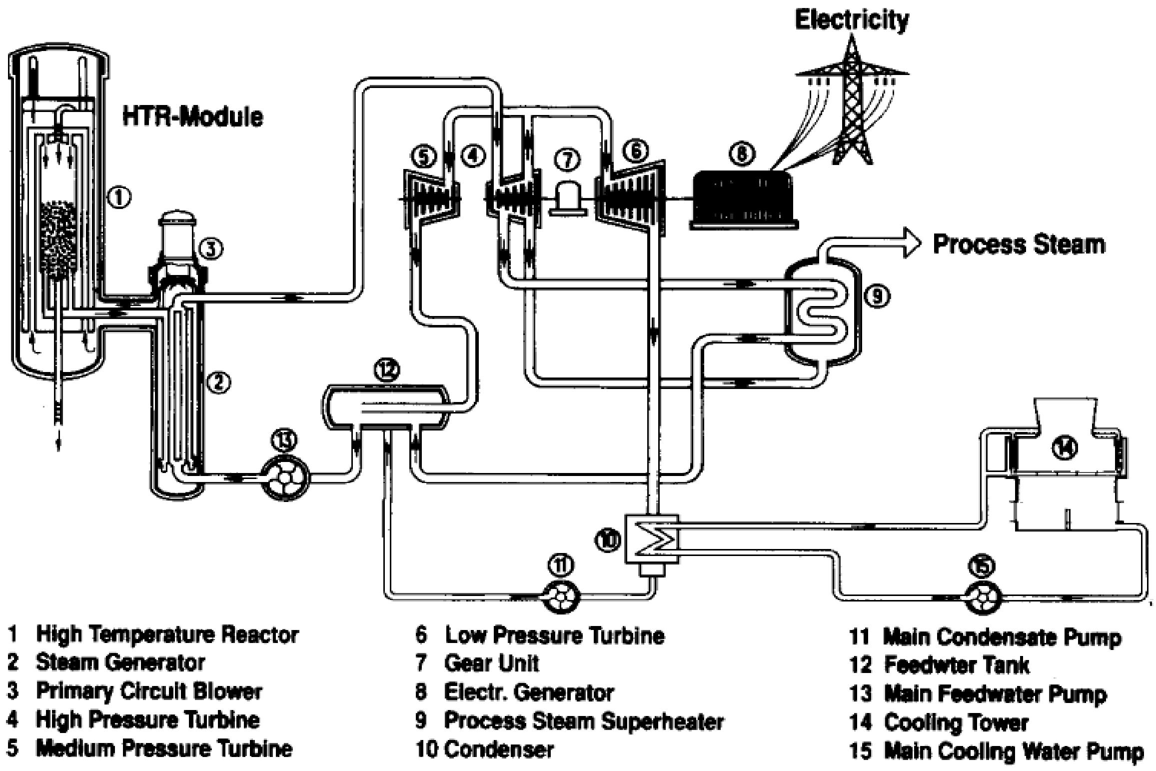

Fig. 6: Schematic scheme of the HTR-MODUL concept

Regarding licensing of the industrial application, the data in the following chapters were the bases in the pre-licensing procedure for the HTR-MODULE foreseen for oil field exploration. Mainly tritium is discussed for the water/steam cycle and the process steam. Those amounts of tritium depend strongly on the given plant design, that means first of the sources in the primary circuit and the distribution in the plant. For other process heat applications the interface to the end user may be implemented by a tertiary circuit depending on the regulation requirements.

3.7.4.1 Tritium in the Primary Circuit

As shown and explained in the Technical Report /7/ the maximum tritium value of about 49 TBq/full power year for one module is taken for the calculations. Therefore against 31 TBq/full power years at equilibrium core condition a safety margin of a factor about 1.6 to the accepted initial phase tritium release rate is given (Tab. 2).

Tritium source	Tritium release rate [TBq/full power year]	
	Initial phase (t=0)	Equilibrium (t=6 years)
Fission, direct	3.3	4.6
6Li, pebble	3.3	8.5
6Li, reflector	3.7	1.9
6Li (maximum after 0.5 years)	15.5	
10B, absorber	1.9	1.9
3He primary	24.4	14.1
Total (maximum after 0.5 years)	36.6 (49)	31.0

Tab. 2: Tritium release into the coolant (design values, data per module)

3.7.4.2 Tritium Distribution in the Plant

- *Primary Coolant*

Neglecting the adsorption of the core graphite as a sink for tritium it remains:

- Helium leakage
- Helium purification system and
- Permeation through the steam generator tubes.

The helium leakage can also be neglected due to the design leak rate of 0.1 % per day against a gas purification constant of 0.05 h^{-1} of the helium inventory (1.2 per day, means purification of more than one complete primary coolant volume per day). The same can be said for the permeation path.

Due to the dominant sink of the helium purification system, the tritium activity of the coolant is

$$A_C \sim A_F/0.05$$

With the given flow through the purification system of A_F = 5.6 GBq/full power hour (per module) the maximum equilibrium activity of tritium in the coolant of one module 0.11 TBq (Ac).

- *Helium Purification System*

Via the purification bypass 99 % of the tritium enters the purification system. Every 1000 full power the molecular sieves and the deep temperature absorber are regenerated and their tritium load of 5.6 TBq in form of HTO) is collected in a special vessel. After 32 full power years the activity in this vessel is about 930 TBq.

- *Water/Steam Cycle*

Because of its property of being able to pass through metals at high temperature (permeation), tritium enters the water/steam cycle even through intact steam generator tubes, governing the activity inventory in the cycle. Calculations based on conservative assumptions indicate that 3.7×10^7 Bq/hour per module enter the water/steam cycle. The maximum activity concentration is reached during operation with a closed water/steam cycle. At a loss of 1.25 Mg/hour (loss of water and steam in the turbine) the activity concentration adjusts to approx. 63 Bq/g.

Besides the tritium permeation neutron activation has to be considered. The thermal neutron flux which reaches the steam generator amounts to approx. 10^7 s^{-1} cm^{-2} at the steam generator tube bundle inlet. The neutron flux decreases rapidly in the tube bundle and the average is therefore lower about one order of magnitude. The saturation activity resulting from neutron capture reactions integrated over the tube bundle is estimated as being less than 2×10^{10} Bq on the basis of the main reactions. Corrosion-induced metal thinning causes less than 1×10^7 Bq per year to enter the turbine building with the main steam from the two steam generators, primarily during start-up and shutdown operation. The variation range of corrosion products of the Co-free steam generator material is limited to a few base elements therefore isotopes with high biological impact as Co-60, Cs or I are no issue. So the dose level of the steam is determined by the concentration of tritium and activated corrosion products released from the secondary side of the steam generator.

Trace impurities in the feedwater and conditioning additives also undergo activation in the steam generator to a certain extent. Extrapolation of the integral activity concentration of all radioactive materials in the main steam, as measured in the AVR during operation, to the HTR-MODULE yields a value not exceeding to 0.4 Bq/Mg.

- *Process Steam System*

The nuclear induced tritium activity in the process steam is of interest when process steam is directly extracted from the water/steam cycle. The considerations given below are related to an HTR process heat application for oil field exploration, using steam directly from the open secondary circuit. For sensitive end user products certainly a third heat exchanger will be applied to comply with the respective nuclear regulations on Tritium emission.

According to conservative calculations by the designer (former Siemens Interatom, now AREVA), the specific activity of the process steam is approx. 0.49 Bq/g in steady state operation for an open secondary cycle (note: average tritium concentration in river water is in the range of 10-20 Bq/l, limit of tritium concentration according to the EC directive on water intended for human consumption (98/83/EC) dated to 5.12.1998: 100 Bq/l; varying national limits from 740 Bq/l (USA) to 76103 Bq/l (Australia), World Health Organization: 10^4 Bq/l). A higher tritium carryover rate is to be assumed during the brief period of start-up operation on account of the reduced obstructive effect of the oxide surface films in the steam generator. A maximum of 30 Bq/g was calculated for this operating condition.

Considering the uncertainties in the calculations (factor of 10), the following data for specific tritium activity in the process steam are stated in the licence application:

- Steady-state operation 5 Bq/g (at normal condition);
- Start-up operation 74 Bq/g (limit for exceptional handling from licensing pursuant to Article 4, Paragraph 2 of the Radiological Protection Ordinance /23/ in conjunction with Annex III Item 2) on a maximum of 20 days per year.

For other process heat applications with closed secondary cycle (accumulation of specific activity) the process heat may be used via a tertiary circuit to satisfy radioactive release limits.

3.7.4.3 Assessment by the Regulatory Body

The Safety Review Report by the TUEV Hannover on behalf of the Federal Minister for Research and Technology (BMFT) was based on the Safety Analysis Report for the HTR-MODULE and hundreds of supplementary reports. Regarding the process steam and the direct use of main steam in industrial facilities the following was assessed in the review report:

> *In principle, it is possible to operate the secondary circuit in the open mode and to divert one part of the main steam as process steam. Accordingly, the activity concentrations in the main steam are lower.*
>
> *We do not see any problems concerning the compliance with the applied data. The difference with the design data for the tritium concentrations in the process steam during open operation mode is with a factor of 10 for steady-state operation and a factor of 2 for the short term start-up operation large enough.*
>
> *Both values meet the demands of Article 4, Paragraph 2 of the Radiological Protection Ordinance /23/ allowing in conjunction with Annex III Item 2 an approval-free operation with the radioactive materials, should be the specific activity be lower than 74 Bq/g.*
>
> *By utilizing the process steam outside the power plant and thus beyond the „controlled area", unlimited further utilization of the process steam condensate must be guaranteed. Should no yearly limits for the drawing off of tritium through the process steam path during a later construction licensing procedure be set, Article 46, Paragraph 4 of the Radiological Protection Ordinance must be complied with, allowing the transportation of the process steam condensate into wastewater channels or into surface waters only if the activity does not exceed a 1.25 multiple of the value stated in Annex 4, Table IV 1, Column 6.*
>
> *If it cannot be generally excluded that the condensate will be used as drinking water, the maximum yearly ingestion of tritium set by Article 46 may be slightly exceeded- based on the intended application values, assuming yearly drinking water consumption of 880 l. This aspect has to be examined during a later construction licensing procedure.*
>
> *Additionally during a construction licensing procedure, it has to be clarified, if requirements have to be applied, when the process steam is used for applications governed by Article 4, Paragraph 4, Item 2b-d (e.g. medicine, essential goods for food).*

Considering the latest version of the German Radiological Protection Ordinance /24/, Appendix VII, Part D, the respective limits for the radionuclide specific concentrations free releasable from controlled areas to conventional sewers has changed to 10^7 Bq/m³ ($7*10^6$ for organic tritium condition) so the acceptable release of Tritium needs to be re-evaluated for formal reasons. But the concentration values at steady state and start-up will be further on below the respective release limits of Tritium dissolved in water. As for a process heat applications steam is treated as product radioactive release limits for aerosols at normal operation are not applicable.

Nevertheless it should be kept in mind that the HTR-MODULE scenario described above only considers 2 coolant circuits. The additional Tritium barrier effect of a third coolant circuit is not credited. While significantly apart from the core in the steam generator will be affected by a low neutron flux causing a certain activity level. But the total activity level of the steam generator is dominated by the activity of the oxide layer. Activation of the steam generator base material usually is not considered.

3.8 China

3.8.1 HTR-10

The HTR-10 reactor located at the Tsinghua University near Beijing, China, is a small HTR prototype reactor (10 MWth, 3 MPa, outlet temperature 700 °C) for development and testing of HTR related materials, technologies, fuel types and operation procedures as electricity production or heat co-generation. Constructed in 1995 and in full operation 2003 the reactor was shut down for maintenance in 2011 and afterwards intended for obtaining experiences for the HTR-PM project (e.g. fuel element tests) and to conduct temperature measurements in the core and high temperature operations.

Depending on the test program the reactor runs not in a continuous mode but is started only for the test series. The heat removal is realized by a once–through steam generator coupled to the primary coolant circuit (inlet, 104 °C, outlet 435 °C) /87/, /88/. Via a steam turbine the energy is transferred to the local electric grid (up to 2.5 MW).

The output of the SG is partially used for supporting the heat grid of the nearby university but detailed information on the level of heat grid supply or specific licensing procedures conducted for this co-generation application are not available.

3.9 Other countries

3.9.1 Turkey: Development of Co-generation Systems

While not in the focus of the NC2I-R project nevertheless some experiences regarding co-generation are investigated for Turkey. These experiences may influence the nuclear options of energy production discussed, planed or already decided.

<u>Overview</u>

Due to the lack of primary resources, Turkey is highly dependent on imported energy. According to the Ministry of Energy, import dependency was above 72 % in 2012. This is underpinned by the dependency on natural gas imports which account for nearly 4 3% of the total electricity production.

To secure the supply of electricity in longer term, the government is tendering the construction of nuclear power plants. In May 2010, Russia and Turkey signed an agreement that Russian ROSATOM would build, own, and operate a nuclear power plant in Akkuyu which consists of four units generating 1,200 MWe each. The first reactor is expected to get online in 2020. Meanwhile a Memorandum of Understanding is signed with Japan for the second nuclear power plant which will be built in Sinop.

Turkey’s total primary energy supply (TPES) was 115 millions of tons of oil equivalent (Mtoe) in 2012. Turkey depends on an import of 72 % of its energy sources including oil, natural gas and high grade coal since the indigenous production constitutes only 28 % of the total energy demand.

In 2011, 44.7 % of the electricity generation was generated from natural gas, 28.8% from hydraulic resources, 16.9 % from lignite, 10 % from imported coal, 1.5 % from fuel-oil, 2.1 % from wind and 1.97 % from other sources. As of 2012, almost 60 % of the electricity is generated by private sector and 39 % was generated by public company (EUAS). Approximately 44 % of the electricity was generated by gas firing power plants (including cogenerated electricity). Imported energy resources are dominating the major part of electricity generation.

Insufficient energy sources, high energy prices, power quality problems, dirty air and global climate change are given as reasons for forcing Turkey to develop its energy saving policies to the maximum extent possible.

Past Developments of Co-generation

Power related investments and operations were under the monopoly of TEK (Turkish Electricity Authority) until 1984. TEK was not only in charge of controlling the frequency and voltage of the electricity, but also of supply, accessibility and availability. Industrial and household consumers had to be satisfied with the available and accessible electricity. In 1984 the law with no: 3096 was enacted by the Turkish Parliament which gave investors the right to build, operate and own power producing facilities (Autoproducers).This regulation was called the "Regulation for autoproduction of Electricity". As a result of the new regulations, the first co-generation plant was built in 1992 in Yalova, Elyaf (4 MW)

At that time the natural gas network was available only in the Marmara Region which covers almost 1 to 6th of the total surface area of the country. Therefore the regional availability of the co-generation plants was limited. Beginning from 1995 the gas network started to develop, which lead to the building of new co-generation plants mainly smaller CHP for hospitals and shopping malls. As of year 2012, natural gas is accessible almost in all 80 cities of Turkey.

The total capacity of CHP plants reached 8.300 MW (end of the first period of 2013). Turkey's total installed capacity reached 60.000 MW and the produced electricity was 240 TWh by the end of 2012. The numbers show that the share of installed capacity of CHP in the total capacity for generation of electricity is 14%.

CHP by Sector

The industries of textile, iron and steel, ceramic, paper mills and chemical plants are target sectors in Turkey for the implementation of co-generation systems. On the other hand, organized industrial zones and custom free zones are the most appropriate industrial facilities where co-generation or combined co-generation design and technology can be implemented successfully.

Incentives for Co-generation

In Turkey the high efficiency co-generation systems can benefit from the following incentives:

- Connection to the grid will be a priority
- High efficiency CHP facilities don't need to provide license from EMRA (Energy Market Regulatory Authority),
- The micro co-generation units don't need to hold a license. However their connection to the grid is still under discussion since new private owners of the electricity distribution system prefer to deal with larger customers
- If the co-generation system uses biomass, an additional cost of 0.4 US$ cent/kWh for steam, 2.0 US$cent/kWh for gas turbine and 0.9 US$cent/kWh for internal combustion engine and Stirling engine will be added to the feed-in tariffs.

Barriers to Co-generation

Energy distribution company (TEDAŞ) and state energy whole sale company (TETAŞ) are not obliged to buy surplus electricity from CHP facilities. Instead of that, co-generation producers can sell the surplus electricity in free electricity market.

Accessibility to the Natural Gas

The main energy source for any co-generation plant is the natural gas. 85 % of the total gas consumption is supplied by Botas. Botas transmits the gas to the RMS station of city gas distribution systems. So that gas can be accessible all around Turkey.

Situation of Natural Gas

Public gas concern Botas dominates the gas market and the gas prices to IPP operators is higher than the EU average (app. 12 $/MM Btu). Although gas is accessible everywhere, the free gas market has not been established yet. Botas does not guarantee to meet demands of new CCGTs and Co-generation plants. The rapid increase of the natural gas demand forces Botas and licensed gas importers (private) to search for new gas suppliers. The construction of new LNG degasification terminals and underground storage is not available.

Development of Micro Co-generation

On July 21, 2011 Turkish Parliament enacted the law "License Exemption for Production of Electricity Legislation" in order to support the establishment of plants producing power below 500 kW by using renewable energy sources and to introduce a new technology which was called micro-co-generation which has a total capacity of 50 kW and below to provide heat and electricity for residential buildings. In this Legislation, the recognition of Micro – CHP was defined as a technology which has an installed capacity of 50 kW and below. It will produce both electricity and heat together to enable the usage by the households in a large extent. Furthermore, the Micro-CHP users are exempt from needing a license from EMRA. According to this legislation, surplus energy which is produced by Micro-co-generation unit will be supplied to the system within the scope of the tariff evaluated by distribution companies holding retail licenses. Today this tariff is applied as the lowest price of feed in tariffs (7.3 $cent/kWh) for renewable energy sources stated in "Renewable Energy Sources Law Nr: 5627". Since the date at which the legislation was published, there is not any progress or acceleration in the use and application of Micro-CHP in Turkey. The high costs of Micro-CHP units and lack of technical knowledge to connect the buildings with the grid are the main reasons of the lack of development of this technology for investors.

Conclusion

In conclusion the future for co-generation applications in Turkey can be summarised as follows:

- Co-generation based investments are slowing down in Turkey. The existing incentives for investment have been cancelled in 2012.
- The legal frame work for auto-production was annulled by the new law Nr. 6446. It is replaced with IPP system. However the rights of old units are preserved.
- Law for the unlicensed power production systems has been enacted. According to the new law (Nr. 6446) high efficiency co-generation investments will be exempted from the license obligation.
- Despite increasing gas prices, CHP undertakings are developing rapidly especially in shopping malls, hospitals, university and school campuses, hotels, holiday resorts.
- Turkey needs to enact "Electricity Directive" in order to decrease the existing efficiency limit of 80 %. Otherwise only a limited number of CHP facilities can be officially accepted as "high efficient CHP".

In addition, for the nuclear new-build's confirmed for Akkuyu and planned for Sinop co-generation applications are not explicitly discussed or part of the respective tender. But this may change during the implementation process of the projects.

Note: The co-generation capability will come along with a higher level of public hearing than for a normal NPP for electricity production because public involvement is mandatory and private investors would be part of such a project. If co-generation capacity would be added later on to the NPP and no safety related issues are feasible and safety cases are checked to be not affected by this add on, the licensing procedure would request a lower level of public involvement.

3.10 Experiences from the EUROPAIRS project

The objective of the EUROPAIRS project was to identify the boundary conditions for the viability of nuclear co-generation systems connected to conventional industrial processes and to initiate the partnership of nuclear organizations and end-user industries, which would be deployed in a further step to develop a demonstrator, coupling a (very) high temperature reactor ((V)HTR) with industrial processes. The boundary condition framework defines technical, industrial, economical, licensing and safety requirements for the nuclear system, the processes that can consume the energy generated and the coupling system.

EUROPAIRS aimed at adding to the knowledge and experience that has been built up in recent (V)HTR related programs by the partners of the European high temperature reactor technology network (HTR-TN) in the Fifth and Sixth Framework Programs (FP5 and FP6 respectively), as well in developments performed on national or company bases, the assessment of the viability of a (V)HTR deployment in concrete industrial environments formed by the end users /2/.

Main findings on the performance window of the HTR technology regarding process heat utilization:

- end user process requirements can be split in medium classes: steam (150 – 600 °C), chemical (600-900 °C), mineral (>1000 °C)
- HTR (modular) nuclear system characteristics (power limited to about 600 MWth per unit, block type): passive decay heat removal capability, inherent safety features, reduced power density, process heat limited by the heat transfer from the primary He coolant (SG well established, gas-gas IHX feasible in medium term, gas-salt IHX may feasible in long term)
- avoiding tritium contamination of the process heat circuits and transient feedbacks from the process heat side by applying a tertiary circuit, environmental impact study as part of the licensing process will cover co-generation processes and products
- HTR specific design basis transients are more stringent than process side transients so critical impact on HTR operation by process side feedback not expected
- confidence by industry on HTR technology to be mature
- feasible near term condition: heat transfer by steam coupling, already considerable CO_2 emission reduction achievable in industrial application for this steam temperature level
- Safety and licensing aspects arising from the coupling of an HTR to a conventional end-user facility by heat/steam transfer: avoidance of mutual feedback, critical distance between facilities regarding individual accident scenarios to be considered, fulfilment of Tritium contamination limitations on the conventional end-user side,
- nuclear sites already located in industrial environment, IAEA methodology available for extended external hazard evaluation, HTR with smaller exclusion zone than LWR,

Main findings on the industrial heat market situation:

- combined heat and power supply for industrial facilities by separate HTR, mostly < 550 °C
- internally operating/embedded HTR in industrial facilities (as boilers, burners)
- hypothetical submarkets: co-production of base raw materials (e.g. industrial gases), pre-heating for high temperature application
- economics of nuclear co-generation in contrast to conventional heat/steam suppliers mainly influenced by the thermal match of heat production and consumption, electricity production of less importance
- risk assessment: most risky during construction (delay, overcosts), very low after commissioning, political and public acceptance risks to be minimized by adequate site selection procedure; in contrast to conventional co-generation risk at operation because of price and supply of gas and CO_2 generation

Main conclusions are, beside the technical aspects, performance and safety characteristics of an HTR, that significant effort is required on the public acceptance. This comprises a reference emergency planning for the area covering nuclear and conventional facilities.

Further technical improvements should address higher temperature levels for the heat process widening the scope of industrial applications.

Besides investigations on requirements and characteristics of relevant industrial processes and the nuclear HTR system, safety and licensing compatibility of the coupled system economic perspectives and target costs, also the deployment of HTR based co-generation in Europe and in the world was evaluated in detail /1/. Next steps towards a nuclear co-generation commercial deployment are summarized in a roadmap; focus on different levels of detail:

- the establishment of a HTR prototype, demonstrating the technical, financial and licensing feasibility (prototype roadmap)
- a broader perspective on risks and critical paths of the prototype, R&D requirements, international cooperation to drive licensing and cost reduction (deployment roadmap)
- a global route towards large scale HTR applications supporting communication processes and strategies for implementation of HTR systems (SNETP roadmap)

For the development of these roadmaps certain assumptions were made:

- HTR should be considered as replacement in a fossil fuel base industrial energy infrastructure (“plug in” strategy)

- delivery of steam at various pressures and temperatures
- economic viability, reliability and availability (e.g. redundant 300 MWth HTR per site)
- extended flexibility by electric co-generation capability
- application of “off the shelf “ technology minimizing risks and deployment duration
- replaced fossil fuel infrastructure kept for backup in case of HTR outage

The roadmaps are based on current-day technical and licensing standards and EUROPAIRS industry specifications. From the economic and the market study it is concluded that HTR is only in the range of economic competiveness if the high temperature heat is used and the power level is maximized. This is in contrast to the need of redundancy which supports several smaller HTR. This conflict is not finally solved and requires more elaboration.

Because end users operate their plants globally an adequate global HTR licensing approach and technology basis is required. After initial demonstration of the HTR performance the entering of the plug-in market would be supported by the potentially very large application possibilities of the HTR technology, in particular in the field of hydrogen production (see effort e.g. at South-Korea (NHDD), Japan (HTTR), USA (NGNP)).

Major R&D developments have been identified focusing on the medium and long term future, e.g.:

- fuel performance optimization (modelling and experiments),
- advanced safety testing to support licensing (e.g. air ingress, water ingress),
- high temperature materials and fuels,
- medium temperature hydrogen generation technique, based on an innovative membrane reforming process
- advanced primary system purification system
- recommendation to apply an intermediate coolant circuit resulting in two heat exchanger barriers to the end user

With regard to the connection of the nuclear community and the non-nuclear industry essential for a HTR co-generation system this relation has to be strengthened by investigations of business modelling, site selection, fixation of system specification and in general to improve end users confidence in the potential of the HTR technology. To overcome additional financial burdens correlated with a prototype system such a project should be supported within the European financial framework program Industrial Initiative ST-plan framework.

Based on the findings of the analysis the first, a very detailed roadmap for a commissioning of HTR prototype within 10 years was specified. The 10 year time frame for establishing an Industrial Initiative to commissioning of a prototype was one of the initial boundary conditions of the EUROPAIRS project. In conclusion it was found that the 10 year time frame may be rather tight, the total prototype cost would be in the range of 2-3 billion € and major technical risks are identified with impact on cost and planning (vessel ordering, fuel manufacturing in industrial scale). In addition, site selection and a fixed system specification are needed in the very beginning stage of the project.

From the prototype roadmap a higher level deployment roadmap was derived, specifying the time needs of:

- an adequate European Framework until 2021 (R&D programs to maintain the knowledge base and in support of prototype and longer term perspectives beyond 2015),
- international cooperation’s beyond the R&D phase until 2031 (prototyping, global licensing approach until 2024, satisfying present end user needs, CO2 target reduction, development of enhanced HTR applications as coal-to-liquid, hydrogen production and advanced steel making),
- the HTR prototype industrial initiative (e.g. detailed design in 2014, construction in 2020, commissioning in 2022, in operation: 2022),
- process heat issues (design of transport and distribution system by 2017, construction by 2020) and
- the commercial HTR market deployment (site selection 2 years after HTR prototype in operation, commissioning by 2030, in operation in 2031).

The different steps of the prototyping should be accompanied by corresponding licensing steps (reactor design, SAR, environmental assessment, process heat application system). A subsequent international or at least WENRA work group based generic licensing (site independent) should be envisaged based on the completed HTR prototype and in particular addressing the interface to industrial processes. A documentation equivalent to the GIF safety design criteria report for SFR'S could be a valuable first step in this direction. For co-generation no specific licensing requirements are identified but covered by the IAEA safety guidelines for new reactor systems.

As a major aspect the deployment roadmap covers the economics of HTR's based on business models and market strategies which address cost estimations, modular approaches, market perspectives, business models of the operator, energy costs and further information determining costs, prices and uncertainties. As most economic aspects EUROPAIRS identified the following topics:

- Redundancy, i.e. multiple HTR units per site, leading to moderately sized HTR systems; needs to be optimized against the economic advantages of maximised HTR units,
- co-production of electricity and heat improving the flexibility of the system, economy of solely electricity production by HTRs may hardly competitive,
- application of "off the shelf" as much as possible minimizing risks and deployment duration
- combination with existing fossil fuel energy infrastructure (if possible), in particular an advantage for backup in case of HTR outage, "plug in" replacement feasible and needed in the next decades
- steam condition of moderate coolant HTR's are satisfying for actual needs, minimizing time and cost for R&D

Finally challenging aspects regarding the global prototype planning time schedule are listed as RPV manufacturing and fuel fabrication capacity.

The set of roadmaps is completed with the high level SNETP roadmap addressing communication purposes and the most relevant and important aspects of HTR co-generation deployment. In particular the following items are considered:

- End user involvement, timing and organization
- Timing of the prototype, connection to the Industrial Initiative
- International Cooperation, timing and organization
- Subsequent steps, in relation to international trends and opportunities

In conclusion the three sequential roadmaps developed in the EUROPAIRS project address the details of the HTR prototyping, higher level constraints of a successful HTR market deployment and general aspects of an HTR nuclear co-generation deployment. The information and conclusions of these roadmaps provide a very good knowledge base for subsequent investigations and are valuable input for specific discussions on the licensing issues of co-generation HTR facility addressed in chapter 5. With respect to the time schedule for the commercial HTR market deployment prototyping, licensing and N-of-a-Kind HTR's new-builds need to be closely connected.

Regarding medium and long term (V)HTR applications, experiences of the conventional industry to high temperature alloys performing well under (V)HTR temperature conditions could be used by the nuclear industry.

Regarding the limiting radiation protection levels for co-generation installations promising results for low tritium diffusion heat exchanger design and tritium reduced primary coolant are observed, but the issues are not yet finally solved. Meanwhile, to comply with the requested low contamination levels at the end user side (note: In France artificial radioactive elements in by-products only allowed with exemption statement of the Authority), an intermediate coolant circuit between the primary coolant systems and the conventional end user systems is still recommended /1/.

4 Recommendations for safety assessment and licensing

Based on the experiences gained in the past on nuclear co-generation capabilities (mostly LWR's) in this chapter the experiences are reconsidered in the light of a HTR co-generation setting. Which experiences can be adapted directly for HTR's? Which can be modified accordingly or may in general be worthwhile even if not directly applicable? And which experience may have no impact at all in an HTR co-generation setting?

Based on the answers of these questions recommendations to safety assessment and licensing issues for HTR co-generation applications are derived. If available domestic visions on licensing requirements for planned HTR with co-generation capabilities are presented.

Recommendations for safety assessment and licensing for nuclear co-generation in a global scale should be correlated with the licensing issues of the prototype. This very important task is addressed in major road map steps of the EUROPAIRS project (/1/) too:

- Development of a generic and global HTR licensing approach to ensure risk and cost reduction for subsequent systems following the prototype.

  Applicants will need a common basis for safety requirements and safety approach, the subsequent development of a way to assess the safety demonstration of the operator ("licensing approach") will be the responsibility of the respective national safety authority. Once a complete safety report is written following the IAEA recommendations, various types of documents required by the authorities can be derived from this safety report.
- Generic licensing for co-generation with HTR systems, based on the IAEA process of drawing safety guidelines for new reactors

Regarding local safety assessment and licensing, no specific requirements for co-generation applications are identified. But it is recommended to pay special attention to the following issues:

- strong radiological interface to the coupled conventional systems because of transfer media:

  intermediate heat exchanger needed: for long distances salt or steam are proposed because of the lower heat transport costs, for closer distances gas-phase media feasible (He, N_2) but would come along with an additional risk for a secondary blowdown which could overcool the primary coolant with subsequent loss of the operational heat sink); IHX conflict of temperature loss vs. additional barrier for radioactive release/contamination needs to be managed
- minimum distance between the nuclear island and the end user installation:
  - consideration of generic hazards proposed to cover different chemical flow sheets /19/
  - for hydrogen production consider additional hazard analysis for additional end products as oxygen (increased explosion risk, oxidation level) or intermediate products (as H2SO4 in particular in concentrated state, corrosion, breathing risk if accidentally released), NGNP down selection report proposes "to conduct both a standard HAZOPS analysis and USNRC SRP/Reg Guide evaluations (NUREG-0800) associated with use, spills, transport of especially sulfuric acid, SO_2, SO_3". /5/
  - hydrogen release less important for external events of NPP because of fast dilution and if still exploding, low thermal flux (/19/, p. 6), in addition a massive knowledge base for hydrogen accidents is available
  - accidental release of heavy (and may be corrosive and/or toxic), ground hugging gases and oxygen are of large concern because of potential major off-site consequences,
  - concentrated sulfuric acid, SO_2, SO_3 (SI, HyS process), H_2, O_2 at high temperatures and pressures (/13/, Tab. 5-3) needs to be considered regarding: explosive, flammable, corrosive or toxic clouds and associated exploding rotating machinery (compare turbine orientation and protection)
  - large oxygen release at normal operation (proposed) has to be considered regarding plume and effect on safety related SSCs /19/, chapter 4.1.2
  - Safety philosophy of NPP and chemical plans fundamentally different /19/:
    - Chem. plant: dilution of hydrogen below burning limit -> outdoor construction, controlled inventory size, adequate separation distances between process and storage facility
    - NPP: Contain radioactive material -> housing recommended

- NPP protected from chem. plants only by reactor building and separation distance, housing of chem. plant not creditable (some additional provisions can be taken either on the nuclear or non nuclear site like ground banks, walls etc.)
- in conclusion of /19/: specific characteristics (material characteristic, temperature, pressure), site layout (distance, barriers for direct impact, wind conditions) and maximum stored inventory can provide a good starting point for safety assessment
- additional safety case: NPP events which impact the chemical plant which in turn may affect the NPP safety SSCs /19/
- standard review plan on fire protection (NUREG-0800)

For economic reasons the following issues has to be considered

- close distance to the end users installation to minimize heat loss
- base load performance, high availability
- backup solutions for the cogenerated energy in case of reactor shutdown

4.1 Feedback from the joint WP2/WP3 questionnairs regarding safety and licensing

This chapter summarizes the information related to safety and licensing from the feedback of the WP2/WP3 questionnaires (see report D2.11 /93/).

As part of the WP2 a questionnaire list was created to identify amongst others the safety and licensing related experience from nuclear installations (realized or only projected) with co-generation or process application. The following facilities/projects were identified and facilities were asked to answer the questionnaire:

- BUNA AG and Leuna-Werke(Germany, former GDR)
- BASF HKV (Germany)
- HOECHST AG / Bayer AG (Germany)
- LURGI AG (Germany)
- HTR-MODULE for the Oil Industry, Hannover (Germany)
- HTR-10, INET (China)
- AVR (Germany)
- THTR (Germany
- Temelin, (Czech Republic)
- Ågesta, AB Atomenergi (Sweden)
- Paks (Hungary)
- Loviisa3 (Finland)
- Bohunice (Slowakia)

Based on the answers to safety and licensing (chapter 5.5 of D2.11) for co-generation or process heat application it can be concluded that:

- Not an explicit licensing issue, initial co-generation capability covered by the usual licensing procedure
- Usually no safety adaptions for the NPP required, Finnish regulator in general requests a new operation license based on the corresponding safety assessment and new environmental impact assessment if co-generation is implemented later on
- Authorization by the regulatory body needed
- Tritium concentration and radiation level in the tertiary circuit and in the final product or service was considered and monitored, partially the artificial radiation monitoring of the secondary circuit was considered to be sufficient for the radiation protection of subsequent systems, Finnish regulator considers two physical barriers and in minimum one pressure grading to be sufficient to avoid radioactive release to the tertiary circuit in case of leakage

- No or only very limited impact on the environmental footprint of the NPP, sometimes a positive impact was identified because of the reduction of CO_2 and NO_x emission, conventional accident scenarios to be considered for the added heat transfer pipeline

In conclusion no co-generation or process heat specific licensing issues beyond the standard licensing requirements of the respective regulatory body are faced. Compliance with the radiation activity limits in the end user product or service is provided by radiation monitoring of the secondary and in the most cases also of the tertiary system. The negative impact on the environmental footprint of the NPP is considered to be negligible/very limited or even positive because of the reduction of CO_2 and NO_x emission. No critical or even significant issues are identified on safety and licensing of NPP co-generation application.

4.2 Applicability of the experiences for HTR cogen setting

This chapter describes the applicability of past/present LWR co-generation experiences for an HTR co-generation setting.

4.2.1 Licensing issues for LWR co-generation installations

Important licensing issues for LWR co-generation installations are the following (to be extended):

- Iodine content in secondary coolant circuits is an important issue because of the acceptance of small leaks in SG and the license to further run the SG with such a leak. If leaking occurs between the secondary and the tertiary coolant circuit and lower pressure in the tertiary circuit this would result in a contamination of the co-generation interface/transfer line leaving the NPP to the end user. This contamination issue is even more important for the tritium generated in the core from the radiological point. Because in water circuits tritium is bounded as HTO (tritiated water) the contribution to the tritium fraction in the secondary system by diffusion of gaseous tritium is comparably small.
- Neutron reactions on boron and lithium are a main contributor to the coolant tritium fraction

4.2.2 Licensing issues for HTR co-generation installations

Based on licensing experiences from LWR co-generation installations these experiences were evaluated with regard to HTR licensing constrains and boundary conditions.

Iodine concentration

Iodine concentration in secondary and tertiary circuits will be no licensing issue for the HTR licensing because of the pressure graded operation condition (secondary circuit runs with higher pressure than the primary circuit). Any leakage will result in an insertion of media to the primary circuit. While for a He-He heat exchanger a small leakage may be acceptable at normal operation condition a small leakage in a SG will result in a water ingress accident scenario and therefore not acceptable for normal operation. But in any small leakage case iodine fractions from the primary circuit will not be transferred to secondary or tertiary systems because of the pressure graded operation condition.

Tritium content

For the licensing of a HTR co-generation installation the contamination risk by a certain tritium level in the coolant gas on the heat transfer medium and finally on the final product are the issues to be considered. Previous concepts of a single interface of the primary coolant directly to the final product (single heat exchanger concept) are no yet finally approved with regards to sufficiently low tritium permeation characteristic despite investigations and experiments already demonstrate promising results. For the licensing of a prototype HTR therefore a single heat exchanger application will be not the reference concept. Future investigations and tests may demonstrate a single heat exchanger concept for commercial HTR facilities but the decision for a two or three cycle system finally depends on the intended type of application and the respective regulations. It should be noted that depending on the nationality of the regulatory body for the maximum allowed tritium concentration in the end IAEA radiation protection limits or the natural tritium background levels may be applied.

Regarding the tritium generation by fission and subsequent nuclear decay the tritium production rate is correlated to the nuclear fuel applied and the targeted final burn-up. For HTR and U-235, U-233 or Pu-239 fuel compositions the most of the tritium will be created by a ternary fission reaction (e.g. U-235 (n, fission product) H-3) and to a lesser extent from n-capture induced decay of Li-6 (n, He-4) H-3. Altogether these reactions create about 50 % of the total tritium /7/. Most of this tritium fraction will be retained by the fuel coating. So for sufficient reduction of the coolant tritium concentration the contribution from boron and lithium reactions have to be minimized or retained and the He purification systems should be optimised.

Regarding the total tritium content and the coolant tritium level in the primary coolant circuit several differences have to be considered between HTR and LWR conditions:

- Compared to LWR for HTR a higher burn-up of the nuclear fuel reflected by a higher total number of fission reactions is intended. This corresponds to a higher total number of ternary fissions which finally results in higher tritium content the coated particles.
- The He purification system of a HTR is significantly more effective than the ion exchanger of an LWR regarding the retention of tritium.
- For a pebble bed HTR the excess reactivity is significantly smaller than for an LWR, therefore the tritium production from boron or lithium reactions (applied as burnable poison) is smaller than in an LWR. The low tritium fraction would be further supported if the coolant flow from the separately located control rods would be filtered by a higher bypass ratio. (Note: The He purification system continuously filters a certain fraction of the coolant inventory e.g. for HTR-MODULE 5% per hour)

  Because of the coolant gas, the tritium production rate from neutron-coolant interactions is smaller in HTR coolant than in LWR coolant because in an HTR only ^{3}He is present as a target while in PWR coolant boron is diluted. But this contribution to the overall tritium level is rather small.
- Because of different operation conditions in a HTR compared to a LWR (graphite fuel matrix and structures contaminated by Li, higher temperature) the safety related impact of a certain hydrogen concentration (and the corresponding tritium fraction) in the coolant gas and the purification system need to be re-evaluated (significant adsorption effect of tritium in hot graphite while desorption effect for cold graphite) /7/. But these issues are related to the safety performance of the HTR itself and not to a co-generation installation.

Additional aspects

For HTR co-generation installations no specific licensing requirements on tritium levels beyond the corresponding radiation protection and standard NPP licensing requirements are defined. Initial evaluations of the contamination levels of the primary coolant, the heat transfer medium and the final end user product or service can be done based on LWR experiences, regarding tritium permeation of course on the basis of HTR HX temperatures, because for the HTR Demonstrator at least three radiological barriers (coating + heat exchangers) are foreseen between the main source (ternary fissions in the fuel kernels) and the end user system. But because of the higher temperature levels in a HTR the permeation performance of tritium through a HTR graded heat exchanger material barrier has to be confirmed (see Archer report D23.71 /95/).

4.3 Domestic visions on licensing requirements

This chapter describes the point of views of national safety authorities, regulators and other nuclear experts on the licensing requirements if an HTR with co-generation capability would be proposed in their country.

- ➔ How the national safety authorities could proceed facing such a project?

4.3.1 Regulation in Lithuania

Heat and power co-generation in Lithuania is regulated by Energy Law /32/, the Law on Electricity /33/ and the Law on Heat Sector /34/. Legislation in district heat supply market, as one of dominant in Lithuania, is well developed. The main authorities regulating the energy sector are the Government, the Ministry of Energy, the Ministry of the Environment, the National Control Commission for Prices and Energy (NCC), the State Energy Inspection and the Council of the 60 Municipalities in Lithuania. The Ministry of Energy implements the State policy in the energy sector, drafts and approves legal acts. The Ministry of the Environment decides on issues relating to environmental protection, construction and fulfils functions, organises and carries out monitoring of environmental effects in increased pollution areas of the energy sector. The National Control Commission for Prices and Energy (NCC) approves methodologies and procedures for setting state regulated prices; approve charges for connection of energy facilities; grant licenses for companies. The State Energy Inspectorate exercises state control over energy facilities and establishes the technical reliability standards for equipment. The management of the energy sector at the municipal level, is allocated to the individual Municipalities. Within their respective territory, municipalities organize the supply of heat to customers including heat price setting, according to methodology issued by NCC.

A municipality, following the heat sector special plan, organises the supply of heat to heat consumers according to their needs for heating and ventilating the premises and for hot water preparation. A heat supplier must hold a licence for the supply of heat. The procedure and rules for issuing licences are approved by the Government. Licences for the supplier of heat supplying at least 10 GWh of heat per year, taking into account the recommendations of the municipal institution, are issued; their validity is suspended and the licensed activities are controlled by the NCC. Licences for the supplier of less than 10 GWh of heat per year are issued, suspended and cancelled and the licensed activities are controlled by the municipality. Licences for the activities involved in heat supply are granted for an unlimited amount of time for one person only in certain specified territories /35/.

The Law on Heat Sector /34/ regulates the activities and responsibilities of state institutions, relations between heat suppliers and consumers, heat price setting procedure and other issues of regulation.

The Law on Electricity /33/ defines public interests in the electricity sector, which means any act or omission in the electricity sector, directly or indirectly related to the public security, environmental protection, and to electricity generation from renewable energy sources, waste or combined heat and power generation. In the list of public interest services in the electric power sector (hereinafter referred to as 'the VIAP') the Government stated that thermal production of electricity in co-generation power plants supplying heat to urban centralised heat supply networks is a service of public interest in the electricity power sector. The Rules establish the general terms and conditions for the provision of electricity sector services related to public security, environmental protection, and to electricity generation from renewable energy sources, waste or combined heat and power generation, and they also regulate the requirements and obligations of the holders of the licence for electricity supply and the operators of the market, transmission and distribution networks to provide these services. The Law on Heat Sector /34/ lays down that co-generation of heat and electricity is a service of public interest.

Regarding **nuclear sector**, in accordance with national legislation and international conventions and treaties, the Republic of Lithuania undertakes appropriate measures to ensure the safety of nuclear installations under its jurisdiction through the establishment of legal framework and infrastructure necessary to maintain the effective nuclear safety regulatory system. The main laws which set out the framework governing the regulation of nuclear energy are:

- Law on Nuclear Energy;
- Law on Nuclear Safety;
- Law on Radiation Protection;
- Law on Radioactive Waste Management.

These Laws are supported by the main Governmental decrees that include legally binding regulations. The term of the nuclear power plant is defined as "*a set of equipment and buildings for the generation of electricity or electricity and heat from nuclear fuel*" in national legislation, and that refers to the use of the nuclear co-generation.

The Law on Nuclear Energy /36/ provides the mandate to Regulatory Body VATESI (State Nuclear Power Safety Inspectorate (to draft and approve the requirements and rules for nuclear safety, radiation protection in the area of nuclear energy, accounting for and control of the nuclear materials, physical protection of nuclear materials the quantity of which exceeds the well-defined quantity. These requirements and rules are mandatory to all the state and municipal authorities, also to all the persons engaged in such activities.

The Law on Nuclear Energy /36/ and the Law on Nuclear Safety /37/ are the main laws that together with the Law on Radioactive Waste Management /38/ and regulations made under these laws establish the licensing system for activities related to nuclear safety of nuclear installations during following life-stages: site evaluation, design, construction, commissioning, operation, and decommissioning. VATESI is a competent authority for the licensing of activities related to nuclear safety of nuclear installations. The following types of licences and permits, related with nuclear installations, are established in order to be issued by VATESI /39/:

- construction licence;
- operation licence;
- construction and operation licence;
- decommissioning licence;
- permit for first carry-in of nuclear fuel to site of NPP or non-power nuclear reactor;

- permit for the first carry-in and testing of the nuclear installation using nuclear and/or nuclear fuel cycle materials;
- permit for first start-up of NPP unit or non-power nuclear reactor;
- permit for industrial operation of the nuclear installation;
- permit for start-up of the nuclear reactor after its short-term shutdown.

Nuclear installation construction licence may be granted only if the Parliament of Lithuania (in case of NPP) or the Government of Lithuania (in case of other nuclear installations) has adopted a legal act on the nuclear installation and State Territorial Planning and Construction Inspection under the Ministry of Environment has issued a permit for construction of the nuclear installation.

As stipulated in the Law on Nuclear Safety /37/, licences and permits shall be issued to legal entities or persons having sufficient technological, financial, management system, human, emergency preparedness, physical security capacities, capacities for safe storage, transportation, accounting for and control of nuclear materials meeting the provisions of IAEA and EURATOM for safeguard, allowing proper fulfilment of the conditions of the licensed activity and ensuring nuclear safety. VATESI is empowered to suspend, amend, revoke suspension and revoke licences and permits that are issued by VATESI.

The Law on Nuclear Safety /37/ stipulates that a licence shall be issued for an unspecified period until the licence is terminated. Nevertheless, it is defined that reassessment of safety at a nuclear facility shall be carried out at regular intervals, at least every 10 years, in order to provide evidence that the nuclear installation is in compliance with the current design basis and to identify further safety improvements by taking into account ageing issues, operational experience, most recent research results and developments in international standards. At this point, after review and assessment VATESI may decide to suspend or amend the licence using established procedures. Full responsibility for the nuclear safety of a nuclear installation solely falls on licence holders. See the following chapter for respective regulations of the regulatory body VATESI.

4.3.1.1 Lithuania: NPP new build

The series of various studies were carried out in order to answer the question what energy sources should replace the lost nuclear electricity capacities in Lithuania (/42/, /43/). The results of various studies concerning the future structure of power plants in Lithuanian energy system have shown that looking from the economical point of view the best option to replace Ignalina NPP are new nuclear unit, new combined cycle condensing units together with the existing and new units of Combined Heat and Power plants (/43/, /45/). As was mentioned above due to climate conditions in Lithuania, district heating presents a notable fraction of energy consumption in winter months, and infrastructure is available in most population centres in Lithuania. So, in the future new co-generation units are likely to be the best alternative for electricity and heat generation in Lithuania.

Taking into account the ongoing nuclear projects (the Visaginas NPP project and decommissioning of the Ignalina NPP) a legal and regulatory reform, in line with the international requirements, is being initiated. The package of new or revised nuclear field related laws were adopted by the Parliament in 2011. VATESI implements annual plans for updating regulation documents according best international practice and changing situation in Lithuania.

Regulatory Body of Lithuania VATESI currently is preparing the nuclear safety requirements "*Design of Nuclear Power Plant*" for new power plant /46/. These requirements are based on IAEA SSR2/1 /47/, includes provisions from WENRA, as well reflects provisions of European "stress tests" specification, prepared reacting to Fukushima Daiichi accident. In the draft version of this document /46/ concerning main requirements for systems used for heat production (co-generation) it is said that:

nuclear power plants coupled with heat utilization units (such as for district heating) shall be designed to prevent processes that transport radionuclides from the nuclear plant to the district heating unit under conditions of operational states and in accident conditions.

VATESI expressed its position on potential development of nuclear co-generation in Lithuania.. The questions (prepared by LEI staff, partially using questionnaire developed under WP2) and answers are presented below.

1. What licensing requirements would be applied for a new NPP with co-generation option? What additional safety objectives/criteria and what specific safety assessments or authorizations would be required?

   VATESI: In the 2nd article of the Law on Nuclear Energy of the Lithuanian Republic /36/ it is stated that “the nuclear power plant is a complex of equipment and buildings intended for generation of electricity and heat by using nuclear fuel”. In such a way it is already foreseen the possibility for co-generation during construction of the new NPP. For NPP co-generation equipment would be applied the same nuclear and radiation safety requirements, criteria, safety assessment principles, licensing and authorization rules as applied for designing, building and operating NPP for electricity production. In nuclear safety requirements “Design of Nuclear Power Plant” which currently is under preparation by VATESI /46/ it is noted that nuclear power plants coupled with heat utilization units shall be designed to prevent processes that transport radionuclides from the nuclear plant to the district heating unit under conditions of operational states, in accident conditions and even during severe accident conditions. Requirements for nuclear safety assessment would be the same as foreseen in the Law of Nuclear Safety of the Lithuanian Republic /37/.

2. If the co-generation will be implemented after Visaginas NPP commissioning, what additional safety objectives /criteria and what specific safety assessments or authorizations would be required?

   VATESI: If the co-generation will be implemented when Visaginas NPP would be in operation (after commissioning), for that case the nuclear safety requirements “Categories of Modifications at Nuclear Facilities and Procedure Regulations for Making these Modifications” approved by VATESI head /52/, and other nuclear safety requirements and rules, which will be applied for new NPP project, would be taken into account. Requirements for nuclear safety assessment would be same as foreseen in Law of Nuclear Safety /37/.

3. What are the key stages (safety report, construction licence, etc.) of the licensing of VNPP installation and what authorities would be involved in case of co-generation?

   VATESI: Co-generation has to be foreseen in the final NPP design. Licensing stages, required licensing documents and institutions involved in licensing processes are defined in Law of Nuclear Energy /36/, Law of Nuclear Safety /37/, Law of Construction /53/ of the Lithuanian Republic and other accompanying legislation, and in “Rules of Procedure for Issuing Licenses and Permits in the Area of Nuclear Energy” /54/ approved by the Government of the Republic of Lithuania.

4. What main initiating events have to be considered during safety assessment of nuclear facility with co-generation option?

   VATESI: Any specific initiating events which are related to co-generation are not predefined. In the nuclear safety requirements of “Design of Nuclear Power Plant”, that is in preparation phase /46/, there is provision that during NPP design stage all the possible initiating events (internal events, internal and external hazards and intentional actions) must be investigated applying systematic approach. Based on those events the list of initiating events should be prepared, and application of this set of events during the designing of NPP should assure the acceptable safety level of NPP.

5. Are there any specific criteria and radiological control requirements applied for the heat transfer fluid which will be provided for end user (district heating system, industry enterprise, etc.) from VNPP?

   VATESI: The radiological characteristics of heat transfer fluid coming from the NPP to the end user required to meet the free level criteria, i.e. heat transfer fluid cannot be contaminated with radionuclides as it is interpreted in hygiene norms.

6. What do you think, will the co-generation influence the environmental footprint of the nuclear plant greatly (effluent quantities, heat rejected through the cold source, quantity of steam released to the atmosphere, etc.)?

VATESI: VATESI do not regulate (effluent quantities, heat rejected through the cold source, quantity of steam released to the atmosphere, etc.) and do not supervise the environmental aspects. The radionuclides discharges from NPP must satisfy nuclear safety requirements BSR-1.9.1-2011 “Limits of Radioactive Discharges into Environment from Nuclear Facilities and Requirements for a Plan for Radioactive Discharges into Environment” /55/.

7. What do you think who in Lithuania should be responsible for a common safety of nuclear and industry/DHS complex, when they are physically connected together by process line (hot water, high temp. steam, and etc.)?

   VATESI: The license holder is responsible for NPP nuclear and radiation safety, including any operating activity connected with nuclear fuel cycle materials. For the heat consumers, being outside of the NPP site, uncontaminated (without radionuclides) heat transfer fluid should be provided, and they would not be the holders of licences or permits as indicated in the Law of Nuclear Safety /37/.

8. Is the VATESI legal basis ready for implementation of nuclear co-generation in Lithuania?

   VATESI: In the 2nd article of the Law on Nuclear Energy of the Lithuanian Republic /36/ it is stated that the nuclear power plant is a set of equipment and buildings for the generation of electricity or electricity and heat from nuclear fuel. In such a way it is already foreseen in the Lithuanian nuclear legislation the possibility and the conditions to construct the NPP with co-generation option. The main nuclear safety principles for such NPP are defined in the Law of Nuclear Safety /37/. If the new NPP project will continue to be developed, the legislation basis will be updated / supplemented by necessary legal documents, for instance, nuclear safety requirements “Design of Nuclear Power Plant”, “Recognition for operation of the Nuclear Power Plant”, “Nuclear Power Plant Operation” and others, where the nuclear and radiation safety aspects related to co-generation will be reflected.

4.3.1.2 Assessment of SMR application

The Lithuanian government is seeking to attract investment in the development of new co-generation power plant projects in most populated areas, as it looks to deliver affordable heat to residents of both cities. These projects are recognised as economic projects of national significance. It should be noted, that this plans are related to the employment of CHPs. Although heat supply sector is large and consisting of many district-heating systems, the district heating network is well developed and very attractive option could be the employment of Small and Medium type nuclear Reactor (SMR) in the new site close to the cities with large heat demand, like Vilnius or Kaunas cities where district heating systems supply heat for 80% building in the cities. In general it could be considered for small countries like Lithuania, as alternative for the big nuclear units due to limitation imposed by the grid size and available financial resources. Such analysis was performed during execution of IRIS project in which LEI took part.

Different scenarios of future energy system development were analysed during this study /48/. Economic modelling and optimisation was concentrating on evaluation of possibilities to construct a new energy source. In this study, the introduced approach was applied focussing on SMR, which could be one of the future energy source options in Lithuania. As an example of SMR, the IRIS (International Reactor Innovative and Secure) advanced modular nuclear reactor was chosen in this study (**Fig. 7**).

Fig. 7: IRIS reactor

The IRIS is a pressurised light water cooled, medium-power (335 MWe) reactor (/49/, /50/). The IRIS designing supports idea of licensing the power plant with reduced or even without the need for off-site emergency response planning /51/. This would allow IRIS to be treated as any other industrial facility, located closer to population centres. This allows better implementation of co-generation option and reduction of transmission costs.

Possible introduction of IRIS nuclear units into Vilnius and Kaunas cities district heating systems was analysed in the study. According study /48/ IRIS reactor could be built only for electricity generation at the existing site of Ignalina NPP in order to utilise existing infrastructure and to lower construction cost. The heat demand in this region is very small (population of Visaginas town near Ignalina NPP site is 20 000 only, no big industry is located here).

The results of the study showed that construction of SMR (IRIS) units is very attractive option (looking from economical point of view) for the future electricity generation in Lithuania. The scenario with IRIS co-generation units in two biggest cities (Vilnius and Kaunas) caused lowest total discounted cost compared to the case without nuclear energy source. Depending on initial conditions, up to five IRIS units could be built in Lithuanian energy system in the period up to 2025, and total installed capacity of IRIS units could be up to 1.7 GW. IRIS reactors can cover up to 50% Lithuania's electricity market and 51-75% of district heating markets in these biggest cities. In the case IRIS co-generation unit should be installed away from existing district heating networks (due to Emergency Planning Zones), the attractiveness of this unit is decreasing gradually with distance, because of investment cost and losses in addition district heating pipelines.

4.3.2 Hungary: The Paks NPP extension

According to the Act of 1996 CXVI on nuclear energy (hereinafter referred as Atv.), the use of nuclear energy in Hungary can only be made with continuous possession of necessary licenses defined by the relevant laws and permanent authority supervision. The licensing of the construction and operation of a nuclear power plant in the sense of nuclear safety, security and proliferation resistance is the dedicated task of the Hungarian Atomic Energy Authority (hereinafter referred as OAH) /18/. Other authorities can participate as specialist authority in the licensing process of OAH. The OAH is responsible for the fulfilment of the requirements of nuclear safety, security and proliferation resistance. The OAH doesn't investigate the fulfilment of other requirements (for e.g.: requirement for environmental protection) because other authorities have to investigate and ensure the fulfilment of other requirements. The OAH hasn't scope of authority in the field of energy policy. If the OAH would have such scope of authority then the OAH wouldn't be independent from those organizations which are interested in the utilization of nuclear energy /18/. More information about the OAH can be found at the www.oah.hu website.

According to the Atv. there are no specific licensing requirements on co-generation installations beyond the general NPP requirements in Hungary /18/. So, the tools for the licensing process are already known. The extension of the Paks NPP is in progress and can be used as the best example describing the licensing process for a HTR with nuclear co-generation purpose in Hungary.

4.3.2.1 The licensing process of Paks NPP extension

The OAH has started the first step, the site investigation and evaluation process in order to license new units at the site of the Paks NPP. The licensing process itself was initiated by the MVM Paks II. Atomerőmű Fejlesztő Zártkörűen Működő Részvénytársaság (MVM Paks II. Nuclear Power Plant Developer Exclusively Limited By Shares company, hereinafter referred as MVM Paks II LTD). This is the first phase from the two of the site selection licensing process /17/.

The clients in this site licensing process are:

- the applicant, the MVM Paks II LTD;
- all owners of immovable within the border of the affected zone around the site (Atv. 11/A §).

The affected zone consists of the area of the site and the area within 500 meter beyond the border of the site (Atv. 11/A § (2a)). There will be open auditions within the process of site licensing /17/.

4.3.2.2 Review of the licensing process of a nuclear installation in Hungary

The first step for the installation of a new NPP or the extension of an existing NPP with a new unit is a principled contribution (in the form of a Parliament decision) made by the Hungarian Parliament in advance. This principled contribution enables the start of the preparation work (Atv. 7. § (2)). This vote has been performed on 30 March 2009 with overwhelming success. So, the Hungarian Parliament has given the principled contribution in advance (25/2009 (IV.2) OGY decision) for the start of the preparation work in order to enable the extension of Paks NPP with new unit(s) /17/. Thus the preparation of the installation of Paks II NPP has been started. Paks II NPP has been foreseen to lays on the same site than the Paks I NPP locates.

A decentralized authority system works in Hungary. The currently operating Paks I NPP is under the supervision of not only one authority. Its reason is that an NPP is a very complex system and its licensing has many dimensions (for e.g.: nuclear safety and security, proliferation resistance aspects, environmental protection, water law aspects, country defence aspects, etc.) can be covered by multidisciplinary approaches. Next to the licensing in the scope (nuclear safety and security, proliferation resistance) of the OAH, licenses (for e.g.: for environmental protection, water law, operational etc.) issued by other authorities are needed as well. Each authority issues its license(s) in its own process and as specialist authority validates its own considerations in licensing processes of other authority /17/.

The OAH keeps tabs on the general directions of international development in the field of peaceful application of nuclear energy with a special emphasize on the development of international regulation. Based on this activity, the OAH proposes measures, necessary changes in current, relevant laws and new laws (Atv. 8. § (4)). The revised version of the Governmental order about nuclear safety of nuclear facilities and relevant authority activities (118/2011, VII. 11., Government order) has been issued in July 2011 as a result of a preoperational work of OAH. The Atv., the above mentioned Governmental order and its appendixes specify the nuclear safety licensing process of nuclear facilities in Hungary /17/.

The nuclear facilities (including the NPPs) are under permanent supervision of different authorities during each phases of their life-cycle in order to ensure the fulfilment of the safety requirements. Thus the licensing process adapts to the phases of the life-cycle of a nuclear facility. These phases of a life cycle of a nuclear facility can be seen on **Fig. 8** /17/.

Fig. 8: The life-cycle of a nuclear facility

The following nuclear safety licenses on a facility-level are required during the different phases of life cycle of a nuclear facility /17/:

a) for site investigation and evaluation (license for site investigation and evaluation);
b) for the characteristics and suitability of the site (license for the site);
c) for construction and extension (license for installation);
d) for set in operation (license for set in operation);
e) for operation and operation beyond the planed life-time (license for operation and license for life-time extension);
f) for modification of the NPP (license for modification);
g) for decommissioning of the NPP (license for decommissioning);

h) for termination of the NPP (license for termination).

Currently, the first step in the nuclear safety licensing process, the licensing of the site (in two steps, see step a) and b) in the list above) is under way in Hungary regarding the extension of Paks site.

4.3.2.3 The licensing process of the site selection for a nuclear facility

All licenses in the process of a nuclear facility installation can be described in details assisted by the Atv. /18/. In the following as an example, the currently ongoing licensing processes, the site selection for Paks II NPP will be showed in more details.

During the investigation and evaluation of a possible site of a nuclear facility, a certain site has to be evaluated concerning its suitability to host the nuclear facility. Furthermore, all characteristics of the site have to be recorded and defined which have to be taken into consideration during the design of the facility. The licensing for site investigation and evaluation is an individual licensing process performed by the client and the OAH /17/.

The aims of the licensing for site investigation and evaluation issued by the OAH are the following /17/:

- the definition of site characteristics to be systematic and performed by a well-established program;
- this program has to cover all relevant fields;
- all methods of the investigation and evaluation to be assessed and approved before the start of the investigation and evaluation.

The request for the license for the site can be submitted to the OAH only after all investigation and evaluation (specified in the license for site investigation and evaluation) have been performed. The complex research report based on the results of site investigation and evaluation has to be attached to the request for the license of the site. By the issue of the license for the site the OAH accepts the followings /17/:

- the absence of such site characteristics which exclude the suitability of the site to host a nuclear facility;
- the conformance of the process, methods applied and results achieved of the site investigation and evaluation;
- the proven suitability of the site.

The advices of OAH (in the form of written guidelines) on the methods and program to be applied during site investigation and evaluation and the suggested content of complex research report of the results can be downloaded from the web page of the OAH (www.oah.hu) /17/.

4.3.2.4 The revision of OAH on the requested licence for the site investigation and evaluation

The MVM Paks II LTD has submitted to the OAH the request (hereinafter referred as program) for license for site investigation and evaluation. The OAH investigates the suitability and completeness of the submitted program in the senses of proposed methods and theoretical considerations /17/. In brief: is the program well established or not?

The authority evaluation of the program covers the following field's /17/:

- geographic investigation and evaluation of the site;
- investigation and evaluation of dangers caused by humans;
- investigation and evaluation of earth science considerations;
- investigation and evaluation of geo-technique (for e.g. earthquake) and hydro-geology (for e.g. water beneath the earth level);
- investigation and evaluation of hydrology (for e.g. flooding, etc.);
- investigation and evaluation of meteorology (for e.g. storms, etc.);
- collection of necessary data for the evaluation of radioactive emission diffusion and designing of accident prevention.

The Mine Captain Authority of Pécs participates in the licensing process as specialist authority from the side of Authority for Mining and Geology of Hungary according to the 112/2011 (VII. 4., Government order). This authority investigates the geology, mining and technical considerations of safety /17/.

The OAH has the right to involve other authorities as specialist authority into the investigation and evaluation of the program if it is necessary (Atv. 17. § (3)) /17/. The time frame of the procedure is 21 days, but some actions do not count into this time frame (for e.g. the treatment of specialist authorities, etc.). The head of competent authority can extend the available time frame of the procedure but only once and with further 21 days (33. § of CXL. law of the year 2004). The OAH considers the suggestions and opinion have been presented during the open audiences and embeds them into the decision on the license. Naturally, the OAH considers all other viewpoints, requests and proposals which were presented during the whole process /17/. The decision on the request for license for site investigation and evaluation will be published on the news board of Self-governance Office of Paks city, at the OAH headquarter (at 1036 Budapest, Fényes Adolf street.4.) and the web-page of the OAH (www.oah.hu) /17/. The decision of OAH becomes legally binding after 15 days of its release. Appeal against the decision can't be submitted, but the clients can initiate a revision by a court. In this later case, the initiation has to be addressed to the Court of Capital but has to be submitted to the OAH within 30 days after the release of the decision /17/.

4.3.3 Poland: HTR-PL

Current Polish atomic law rules are suited for licensing of light water reactors, and for this reason they are not fully adequate for high-temperature reactors, especially for the co-generation applications.

In recent decades, the practice of licensing HTR reactors, prompted by the need to accelerate this process, has placed the main emphasis on the adaptation of existing regulations by formulating appropriate guidelines. Currently, the in-depth analysis of the usefulness and shortcomings of the existing regulations are required. The most recent example is the scope of the tasks carried out in licensing phase of HTR reactor in the US project: “Next Generation Nuclear Plant Licensing Strategy” (NGNP) /56//57/.

This study is an overview of the legal acts, relevant to the licensing process, in force in Poland, namely the Polish Atomic Law and the three regulations of the Council of Ministers.

In this work we don't deal with other acts quoted in the Polish Atomic Law, i.e.: the Construction Law, the Spatial Planning and Land Development Act and the Environmental Law.

4.3.3.1 The Polish Atomic Law

The Polish Atomic Law which came into force on 29 November, 2000, established a unified legal system to ensure nuclear safety and radiological protection of workers (NSRP) and the general population in Poland. The most important of its regulations are:

- authorization to carry out activities involving exposure to ionizing radiation,
- responsibilities of managers of organizations dealing with radiation,
- entitlements of the President of the National Atomic Energy Agency (PAA – Państwowa Agencja Atomistyki) to carry out inspections and supervision of these activities.

In connection with the start of the Polish Nuclear Power Program and the need to adapt the Polish legal system to the Council Directive 2009/71/Euratom modified by 2014/87/EURATOM, establishing a Community framework for nuclear safety, the amendment to the Atomic Law was made, which came into force on 1 July 2011.

The regulations of the Council of Ministers contain specifications of the requirements concerning nuclear safety and radiological protection, siting, design, construction, commissioning, operation and decommissioning of nuclear facilities.

4.3.3.2 The Question of the Definition of "Nuclear Facility" and "Nuclear power plant"

Nuclear facilities are defined in Art. 3 Section 17 of the Polish Atomic Law /61/:

"Nuclear power plant, research reactor, isotopic enrichment plant, nuclear fuel fabrication plant, reprocessing of spent nuclear fuel, spent fuel storage installation, as well as radioactive waste storages directly associated with any of these objects, located on its premises "

As we can see, nuclear power plants for the production of thermal energy were not formally included in this text /61/.

The similar problem is with the definition of **"nuclear power plant"**, used in regulations associated with the Polish Atomic Law, which does not include the reactor installation for the production of thermal energy. According to this definition - repeated in each of the three regulations of interest - power plant unit is:

"The system consisting of nuclear steam generation device, coolant circuits, one or more turbine generators, i.e. coordinated system for conversion of thermal energy into electrical energy."

Thus, an inherent feature of the nuclear power plant is the conversion of thermal energy into electricity. In this situation, formally speaking, the Atomic Law and the regulations of the Council of Ministers cannot be currently used for the reactor installation intended solely for the production of heat, regardless of the type of reactor. This applies also to high-temperature reactor (HTR) working in co-generation mode.

In conclusion adjustment is necessary for nuclear facility definition in the Polish Atomic Law and definitions of atomic nuclear power plant unit in the regulations of the Council of the Ministers, so that these regulations may be applied to reactors for the production of heat, especially in co-generation applications.

4.3.3.3 Siting Progress of Nuclear Facility

The amended Law contains the principle that "a nuclear facility is located in the area, which makes it possible to ensure nuclear safety, radiation protection, physical protection during start-up, operation and decommissioning of the facility, and to performing efficient emergency action during the radiation accidents."

The above-mentioned principle applies to the location of all types of nuclear facilities, and therefore also the installation with high-temperature reactor providing heat for industrial plants.

The Act states also that before the choice of the location the investor conducts feasibility study of several locations and on this basis prepares the evaluation of the area designated for the nuclear power plant. Among the elements of the assessment are external hazard resulting from human activity.

The regulation details the external threats, among other: "the threat of industrial plants which could affect nuclear facility chemically, biologically or mechanically". The analysis of this kind of threat is of course particularly important for co-generation applications of nuclear reactors and its actual regulations need elaboration.

The Act states that the **siting report** submitted by the investor is evaluated by the President of PAA during the proceedings **for a permit for the construction of a nuclear facility**.

The latest version of the Act states that, before applying for a permit for the construction, the investor of a nuclear facility may request the President of PAA **for the pre-emptive opinion on the proposed nuclear facility location**. The application shall be accompanied with siting report.

4.3.3.4 Design of a Nuclear Facility

The Act does not provide for a separate license for the design of a nuclear facility, as it is in the US regulations. Polish law defines only the basic conditions to be met by the project of a nuclear facility from the point of view of nuclear safety and radiological protection, and also safe operation of technical equipment installed and operated in a nuclear facility.

The basic requirement of the Polish Atomic Law is that the design of a nuclear facility must take into account the sequence of five levels of defense in depth (i.e. the concept of "defense in depth"). Way of implementation of this principle, in specific projects, may vary. There arises the question whether the concept of HTR reactor, coupled with an industrial installation, requires a change in the strategy of “defense in depth”.

In the Art. 36b of the Polish Atomic Law the postulate concerning the design and construction process of a nuclear facility was formulated, stating that *"solutions and technologies that have not been proven in operation of nuclear facilities or by means of tests, experiments and theoretical analysis are not to be applied"*. The fulfilment of this postulate may be difficult, because it is not sufficiently precise and leaves room for interpretation by the nuclear regulatory commission. This situation can occur for example during efforts to receive the permission to build a pilot installation equipped with high-temperature reactor.

4.3.3.5 Safety Analysis – Preliminary Safety Report

The revised Polish Atomic Law obligates the investor to carry out safety analysis, taking into account the technical and environmental factors, before the submission of the request for permit for the construction of a nuclear facility.The entities participating in the design of nuclear facility cannot take part in the safety analysis. The results of the safety analyses are the basis to prepare the **preliminary safety report** to be presented to the President of the PAA, together with the application for a permit for the construction of a nuclear facility.

The general scope and manner of conducting safety analysis, as well as the contents of the preliminary safety report is determined in the regulation of the Council of Ministers.

4.3.3.6 The Area of Restricted Use

The Atomic Law states that the area of restricted use around a nuclear facility is set up on terms specified in the Environmental Protection Law of 27 April 2001. The size of the restricted use area is determined on the basis of an estimate of the annual effective dose, which outside of this place should not exceed 0.3 mSv in normal operation and during operational incidents, and in case of an accident, without meltdown, does not exceed 10 mSv.

In the case of HTR reactor, in which a failure scenario with core melt cannot be taken into account, the criterion of not exceeding the annual effective dose 10 mSv outside the restricted use area concerns all accident scenarios.

4.3.3.7 The Issue of Charging “Decommissioning Fund”

The Act (Art. 38d) states that the costs related to the financing of the final management of spent fuel and radioactive waste and decommissioning costs are covered from the special separate fund, called the "decommissioning fund", which is to be formed by the company that has received the license to operate nuclear power plant. Payment for this fund is made of each sold MWh of electricity produced in nuclear power plant.

The Act does not specify how the decommissioning fund would be charged for the power plant unit intended for the production of thermal energy supplied to the district heating network or industrial plants. The omission of this issue in the law is a consequence of not recognizing an operation mode of a nuclear power plant producing mainly as the end product. .

4.3.3.8 The Classification of Safety Systems, Components of Constructions and Equipment

The article 36j of the Act declares that "for each element of the system, construction and equipment of a nuclear facility, which are important from the point of view of the nuclear safety and radiological protection, safety classes are established, related to the significance of these systems and components for NSRP. The safety classification shall be submitted for approval to the President of the PAA, together with the application for a permit for the construction of a nuclear facility.

The inherent HTR safety features, in particular reduction of the impact on the containment of internal events, allow to limit the development of technical safety systems and the number of components categorized as safety related.

4.3.3.9 The Regulations of the Council of Ministers

The opinions presented in this study concern the following regulations (which follow from the statutory authorization arising from the Polish Atomic Law of 29th November 2000 /61/):

- on detailed scope of the evaluation of the area designated for the location of nuclear facility and cases which exclude the possibility of recognizing area as a potential location of nuclear facility and on requirements for siting report,
- on requirements of nuclear safety and radiological protection which the project of a nuclear facility should take into account,
- on scope and manner of conducting safety analysis carried out before applying for a permit for the construction of a nuclear facility and the scope of the preliminary safety report for a nuclear facility.

These regulations implement into Polish law the provisions of the Council Directive 2009/71/Euratom, establishing a Community framework for safety of nuclear installations.

These regulations may also concern HTR reactor used for co-generation, provided the Polish Atomic Law will be supplemented with the definition of nuclear facility extended to the production of thermal energy.

4.3.3.10 The Regulations on the Location

The regulation was issued on the basis of section 4 of the Polish Atomic Law, art. 35b. The scope of the detailed evaluation and survey of the area designated for the location of a nuclear facility is formulated in this document in the categories of groups of threats. These threats are common and relate, without distinction, to all types of nuclear facilities listed in the definition of art. 3 Section 17 of the Polish Atomic Law. The evaluation methods of these risks differ, depending on the specific characteristics of the object and the place of its location. In contrast, methods for assessing these risks vary according to the specific characteristics of the object and the place of its foundation. It should be noted that the regulation, as the implementing act, does not impose any restrictions on the methodology and scope of these analyses and investigations. Decisions about these methods are made by the investor, who submits the siting report as part of the documentation required to obtain a permit for the construction of a nuclear facility.

The external hazard in potential location, both natural and human-induced, will be the same regardless of the type of nuclear facility, although they may have different weight compared to different types of objects. For example, for HTR reactors used for co-generation, the factor can be crucial which adjudicates that the area cannot be recognized as meeting the requirements of location when:

"in a vicinity of nuclear facility there is a plant which can have the negative chemical, biological or mechanical impact on its safety". But there is the conditional criterion: *"if this negative influence cannot be compensated by design"*. The important task of the investor submitting the siting report is to show that design features of the HTR reactor are sufficient to compensate the known threats from an industrial facility supplied with heat from this reactor.

The scope of the evaluation of location specified in the regulation (a group of risk factors enumerated in § 2.) as well as the contents of the siting report, described in § 6, relate to national conditions and do not always take into account the full list of threats. Each location has a different, individual characteristics, therefore, at the national level it is impossible to create a catalogue of all external factors which should be evaluated anywhere in the country and at the same time take into account the particular requirements and risks arising from the nature of the nuclear facility.

In the case of HTR reactors in co-generation, the specific risks would be mainly due to the potential location of the reactor as a source of heat at the border of the industrial zone (risk of: fire, accident with emission of chemical poisoning, explosion of chemical materials, etc.). The factor influencing the magnitude of the radiological consequences of potential external failure is not only the type and intensity of the possible external event, but, primarily, the amount and type of radioactive substances used in the nuclear object, and design solutions applied to limit the consequences of potential failures of the object.

High-temperature modular reactor, using relatively small amount of ceramic fuel resistant to melting, has a potentially lower risk of accident than e.g. LWR.

This regulation does not make distinction between different types of nuclear facilities, as it is impossible to a priori evaluate the radiological risks of these objects. Such evaluation and its validation are the responsibility of the investor. In the case of HTR reactor operating in co-generation system, the investor is obliged to provide, in the siting report, the information on technology, the type and quantity of radioactive substances used and the proposals of design solutions that are directly related to the specific features of the location.

The aim of the entire siting process is to create the catalogue of the external hazard for the selected location (crisis management plans, planning and zoning, etc.).

According to the international practice, § 1 of the regulation introduces the universal ranges of the location evaluation namely:

- the region of the location,
- the location area and
- the boundaries of the planned location of a nuclear facility.

The region of location extends to 30 km, and the location area extends to 5 km from the border of the planned location of a nuclear facility. However, the definition of the “boundary of the planned location” (par. 1 p.2) can pose some difficulties, because it confuses the boundaries with the area, but more importantly, it introduces some flexibility in setting this limit, because it defines it as measured from the centre of the proposed location, which is nowhere defined in the regulation. The factors listed in § 2 are grouped in § 3 of the regulation, together with an indication in what ranges they should be determined. The most detailed evaluation, in the process of location, is carried out for the place where the facility is planned. These are, mainly, detailed studies of ground conditions.

According to the international practice, before the start of the construction, the radiological data should be collected and then the models of atmospheric dispersion developed, within a radius of 30 km from the center of the region of the location. Other factors evaluated in this area are spatial planning and geological structure of the foundations base. The collected data and analysis results are published in reports on environmental impact assessment and submitted for the trans-boundary consultations. It can be assumed that in the case of HTR, despite its relatively much lower power, the range of the required data and research submitted to the trans-boundary consultations will also include the area within the radius of not less than 30 km.

Paragraph 6 of the regulation refers to the content of the siting report. In the general section of this report the zoning plan and location of the infrastructure within the planned place of the facility are listed. The question arises **whether the industrial installation powered by heat from the HTR reactor may be located within the boundaries of the planned location of the reactor**.

The content of the § 2 point 7, where the term "scenarios of a nuclear accident" is used, should also be given in a more precise form. You may mistakenly think that this pertains to scenarios of analysis of nuclear accidents in the nuclear facility. In fact, at this point of regulation the requirement to evaluate the size of the potential release of radioactive substances to the environment is introduced, together with the examination of the dispersal of these substances in emergency situations, and also the need to evaluate possibility of emergency actions. For example, the location of the HTR reactor too close to the industrial installation supplied with its heat may have a negative impact on the possibility of emergency actions. This issue should be particularly carefully examined in the framework of the siting report.

The regulation takes into account the recommendations laid down in International Atomic Energy Agency (IAEA) document - NS-R-3 /66/ and in a series of guidelines, which contain detailed description of the particular location features that needed to be evaluated.

4.3.3.11 The Regulation on the Nuclear Facility Project

The layout and content of the regulation are modelled, primarily, on the structure of the IAEA safety standards published in the documents NS-R-1 and DS414. The regulation is also based on the regulations approved in the world, and more specifically on the European Utility Requirements (2001) prepared for 3rd generation nuclear power plants with light water reactors /62/, on the US Nuclear Licensing regulations (10CFR50), guidelines of WENRA published in 2008 /63/ and 2009 /64/ and other requirements in force in EU countries.

Although the formulation of certain requirements indicates that they may be applied only to light water reactors, the dominant part of the report, containing the basic design requirements, is general enough to encompass other types of reactors and nuclear facilities. These requirements may be also applied to reactors used in co-generation, in particular the HTR. The condition for their applicability to HTR is a prior adjustment of the definition of "nuclear facility" in the Polish Atomic Law and the definition of "nuclear power unit" which is used in the regulations.

§ 46 is the only place in the regulation addressed directly to the nuclear facility (to be precise to a nuclear power plant) as a heat source coupled with the industrial installation. The design requirement rules out the possibility of transfer of radioactive substances from a nuclear facility to the industrial installation. This requirement is formulated so that it may be applied only to nuclear power plants and for this reason the nuclear thermal power station, in particular HTR reactors in co-generation, should not be taken into account.

As an example of a generally formulated postulate which would be impractical considering a pebble bed HTR reactor the requirement in § 52 p.3-2t requests: *"The reactor is designed so that to ensure that during the design based accident the fuel elements stay in its place and do not undergo deformation..."*.

From the viewpoint of HTR reactors with ceramic fuel in the form of coated particles, the "design based accident" should be defined as heat up of the core not threaten the integrity of the particular coating (§1 p. 36).

In Section II, Chapter. 1 of the regulation includes elaboration of the concept of “defense in depth” (defined in art. 36c of the Polish Atomic Law). From defined by the IAEA five levels of “defense in depth”, the first three levels are evident in the case of the HTR reactors.

A severe accident with significant release of radioactivity of a HTR would be equivalent to the loss of the first (fuel matrix) and second (particle coating) barrier e.g. by overheating of the core associated with a loss of integrity of the primary pressure system. Because of the intended HTR demonstrator design features e.g.:

- passive decay heat removal at temperature limits not threaten the integrity of the particle coating (e.g. 1600 °C)
- limited power density
- large heat capacity

This scenario is highly unlikely and to be covered by emergency planning. For this reason the planned emergency area may be reduced to the size of exclusion zone, what would essentially facilitate the application of the reactor HTR in co-generation.

§ 67-80 contain requirements for the design of the reactor containment system, in particular, define the containment system configuration (§67.2), requiring that it has to consist of primary and secondary containment.

The requirement of double containment was specified as mandatory in EUR document requirements for Generation III light water reactors /62/.

Favourable HTR reactor safety features, in particular the reduction of the burden of the third barrier (containment), caused that participants of the HTR-L EU project /60/ undertook investigation of the possibility of replacing the full containment building, with a lighter structure (confinement) which allows to control the leak of radioactive substances and keep them below the permissible limits.

4.3.3.12 The Regulation on Safety Analysis and Safety Report Content

The Polish regulation defines the basic requirements for the scope and method of conducting safety analysis of nuclear facilities and the scope of the preliminary safety report, which shall be submitted to the President of PAA with a request for a permit for the construction of a nuclear facility, according to the provisions of Art. 36d of the Atomic Law.

This regulation was developed based primarily on the current guidelines of the International Atomic Energy Agency /65/, taking into account: the guidelines of WENRA /63//64/, the requirements of the EUR document /62/ and the rules and guidelines of several countries, especially the USA, Germany and Finland.

In Chapter 1, a series of specialized terms used in the analysis of nuclear safety and radiation protection was defined. The definitions of these terms are consistent with the definitions introduced in the regulation on the nuclear facility project. Worth noting is, introduced in this document, specification of two categories of design failure, as in EUR document and Finnish regulations.

Sections 2 - 4 and Annex 1 concerns the scope and methodology of safety analysis and Chapter 5 and Appendix 2 concerns the content of the preliminary safety report.

In the case of modular high-temperature reactors, the wider application of probabilistic analysis still encounters difficulties. In the documents of HTR-L project /60/ the lack of definition of risk measure is mentioned, in a situation when the definitions of the core destruction frequency (CDF) and large early release of fission products (LERF) are not used.

In Appendix 1, the categorization of postulated initiating events (with different frequency) leading to certain states of the object is presented in tabular form. This document defines also the acceptance criteria of the results off analyses performed, concerning fuel damage, the condition of cooling circuit systems, reactor containment and the releases of radioactive substances to the environment.

In the case of the HTR reactor with ceramic fuel in the form of coated particles, the term *"damage of fuel clad"* should be interpreted as *"loss of the ability to contain the radioactive fission products within the coated particles and the graphite matrix"*. In addition, highlighting the cases with and without core melting is redundant. Appendix 2 contains the specification of the required content of the Preliminary Safety Report (PSR) for a nuclear facility. But this specification refers mainly to nuclear power plants with light water reactors.

PRS has to contain a comprehensive description of the nuclear power plant, covering all aspects of nuclear safety, radiological protection and safety analysis results (probabilistic and deterministic), showing compliance with established safety criteria. PRS is divided into 15 chapters. One of them is the evaluation of the potential location of the nuclear facility which should be done on the basis of the siting report.

"The threats from nearby industrial facilities" are mentioned in section 4.2.4 of Annex. This is the right place where one can place evaluation of the specific threats arising from the location at the boundary of the reactor site with industrial facility. Section 4.10 is the right place for the evaluation of the specific circumstances of the emergency planning and the emergency activities in the situation where the industrial plant working in co-generation system with the reactor is located at the boundary of the area of restricted use.

In conclusion in Poland the Atomic Law defines no specific licensing regulations or requirements for the co-generation or process heat application of a NPP beyond the standard licensing requirements. The additional applications have to comply with the radionuclide specific radiation limits for materials released/transferred from the NPP site, the external hazard requirements, the environmental impact limits and the safety requirements for the NPP itself.

4.3.4 Germany

In Germany no new nuclear newbuilds are possible because of the change made in the atomic energy act, following the Fukushima accident, to phase out nuclear energy production until 2022. Nevertheless because of the NPP's still in operation in Germany the domestic visions on licensing requirements will be analysed in the following chapter. In particular the chapter will focus on specific regulations of the safety of HTR or co-generation capability.

4.3.4.1 Hierarchy of regulations

Source for the following explanation is the "Bundesamt für Strahlenschutz" /67/. The different national legal and technical regulations in the field of nuclear law can be arranged into a hierarchal system. This is usually demonstrated in the form of a regulatory pyramid:

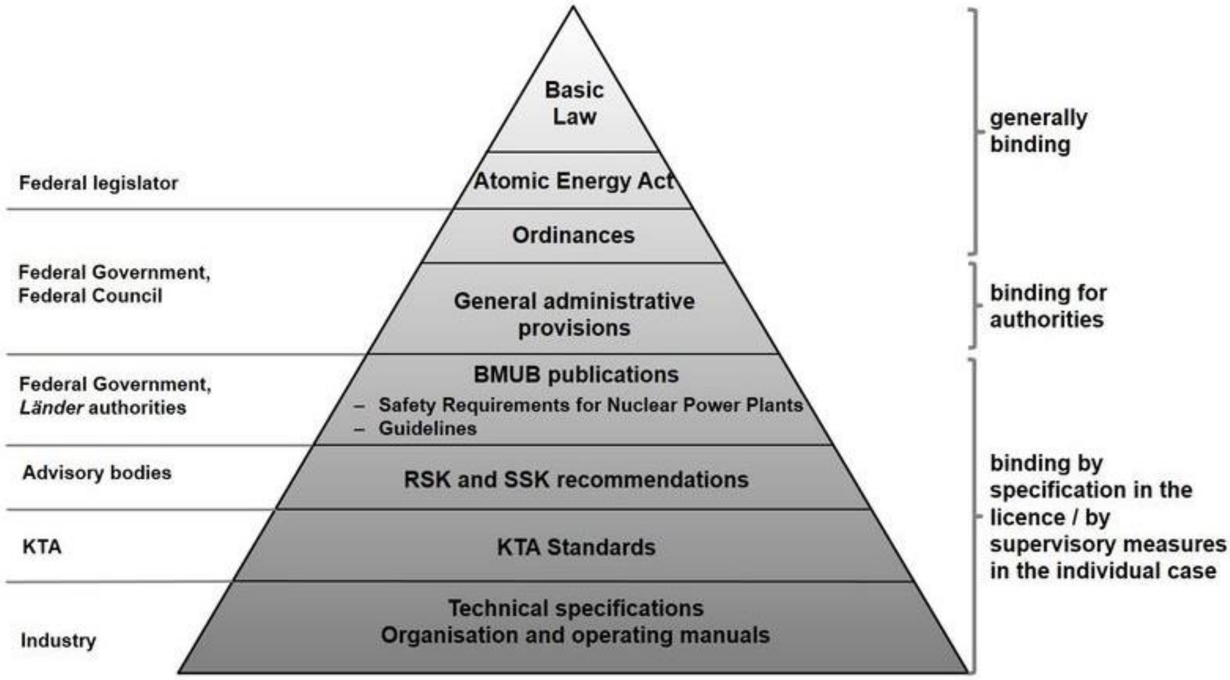

Fig. 9: Regulatory pyramid. (Source: Bundesamt für Strahlenschutz)

The regulatory pyramid (Fig. 9) deals with regulations originating from the Atomic Energy Act /68/. The national legislation on nuclear safety and radiation protection contains other acts, which are important for NPP newbuilds, too. Important in this context is mainly the "Act on the Precautionary Protection of the Population against Radiation Exposure (Precautionary Radiation Protection Act)" /70/.

Ordinances require approval of the Federal Council (Bundesrat) in which the governments of the federal states are represented. They substantiate the protective and preventive measures. At nuclear power plants the following ordinances are effective:

- Radiation Protection Ordinance
- Nuclear Licensing Procedure Ordinance
- Nuclear Safety Officer and Reporting Ordinance
- Nuclear Reliability Assessment Ordinance
- Nuclear Financial Security Ordinance
- Cost Ordinance under the Atomic Energy Act
- Ordinance concerning Potassium Iodide Tablets
- Nuclear Waste Transfer Ordinance
- Repository Prepayment Ordinance

The national laws and ordinances are supplemented by

- multilateral treaties in the fields of nuclear safety, radiation protection and liability,
- the EURATOM Treaty as well as
- regulations of the European Union (EU).

In the hierarchy of legislation, the international treaties, i.e. conventions under international law, are on the same level as is equated to formal federal law. Rights and obligations under the treaty apply only to the Federal Republic of Germany as contracting party.

Additionally, any binding requirement from EU regulations must be considered in the legislation and administrative activities in Germany. With some exceptions, EU law does not apply directly to national procedures of nuclear licensing and supervision but must first be implemented into national law within specified time limits.

4.3.4.2 Licensing requirements for the nuclear part

The national regulations as described above have been screened for relevance concerning licensing requirements in Germany. The following regulations are of high importance:

- Basic Law (Constitution) /68/
- Atomic Energy Act /69/
- Precautionary Radiation Protection Act /70/
- Act on the Environmental Impact Assessment /71/
- Ordinance on proceedings according to nuclear law (AtVfV) /72/
- Radiation Protection Ordinance /73/
- Safety Requirements for Nuclear Power Plants /74/
- BfS: Compilation of Information Required for Review Purposes under Licensing and Supervisory Procedures for Nuclear Power Plants /75/
- RSK and SSK recommendations /76/
- KTA standards /77/

In the following indications are given on the base of the above given regulations addressing in particular aspects in general related to safety of HTR or co-generation capability.

Administrative Authorities

According to §7 Column (4) of the Atomic Energy Act /69/

> *all federal states, local and other regional authorities whose jurisdiction is involved shall take part in the licensing procedure.*

The Nuclear Licensing Procedure Ordinance is based on §7 Atomic Energy Act and describes the licensing procedure for the new-built for a nuclear power plant in Germany. Furthermore the “Compilation of Information Required for Review Purposes under Licensing and Supervisory Procedures for Nuclear Power Plants” from 7 September 1982 will be binding.

The Verwaltungsverfahrensgesetz (VwVfg) “Administrative Proceedings Act” /79/ is binding where more specific regulations (here mainly the AtVfV /72/) don’t apply.

Basic Law

The Basic Law /68/ is the constitution of Germany and is therefore most profound legislation and deals with duties and organization of the state authority and with the relation of the state and the individual. It has a higher rank than any other law or norm in Germany.

Important Paragraphs

> *Article 2 (2)*
>
> *Every person shall have the right to life and physical integrity. Freedom of the person shall be inviolable. These rights may be interfered with only pursuant to a law.*

From this paragraph results a state duty of protection of the individual. To guarantee the rights of the individual from this paragraph legislation for the usage of atomic energy is necessary.

Article 20a

[Protection of the natural foundations of life and animals]

Mindful also of its responsibility toward future generations, the state shall protect the natural foundations of life and animals by legislation and, in accordance with law and justice, by executive and judicial action, all within the framework of the constitutional order.

Article 73

[Matters under exclusive legislative power of the Federation]

(1) The Federation shall have exclusive legislative power with respect to:

[…]

14. the production and utilisation of nuclear energy for peaceful purposes, the construction and operation of facilities serving such purposes, protection against hazards arising from the release of nuclear energy or from ionising radiation, and the disposal of radioactive substances.

Because of §2 (2) Basic Law the question arose in the 1970s if it is concurrent with the Basic Law to use Nuclear Plants for the generation of electric power. The Federal Constitutional Court has decided on 1979-12-20, concerning the NPP Mülheim-Kärlich, that by §73 (14) Basic Law the Basic Law itself approves of the utilization of nuclear energy. Because of that the concurrency could not be doubted because of another provision of the Basic Law.

In conclusion it is not excluded by the Basic Law to build a new Nuclear Power Plant of any type in Germany.

Atomic Energy Act (AtG), Edition 08/13

The Atomic Energy Act /69/ is a federal law on the basis of §73 (1), Point 14, Basic Law.

Important Paragraphs

§ 1 Purpose of the Act

The purpose of this Act is

> *"...to phase out the use of nuclear energy for the commercial generation of electricity in controlled manner, and to ensure orderly operation up until the date of termination,..."*

§7 deals with the licensing of installations:

> *(1) Whoever erects, operates or otherwise holds a stationary installation for the production, treatment, processing or fission of nuclear fuel or the reprocessing of irradiated nuclear fuel or essentially modifies such installation or its operation, shall require a licence. No further licences will be issued for the construction and operation of installations for the fission of nuclear fuel for the commercial generation of electricity or of facilities for the reprocessing of irradiated nuclear fuel. This shall not apply to essential modifications of installations or the operation thereof.*

The cited excerpts from AtG make clear, that it is not possible to erect a NPP in Germany according to the current AtG (Edition 08/13). Prerequisite for the erection of a High Temperature Reactor (HTR) with co-generation would be an amendment of § 1 and § 7 (1) Atomic Energy Act, which would have to be enacted by the federal parliament. It is highly improbable that this could happen in the near future, because there is no political will to do so and the resistance in the population of Germany may be significant.

In conclusion according to AtG no further licenses will be issued for the construction and operation of NPPs. The purpose of the AtG is to phase out the use of nuclear energy for the generation of electricity. Profound changes to this law would be the prerequisite for the licensing of a new NPP in the FRG.

Secondary regulations

Precautionary Radiation Protection Act (StrVG)

The Act on the precautionary protection of the population against radiation exposure (Precautionary Radiation Protection Act) of 19 December 1986, last amendment of 8. April 2008 /70/ determines the duties for the supervision of the environment and the minimization of the radiation exposure of humans and the radioactive contamination of the environment in the case of events involving potential radiological effects (See §1 StrVG).

No specific regulations concerning the erection of an HTR with co-generation were found in this act.

Act on the Environmental Impact Assessment

Acc. to Atomic Energy Act, §2a an Environmental Impact Assessment has to be performed if necessary according to the Act on Environmental Impact Assessment (UVPG) /71/.

An Environmental Impact Assessment according to the Environmental Impact Assessment Act would have to be performed for the new-build of a NPP. This would include a participation of the public.

Because severe accidents could have effects on neighbouring countries, too, these would have to be included in the Environmental Impact Assessment.

Ordinance on proceedings according to nuclear law (AtVfV)

In the AtVfV /72/ the procedure of issuing a license according to §7 AtG is laid down.

In addition the ordinance, in §1a, names properties to be protected which shall be identified, described and assessed in the Environmental Impact Statement (EIS, as part of the licensing procedure) with regard to the impact of the planned nuclear installation:

- human health
- animals
- plants and biological diversity
- soil
- water
- air
- climate and scenery
- cultural properties and other physical products

The licensing authority has to publish an announcement of the project as soon as the documents required for the public inspection are complete. Objections against the project may be submitted during the public inspection period according to §7 AtVfV. If a project requiring an EIS might have major impacts on the above listed properties to be protected in other states, or upon request of another state, §7a EIS initiates the information of the authorities designated by the other states.

Objections are discussed at a hearing which shall not be public but shall be constricted to the applicant, the persons who raised objections in time and other persons whose presence may be decided by the licensing authorities in its function to chair the hearing. If objections are based on special titles under private law, they shall not be discussed at the hearing, but shall be referred by written notice to the procedures before the courts of law.

No ruling concerning specifically the new-build of an HTR in Germany was found.

Radiation Protection Ordinance (StrSchV)

The Radiation Protection Ordinance /73/ has according to § 1 the following purpose:

> *The purpose of this Ordinance is to regulate principles and requirements of preventive and protective measures which apply to the use and effects of man-made and naturally occurring radioactive substances and ionizing radiation in order to protect man and the environment from the harmful effects of ionizing radiation.*

The ordinance originates from §53 AtG and will apply during all stages of the plant lifecycle. Licensing issues are laid down in this ordinance. No ruling concerning the participation of the public is contained in the ordinance.

No ruling concerning specifically the new-build of an HTR in Germany was found.

Publications of the Federal Environmental Ministry (BMUB)

The BMUB publishes regulatory guidelines in the form of:

- Requirements,
- Guidelines,
- Criteria and
- Recommendations.

The most important publication in the nuclear field of the recent years is the Safety Requirements for Nuclear Power Plants of 22 November 2012 /74/ which are addressed in more detail in the next paragraphs. No explanations concerning specifically the new-build of an HTR in Germany were found.

Safety Requirements for Nuclear Power Plants

In the Safety Requirements for Nuclear Power Plants of 22 November 2012 /74/ and their interpretations dated 29 November 2013 the safety criteria and requirements defined for LWR, especially for PWR, in the last decades, have been consolidated, enhanced and updated. In line with the German legislative decision to phase out the use of nuclear energy, the Safety Requirements only apply to modification licenses and to safety-related assessments within the framework of §§ 17, 19 AtG (e.g. periodic safety reviews). Since only LWR are in operation in Germany, the exemption made in the preceding “Safety Criteria for Nuclear Power Plants” of 1977, that the criteria/requirements apply literally to power plants with LWR and only analogously to other reactor concepts, has been omitted in the current document. Commensurate with the current use of nuclear power for generation of electricity, the Safety Requirements also do not address co-generation or supply of process heat outside the plant.

The licensing authorities can assume that the appropriate provisions against design basis accidents have been made, if the Safety Requirements are obeyed. It is important to point out, that this is not binding, but for the owner of the plant it is the simplest way to prove the appropriate provisions. Lots of provisions for nuclear safety are laid down in the safety requirements. In the following some important statements are summarized.

- According to § 2.1 of the Safety Requirements for Nuclear Power Plants the enclosure of radioactive material and shielding of radiation of this material has to be ensured. Therefore a safety concept has to be established assigning measures and installations to graded safety levels (normal operation, abnormal operation etc.; see also IAEA safety standards on this topic).
- To ensure the enclosure a barrier concept according § 2.2 has to be established. The containment has to function as a barrier up to safety level 4a (highly unlikely events).
- § 2.3 requires that the fundamental safety functions:
    - Control of radioactivity,
    - cooling of the fuel elements and
    - confinement of radioactive material

  shall be achieved and provides respective detailed requirements for the different levels of defence in depth.

- According to § 2.4 a protection concept against internal and external hazards as well as against very rare human induced external hazards has to be established.
- § 2.5 provides radiological safety objectives for the different levels of defence in depth.
- Concerning technological requirements general requirements are formulated in §3.1.
- Requirements for the “pressure holding encirclement” are defined in §3.4. The requirements for outer systems that hold pressure are high. For example the basic safety and the absence of interaction with safety relevant plant parts up to safety level 4a has to be shown.
- § 3.6 provides requirements for the containment system of the plant. Penetrations of the containment have to be closable according to § 3.6 (3).

As stated above, the Safety Requirements are specific for the reactor types currently operating in Germany and for their application to the generation of electricity. Nevertheless they comprise many requirements which are independent of the reactor type and would therefore be directly applicable to a HTR for co-generation. Regarding co-generation and HTR-specific safety features (especially requirements on confinement), the Safety Requirements would have to be applied analogously and to be developed accordingly.

RSK and SSK recommendations

Recommendations of advisory committees, like the Reactor Safety Commission (Reaktor-Sicherheitskommission - RSK) and the Commission on Radiological Protection (Strahlenschutzkommission - SSK) /76/, can be implemented by the BMUB in the respectively appropriate manner.

Besides a number of expert’s statements and recommendations during the construction phase of the AVR and the THTR, the RSK also issued a recommendation on the safety concept of an HTR-MODULE power plant /22/.

KTA safety standards

The Safety Standards of the Nuclear Safety Standards Commission (KTA) /77/ are used in German procedures. They specify those safety related requirements which shall be met with regard to precautions to be taken in accordance with the state of science and technology against damage arising from the construction and operation of the facility (Sec. 7 para 2 subpara 3 Atomic Energy Act -AtG-) in order to attain the protection goals specified in the Atomic Energy Act and the Radiological Protection Ordinance (StrlSchV) /73/.

Since most of the NPP built in Germany were LWR, the KTA safety standards are focussed on these reactor types with the consequence that they are not applicable to HTR core design and core assembly. For these items specific KTA safety standards have been drafted in the 1970's and 1980's but, with the decommissioning of AVR and THTR, have not been finally enacted.

For a new-built HTR with co-generation the following standards are of higher interest (not complete):

KTA 1201/2	Requirements for the operating manual/testing manual
KTA 1301.1/.2	Radiological protection: Design/Operation
KTA 1401/8	Quality assurance general/for welding issues
KTA 1501/2/3/4/6	Measuring, monitoring and assessing of radioactive dose levels in the nuclear and for discharge issues
KTA 3206	Verification Analysis for Rupture Preclusion for Pressure Retaining Components in Nuclear Power Plants
KTA 2101.1	Fire protection
KTA 2201.1/.2/.5	Design of nuclear power plants against seismic events
KTA 2207	Design of nuclear power plants against floods
KTA 3102.1 to .5	Reactor core design for high temperature gas cooled reactors
KTA 3103	Shut down system (no design, only general requirements)
KTA 3201.1/.2/.3/.4	Components of the primary circuit (applicable up to 400 °C, for higher temperatures individually to be agreed by the authority)
KTA 3203	Monitoring radiation embrittlement of materials of the reactor pressure vessel
KTA 3205.1	Component support structures with non-integral connections
KTA 3211.1/.2/.3/.4	Pressure and Activity Retaining Components of Systems Outside the Primary Circuit (partly)
KTA 3404	Isolation of operating system pipes penetrating the containment vessel (if containment vessel is considered)
KTA 3407	Pipe penetrations through the reactor containment vessel (if considered)
KTA 3501	Reactor protection system and monitoring equipment of the safety system
KTA 3502	Incident Instrumentation (general principles)
KTA 3503	Type testing of electrical modules of the reactor protection system
KTA 3505	Type testing of measuring transmitters and transducers of the reactor protection system
KTA 3506	System testing of the instrumentation and control equipment of the safety system
KTA 3507	Factory tests of equipment for instrumentation and control
KTA 3603	Facilities for treating radioactively contaminated water
KTA 3604	Storage, handling and plant-internal transportation of radioactive materials
KTA 3701.1/.2	Basic requirements for the electrical power supply of the safety system
KTA 3702.1/.2	Emergency power facilities with diesel generator
KTA 3703	Emergency power facilities with batteries and rectifiers
KTA 3704	Emergency power facilities with DC/AC converters
KTA 3901	Communication devices
KTA 3902	Lifting equipment
KTA 3903	Testing and operating of lifting equipment

In these papers no ruling was found that hinders the new-built of a NPP with co-generation. The KTA safety standards will in any case have an impact on the design of the nuclear parts of a co-generation facility.

Other regulations

The Water Resource Act (Wasserhaushaltsgesetz, WHG) /80/ in combination with the federal state acts on water resources rules the licensing or permission of the usage of local water reservoirs. The regulations comprise provisions on the protection of surface and ground water reservoirs, development of the resources and the local water planning policy.

With respect to NPP operation the permission on usage of local water reservoirs is mandatory. Beside conventional constraints this permission comprises limits on the radionuclide discharge to these water reservoirs based on the regulations of the Radiation Protection Ordinance /73/. These permissions were given by the regulators supported by assessments of experts.

Additional constraints are defined by the Federal Immission Protection Act (Bundes-Immissionsschutzgesetz, BImSchG) /81/ addressing the protection of human, flora, fauna, soil, water, atmosphere and cultural installations from harmful influences correlated with the operation of NPP's. The general constraints of this act are specified in more detail in several additional regulations, rules and instructions.

For specific nuclear facilities approved according to § 7 AtG in addition the approval according to § 4 BImSchG has to be achieved.

4.3.4.3 Co-generation of district heating

Atomic Energy Act and Radiation Protection Ordinance

To provide district heating hot water has to be to be transported from the plant to the customer. As an additional material barrier against contamination this water has to be separated from the secondary circuit of the NPP by the tube walls of a heat exchanger. Existing plants in Germany only discharge water from the plant but don't have a circuit bringing the water back (reducing the effort for cleaning measures).

The regulatory frame consequently doesn't have specific regulations for this case. In the StrSchV /73/ the word “Discharge” is defined as follows:

> *Surrender of liquid, aerosol-bonded or gasiform radioactive substances from installations and facilities on ways provided for this purpose;*

It can be concluded, that district heating will be considered as hot water discharge from the plant (confirmed by §3, (2), Point 2 StrSchV), initially considered as release radioactive substance according AtG /69/, §2 (1) which defines:

> *Within the meaning of this Act the term "radioactive material" (nuclear fuel and other radioactive substances) refers to all material containing one or more radionuclides and whose activity or specific activity in conjunction with nuclear energy or radiation protection cannot be disregarded under the provisions of this Act or a statutory ordinance promulgated on the basis of this Act.*

The following paragraph §2 (2) AtG regulates when the activity or specific activity of the substance can be disregarded e.g. for activity levels below specified exemption levels of an ordinance on the basis of AtG. The StrSchV is such an ordinance. Exemption levels are according to §3 StrSchV:

> *Values of activity and specific activity of radioactive substances as specified in Appendix III, Table 1, Columns 2 and 3, the exceeding of which entails that practices involving these radioactive substances are subject to surveillance under this Ordinance;*

The best case would be that the water would not be subject to surveillance of the StrSchV, because the exemption levels of Appendix III, Table 1, Columns 2 and 3 are not exceeded. For this the water had to be cleared. Clearance is defined in §3 StrSchV (2) as follows:

> *Clearance:*
>
> *Administrative act which effects the exemption of radioactive substances and any movable goods, of buildings, soil areas, installations or parts of installations which are activated or contaminated by radioactive substances and which originate from practices pursuant to § 2, para. (1), subpara. 1, character (a), (c) or (d), governed by the scope of application*
>
> *a) of the Atomic Energy Act and*
>
> *b) ordinances based on it and decisions of administrative authorities*
>
> *for the use, utilization, disposal, possession or their transfer to third parties as nonradioactive substances;*

According to §29 (2) StrSchV *the competent authority shall grant clearance in writing, if the effective dose occurring for members of the public may only be in the order of 10 µSv per calendar year.*

Further on in §29 conditions are mentioned for which the competent authority may consider this requirement to be met. One of these conditions is unrestricted clearance (no limitation regarding the mass discharged per year) addressed in Appendix IV, part B of the StrSchV. For the release of tritium as part of liquid or solid material the level for unrestricted release would be 10^3 Bq/g (for aerosols the limit is 100 Bg/cm³). In this case no specification with regard to future use, utilization, recycling, re-use, disposal or the final disposal of the substances are required. Appropriate clearance measurements (equipment according to §67) have to be performed according to §29 (3) StrSchV. But because values for specific activities or activities of an unrestricted clearance are only given only for individual isotopes this condition is not applicable for water.

Another option to handle hot water “release” for district heating is described by appendix VII of the StrSchV. which gives assumptions, how to determine the radiation exposure.

Appendix VII of the StrSchV deals with assumptions with the determination of radiation exposure. The listed assumed exposure pathways (exposure through stay on sediment, through ingestion) are not applicable for district heating. But according the statement in Appendix VII other exposure pathways shall be taken into consideration if this is justified as a result of local characteristics of the location, or as a result of the type of installation or facility, heat transfer pipes of district heating should be handled as possible exposure pathway.

Concluding the topic of clearance it can be said that it lies within the duty of the owner of the co-generation plant to convince the competent authority, that

- the requirements for an unrestricted clearance are met;
- the intended measuring assemblies for clearance are appropriate. This can theoretically mean that no measuring is intended. The KTA /77/ safety standard 1504 provides specific data on this matter, too.

It is unclear, if hot water would be classified as consumer good according to StrSchV, in particular if a closed circuit is applied connecting to the secondary NPP system and the district water grid by separate heat exchanger. The definition is (§3 StrSchV):

> *Consumer goods as defined in the Food and Feed Code which are intended for ultimate-consumer use and goods and other items of everyday use intended for household or occupational purposes, except for building materials and devices whose type has been approved into which any other radioactive sub-stances pursuant to § 2, para. (1) of the Atomic Energy Act have been introduced;*

The provision of district heating is not listed as unauthorized type of activity in Appendix XVI, StrSchV and therefore could be justified according to §4 StrSchV.

There are no examples in Germany of co-generation for district heating. There are experiences with waste water removal from supervised areas and controlled areas. The regulations for controlled areas usually require the collection and storage of waste water before clearance. Hence new concepts had to be developed which include storage of heat water before transport from the premises.

Because of the large amount of hot water to be discharged (means transferred for district heating) an unrestricted clearance of the water would be necessary therefore the maximum release levels of the radioactive components/impurities have to maintain (StrSchV, Appendix III, Table 1, Column 5).

Regarding the respective measurement system KTA 1504 will have to be obeyed, that deals with Monitoring and Assessing of the Discharge of Radioactive Substances in Liquid Effluents in particular from water or steam circuits of the power house (KTA 1504 §3.1/§3.4). It requires a continuous, integral measurement of the gamma activity (Cs-137 equivalent applied), a monthly check of the tritium content in the secondary circuit (balance recording requested for tritium activity release > 10^6 Bq/m³) and a radioactivity concentration levels in maximum below $2*10^7$ Bq/m³. For BWR in addition a balance recording may be requested if the continuous release measurement of the activity concentration exceeds $4*10^4$ Bq/m³. If water/steam separation from auxiliary steam system is considered additional measurements are requested for activity levels (Cs-137 equivalent) in the auxiliary steam system above $4*10^4$ Bq/m³

According to KTA 1504, § 3.8 no measuring of radioactivity is necessary if between the relevant system (secondary circuit) and the environment two material barriers or one material barrier and one pressure barrier exist. Assuming the layout (already mentioned) of the heat transfer system connected to the secondary NPP circuit and the district heating grid by two heat exchangers a release of contamination in the hot water has not to be considered. In particular if in addition pressure barriers are applied. The duty of measuring the activity constantly could be omitted. Because the KTA rules are not legally binding and the authorities have to decide about the usage this is only estimation.

The water act of the concerned federal states has to be obeyed, if water is discharged from the district heat generation plant to the environment. No larger problems have to be expected because of that.

If the water is not delivered to an industrial company the ordinance “Verordnung über Allgemeine Bedingungen für die Versorgung mit Fernwärme (AVBFernwärmeV)“ /78/ will be used. No regulations concerning radioactivity are contained in this ordinance. There are regulations containing the right to interrupt or stop the supply of the customer with district heating. For example the district heating company is allowed to stop the supply if persons or installations shall be protected from immediate danger for their safety. This corresponds with the provisions against design basis accidents according to the Safety Requirements for Nuclear Power Plants. The supply with district heating could be stopped in the case of design base accidents in full compliance with the ordinance AVBFernwärmeV.

4.3.4.4 Desalination

Concerning desalination of water to generate drinking-water in the first the same considerations apply as for district heating. The water is also transported continuously from the production plant to the customers.

The water act of the concerned federal state has to be obeyed, too. Additionally for drinking-water the “Trinkwasserverordnung” (TrinkwV, drinking-water ordinance) /83/, which bases on the “Lebensmittelgesetz” (food act) /84/ has to be obeyed. The TrinkwV doesn’t have the aim to protect people from the incorporation of radioactive substances. For this the StrSchV /73/ applies. Anyway for the purpose of co-generation of desalinated drinking-water some aspects have to be considered:

Concerning the materials in contact with water the BMUB has regulated the requirements for materials. European law is binding, too. Diverse chemical parameters of the water are regulated in the TrinkwV. Of interest are:

- The Uranium concentration must not exceed 0.01 mg/l.
- The Tritium activity must not exceed 100 Bq/l. The regulator has the duty to control the radioactivity. Under certain circumstances this duty can be omitted.
- The dose limit (Gesamt-Richtdosis) is 0.1 mSv/year

(Note: These limits are of generally valid protecting population from drinking water pollution, they are not limited to regions of nuclear facilities)

If according to KTA /77/ 1504 monitoring of the water is not necessary it seems to be achievable to comply with the TrinkwV, too. The company owning the desalination plant would be a water supply company and the plant would also fall under the “Verordnung über Allgemeine Bedingungen für die Versorgung mit Wasser“ (AVBWasserV) /85/. Assuming a heat exchanger between heat transfer circuit and water pipe system to be desalinated again according KTA /77/ 1504 monitoring of radioactivity may not necessary.

4.3.4.5 Other Co-generation applications

Other possible uses are co-generation of hydrogen, coal gasification or high temperature process heat. These uses require the coupling of a Nuclear Plant and a Chemical Plant.

The first problem would be the different safety philosophy. A nuclear power plant in Germany would be designed with a barrier concept according to § 2.2 Safety Requirements for Nuclear Power Plants /74/ while for chemical plants usually an open layout is favoured. This is especially the case when combustibles are processed. Because of this, to obey the legislation it will be necessary to separate the operations of the nuclear power plant and the chemical plant. For the chemical plant a different legislation will be valid.

It is highly recommendable not to treat the chemical plant as an extension of the nuclear plant, but as an external facility. The conventional law would be applicable for the chemical plant, but the StrSchV would have to be applied where applicable, too.

Possible effects from the chemical plant on the nuclear plant have to be considered. Events in the chemical plant have to be considered according to § 2.4 of the Safety Requirements for Nuclear Power Plants /74/. All installations that ensure and keep the safe shut-down of the nuclear power plant, that are necessary to remove residual heat or hinder a release of radioactivity have to be designed against hazards from the chemical plant. Regarding the protection of the NPP against blast waves from chemical reactions, Annex 3 of the Safety Requirements makes reference to a respective rule issued by the Federal Ministry of Internal Affairs /82/ which provides a design curve of the overpressure to be applied at the outer surface of the buildings to be protected and a relation for determination of the minimum distance between power plant and chemical plant, depending on the mass of explosives handled there.

According to “Compilation of Information Required for Review Purposes under Licensing and Supervisory Procedures for Nuclear Power Plants” from 7 September 1982 /75/, a statement of industrial facilities affected by the nuclear power plant had to be provided (1.3.3.1) as well as a statement of potentially hazardous industrial facilities or plants (1.3.3.2).

4.3.4.6 Conclusion

The political consensus in Germany after the Fukushima accident to phase out the use of nuclear energy was cast in a revision of the Atomic Law /69/, in which new builds were excluded and final shutdown dates for the existing NPP were fixed. For any new-build of a facility providing nuclear energy for generation of electricity, process heat or a combination of both (co-generation), the Atomic Law would have to be revised again. Looking at the public opinion in Germany, it is highly unlikely that this will happen in the near future. It is, however, worth noting that the German constitution, the Basic Law, still does not rule out nuclear energy. A revision of the Basic Law would only be possible with a two-third majority in the federal parliament while a revision of the Atomic Law could be achieved with an absolute majority.

Since existing German NPP will stay in operation for some years, the entire system of lower tier nuclear ordinances, regulations and safety standards is still in force. While this system is tailored more or less to the needs of regulating operating the existing NPP which are all LWR, much of them could be applied analogously to an HTR.

Neither the Atomic Law nor the lower tier system of rules defines specific requirements beyond the usual licensing regulations for co-generation in general. A circuit for heat transfer off-site of the NPP is not considered in the German regulations. <u>But considering the heat transfer as discharge of hot water/steam requirements regarding the radionuclide inventory and the nuclear activity are defined which could be used to design a co-generation option within the lower tier system of rules.</u> The requirements address specific activity limits for individual isotopes as tritium as well as total activity concentrations for discharged water.

Assuming additional physical (heat exchanger) and pressure barriers between the secondary NPP circuit and the district heating or industrial application grid, based on German nuclear standards even the measurement of the radioactivity in the heat transfer circuit may not be mandatory.

Nuclear co-generation in connection with chemical plants will raise additional issues with regard to reciprocally interactions on the respective safety cases as part of “external event evaluation”.

4.3.5 Other Countries

4.3.5.1 USA: NGNP project

The Next Generation Nuclear Plant (NGNP) will be a licensed commercial HTGR plant capable of producing the electricity and high temperature process heat for the industrial markets. The NGNP Project will design, construct, and operate the HTGR plant and associated technologies to establish the technological basis for commercialization of this new generation of advanced nuclear plants /58/.

<u>Licensing Strategy</u>

NGNP has developed a strategic implementation plan to establish the regulatory licensing basis and COL application (Combined Licenses) for an HTGR to result in review/approval by the NRC. This plan focuses on three key elements of plant development and licensing:

1. The development and understanding of the radiological source term (based primarily on fuel design, qualification testing results, and analytical methods development)

2. The prevention/mitigation of the release of this source term to the environment (including definition of licensing basis events and design/implementation of multiple release barriers, consistent with defense in depth strategies and requirements)

3. The development of an updated emergency planning structure that considers potential radiological releases from the HTGR, coupled with various industrial application configurations, in order to assure the protection of the health and safety of the public in the unlikely event of a release /58/.

<u>Method for applying LWR Regulations</u>

Historically, a range of approaches have been taken to adapt LWR licensing technical requirements for advanced reactor designs. These approaches have mostly used deterministic methods. Fort St. Vrain (an HTGR located in Colorado) was licensed in the early 1970s using a traditional deterministic approach to adapting (then) existing LWR requirements. At Commission direction, since the early 1990s, greater use of probabilistic methods has been employed, resulting in a continuing effort to risk-inform the traditional deterministic requirements. Existing LWR licensees as well as new LWR COL applicants have availed themselves of the expanding use of risk-informed and performance-based methods. However, the approaches have varied in the extent to which probabilistic information was used for establishing the licensing basis

The approved approach would allow adaptation of LWR requirements by applying traditional deterministic engineering judgment, complemented by PRA information and insights. Specifically, in SECY-93-092, the NRC staff proposed an approach that had the following eight characteristics /58/:

1. Events and sequences would be selected deterministically and would be complemented with insights from PRAs of the specific designs
2. Categories of events would be established according to expected frequency of occurrence. One category of events that would be examined was accident sequences of a lower likelihood than traditional LWR design-basis accidents (DBAs). These accident sequences would be analysed without applying the conservatisms used for DBAs. Events within a category equivalent to the current DBA category would require conservative analyses, as is presently done for LWRs.
3. Consequence acceptance limits for core damage and onsite or offsite releases would be established for each category to be consistent with Commission guidance.
4. Methodologies and evaluation assumptions would be developed for analysing each category of events consistent with existing LWR practices.
5. Source terms would be determined as approved by the Commission.
6. A set of events would be selected deterministically to assess the safety margins of the proposed designs, to determine scenarios to mechanistically determine a source term, and to identify a containment challenge scenario.
7. External events would be chosen deterministically on a basis consistent with that used for LWRs.
8. Evaluations of multi-module reactor designs would be considered as to whether specific events apply to some or all reactors on site for the given scenario for all operations permitted by proposed operating practices.

Report to Congress Recommendation

Within this regulatory framework, there are several technical options for establishing the NGNP licensing basis, each placing progressively greater emphasis on the use of probabilistic risk assessment (PRA) techniques and risk insights. With the exception of the last option (option 4), all other options adapt existing LWR requirements for licensing an NGNP. The last option entails rulemaking to develop a new body of risk-informed and performance-based regulations. These options are described below /58/.

1. Deterministic Approach.

   This option uses deterministic engineering judgment and analysis to establish the licensing basis (including selection of events) and licensing technical requirements. This approach has been used for licensing operating LWRs and involves no use of PRA information and insights.
2. Risk-Informed and Performance-Based (RIPB) Approach.

   This option uses deterministic engineering judgment and analysis, complemented by NGNP design specific PRA information, to establish the licensing basis (including selecting licensing basis events) and licensing technical requirements. The use of the PRA would be commensurate with the quality and completeness of the PRA presented with the application.
3. Risk-Informed and Performance-Based Approach (with greater emphasis on PRA).

   This option places greater emphasis on the use of the NGNP design-specific PRA in complementing deterministic engineering judgment and analysis, to establish the licensing basis (including selecting licensing basis events) and licensing technical requirements. As in Option 2, the use of the PRA would be commensurate with the quality and completeness of the PRA presented with the application.
4. New Body of Risk-Informed and Performance-Based Regulations.

   This option would use a new body of regulations to establish the licensing basis (including selecting licensing basis events) and licensing technical requirements. The new body of regulations would make extensive use of the risk-informed and performance-based regulatory structure, and would require rulemaking to be implemented.

The Report to Congress recommended that the best option for licensing the NGNP prototype would be to start with a risk-informed and performance-based technical approach, Option 2 (as described above), and then adapt the existing LWR technical requirements and establish any NGNP-unique requirements that are needed. The risk-informed approach reflects a decision making based on the consideration of insight risks identified, engineering judgement, safety limits and redundant and diverse safety systems. It allows the establishment of requirements to better focus licensee and authority attention on design and operational issues commensurate with their importance for public health and safety. Insight risks are derived from risk assessments and may include improved understanding of the likelihood of possible outcomes, sensitivity of the results to key assumptions, relative importance of the various system components and their potential interactions, and the areas and magnitude of the uncertainties.

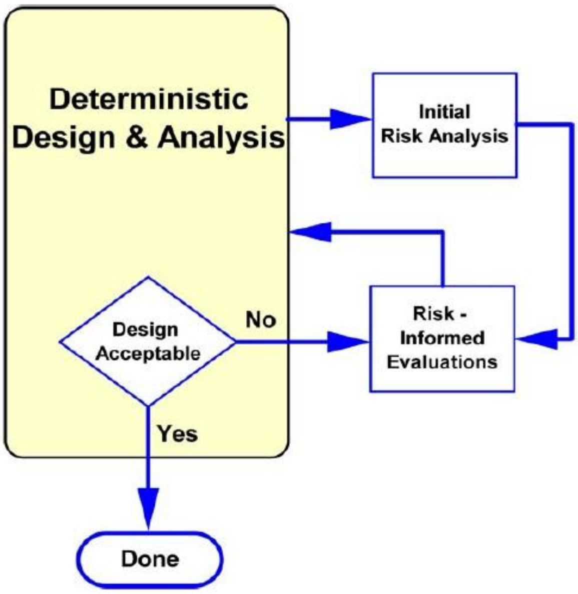

Fig. 10: Risk informed approach

NGNP Approach for Adapting LWR Regulations

Given the limited regulatory experience with gas reactor technology as well as its deployment in non-traditional process heat applications, there is not an existing and complete body of regulations directly suited to the NGNP design. Consequently, for a license application to be successfully prepared, reviewed and approved, updated regulatory guidance (or an agreed framework) derived from the existing LWR regulations will have to be proposed and agreed upon to guide the approval of the NGNP COL application. Subsequently, it would be expected that such updated guidance would guide future applicants and the NRC staff in preparing and reviewing HTGR license applications.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units. It is therefore clear that the current set of regulations and guidance should be reviewed for applicability to a high-temperature gas reactor design like the NGNP. Engagement with the NRC is needed to reach agreement on the scope and development of new regulatory policies or to change existing regulatory policies that support the high-temperature gas reactor design so that the NGNP licensing process is successful.

Deterministic Elements of Adapting LWR Regulations

The NGNP safety approach is rooted in deterministic engineering principles. However, as noted above, the licensing approach for determining licensing basis events (LBEs) is risk-informed, and thus is based on both deterministic and probabilistic elements. Other areas where a combination of deterministic and probabilistic analysis will play a role include the definition of NGNP safety functions and success criteria, the prediction of plant response to initiating events, and the development of mechanistic source terms. Traditional regulatory objectives that are not generally amenable to probabilistic treatment include issues such as occupational exposure minimization, environmental impacts other than radiological, and security and safeguards, which will thus be addressed in a deterministic manner.

Risk Assessment Elements of Adapting LWR Regulations

Consistent with the Report to Congress (Ref 2) recommendation, the NGNP Project is developing a risk-informed and performance-based technical approach that adapts existing NRC LWR technical licensing requirements in establishing NGNP design-specific technical licensing requirements. The rationale for use of these risk-assessment techniques includes:

- Using a PRA to aid in the development of events that are included in the licensing basis maximizes the probability of establishing a comprehensive safety basis. By its nature, PRA development is a rigorous process that considers the comprehensive performance of the facility design and safety margins.
- The PRA development process includes evaluations and analyses that are deterministic in nature (e.g., failure modes and effects analysis and reactor system performance analyses).
- Probabilistic methods for event selection, SSC classification, special treatment identification, as well as integration and evaluation of defence-in-depth strategies will take advantage of the safety characteristics provided by gas-cooled reactor designs.
- Integrating PRA insights into the design provides a more structured means for both assessing single point vulnerabilities as required by the single failure criterion used in current LWRs and providing more robust capability for considering functional vulnerabilities and diversity for all portions of the event frequency spectrum. This is of considerable value, particularly for designs that are new or contain novel features.
- Using a PRA provides a rational method for selecting design features and resolving safety issues; hence avoiding conflicting requirements that can arise from a purely deterministic design and safety analysis approach.
- The PRA provides a rational approach for quantitatively addressing uncertainties.

For the licensing of HTGR facilities NRC performed in 1998 a regulatory gap analysis (**Fig. 11**) to identify gaps to be closed for a successful HTGR licensing /59/. The following approach was proposed by Exelon in 2001 to close the gaps,

- identify range of “applicability” conditions
- utilize design specific details and risk techniques for evaluation
- create basis for “not applicable” determinations
- provide exemption basis
- identify a basis for policy development where needed
- feed these information into HTGR-specific application specification

The Proposal by Exelon was evaluated by NRC in 2002 and reported as NGNP Licensing Strategy Report to the Congress in 2008. Additional requirements are defined in respective reports (SECY-05-0006: Source term definitions, SECY 03-0047/04-0103/05-0006: policy issues for non-LWR licensing)

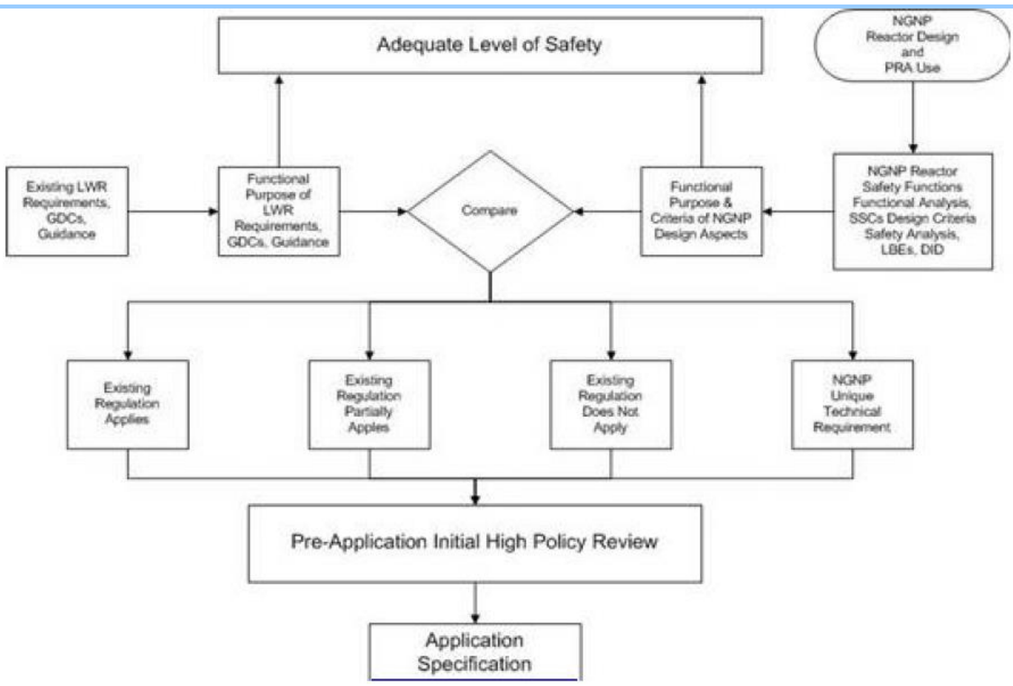

Fig. 11: Regulator gap analysis process

In conclusion also the intended NGNP licensing process is based on the general design criteria for nuclear reactors (base on general safety objectives) which define the minimum requirements for the principal design criteria for water-cooled nuclear power plants. Hence the licensing process shall be based on LWR licensing regulations and requirements. NGNP-unique requirements shall be identified by a risk-informed and performance-based (RIPB) PRA, adaption of existing LWR technical requirements and an elaborated pre-application phase.

This licensing approach will cover the three key elements of a HTRG specific licensing procedure and also requirements with regard to the co-generation or process heat applications:

- development and understanding of the radiological source term,
- prevention/mitigation of the release of this source term to the environment and
- development of an updated emergency planning structure that considers potential radiological releases from the HTGR.

For the NGNP project a well-defined licensing strategy is developed:

1. Submission of a combined license application
2. Provision of analytical tools, models and associated data for major technical areas of NGNP
3. Establishment of the license and safety basis by a risk-informed and performance based technical approach
4. Application of LWR based regulatory guidance for the prototype NGNP, to be upgraded regarding NGNP special issues involving security, safeguards, spent fuel , environmental matters, inspection and start-up testing
5. Other issues identified during the pre-application phase
6. Provision of additional guidance as
   a. focus on key areas, including planning, training, design familiarization, and identification of programmatic and technical issues,
   b. identification of the specific reactor technology to be built in advance of the pre-application review,
   c. expanded 3-year period for the pre-application review for the specific NGNP design.
7. Utilization of previous pre-licensing efforts and NRC interactions associated with gas-cooled reactor technology
8. Identification of earliest and highest priorities for the pre-application period as reactor design selection, siting and early site permission

The applicant will face the usual licensing requirements as:

- top level regulatory criteria (TLRC) must be met for licensing basis events
- TLCR must be met by safety functions, SSC safety classification and regulatory design criteria
- Fulfilment of TLRS must be demonstrated by deterministic DBAs, defense-in-depth (DiD) and regulatory special treatments

4.3.5.2 Republic of South Africa: PBMR project

In response to the early investigation from Eskom the NNR had been reviewing aspects of High Temperature Gas Cooled Reactor (HTGR) technology from 1998-99, in anticipation of the PBMR license application. In July 2000 the NNR received a Nuclear Installation Licence (NIL) application from Eskom for a PBMR Demonstration module electricity generating power station. The principal safety requirements formulated in the Regulations in terms of section 36, read with Section 47 of the NNRA (Act No. 47 Of 1999) on Safety Standards and Regulatory Practices form the basis for the stipulation of the Licensing Requirements for the PBMR. The Licensing Requirements have been further elaborated in a suite of license documents /90/.

Licensing requirements and criteria comprise, besides the general requirements to respect good engineering practice, ALARA and defence-in-depth principle, specific radiation dose limits. A categorization for normal operation, operational occurrences, as well as for design basis events for workers and the public is applied. The safety criteria also stipulate occupational risk limits for the workers as well as risk limits for the public for all possible events that could lead to radioactive exposure. The dual nature of the NNR safety criteria implies that the safety analyses for demonstration of compliance of the Safety Case with the licensing criteria for the PMBR have to comprise both deterministic and probabilistic analyses.

Multi-Phase Licensing Approach

In view of the complexity of this project and acknowledging the developmental nature of the PBMR DPP, a multi-staged licensing process has been adopted. This licensing process provides a logical link between the various steps of the design process, the safety assessment and the development of operational support programmes. Following a satisfactory regulatory review of the safety case by the NNR an initial Nuclear Installation License (NIL) will be granted (or refused) to the applicant for the first stage of the process. A variation to this NIL will be requested by the applicant, and issued by the NNR following its satisfactory regulatory review, at each of the subsequent Licensing Stages.

The envisaged Licensing Programme would include the following major licensing stages /90/:

- Stages 1&2: Site preparation, construction and manufacturing phase

  (Stage 1: assessment of the applicant's organisational and management processes and the evaluation of the applicant's ability to perform the envisaged design development and control processes

  Stage 2: assessment of the Reference Design against safety criteria and safety issues associated with the design including identification of phenomena and innovative SSC that may require specific test and qualification.)
- Stage 3: Nuclear Fuel on Site/ Commissioning and Start-up
- Stage 4: Plant operation (commercialization)
- Stage 5: Eventual decommissioning

With the introduction of the licensing of this "new" reactor technology in South Africa major challenges faced by the NNR were the adjustment of the regulatory philosophy and processes to the licensing of a "first of a kind" reactor project and the improvement of the in-house expertise on gas/graphite reactor technology.

One of the major aspects of the PBMR licensing process was the credibility of the PBMR design and licensing basis. Unlike LWRs for which well researched and documented design criteria, rules, codes and standards, operational data etc. are readily available, broad international consensus has not been developed on inter alia general design criteria, design rules, codes and standards for the PBMR. One of the first challenges faced by the NNR was to develop relevant licensing requirements taking cognizance of reactor operating experience, developments in international safety standards and application of these in the design of new generation of reactors such as for example EPR. Finally the NNR developed and published the first revision of the "Basic licensing requirements for the PBMR" in 2000 followed by various other licensing documents.

<u>Licensing Framework (Safety Case)</u>

Within this framework the main components for the development and review of the PBMR safety case are /90/:

1. The Safety Case Philosophy (SCP):
   - intellectual and philosophical arguments of how PBMR safety will be demonstrated to meet the safety requirements set by the NNR
   - consideration of the broad safety objectives of the PBMR
   - systematic identification of Key Licensing Issues (KLIs) which will need to be addressed as part of the demonstration of the PBMR safety objectives in the Safety Analysis Report.
2. The Safety Analysis Report (SAR) and other supporting documents providing a detailed justification of how the safety arguments/objectives presented in the SCP are or will be demonstrated.
3. The General Operating Rules (GOR) and additional Development / Support Documents e.g. on Project and Licensing Management and Test and Commissioning.

The licensing process continued with an evaluation in July 2000 and the definition of the safety case for stage 1 of the multi stage licensing process in November 2001. Issues relating to the organisation, design, documentation, etc. were identified as part of the review and summarised in a number of Pre-conditions and Conditions for licensing. These conditions were then combined resulting in some 25 Key Licensing Issues that required resolution /90/.

In conclusion for the prototype HTR the regulator evolved a licensing framework and corresponding licensing requirements based on LWR experiences and identified HTR specific features (key licensing issues). The licensing process was supported by external experts.

The licensing was organised in a multi-phase approach allowing the evaluation and optimisation of HTR design features, manufacturing processes and safety culture considerations developed by the applicant ESCOM during the licensing process (comparable to the graded approach as described in the IAEA report on licensing process /44/). These accompanying measures aim at the demonstration (not only the assumption) of safety performance and features which is one of the first principles of a licensing process. Lessons learned relating to the Licensing Framework, Design Assessment and the requirements correlated with a “first-of-a-kind” nuclear facility have been captured in the Design Engagement Framework (PP-0008) and relate specifically to:

- close interaction with the designer/architect engineer,
- identification and agreement on key safety issues and the proposed technical resolution,
- adequately developed and stable design,
- agreement on a safety concept considering safety principles (e.g. DiD, redundancy, safety and quality classification (reliability)) and
- application of sound system engineering principles, past experience, robust research, test and qualification to demonstrate that the design will survive all postulated transient and accident conditions.

The PBMR Company changed business strategy and design of the prototype HTR in 2008 to make provision for process heat application. The design change also considered the technical issues associated with the direct Brayton cycle design. Until the end of the project PBMR did not have the opportunity to implement PP-0008

4.3.5.3 Republic of South Korea: NHDD

Regarding the energy needs South Korea identified nuclear sources for process heat application as promising and valuable opportunity. For the long term the intended HTR application ranges from Hydrogen production (800 – 950 °C) and consumption for advanced steel making process, synthetic fuel (500 – 900 °C), electricity (around 750 °C) to process heat (300 – 600 °C) /91/. For these applications the NHDD project (Nuclear Hydrogen Development and Demonstration) was initiated.

Regarding the licensing of a prototype NHDD system the following key technology developments are identified for R&D:

- Design and analysis codes
- Helium Test Loop
- Material & Components
- TRISO Fuels
- Hydrogen Production Processes

For the preparation of the NHDD licensing process the Nuclear Hydrogen Key Technology Development Project was initiated and development and for the accompanying licensing a step-wise approach is applied

The licensing process will follow the strategy of safety by demonstration and start with a pre-licensing review. The step-wise approach will address in the first stage residual R&D needs and key technologies (until 2027). The overlapping second stage will start 2015 and focus on a demonstration plant:

- Design (conceptual, basically, detailed)
- Licensing (regulations, pre-application, PSAR, FSAR)
- Supply Chain (regulations, SSC, fuel plant)
- Construction (siting, construction)
- Demonstration

The subsequent third stage of the licensing strategy will focus on commercial plants based on the experiences of the demonstrator program and a standard plan design.

In conclusion South Korea applies a step-wise licensing approach for the VHTR demonstration plant on existing regulations accompanied by R&D activities on VHTR key technologies essential for the realisation of the demonstrator. Specific licensing procedures for co-generation and in particular process heat applications are not defined.

4.3.5.4 China: HTR-PM

The licensing procedure of the prototype HTR-PM based on the following licensing base /89/:

- HTR-10 licensing experience of PSAR, FSAR, commissioning and testing
- HTR design criteria in particular for key systems after review of the corresponding HTR-10 documents by the licensing authority
- important licensing criteria as codes, standards, safety goals, key issues, reviewed and accepted by the licensing authority before the formal start of the licensing
- licensing experiences from French licensing authority and from other NPP projects already under construction in China considering the differences in safety features between LWR and HTR concepts

Formally the licensing of the HTR-PM follows the standard procedure:

- submission of a PSAR
- technical briefing
- review conferences (2 times)
- key issues discussion and PSA review
- key issues discussion (5 times)
- key issues concluding discussions
- safety advisory committee
- answering to identified questions (more than 2000)
- proposal for additional testing related to 16 issues
- environmental impact analysis
- emergency planning
- preparation and submission of FSAR

In conclusion any specific requirement or condition related to HTR technology or co-generation and process heat application is covered by the standard licensing procedure applying experiences from the previous HTR-10 project and specific investigations on HTR specific key issues. Key issues and the need for further investigations are identified by the review of the HTR-10 documentation, other licensing regulations and the actual literature.

5 Derived licensing requirements for new HTR with co-generation installations

The nuclear energy industry has been traditionally using light water reactor (LWR) technology for the generation of electricity. This technology limits process steam temperatures to approximately 300°C. Alternatively, high temperature gas-cooled reactor (HTGR) technology can provide both electricity and the high temperature heat needed for industrial processes. In fact, HTGR technology is expected to supply process heat as an alternative to existing carbon-based sources in commercial applications. These potential applications include co-generation of electricity and steam supply or high temperature gas supply to petrochemical and refining plants, electricity and steam supply for oil sands oil recovery, and high temperature steam or gas and electricity for hydrogen production. Applied in this way, HTGR technology can significantly reduce the use of premium fossil fuels for the production of process heat and reduce greenhouse gas releases, thus providing a significant competitive advantage for industrial markets.

Licensing of a nuclear installation is the method of the regulatory body to ensure the fulfilment of all regulatory and legal requirements in a country to authorize the establishment of a nuclear installation or initiation of its activities. This means all technical and organizational issues related to the design and construction of the facility have to be clarified and solved by the customer. In case of new technologies beyond the respective standard technique - as to be expected for a prototype – the licensing procedure can be realized by a stepwise approach of presentation of an individual technology or operation by the applicant and by the corresponding stepwise licensing by the regulatory body. But this step-wise procedure usually is correlated with a longer total licensing period. Licensing of technologies or operations not yet finally fixed can also be handled by licensing conditions. This may allow proceeding with the licensing process but creates the risk for the applicant that the licensing conditions finally may not met by the respective technology.

In addition the licensing procedure may be split in different sections with respect to the task to be licensed as construction, operation or decommissioning (graded approach). This provides some flexibility for the regulatory body and some reliability for the applicant on the validity of the applied regulations.

It should be noted, however, that a graded approach may bear specific risks, some of which showed up in the licensing process of the THTR. If a plant design still needs to be significantly detailed during a multi-step licensing scheme, a time-consuming process is likely to develop bearing additional risks of tightening regulatory requirements over the licensing period, resulting in expensive backfitting measures during construction.

The licensing of nuclear facilities follows in general the same procedure covering the main aspects as:

- Definition of the nuclear facility, its activities and the respective boundary conditions (e.g. dose and discharge limits, action levels)
- Siting and site evaluation
- Safety assessment and environmental impact assessment
- Safety demonstration of the proposed technology, SSC's and the complete facility for all operation stages and design based accident conditions
- Public participation
- Construction
- Commissioning
- Operation
- Decommissioning

Based on basic safety requirements defined by the regulatory body it is recommended to establish a pre-licensing process to organize e.g. an early approval of the sites and certifications of standardized plant designs in advance to the authorization by the regulatory body. This pre-licensing process will be in particular useful for a prototype facing multiple technical options to reach certain goals. For each of these technical options a well prepared documentation of the involved materials, manufacturing procedures and test and control requirements will be needed to reach a certification status sufficient for a licensing. Further details to the general licensing process can be found in the corresponding IAEA report /86/.

The contributions from the European partners on past nuclear co-generation applications demonstrate that no specific licensing requirements are defined for a co-generation application beyond the general licensing requirements for a nuclear facility. For an HTR co-generation application only quantitative differences in the respective operation limits and operation procedures to standard LWR facilities have to be considered but the licensing procedure in general remains unaffected.

Nevertheless the following licensing requirements need some more attention for an HTR co-generation application:

Minimum Distances

The HTR and the conventional end user facility shall not influence each other in particular in case of severe accidents as explosions or release of corrosive materials. The safety related risk by external hazards for the HTGR defined in the SAR shall be independent from the co-generation components and the end user facility. While this is usually covered for a standard NPP because of the requirement “as much as possible away from any external hazards”, for co-generation applications a limited distance to the end users facility is mandatory for economic reasons (investment for the transfer line). Therefore any external hazard for the reactor by the end user facility has to be evaluated in particular.

Regarding the potential impact of the reactor on the end users facility it should be noted that because of the passive safety characteristics of HTR’s (below 600 MWth), the low power density and the consequential elimination of scenarios comparable to LWR core melting scenarios, the derived HTR exclusion zone is smaller than for other NPP types, allowing a closer distance to the end user facility or other public infrastructure (schools, stadium, etc.). Demonstrated by investigations on the size of the exclusion zone this area may be limited to the NPP/industrial site itself.

Radionuclide release limits

Main licensing criteria of NPP is the compliance with radionuclide release dose limits defined by the regulatory body. These limits are given for gaseous, liquid and solid conditions. For a gaseous release the nuclide specific limits to be considered are requested for the closest distance between release stack and nuclear island boundary.

For co-generation application radionuclides may transported by the transfer medium in constant concentrations a longer distance than considered by standard release calculations. Therefore the radionuclide release scenarios have to consider this release path way option adequately (at HTR conditions) and design and safety systems of the secondary circuit and the transfer system has to limit the radionuclide concentration in the transfer medium below the regulatory limits in any case. This will usually result in technical measures to quickly isolate the transfer line in case of accidental contamination.

It should be mentioned that the radionuclide limits applied by the regulatory body may vary between the values defined by the European Directive on Radiation Protection /96/ and the natural background level of the radionuclide (e.g. in France, any addition of artificial nuclides to byproducts is forbidden and for other cases (like intermediate steam in closed circuit) European directive is to be applied at least)..

Besides the regulatory requirements the compliance of the radionuclide concentration in the transfer medium and finally in the end user facility/product is probably the most important aspect to reach positive acceptance levels in the public for a co-generation system.

Thermal hydraulic feedback/transients

For HTR process heat applications the end user will be the main consumer. Varying operation conditions at the transfer system or the end user side will generate feedback/transients to the HTR. These feedbacks and transients have to be considered in the safety analysis report of the HTR and covered by corresponding safety systems (e.g. compressor chambers).

In case of co-generation systems with a limited power share (heat or electricity production in the range of 10 % total power), the co-generation system shall be designed in a way that any end user operation condition will not create a safety related thermal hydraulic feedback or nuclear transients (e.g. cold source) to the nuclear systems of the HTR in order to avoid additional safety discussion for the HTR.

For economic reasons the impact from the HTR operation to the end user system should be avoided/limited and a reliable heat/steam source should be provided (e.g. by a modular HTR concept or conventional backup systems).

6 Preliminary guidelines to support licensing process

This chapter covers preliminary guidelines to support a licensing process from the technical side to provide a consistent proposal for a HTR co-generation installation to the regulatory body.

Usually the licensing procedure for a nuclear reactor follows national laws and regulations and needs to be confirmed by the regulator authority. The main topics of the licensing procedure are given in the previous chapters. Depending on national regulations which might facilitate or impede this, it is recommended to support the licensing process by an initial pre-licensing phase to clarify the conditions for approval of the intended site and for acceptance of the key plant safety design features by the regulatory body in advance of the formal licensing process. For a co-generation or process heat application a public participation seems to be essential to demonstrate the gain of such an application for the local population and to discuss public concerns on a very early stage.

For a prototype HTR no standardized plant design will be available in the beginning of the licensing process. Therefore the report on the preliminary safety analysis (PSAR) will need more effort on technical aspects than for a standardized NPP. But technical aspects are not the only one part of the licensing process and the PSAR addresses usually a wider range of issues as:

- Location and environmental conditions
- Detailed description of the nuclear facility and applied safety technologies (nuclear grade certifications of materials, manufacturing processes and control procedures)
- Measures for radiation protection and minimization of radioactivity levels
- Operation modes of the nuclear facility (e.g. cold sub-critical, hot sub-critical, partial and full load level, abnormal operation limits, safety limits)
- Safety cases for design base accidents (adequate design of systems, structures and components (SSC) with regard to loads and requested safety margins)

In contrast to standardized NPP an HTR system offers some specific features which may need additional licensing support in terms of documentation or evaluations:

- higher temperature regime of coolant and SSC's
- Graphite structures (fuel, core structure, thermal shield)
- behaviour of gaseous He coolant (e.g. requirements on leak tightness and, bypass flows, impact on friction behaviour, purification system)
- consideration of HTR specific accident scenarios (e.g. water ingress, air ingress, dust release after leak or breaks in the primary pressure boundary, reactivity transients initiated by a failing top reflector, failure in the continuous fuel loading (specific for pebble bed cores))
- demonstration that severe LWR accident scenarios are not applicable to HTR's (e.g. core melting)
- advantageous passive heat removal concept
- thermal self-regulation feature of the criticality by strong negative temperature coefficient
- specific control and shut down systems

In preparation of the PSAR, related information for proposed materials, components and technologies should be collected and compared from past projects and experiments. Based on this information gaps should be identified and respective qualification measures have to be taken.

In conclusion the licensing process of a prototype HTR should be supported by:

- demonstration on the reliability of the technical approach of the HTR based on past experiences (various HTR designs, specific experiments on e.g. flow dynamic, fuel performance or accidental behaviour and corresponding numerical models for calculations) as "proven technology"
- documentation and evaluation of HTR specific features (e.g. materials, temperature levels, coolant, accident scenarios, system behaviour)
- clarification/confirmation of the fuel supply on industrial scale
- fixed specification of design, power level and cogenerated energy level in an early stage, preferably in the pre-licensing phase
- licensing and ordering of main components in an early stage because of the long delivery times

- concept of decommissioning and final disposal of the burned nuclear fuel and of the graphite of the core assembly
- efforts to gain or improve public acceptance of co-generation, consideration of end users needs

Activities on the optimization of HTR components to improve the economic efficiency (e.g. temperature level, tritium retention by the heat exchanger, final nuclear burnup, nuclear fuel material) should be addressed in a later stage.

7 Road map for licensing of a prototype HTR

The derived licensing requirements for an HTR co-generation capability in chapter 5 will be integrated in a generic road map for licensing of a prototype HTR. For details of the demonstrator see WP 4.4.

The licensing roadmap of a prototype HTR will be very similar to the licensing roadmap for a next-of-a-kind (NOAK) HTR because a licensing procedure of a nuclear facility has to consider in general the same topics. In addition national regulations on licensing of a nuclear facility usually do not contain specific regulations for prototypes. The main difference to a standard licensing roadmap will be considerations of technical and operational options for testing and optimization forward to the N-of-the-kind of this reactor type.

The licensing procedure of a prototype HTR should follow the road map given below:

0. Pre-licensing process
1. Preliminary Safety Report and review (e.g. definition of the nuclear facility, activities, boundary conditions, design accidents base)
2. Site selection
3. Public Inquiry (safety assessment and environmental impact assessment)
4. Final Safety Report (e.g. safety demonstration of the proposed technology, SSC's and the complete facility for all operation stages and design based accident conditions), review
5. Construction
6. Commissioning
7. Authorization of continuous reactor operation

<u>0. Pre-licensing process</u>

While not directly a part of the licensing procedure the pre-licensing process should be initiated in the beginning of the licensing roadmap to clarify the general conditions of the prototype HTR as power and temperature level range, cooling conditions/ultimate heat sink, expected service for co-generation or process heat end users, nationality of regulatory body, codes and standards requested by the regulator, identification of long lead items (LLI, large components with long lead and manufacturing time, e.g. RDB, heat exchanger).

<u>1. PSAR</u>

Without the definition of the design values of the prototype no subsequent investigations or evaluations or other steps of licensing procedure can be started. This information will be given in the PSAR.

Because of the long lead time of the LLI's the design of the prototype has to be defined and ordered as early as possible in the project. The fixed design is also very important for any discussion on end user applications and corresponding exclusions zones. With a fixed design also the way for a fuel supply chain at industrial scale can be initiated.

Knowledge, technology and materials for different prototype HTR designs below 600 MWth and maximum He coolant gas temperature of 750 °C are available. For selected designs even SAR are available which while outdated may nevertheless provide useful guidance for the preparation of the PSAR.

<u>2. Site selection</u>

For a prototype HTR with co-generation capability in particular the site selection is different from the usual guideline "as far as possible from external risks" because the HTR should be located as close as possible to the end user facility to minimize the costs for the respective transfer lines. Certainly the minimum distance between the HTR and the conventional facility (in case of process heat) must be determined and maintained in the site layout.

<u>3. Public inquiry</u>

For a positive acceptance of a nuclear new-build by the public the licensing process should be performed with high transparency and strong public participation. This offers the opportunity to demonstrate the advantages of the project and to discuss opportunities, risks and clarify preconceptions. For locations close to national borders the neighbouring countries should be involved in the inquiry.

Because of the interface to conventional systems in the local area a co-generation or process heat HTR will need particular effort for the public inquiry. A radiological study of the region with an appropriate baseline survey will provide reference information for future investigations. An environmental impact study will be performed analysing the impact of the nuclear facility on the local region. The study will have to address:

- risks for the nuclear installation
- risks for human health and environment, considering:
    - local population, population density, health and socio-economic aspects
    - impact on emergency preparedness
    - existing environmental conditions (air ,water earth, flora, fauna) and pre-existing contamination
- climate and scenery
- cultural properties and other physical products

With respect to the co-generation application the environmental impact study will also discuss effects related to end user products or services (e.g. concentration of radionuclides in syngas for cooking, heating, fuel). Therefore the range of the impact study should be increased to regions affected by distributed end users products or services.

4. FSAR

The minimization of any risk for the public should be demonstrated by detailed risk management studies based on the preliminary safety report of the nuclear facility. To demonstrate transparency and reliability of the applied methods and results of these studies an independent assessment (e.g. by NGO's etc.) should be organized after the finalization of the SAR.

5. Construction

While specific nuclear SSC's of the prototype HTR can only be manufactured by nuclear certified companies nevertheless a lot of conventional work has to be done for the construction. The economic advantage for the local industry should be stressed and presented already in the very beginning of the project to improve the public acceptance level. In addition the opportunity of local supply chains and should be evaluated and presented to the public.

6. Commissioning

For the commissioning of a prototype HTR a graded approach seems to be beneficial allowing the step-wise testing, optimization and approval of systems, components and operations procedures. In addition with a step-wise commissioning the progress of the project can be demonstrated illustratively.

7. Reactor authorization of operation

Finally the authorization of the continuous reactor operation (based on the safety assessment of the Safety Report) should cover also the safety tests intended for the prototype as far as not conducted in the initial operation procedure. A demonstration of the safety performance in an early licensing stage may improve in addition the public acceptance. Results of the safety test should be presented to the public to demonstrate transparency and the confirmation of the designed safety behaviour.

A short roadmap for licensing a prototype HTR is also given in the EUROPAIRS deliverable 3.2 /1/, chapter 6.2.6 (see EUROPAIRS Table below). According to this roadmap the licensing starts with SAR and environmental assessment. Nearly in parallel to the SAR the construction starts indicating a graded approach for the licensing procedure. For a more detailed roadmap the final design on the prototype HTR needs to be specified.

Reactor	2015	2018	Starts 1 year after start construction
Safety Analysis Report	2014	2017	Starts 1 year before reactor license
Environmental Assessment	2014	2017	Starts at site evaluation, 1 year before reactor license
Process Heat Application	2015	2018	Starts with Reactor license
Generic License	2022	2025	Starts at the end of prototype construction

8 Summary and Conclusion

In conclusion, in the screening of past and recent co-generation applications with a nuclear heat source no specific licensing issues beyond the standard licensing requirements of the NPP were identified. For some standard requirements a higher effort will be needed to address specific co-generation related aspects (e.g. evaluation of external hazards by nearby industrial facilities, quick isolation options for transfer lines out of the NPP site). Regarding other standard requirements the specific safety features of modular HTR will lead to a significant improvement of the nuclear safety (e.g. reduced exclusion zone possible because of limited radioactive releases even during beyond design basis events) and of the economic conditions (maximum achievable temperature level).

While emissions or releases of radionuclides from nuclear facilities are in general limited by the respective national regulations, for co-generation applications the aspect of a product free (related to international or national limits) from artificial radioactive contamination will be even more in the focus of the public. This strong interest should be met by promoting the radionuclide barriers already applied in modern NPP for normal operation and design based accidents and the additional advantages given by the HTR technology with its primary safety feature to retain the vast majority of fission products in the coated particles, both under operating and accident conditions.

Nevertheless the tritium issue related to gaseous primary coolant circuit needs to be discussed. The gaseous coolant allows the existence of elementary tritium in the primary coolant. Tritium, as well as hydrogen, is able to diffuse through the metallic walls of heat exchanger tubes or sheets at comparably high rates, especially at the high temperatures which are aimed at for process heat applications. Therefore the contamination of the secondary coolant circuit with tritium has to be considered in the radiological assessment. On the other hand the He purification systems are very efficient regarding tritium retention and additional contamination of the secondary circuit through leaks can be avoided by an overpressure in the secondary circuit (which will usually be the case for co-generation since efficient production of electricity requires high steam pressure). An additional barrier against the contamination of product stem or product gas with tritium can be provided by a tertiary circuit. In order to minimise the effort for the licensing process of the prototype HTR, such an additional circuit has been proposed in the EUROPAIRS project and is also supported in this project, although recent investigations in the ARCHER project indicate that reasonable limits of tritium contamination in process steam might also be met without a tertiary circuit. For a final assessment, a detailed investigation considering the applicable regulations of the country hosting the prototype HTR will be necessary. To improve the economy of an HTR co-generation application in the future, the promising results on heat exchangers with reduced tritium permeation should be evaluated and confirmed by additional R&D activities to finally demonstrate that tertiary heat exchangers are no longer required.

To effectively support the licensing of HTR based co-generation application and in particular a prototype facility the following activities should be conducted in addition and in advance (if possible) to the standard licensing procedure:

- in a pre-application phase, early discussion of the safety features specific for a modular HTR (e.g. passive decay heat removal, “vented containment”) with the regulator of the country hosting the Demonstrator with the aim to achieve clarity about their consideration in the licensing process
- demonstration that co-generation or process heat application issues are covered by the licensing procedure
- gap analysis for further R&D needs under consideration of the results achieved in the gap and SWOT analyses in the ARCHER project, see also deliverable D3.31 of this project.

The licensing of the HTR will follow the general licensing procedure covering the main aspects as:

- Definition of the nuclear facility, its activities and the respective boundary conditions (e.g. dose and discharge limits, action levels)
- Siting and site evaluation
- Safety assessment and environmental impact assessment
- Safety demonstration of the proposed technology, SSC's and the complete facility for all operation stages and design based accident conditions
- Public participation
- Construction
- Commissioning
- Operation
- Decommissioning

Because of the prototype issues and the strong interface to the local public a road map should include an extended pre-licensing phase with a strong public participation to force a positive acceptance level in the local public. An extended environmental impact study should be included analysing e.g. the impact of the nuclear facility on the local region (safety issues, economic advantages/opportunities during construction and operation), the status of the radiological background of the whole area and wide field impact by produced end user products or services (syngas, hydrogen or methanol based fuel, process steam, district heating).

For a prototype a fixed licensable design is essential. Open questions can be clarified during the pre-application phase, which is very important for a FOAK facility to receive a positive acceptance by the regulator.

The licensing process of a HTR powered co-generation installation should be supported by the typical actions for a NPP addressed in a PSAR. For the co-generation systems and components and in particular for the interface to the local public infrastructure or the conventional end user facility the licensing issues are covered by the nuclear standard regulations. But because of the larger and more visible interface outside the nuclear site, additional effort should be invested in the transparent description of the facility in an early stage of the project, discussing public concerns and promoting the advantages of the HTR based co-generation (safety behaviour, fossil fuel independency, wide range of temperature and pressure condition).

From the technical point for the licensing procedure an "accepted" HTR design (widely discussed and investigated, supported by test results from previous HTR prototypes) should be selected to confirm a can-be–build status close to a "proven technology" status. This comprises a reliable retention performance for any artificial radionuclide, in particular for tritium.

Beside the support of the licensing process regarding the technical issues, the feasibility of a fuel production in industrial scale should de demonstrated. This certainly comprises a demonstration of the fuel performance assumed in the safety case for all operation modes and all design based accident scenarios. While far in the future of the project nevertheless also concepts for the disposal of the finally burned nuclear fuel and the decommissioning of the nuclear facility should be prepared and presented.

The road map for licensing of a prototype HTR follows very closely the usual licensing procedure of an NPP because a prototype NPP has to be demonstrating the same level of safety as an NOAK NPP. Due to its FOAK nature, increased efforts for SSC testing and qualification have to be provided for.

The road map should also consider HTR specific long lead items because the manufacturing process of large HTR specific components may differ significantly from respective LWR components.

9 Acronyms and definitions

Acronym	Definition
EC DG RTD	European Commission – Directorate General for Research and Innovation
DiD	Defense-in-depth
HTR	High Temperature Reactor
NPP	Nuclear Power Plant
NOAK	Next of a kind
PRA	Probabilistic Risk Analysis
RIPB	Risk-informed and performance-based process
SSC	Systems, structures and components
TLRC	Top level regulatory criteria

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