2 CHARACTERISTICS OF THE REPORT

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TABLE OF CONTENTS:

2.1 Formal and legal grounds for the Report

The Strategic Environmental Assessment Report for the Polish Nuclear Programme (hereinafter referred to as the "Report" or "SEA Report") was prepared in accordance with the agreement signed by and between the Minister of Economy and the company Fundeko Łukasz Szkudlarek.

The Report was based on draft Polish Nuclear Programme dated 16 August 2010 and the position of the General Director of GDOS (General Directorate for Environmental Protection) and Chief Sanitary Inspectorate regarding the scope of the Strategic Environmental Assessment Report for this document.

The decision to develop the Polish Nuclear Programme was adopted by resolution of the Polish Council of Ministers No. 4/2009 of **13 January 2009** on actions taken to develop the Polish nuclear power sector, which provided that:

"To ensure the national energy security, and taking into account the economic development, a Polish nuclear power program shall be developed and implemented. The draft of such program shall be developed and submitted to the Council of Ministers by the government's plenipotentiary; this program shall determine the nuclear power plants' number, size and possible sites. Moreover, the government obligates the National Treasury Minister to ensure that PGE Polish Energy Group SA shall cooperate on the program's development and implementation. "

On 10 November 2009, the Council of Ministers adopted Poland's Energy Policy until 2030. The document provides that one of the main directions of development of Poland's energy policy is *"diversification of the generation structure by introducing nuclear energy."* The adoption of the policy took place after the strategic environmental assessment for the effects of implementation of the Polish energy policy until 2030 was prepared, which included public consultations. Therefore, it must be emphasized that the present Strategic Environmental Assessment Report for the Polish Nuclear Programme is not a document that was meant to introduce nuclear energy in Poland or to justify such actions (the rationale for the Polish Nuclear Programme has already been presented in the Strategic Environmental Assessment for the Polish Energy Policy until 2030).

The Polish Nuclear Programme only provides the framework and the schedule of actions needed to introduce nuclear energy and, thus, to achieve the objectives of the Polish Energy Policy until 2030. It presents the scope and the organizational structure of the actions that must be taken to implement nuclear energy and to assure safe and effective operation of nuclear energy facilities. A result of such implementation will be the commissioning of the first two nuclear power plants in Poland, which will have some environmental impacts. Therefore, this Report focuses on analyzing and assessing such impacts.

The report does not avoid the issue of impacts resulting from the expansion of the necessary power distribution infrastructure, the fuel cycle including the generation, transport, and storage of radioactive waste. However, this is not included in the subject matter of the Report. It must be emphasized that a detailed discussion of the aforementioned issues related to the environmental impact of expansion of the necessary power distribution infrastructure and the selection of the fuel type and the best, in the Polish conditions, method of storage of radioactive waste will be included in the strategic environmental assessment reports for the updated Development Plan for Satisfying the Present and Future Demand for Electric Energy and the National Radioactive Waste and Spent Nuclear Fuel Handling Plan.

This approach is in compliance with **art. 5.2 of Directive 2001/42/EC of the European Parliament and of the Council of 27 June 2001 on the assessment of the effects of certain plans and programmes on the environment:**

"The environmental report prepared pursuant to paragraph 1 shall include the information that may reasonably be required taking into account [...] the contents and level of detail in the plan or programme, its stage in the decision-making process and the extent to which certain matters are more appropriately assessed at different levels in that process ***in order to avoid duplication of the assessment.****"*

Given the above, the following Report was elaborated.

2.1.1 The basic sources of community law regulating the Environmental Assessment procedure:

- Directive 2001/42/EC of the European Parliament and of the Council of 27 June 2001 on the assessment of the effects of certain plans and programmes on the environment (SEA directive);
- Council Directive 85/337/EEC of 27 June 1985 on the assessment of the effects of certain public and private projects on the environment (EIA directive);
- Council Directive 92/43/EEC of 21 May 1992 on the conservation of natural habitats and of wild fauna and flora (Habitat directive);
- Council Directive 79/409/EEC of 2 April 1979 on the conservation of wild birds (Bird directive).

2.1.2 The basic sources of Polish law regulating the Environmental Assessment procedure:

- Act of 27 April 2001, Environmental Protection Law (AEPL);
- Act of 16 April 2004 on the protection of the environment (APE)
- Regulation of the Council of Ministers of 9 November 2010 concerning undertaking with potential high environmental impact (EA Regulation);
- Act of 3 October 2008 on Access to Information on the Environment and Its Protection, Public Participation in Environmental Protection and on Environmental Impact Assessments (the EA Act);
- Act of 27 March 2003 on spatial planning and management;
- Act of 7 July 1994 – Building Code;
- Act of 18 July 2001 – Water Code;
- Act of 14 June 1960 – Code of administrative procedure (CAP).

2.1.3 The basic sources of international law that regulate the environmental impact of nuclear energy facilities

European Union law:

- The EURATOM treaty;
- Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear installations;
- Council Directive 96/29/Euratom: basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation;
- Council Decision 87/600/Euratom of 14 December 1987 on Community arrangements for the early exchange of information in the event of a radiological emergency (6) which established the

framework for notification and transfer of information to be used by member states to protect the population in the case of a radiological emergency;

- Council Directive 89/618/Euratom of 27 November 1989 on informing the general public about health protection measures to be applied and steps to be taken in the event of a radiological emergency (7) which imposes on member states the duty to inform the public about radiological emergencies;
- Directive 2001/81/EC of the European Parliament and of the Council of 23 October 2001 on national emission ceilings for certain atmospheric pollutants (so-called NEC directive);
- Directive 2008/1/EC of the European Parliament and of the Council of 15 January 2008 concerning integrated pollution prevention and control and Council Directive 96/61/EC of 24 September 1996 concerning integrated pollution prevention and control (the "IPPC" directive);
- Directive 2001/80/EC on the limitation of emissions of certain pollutants into the air from large combustion plants (the LCP directive);
- Directive of the European Parliament and of the Council of 7 July 2010 on industrial emissions.

International conventions signed and ratified by Poland

- United Nations Framework Convention on Climate Change made in New York on 9 May 1992;
- Kyoto Protocol to the United Nations Framework Convention on Climate Change, made in Kyoto on 11 December 1997;
- Convention on Nuclear Safety (1996);
- Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (2001);
- Convention on Early Notification of a Nuclear Accident (1986);
- Convention on Assistance in the Case of Nuclear Accident or Radiological Emergency (1987);
- Convention on the Physical Protection of Nuclear Material (1987);
- Nuclear Non-Proliferation Treaty (NPT).

2.1.4 International requirements, guidelines, recommendations, and standards

- *European Utility Requirements for LWR Nuclear Power Plants, "EUR" 2001;*
- IAEA standards[2];
- WENRA recommendations[3].

2.1.5 Selective secondary legislation to the Atomic Law (currently in force):

1. Act of 29 November 2000 – Atomic Law;
2. REGULATION OF THE COUNCIL OF MINISTERS of 21 April 2004 amending the regulation on the documents required upon the submission of a licensing request for operations which involve exposure to ionising radiation or upon reporting such operations;
3. Regulation of the Council of Ministers of 21 October 2008 concerning the permit and approval for importation into the territory of the Republic of Poland, exportation from the territory of the

Republic of Poland, and transit through this territory of radioactive waste and spent nuclear fuel – in force since 25 December 2008;

4. Regulation of the Council Of Ministers of 4 November 2008 on the physical protection of nuclear materials and nuclear facilities – in force since 25 December 2008;

5. Regulation of the Council of Ministers of 18 January 2005 on ionizing radiation dose limits;

6. Regulation of the Council of Ministers of 6 August 2002 on nuclear regulatory inspectors;

7. Regulation of the Council of Ministers of 20 February 2007 on the basic requirements for controlled and supervised areas;

8. Regulation of the Council of Ministers of 12 July 2006 on detailed safety requirements for work involving ionising radiation sources;

9. Regulation of the Council of Ministers of 23 March 2007 on the requirements for the individual dose registration;

10. Regulation of the Council of Ministers of 3 December 2002 on radioactive waste and spent nuclear fuel;

11. Regulation of the Council of Ministers of 23 December 2002 on the requirements for dosimetric equipment;

12. Regulation of the Minister of Environment of 30 December 2002 on detailed rules for the creation of a restricted-use area surrounding nuclear facility, indicating relevant restrictions concerning its uses;

13. Regulation of the Council of Ministers of 18 January 2005 on emergency plans for radiological emergencies;

14. Regulation of the Council of Ministers of 20 February 2007 amending the regulation on emergency plans for radiological emergencies;

15. Regulation of the Council of Ministers of 27 April 2004 on intervention levels for various intervention measures and criteria for cancelling intervention measures;

16. Regulation of the Council of Ministers of 27 April 2004 on prior information to the general public in the event of a radiation emergency;

17. Regulation of the Council of Ministers of 27 April 2004 on the protection against ionising radiation of outside workers exposed during their activities in controlled areas;

18. Regulation of the Council of Ministers of 27 April 2004 on the determination of entities competent to inspect maximum permitted levels of radioactive contamination of foodstuffs and feeding stuffs following a radiation event;

19. Regulation of the Council of Ministers 20 February 2007 on the terms for import into the territory of the Republic of Poland, export from the territory of the Republic of Poland and transit through this territory of nuclear materials, radioactive sources and equipment containing such sources.

2.1.6 **Polish regulations in the course of elaboration (with agreed essential contents)**

- Draft amendment to the Atomic Law of 2010:

  - Assumptions of the draft act on amendment of the Atomic Law and on amendment of some other statutes, constituting a transposition of the Council Directive 2009/71/Euratom of 25 June 2009 (EU Official Journal L 172 of 2 July 2009, p. 18, and EU Official Journal L 260 of 3 October 2009, p. 40), version dated 31 May 2010 - adopted by the Council of Ministers on 22 June 2010;
  - Draft act on amendment of the Atomic Law and on amendment of some other statutes – version dated 26 October 2010 (elaborated by the Government Legislation Centre).

- Draft Regulation of the Council of Ministers of ... on the factors considered when performing an evaluation of the site intended for location of a nuclear facility and on the requirements for the siting report for a nuclear facility;
- Draft Regulation of the Council of Ministers of ... on the basic nuclear safety and radiological protection requirements that must be taken into account in designs of nuclear facilities;
- Draft Regulation of the Council of Ministers of ... on the nuclear safety and radiological protection requirements at the commissioning and operation stages of nuclear facilities;
- Draft Regulation of the Council of Ministers of ... on the nuclear safety and radiological protection requirements at the decommissioning of nuclear facilities and on the content of report from the decommissioning of nuclear facilities;
- Draft Regulation of the Council of Ministers of ... concerning requirements for safety analyses conducted prior to applying for a permit to build a nuclear facility and for the content of safety report for a nuclear facility.

2.2 Content and main objectives of the program

2.2.1 Content of the Program draft

Chapter I presents an introduction to the nuclear energy programme in Poland and the main reasons for its implementation.

Chapter II presents the schedule and the detailed lists of preparatory tasks necessary to implement nuclear energy programme in Poland, divided into the individual states and with the dates of their implementation.

Chapter III refers to the European Energy Policy and the Polish Energy Policy until 2030 and presents important decisions that have been made concerning the development of the nuclear energy programme in Poland.

Chapter IV presents an analysis of the costs and the economic justification for development of the nuclear energy sector. The analysis assumes the individual costs of electricity generation based on the forecasts of global research centres, with a very conservative approach to state-of-the-art technologies (mostly nuclear power plants). The results of the analysis, in line with the facts presented in the program, confirm the significant advantage of nuclear power plants compared to conventional power plants and RES. Moreover, the analysis confirms the need to implement the nuclear energy programme due to the need to assure the operation of the Polish energy system after 2020 with Poland fulfilling its obligations.

Chapter V describes the organization of the works related to the Programme's implementation.

Chapter VI covers the matter of establishing the conditions for safe implementation of the nuclear energy sector.

Chapter VII discusses the cost of performance and sources of financing of the Programme.

Chapter VIII discusses the issue of selection of the locations for the future nuclear plants in Poland. The chapter presents a review of the siting studies for nuclear power plants in Poland performed before 1990 and information on the current status of the works related to the updates of the previous studies and the new research. The results of the studies was the ranking list prepared by the Ministry of Economy, which included 28 locations. The results of the analysis were submitted to the future investor, the Polska Grupa Energetyczna S.A. (PGE) which selected 4 main locations and 2 backup locations for further detailed studies.

Chapter IX discusses the matter of preparation and the required changes in the national power grid system. It mentions the need to expand the national grid system, in particular the 400 kV lines. Upgrade of the system was included in the development plan, to the extent necessary to satisfy the current and future demand for electricity in the years 2010-2025, prepared by the PSE-Operator S.A. The Polish Nuclear Programme underscores the fact that the solutions proposed in the development plan are inadequate and that it is necessary to determine the basic criteria that the connection of the nuclear power plant to the National Power Grid must meet. The Programme indicates that this task should be performed in close cooperation between the PSE-Operator S.A, the Investor, and the Energy Regulatory Office and with the support of independent consultants and experts. The chapter also points at important problems that must be resolved on the occasion of expansion of the National Power Grid, which are connected mostly to the long and excessive administrative procedures.

Chapter X focuses on environmental protection. It discusses mostly questions related to CO2 emissions and, eventually, it will be replaced with this Report.

Chapter XI emphasizes the need to assure an appropriate number of qualified staff at the project preparation stage, construction phase, and operation phase. It was clearly stated that failure to complete the basic intent described in the chapter will constitute a serious risk to timely completion of the Programme.

Chapter XII describes the formation of the National Centre for Nuclear Research as the technical and scientific-research support for the Polish nuclear energy programme. The Centre will support the government and the nuclear regulatory office in the area of safe operation of nuclear facilities.

Chapter XIII pertains to the very important matter of assuring safe supply of nuclear fuel. It provides the basic information on the capacity and availability of uranium deposits in the world and, potentially, in Poland.

Chapter XIV pertains to the management of radioactive materials at various stages of the fuel cycle. The chapter describes the basic methods of handling spent nuclear fuels used in the world. The chapter also describes Poland's experience with radioactive waste that has been collected in the course of operation of the radioactive waste repository in Różan. It also includes a brief presentation of the actions planned with regards to management of radioactive waste in Poland related to the nuclear energy programme.

Chapter XV discusses the potential benefits to the domestic industry as a result of implementation of the Polish Nuclear programme. The chapter discusses the Investor's efforts to involve the Polish industry as well as the future benefits to the Polish economy.

Chapter XVI focuses on the issue of public participation and support for the Polish Nuclear programme. The chapter presents a list of actions needed to win public support for the location of the first nuclear power plant in Poland.

2.2.2 The main objectives and directions of actions assumed in the Program draft

The preparatory activities related to the implementation of the Polish Nuclear Energy Program are to be performed in compliance with Poland's domestic laws and with full respect to international laws and EU regulations, as well as the recommendations of the International Atomic Energy Agency (IAEA).

The schedule of the activities results from the provisions of Poland's Energy Policy until 2030, the Strategic Plan of the Ministry of Economy, and the Framework Schedule of Actions for Nuclear Energy. It is also one of the necessary tools to assure the economic growth by increasing the potential of the Polish energy sector which was described in the report titled "Poland 2030. Development challenges." The schedule of the Polish Nuclear programme is shown in the table below.

Table 2.2.1 Schedule of the Polish Nuclear Programme

PHASE	DATES	TASKS
Stage I	by 30 June 2011	elaboration and adoption by the Council of Ministers of the Polish Nuclear Programme by 31 December 2010 adoption and entry into force of the laws required for the development and functioning of the nuclear energy sector by 30 June 2011
Stage II	1.07.2011 - 31.12.2013	determination of the location and conclusion of the contract for the construction of the first nuclear plant
Stage III	1.01.2014 - 31.12.2015	elaboration of the technical design and obtaining of all the legally required approvals
Stage IV	1.01.2016 - 31.12.2022	building permit and construction of the first power unit of the first nuclear plant; commencement of construction of further power units/nuclear plants
Stage V	1.01.2023 - 31.12.2030	construction of additional power units/nuclear power plants

Of key importance to timely completion of the individual stages is timely completion of the most important actions comprised in Stage I, in particular entry into force of the laws required for the development and functioning of the nuclear power sector in Poland. Any delays in this area will result in postponement of the dates of completion of the successive stages. At the time of writing of this Report, it is known that the Polish Nuclear programme will not be elaborated and adopted by the Council of Ministers by 31 December 2010 due to the need to perform the strategic environmental impact assessment. Consequently, the adoption of the programme will most likely be delayed by approximately six months, i.e. until 30 June 2011. Thus, the successive stages will be pushed back by the same periods of time.

Table 2.2.1 Description of the actions enumerated in the Polish Nuclear programme

ACTION NUMBER	ACTION NAME	ACTION OBJECTIVE
Action 1	Legal framework for the construction and functioning of the nuclear energy sector in Poland	The objective is to elaborate, adopt, and implement laws that are necessary to allow building of the nuclear energy sector and the associated infrastructure, as well as its functioning. The application of such laws will be systematically monitored and evaluated. The necessary changes will be introduced in an ongoing fashion.
Action 2	Elaboration and implementation of the Polish Nuclear programme	The objective is to elaborate and implement the PNE Programme as the necessary comprehensive basis for all the efforts related

ACTION NUMBER	ACTION NAME	ACTION OBJECTIVE
		to introduction of the nuclear energy sector into Poland in the fastest and most effective way
Action 3	Nuclear power plants siting analyses	The objective is to select potential locations for nuclear power plants
Action 4	The final phase of the fuel cycle – management of radioactive waste and spent fuel – Analysis and studies concerning the location for low- and moderate-activity radioactive waste repository, preparation of the design of the repository, and construction of the repository.	The objective is to determine the location for a new low- and moderate-activity radioactive waste repository due to the fact that the currently used repository is nearly completely full. Preparation of the design of the new repository and construction of the repository.
Action 5	The final phase of the fuel cycle – management of radioactive waste and spent nuclear fuel – National radioactive waste and spent nuclear fuel handling plan	The objective is to prepare and to implement a technically and economically reasonable and socially acceptable management of radioactive waste and spent nuclear fuel, which is one of the key elements related to the functioning of the nuclear energy sector
Action 6	Education and training of staff for institutions and businesses involved with the nuclear energy sector	The objective is to prepare the staff for the Polish nuclear energy sector, both for the preparation and development of the infrastructure, and for the operation of nuclear power plants
Action 7	Information and educational campaign	The objective is to present to the public a credible and reliable information on nuclear energy and to improve the public's knowledge of this matter by education activities
Action 8	Scientific and research institutions	The objective is to form strong scientific and research institutions working for the nuclear energy sector, which is necessary for Poland to take full advantage of the advantages and opportunities related to its introduction
Action 9	Participation of Poland’s industry in the nuclear energy programme	The objective is to assure the broadest possible participation of the Polish industry in the supply of equipment for the nuclear energy sector and of Polish companies in the construction of nuclear power plants in Poland and in other countries
Action 10	Initial phase of the fuel cycle – assuring the supply of uranium from foreign and domestic sources	The objective is to obtain data about the deposits of uranium present in the territory of Poland and about the possibilities of its use, as well as to obtain information on the most advantageous potential supply of uranium for the Polish nuclear power plants
Action 11	Functioning of the nuclear and radiological regulatory institution	The objective is to assure the functioning of an independent, modern, and professional nuclear and radiological regulatory institution which, as a public-trust institution, will be able to meet the challenges related to the development of the nuclear energy sector in Poland

ACTION NUMBER	ACTION NAME	ACTION OBJECTIVE

The presented actions constituting the basis of the Polish Nuclear Programme cover mainly legal, organisational, and formal actions, and as such have no negative impacts on the natural environment. The result of these activities will be the launch of the first nuclear power plants in Poland. This Report focuses on the environmental impacts of this outcome.

2.3 Assumptions made in the Report

2.3.1 The objective and the scope of the Report

The purpose of this document is to analyze the potential environmental impact of the results of the implementation of the Polish Nuclear Programme. The Strategic Environmental Assessment Report contains:

- information on the contents, the main characteristics of the proposed document, and its relations with other documents;
- information on the methods used when preparing the Report;
- proposals regarding the anticipated methods of analyzing the outcomes of implementation of the provisions of the proposed document and the frequency of the analysis;
- information on possible trans-border environmental impact of the programme;
- a non-technical summary.

This Strategic Environmental Assessment Report also defines, analyzes, and evaluates:

- the present condition of the environment and the potential changes to the conditions in the event that the proposed document is not implemented;
- the condition of the environment in the areas where significant impact is anticipated;
- existing problems with environmental protections that are important from the point of view of implementation of the proposed document, in particular those pertaining to the areas that are protected under the Act of 16 April 2004 on environmental protection;
- the environmental protection objectives set forth on the international, community, and national level that are important from the point of view of the proposed document and the ways that these objectives and other environmental problems have been considered during the preparation of the document;
- the anticipated significant impacts, to include direct, indirect, secondary, cumulative, short-term, mid-term, and long-term, permanent and temporary, as well as positive and negative impacts on the objectives and object of protection of Natura 2000 areas and the integrity of such areas, as well as on the environment, in particular on biodiversity, people, animals, plants, water, air, Earth surface, landscape, climate, natural resources, historical monuments, and material goods, taking into account the relations between these elements of the environment and between the impacts on these elements.

To the extent that is appropriate in the case of this analysis, the following were also taken into account:

- solutions intended for prevention, limitation, or environmental compensation of negative environmental impacts that may result from the implementation of the proposed document, in particular the impacts on the objectives and objects of protection of Natura 2000 areas and their integrity;

- the objectives and the geographic range of the document and the objectives and the object of protection of Natura 2000 areas and their integrity – solutions that are alternative to those presented in the proposed document, with justification for their selection and a description of the methods of evaluation resulting in the selection, or explanation of the reasons for the lack of alternative solutions, to include information on encountered difficulties resulting from shortcomings of the technology or inadequate knowledge.

The table below shows how the content of this Report will be made to conform to Art. 51 of the Act of 3 October 2008 on Access to Information on the Environment and its Protection, Public Participation in Environmental Protection and on Environmental Impact Assessments (Journal of Laws No. 199 item 1227).

Table 2.3.1 Description of the actions enumerated in the Polish Nuclear Programme

REPORT CONTENTS REQUIRED BY THE STATUTE		CHAPTER
information on the contents, the main characteristics of the proposed document, and its relations with other documents		2, **3**, **6.3**, **6.6**
information on the methods used when preparing the Report		**2.3**, 7, 8, 10
proposals regarding the anticipated methods of analyzing the outcomes of implementation of the provisions of the proposed document and the frequency of the analysis		7,8, 10.4, **11**
information on possible trans-border environmental impact of the programme		**9.5**, 10.3
a non-technical summary		**1**
ANALYSES AND EVALUATIONS		**CHAPTER**
the present condition of the environment and the potential changes to the conditions in the event that the proposed document is not implemented		**4**, **5**, 8.3.2, 10.3
the condition of the environment in the areas where significant impact is anticipated		**4**, **10.3**
existing problems with environmental protections that are important from the point of view of implementation of the proposed document, in particular those pertaining to the areas that are protected under the Act of 16 April 2004 on environmental protection		**4**, 5, 7, 8, **10.3**
the environmental protection objectives set forth on the international, community, and national level that are important from the point of view of the proposed document and the ways that these objectives and other environmental problems have been considered during the preparation of the document		**3**, **6.3**, **6.6**
the anticipated significant impacts, to include direct, indirect, secondary, cumulative, short-term, mid-term, and long-term, permanent and temporary, as well as positive and negative impacts on the objectives and object of protection of Natura 2000 areas and the integrity of such areas, as well as on	biodiversity	4.9, **4.10**, **8.5**, **9.3**, **10.3**
	people	5, 7, 8, **9.1.1**, **9.6**, **10.3**
	animals	4.9, **4.10**, **8.3.2**, 8.3.5, 8.3.7, **8.5**, **9.3**, **10.3**
	plants	4.9, **4.10**, **8.3.2**, 8.3.5, 8.3.7, **8.5**, **9.3**, **10.3**
	water	4.3, 4.4, **7.6**, **8.2.1**, **8.3.2**,

the environment, in particular on:		**8.3.3**, 8.4, **9.1.2**, **9.1.3**, 10.3
	air	4.5, **5**, **7.2**, **8.2.2**, **8.3.2**, **8.3.4**, **9.1.4**, **10**
	Earth surface	4.1, **8.3.6**, **9.1.6**, **10.3**
	landscape	4.1, **4.9**, **8.3.8**, **9.1.7**, **10.3**
	climate	**5**, **8.2.2**, **9.1.5**, **10**
	natural resources	**8.3.1**, **9.1.8**, 10.3
	historical monuments	4.8, **9.1.9**, 10.3
	material goods	4.8, **9.1.10**, 10.3
taking into account the relations between these elements of the environment and between the impacts on these elements		4, 5, 6, 7, 8, 9.1, 9.2, 9.3, 10, 11

THE WAY THAT THE FOLLOWING ELEMENTS WERE CONSIDERED	CHAPTER
solutions intended for prevention, limitation, or environmental compensation of negative environmental impacts that may result from the implementation of the proposed document, in particular the impacts on the objectives and objects of protection of Natura 2000 areas and their integrity	6.3, 6.6, 10, **11**
the objectives and the geographic range of the document and the objectives and the object of protection of Natura 2000 areas and their integrity – solutions that are alternative to those presented in the proposed document, with justification for their selection and a description of the methods of evaluation resulting in the selection, or explanation of the reasons for the lack of alternative solutions, to include information on encountered difficulties resulting from shortcomings of the technology or inadequate knowledge	**4.9**, 4.10, 6.4, 8.5, 10, **10.3**, 11

2.3.2 **Methods applied in the Strategic Environmental Assessment**

There are two basic methods applied to conduct a strategic environmental assessment[4]:

- **The first method** is based on the procedure of environmental impact assessment that is used for specific projects in the course of an administrative process resulting in the issue of a permit for the performance of the project. It is based on a formal procedure, which is often separate from the procedure of the strategic document itself which is the object of the report, and which separately lists each project whose framework is defined in the proposed document. As a result, environmental impacts of a project are defined as precisely as possible and proven in a scientific manner. The review of alternatives is based mainly on location or technology alternatives within the adopted or evaluated option. The model works well in the case of documents that define the framework of performance of specific projects with a similar form and scope at the evaluation stage.

- **The second method** is based on the British experiences with policy appraisal. The main element of this method is to define the objectives of the document itself and to evaluate their implementation – as opposed to evaluating the direct environmental impact of individual projects. This procedure is much less formal and more condensed than the first model. It focuses more on the relationship between the assessment and the decision-making process that includes the assessment as its integral part. This model works well in appraisal of policies, development strategies, and statutes – documents that do not define the framework for implementation of the different documents but rather the frameworks and directions of development of various processes in the social, economic, legal, and environmental arena.

In principle, the Report applies the first method to analyse the possible environmental impacts resulting from the construction of the first nuclear power plants in Poland as thoroughly as possible, based on the available information regarding both the environmental impact of nuclear plants and their potential locations.

For this purposes, a matrix of environmental impacts that may result from the planned installation of different types of nuclear reactors in Poland was elaborated. The Report also focuses on the analyses of potential locations of nuclear power plants recommended by the Ministry of Economy, as well as their possible environmental impacts. For each of those locations, their anticipated significant environmental impacts were evaluated to the extent that was possible with the information available as of the date of the Report. The following elements were also considered:

- the nature of the impacts (direct, indirect, secondary, and cumulative);
- the duration of the impacts (short-, mid-, and long-term);
- the frequency of the impacts (continuous and momentary).

Also, the likelihood of occurrence of the anticipated significant impact on the environment and on Natura 2000 areas was analyzed. In determining the anticipated impact on Natura 2000 areas, the impact on the object of their protection, their cohesion, and their integrity was taken into account.

However, it must be emphasized that the adopted methodology that focuses mainly on **the negative environmental impacts** may be misleading both for the reader and for authorities that will evaluate the Report. Therefore, the Report also evaluates and presents certain positive environmental impacts of the Programme.

2.3.3 **The structure of the Report**

The distribution of the contents of this document, in general terms, is based on a typical environmental impact assessment (EIA) prepared in accordance with Art. 51 of the Act of 3 October 2008 on Access to Information on the Environment and its Protection, Public Participation in Environmental Protection and on Environmental Impact Assessments (Journal of Laws No. 199 item 1227). However, the complexity of the task at hand and the diverse nature of the environmental impacts that need to be studied have required some modification of the typical distribution of the contents used in strategic environmental assessment reports. This is mostly due to the fact that the analyzed Nuclear Energy Development Program covers a number of actions aimed to determine the location for the first nuclear power plants in Poland. The actions include not only performance of a specific project by way of building (two) nuclear plants but also a number of formal-legal and organizational actions and performance of associated projects that are necessary to assure the functioning of the nuclear energy sector in Poland (e.g. acquiring the raw materials, development of the power grid, location of a nuclear waste repository, etc.).. A description and an analysis of the different environmental impacts related to the full spectrum of the actions that are to be undertaken have turned out to be difficult and incomprehensible when using the structure of the Report as defined in the EIA Act without any changes. At the same time, the requirements for the Report given by the General Director for Environmental Protection have imposed on the authors a duty to analyze in detail numerous aspects contained in the Programme while maintaining the structure.

The complexity of the problem at hand required individual approach to the study. Thus, an broadened model of description of environmental impacts was developed. It was based on a multi-level analysis of impacts related to the operation of nuclear power plants. Then a conclusion chapter was prepared where the radiological and non-radiological impacts that had been identified were collected and assigned to appropriate statutory elements.

Below you can find a discussion of how the adopted method influenced the structure of the Report.

The first part of the Report describes a study of the relations between the Polish Nuclear Program and other strategic documents (**chapter 3**). Next, in accordance with the strategic environmental assessment procedure, the current condition of the environment is described (**chapter 4**). Because the Program in question concerns, in a way, the whole territory of Poland and the final locations of the individual projects have not been identified, the chapter refers to the condition of the environment in Poland, with the level of detail appropriate for further analyzes conducted in the Report. Then, in accordance with the strategic environmental assessment procedure, the zero-option, i.e. the consequences of withdrawal from implementation of the program, was analyzed (**chapter 5**).

In the next part of the Report, in accordance with the method selected, the reader is familiarized in detail with the technical aspects of the nuclear energy sector, to include the issue of nuclear safety and possible breakdowns (**chapter 6**). This approach facilitates understanding of the complex analyses presented in the following chapters.

The next chapters consider in detail the individual environmental impacts of nuclear power plants. First, an analysis and evaluation of the impact of radioactive emissions from nuclear power plants were performed. Because the impact is unique to nuclear power plants and raises the greatest controversies in the public, a separate chapter was devoted to it (**chapter 7**). All the data in this chapter is numerical and accurate and presented as objective values, without interpretation concerning the consequences of the different consumption and emission values in the form of environmental impact.

Chapter 8 discusses all the remaining impacts related to the operation of nuclear plants. A separate subchapter (8.5) discusses, in accordance with the recommendation of the General Director for Environmental Protection, the impact on biotic elements of the environment, to include Natura 2000 areas.

In order to meet the statutory requirements of environmental impact assessments, **chapter 9** identifies and characterizes the impacts (described in detail in the preceding chapters)with respect to their effects to the different elements of the environment. The results of such analyzes were presented in tables, for the sake of transparency. Sub-chapter 9.1 presents all the impacts identified in **chapters 7 and 8**, divided into impacts on the different elements of the environment (biodiversity, people, animals, plants, water, air, Earth's surface, landscape, climate, natural resources, historical monuments, and material goods). Sub-chapter (9.2) presents the characteristics of such impacts with regard to their scale, nature, duration, continuity, and likelihood of occurrence. Sub-chapter (9.3), on the other hand, presents a total balance of impacts, both positive and negative.

In the successive sub-chapter (9.4), the reader is made familiar with the possibility of cumulative impacts, while sub-chapter (9.5) presents, in accordance with the statutory requirement, an analysis of the possibility of transborder impacts. The last sub-chapter (9.5) comprises an analysis of the likelihood of public conflicts.

Chapter 10 presents analyses of possible alternative scenarios. Because of the unique assessment of the strategic document, in addition to the analyses of possible technical and location scenarios, an additional analysis was performed of the possible scenarios of the strategy to acquire energy for Poland and to assure the country's energy safety. The analysis of the location scenarios focused most of all on the six most likely locations, as the impact matrixes prepared in the previous chapters were superimposed on them. Also, less detailed references to the remaining locations were made (the expert opinion on the location, prepared by Energoprojekt Warszawa S.A. defines them as not suitable as a site of a nuclear power plant.

The document ends with a concluding chapter which presents the conclusions and recommendations, as well as the anticipated methods of analysis of the consequences of the program's implementation (**chapter 11**).

2.3.4 Description of the assumptions made and the methods of the individual analyses

The reference objects method

The reference objects method was selected to determine the anticipated significant impacts related to the performance of the Polish Nuclear Program. The method consists in applying the impacts of a specific implemented project to the location of the proposed investment. For this purpose, the monitoring data and the relevant EIA reports are used for this purpose.

No data on the implemented nuclear power projects including Generation III EPR, AP1000 and ESBWR reactors that will be potentially used in Poland was available during the preparation of this Report. The author of the Report has, however, obtained access to safety analyses of such nuclear power plants, which define the radiation impact of the plant on the environment and on people during regular operation and during emergencies.

Also, monitoring data for Generation II nuclear power plants that has been built in other countries was used. Because Generation III nuclear power plants will have all the good characteristics of the operating Generation II plants, the monitoring data from the existing plants can be used to determine the likely impacts of Generation III plants.

Thus, to determine the consequences of the implementation of the Programme, a mixed method was used, which consisted in extrapolating the monitoring data for Generation II plants to Generation III plants and in using the data from the safety analyses. Based on this, a model of Generation III nuclear power plant's impact on the environment was elaborated; this model will be applied to the proposed locations.

Analysis and evaluation of the impact of emissions from nuclear power plants

A separate chapter (**chapter 7**) is dedicated to the matter of radiological impact, which is the single impact causing the greatest concern.

The information presented in that chapter is based on the data that has been published and verified by nuclear regulatory offices. In reference to Generation II reactors – i.e. reactors being in operation currently – historical data, concerning both regular operation and emergencies, was used. The data covers a total of over 12,000 years of operation of the reactors, i.e. it is based on very extensive statistics, collected over a period of 50 years. Because, over the course of all these years, no radiological emergency has taken place in civilian power generation reactors (Generation II and III), which led to the loss of life or health of any staff member or anyone in the local population, the historical data does not allow for presenting any statistics regarding the loss of human lives per year of reactor operation - there has been no such loss at all. Nevertheless, both for existing reactors and for Generation III reactors being built there are probability assessments that can be used to determine the conditions for safe operation of nuclear power plants to be built in the future.

The probability data and the results of the safety analyses for the three types of reactors recommended for Poland, i.e. the EPR reactor by AREVA, the AP1000 reactor by Westinghouse Toshiba, and the ESBWR reactor by General Electric-Hitachi, has been used to determine the characteristics of the impact of these reactors on the ecosystems during regular operation and in emergencies. Because no tender has been announced in Poland and the documentation for these reactors has not been submitted, the information on their parameters and behaviour has been taken from the extensive – albeit not complete – documentation presented for evaluation to the nuclear regulatory offices in the United Kingdom and the United States. The data in the documentation is

true, because it is checked by the nuclear regulatory offices of two very competent countries; however, it is not complete in the sense that the reactor suppliers can still be requested to answer any and all the questions that the Polish party will definitely have in the process of licensing of one of these reactors in Poland. Consequently, the analyses are not based on a complete sate of data, for example while full data concerning emissions in emergencies is not available, the doses that may be received by the local population are stipulated. Because the doses are of key importance to Poland's population, as they determine the health consequences of any failures or make reactor failures harmless, the document provides more extensive information about the doses and the emission data is limited to that which has been taken from available literature.

Also, preliminary evaluations of the atmospheric dispersion in Poland have been prepared using the method adopted by the American nuclear regulatory office, NRC; they were compared with the values presented by the reactor suppliers. This has made it possible to find a common denominator in the assessment of the various reactor manufacturers and to determine whether construction of a nuclear power plant in Poland will involve a radiation exposure outside of a very limited zone with radius estimated to be approx. 800 m from the reactor.

Chapter 7 also includes information on possible decommissioning of the nuclear power plant, defines the costs of such an operation and the potential radiation doses.

At the end of the chapter there is information of the impact on health of small radiation doses and on whether radiation from nuclear power plants may cause damage in ecosystems and people involved in the operation of nuclear power plants.

The remaining impacts are analyzed in chapter 8. As far as cooling water is concerned, the demand for water has been analyzed on the basis of information on the quantity of heat to be transferred in water (based on the heat balance of the different types of power units) and the irreversible losses. The data was compared with hydrological data in order to perform a preliminary assessment of sufficiency of the cooling water supply, for all the possible (and reasonable) types of cooling systems and locations.

The analysis of the consequences of discharge of heat into the water and the air and of chemical substances into the water was performed on the BAT document issued by the European Commission[5] and on relevant environmental impact reports for the UK EPR and UK AP1000 reactors. The analysis of non-radiological emissions was performed based on the UK EPR report. The demand for raw water has been estimated based on the environmental assessment report for the UK EPR report by comparing it with the demand for nuclear power plant formerly planned to be built in Żarnowiec. The emission of substances was calculated per 1,000 MWE generated by a nuclear power plant unit with a pressurized water reactor.

Analysis of impacts on Natura 2000 areas

Due to the lack of accurate data on the fauna and flora for the specific locations, the fullest possible assessment of the natural assets was performed based on the available scientific data. The basis was an analysis of various forms of environmental protection present in the vicinity of the identified sites. Most of all, data contained in the standard forms for the specific Natura 2000 (habitats and bird-protection areas) and characteristics of natural preserves was used.

In the part concerning the plant cover, data from the document titled "Plant reserves in Poland" was used, together with other available basic data concerning the distribution of plant species and plant communities in Poland.

The characteristics of the plants were determined based on the description of the site contained in the previous expert opinion, the map of potential natural vegetation of Poland, the listed documents

concerning protected areas and plant reserves, and the “Habitat and specie protection manuals - methods", vol. 1-5. Based on the data, a list of protected natural habitats present in the vicinity of the individual sites was prepared.

The presence of rare species of vascular plants in the vicinity of the recommended locations has been determined on the basis of the Atlas of Distribution of Vascular Plants in Poland. The Atlas shows the presence of all the species in the form of frequency values in the atlas fields sized 10 x 10 km. Consequently, when evaluating the locations, we initially determined in which atlas fields the individual sites are located. Then, we checked whether and what species are present in the direct vicinity. Two species from Appendix 2 to the Natura 2000 Habitat Directive, species protected by a national Regulation, and species enumerated in the Red List Index of extinct and endangered species were selected for the analysis. Also, data contained in the following specialized literature was used: the Polish Red Book of Plants, the Habitat and Species Protection Handbook – the Methods Handbook, vo. 9 Plant species, and information published in other scientific literature. Based on the above-mentioned sources, it was possible to fairly accurately determine the rare and protected flora species in the different locations and their vicinity.

The data pertaining to the plant cover was presented in the form of a synthetic analysis for all the locations collectively, and a detailed analysis for each individual location.

The landscape conditions in the individual locations were also evaluated based on their location in relation to the existing Landscape Parks and Protected Landscape Areas.

Location analysis

In order to perform the location analysis and the cartographic visualization, the GIS techniques and the Geoxa Editor 2.0 software were used. Selection of the GIS software instead of a standard graphic application has made it possible to precisely present the source data as well as to use the existing data bases (to include those accessible through WMS servers) and the quantitative analyses of selected occurrences. This was possible thanks to the increase of the final essential value of the study with timely completion of the Report.

A number of maps in the 1:100,000 scale were prepared for the 28 potential locations of nuclear power plants included in the Report; the maps show the areas within a 10 km radius of the nuclear plant sites and the environmental protection measures present there. For this purpose, topographic data coming from generally accessible WMS servers and vector data concerning the range of environmental protection measures from the EEA and the General Directorate of Environmental Protection were used. The precise data concerning the location of the individual sites was taken from the document titled “Expert opinion concerning the siting criteria for nuclear power plants and the evaluation of the agreed locations.” Also, in order to forecast potential transboundary impacts the distances between the sites and Poland's national borders were calculated.

The GIS techniques were also used to prepare maps in the 1:4,000,000 scale, showing the location of the recommended sites in the territory of Poland; this was done both for the purpose of their general presentation and for the purpose of comparison with the location of environmental protection measures and the location of areas affected by seismic events (point data from the IRIS database was used). The topography backgrounds for the aforementioned maps were prepared based on the generally accessible WMAP Level 0 database.

Analysis of the Programme’s relations with other documents

The examination of the relations between the Polish Nuclear Programme and other strategic documents was performed based on an analysis of community, national, and local documents (on

the province level); for this purpose, regions with the highest likelihood of location of nuclear power plants were selected (the Mazowieckie, Wielkopolskie, Pomorskie, and Zachodniopomorskie provinces). This was supplemented by the required analysis of documents constituting the basic legislative achievements of the global nuclear energy sector, for example the International Atomic Energy Agency (IAEA), US federal regulations, or the Convention on Nuclear Safety. The first step of the analysis was to become familiar with the contents of the documents. Then, the main provisions concerning the problems of the nuclear energy sector and the environmental protection were identified, especially in areas where these problems overlap. To achieve the objective of defining the relations between these documents and the document in question, also the background of the proposed transformations in the energy sector in Poland, to include provisions concerning the development of renewable energy sources technologies, were analyzed. The work was concluded by preparing an evaluation of the provisions of the analyzed documents and the Programme with regards to the environmental protection aspects.

Analysis of potential social conflicts

The analysis of potential social conflicts focused on the history of such conflicts in other countries and in Poland – in the past during construction of the Żarnowiec nuclear power plant and currently in connection with information on resumption of the works aimed to build a nuclear power plant. The program, objectives, and actions of the best known organizations and persons who opposed the development of the nuclear energy sector were studied. The arguments against construction of a nuclear power plant were collected and compared with the opinions of professionals involved in promoting this source of energy. The key problems related to construction of nuclear power plants were identified.

The analysis was based on the source materials that have been collected, which included books, official documents, programmes and petitions published on the Internet, contents of official web sites of environmental organizations, and articles published in the press. The broad spectrum of materials was needed to present the point of view of the parties in potential social conflicts. The analysis also used the results of completed studies, among others the data collected in the course of the social studies performed by Public Opinion Research Centre (Centrum Badania Opinii Społecznej – CBOS) in the period of 26 August – 2 September 2009 which involved a national representative sample PESEL (15+) of 1,181 persons, or conclusions from the Subject Study OT-575 of October 2009 titled “Attitudes of local communities in European countries to location of nuclear plant in their neighbourhoods” conducted by the Office of Analyses and Subject Studies of the Senate Chancellery.

Economic aspects

The Report does not include any economic analyses prepared by its authors – it was not the basic purpose of the Report. However, economic analyses presented in the existing publications were quoted and used in the Report to discuss certain aspects relating to environmental changes both for the no-action alternative and for all other alternatives. Due to the high extent of interdisciplinarity of the matters discussed in the Report, the anticipated economic effects, inherently connected with the impact on people and on the natural environment (in particular with regard to use of natural resources) are a valuable indicator that makes it possible to perform a measurable assessment of the impacts in question and to compare the scenarios being considered. However, the authors were not able to verify the economic calculations quoted in the Report. All the sources of information presented in the Report were analysed for their reliability, based on the quality of the publications (references to source data and detailed description of the methodologies applied) and the composition of the team of authors (consisting of experts in the field).

2.3.5 The difficulties encountered due to the technology shortcomings or insufficient scientific knowledge

What makes nuclear power plants different from other power plants is the fission reactions generating heat and constituting a potential hazard of radiation by the fission products. The analysis of the environmental impact of nuclear power plants consists in evaluating this hazard and assessing all the technical measures and natural phenomena that are used to limit the exposure of people and the whole ecosystem to radiation. The general safety principles as well as the structures, equipment, and elements used in nuclear power plants to eliminate or at least reduce this hazard are known.

Nuclear power plants are very complex and expensive facilities. Reactor suppliers focus on presenting the aspects of their products that emphasize the strengths of the adopted solution and avoid presenting any information that could limit their chances to win the contract. The reactor suppliers must provide full information on all the solutions adopted only at the stage of safety analyses conducted by the nuclear regulatory authorities. At the stage of preliminary analyses of the nuclear energy program the reactor suppliers are not required to answer questions asked by independent experts who may not even have any connections to the future investors (as the persons hired to prepare this Report). Thus, some information concerning the reactors that may be selected was hard to access.

This was an important difficulty in the process of evaluating the offered reactors. Nevertheless, the problem was overcome thanks to the fact that one of the experts involved in the project is an expert of the International Atomic Energy Agency who participated in the detailed safety analysis of the EPR, AP1000, and ESBWR reactors performed by the IAEA upon request of the nuclear regulatory authorities of the United Kingdom. The reactor suppliers who applied for licenses of the British nuclear regulatory authorities were required to submit to them – and, consequently, to the IAEA – a complete set of documents and to answer the questions they were asked. The persons who participated in these analyses had access to extensive documentation for these three types of reactors. Further information on the licensing process for the EPR and AP1000 reactors in the United Kingdom and the USA was received in an ongoing manner from the nuclear regulatory authorities of these two countries. Altogether, the information made it possible to conduct a thorough evaluation of the radiological effects of normal operation, incidents, and breakdowns in EPR reactors as well as in AP1000 and ESBWR reactors. Some data was not available, but the key results, i.e. the information on the radiation doses related to all the phases of operation of the aforementioned reactors, were available and used in the project.

Another difficulty was the sheer volume of the material to be studies in the relatively short time. Moreover, coordination of the work of many persons, who are experts in different fields and are located in various parts of Poland, and reaching common conclusions on difficult and sometimes controversial issues, was quite difficult. Also, it was not possible to clarify all the questions that the team working on the Report had concerning the documentation submitted by the reactor manufacturers. This has led to differences in the level of detail of the different analyses – they are more detailed for the EPR reactor, in which case the team of experts had three sets of safety documentation which were prepared for the UK and the USA, and less detailed for the other two reactors.

An important factor affecting the evaluation of the impact of potential failures on the hazard to people is the weather conditions, which determine the atmospheric dispersion factor and, consequently, the concentration of radioactive substances at the boundary of the exclusion zone. The studies of atmospheric dispersion of radioactive substances are usually performed for selected sites for periods of many months so as to obtain full information on the weather conditions that may occur during an accident. No such studies have been conducted in Poland and, considering the short time, it was impossible to obtain data for typical sites inland, in coastal areas, in the vicinity of lakes,

hills, etc. Therefore, uniform weather conditions (plus a safety margin) were provisionally assumed for a typical location in Central Europe. These conditions are not representative of all the sites and this shortcomings will need to be compensated at the next stage of works. One must keep in mind that this is a very serious and time-consuming task.

Another significant problem was the fact that the Atomic Energy Act and the Resolutions of the Council of Ministers on nuclear energy have not been finally approved. If more time was available, we could have determined the items that required a decision by the nuclear regulatory authorities, submitted those items to them, and most likely obtained valid answers. However, given the short time we had to complete our work, such additional consultations were impossible. Consequently, the proposed regulations were assumed as the applicable guidance and were used to evaluate the impact of the future Polish nuclear power plant on the ecosystem and human health.

Other important issues were presented in the table below.

Table.2.3.2 Description of encountered problems and their solutions

PROBLEM	SOLUTION
• A very extensive scope of issues to include in the Report	• A team of 14 recognised experts in various fields of study (connected with scientific circles) was appointed. • Analyses conducted as part of other studies and expert reports were used. • The reference objects method was used.
• Lack of hydrological data necessary to evaluate the sufficiency of cooling water resources for some locations (Bełchatów, Pątnów, Krzywiec, Lisowo, and Wiechowo). • Lack of detailed alternative location analyses (for all locations). • The dimensions and location of the nuclear power plant lot and the positions of the main facilities with reactors of different types (preliminary general layout). • Solutions adopted for the cooling system, with the concept of cooling and raw water supply. • Lack of sufficiently detailed information concerning experience with operation of large hybrid wet-dry tooling towers.	• The available data from various sources (to include the Internet) were used, as well as hydrographical data, brochures, and reports of suppliers of nuclear plant technology. • The authors used the knowledge of experts and publications on the cooling systems as well as water and effluent management systems in nuclear power plants.
• No data on the implemented nuclear power projects including Generation III EPR, AP1000 and ESBWR reactors that will be potentially used in Poland was available during the preparation of this Report.	• Information and knowledge offered by the IAEA expert was used – based on the analysis of reactors similar to those proposed for Poland. The author of the Report has gained access to safety analyses of Generation III nuclear power plants, which define the radiation impact of the plant on the environment and on people during regular operation and during emergencies. Monitoring data for Generation II nuclear power plants that has been built in other countries was used and by using the mixed method consisting in extrapolation of the monitoring data from Generation II plants to

PROBLEM	SOLUTION
	Generation III plants and the data from safety analyses, a model of environmental impact of Generation III plants was prepared.
• An important factor affecting the evaluation of the impact of potential failures on the hazard to people is the weather conditions, which determine the atmospheric dispersion factor and, consequently, the concentration of radioactive substances at the boundary of the exclusion zone. The studies of atmospheric dispersion of radioactive substances are usually performed for selected sites for periods of many months so as to obtain full information on the weather conditions that may occur during an accident. No such studies have been conducted in Poland and, considering the short time, it was impossible to obtain data for typical sites inland, in coastal areas, in the vicinity of lakes, hills, etc.	• Uniform weather conditions (plus a safety margin) were provisionally assumed for a typical location in Central Europe. These conditions are not representative of all the sites and this shortcomings will need to be compensated at the next stage of works. One must keep in mind that this is a very serious and time-consuming task.
• No binding acts of law. The Atomic Energy Act and Resolutions of the Council of Ministers on nuclear energy have not been finally approved.	• The proposed regulations were assumed as the applicable guidance and were used to evaluate the impact of the future Polish nuclear power plant on the ecosystem and on human health.
• The authors had no uniform fauna and flora data for the specific locations that would make it possible to compare and evaluate them directly and reliably.	• The evaluation of natural resources is as complete as possible, based on a very detailed analysis of the available data presented in the relevant publications. Moreover, data from inventories performed as a part of other studies for similar locations was used.
• Authors were not able to verify economic calculations quoted in the Report.	• All sources of information presented in the Report were thoroughly analysed for their reliability, based on the quality of publications (references to source data and detailed description of methodologies applied) and the composition of the team of authors (consisting of experts in the field).

No data on the implemented nuclear power projects including Generation III EPR, AP1000 and ESBWR reactors that will be potentially used in Poland was available during the preparation of this Report. The author of the Report has, however, obtained access to safety analyses of such nuclear power plants, which define the radiation impact of the plant on the environment and on people during regular operation and during emergencies.

Also, monitoring data for Generation II nuclear power plants that has been built in other countries was used. Because Generation III nuclear power plants will have all the good characteristics of the operating Generation II plants, the monitoring data from the existing plants can be used to determine the likely impacts of Generation III plants.

2.4 The team who prepared the Report

Prof. dr inż. Andrzej Strupczewski – Vice-President of the Environmentalists for Nuclear Energy (Stowarzyszenie Ekologów na Rzecz Energii Nuklearnej) association, president of the Nuclear Safety Committee in the POLATOM Atomic Energy Institute, expert in nuclear safety of the European Commission and the International Atomic Energy Agency, and expert of Austria's Ministry of Environment for nuclear reactor safety.

For 50 years worked in the Atomic Energy Institute, to include 6 years in the IAEA in Vienna. He designed the MARIA research reactor and managed its process start-up, conducted the pioneer heat transfer tests in the core of the EWA reactor. As a deputy head of the Atomic Energy Institute, until 1992 he managed an international research programme on reactor safety and, for a period of 20 years after the construction of the Żarnowiec nuclear power plant was cancelled, he conducted continuous safety analyses of nuclear power reactors in various countries.

He studied the safety of nuclear power plants in Armenia, Bulgaria, and Slovakia, lead IAEA missions charged with assessing the safety of the Paks nuclear power plant in Hungary and the Dukovany and Temelin plants in Czech Republic, and performed analyses for the Temelin and Mochovice nuclear power plants in Czech Republic for the government of Austria. Also, he conducted an assessment of the impact on European Union's assistance on the safety of nuclear power plants in Russia and Ukraine and, lately, an assessment of the latest Generation III nuclear reactors offered to be built in the United Kingdom. His competences and impartiality are confirmed by the fact that he continues to conduct analyses for both the IAEA and the government of Austria which is against nuclear power plants. Currently, Professor Strupczewski is involved in the siting project for the largest contracted nuclear power plant in the United Arab Emirates.

He is the author of 4 books and 250 papers on nuclear power plants and the holder of 6 patents. He was the member of the Committee for Power Generation Problems and the Committee of Radiation Sciences of the Polish Academy of Sciences and Poland's representative in the Nuclear Energy Committee of the UNIPEDE and in the ISO Nuclear Energy Committee. Professor Strupczewski has been awarded the Knight's Cross of the Polonia Restituta order for his merits.

Prof. dr hab. Andrzej Solecki – a geologist, graduate of the University of Wrocław. In 2009 he completed the "Short Course in Economic Geology: Metallogeny and Exploration of Uranium Deposits" organized by the TU Bergakademie Freiberg. In the years 1999-2000 he completed the course titled "European Standards of Environmental Impact Assessment of Mineral Projects" and the course titled "European Standards for the Evaluation of Raw Materials Projects and Investments," both organized by the Centre for Continuing Education, Imperial College London.

Since 2004, he has been the head of the Institute of Mineral Materials Department of the Institute of Geological Sciences of the University of Wrocław. In the years 2009-2010 he was the head of a team of experts hired by Atkins-Polska Sp. z o.o. to conduct a study of uranium deposits in Poland for the Ministry of Environment. Since 2007, he was a member of the Fossil Resources Committee (Ministry of Environment) and a member of the Committee for Sustainable Management of Mineral Resources (Polish Academy of Sciences, Kraków). In the years 1996-1997, he was an expert of the consortium of Consulting&Engineering, Uranerz, and Iwaco companies which prepared a database on the Central European uranium mining. In 1996, he represented Poland in the IAEA project titled "Technologies for cleanup and remediation of radioactively contaminated sites." In 1995, he represented Poland in the IAEA project titled "Environmental Restoration in Central and Eastern Europe (uranium mines)."

He is the author of several dozens of publications, to include monographs on radiometric anomalies in the central part of the Sudeten Foreland, a popular book on radioactivity of the geological environment, and one of the co-authors of the IAEA technical document concerning reclamation of radioactive contaminated land. He is the author or co-author of numerous geological documents and

expert opinions concerning management of mineral resources performed for domestic and foreign companies, to include Longview Capital Partners, Puma Resources Limited, Micon International Ltd, and Knauf Engineering.

Dr Wojciech Drzewicki – educated in the field of geology (search for and exploitation of fossil deposits, geochemistry) and environmental protection (geoecology and protection of fossil deposits). He completed graduate studies in law in environmental protection at the University of Wrocław. He is an assistant professor at the Applied Geology and Geochemistry Department of the Institute of Geological Sciences of the University of Wrocław. In 2008 he received a scholarship from the Wrocław's local government, as a part of the project titled "GRANT – support of doctoral students' research," financed by the European Social Fund. Distinguished expert in the field of impact of investment projects on water and soil environments.

Dr Dominika Lewicka-Szczebak – educated in the field of environmental protection (licentiate degree in environmental protection at the University of Wrocław) and geology (master's degree in geology at the University of Wrocław, specialty: geochemistry of environmental and waste management). Holder of a Doctorate in Earth Science, specialized in geology. Currently, she is an assistant professor at the Applied Geology and Geochemistry Department of the Institute of Geological Sciences of the University of Wrocław. Winner of the Scholarship of the Minister of National Education (2005), the award for the best geology graduate in 2006, Scholarship of the Local Government of Wrocław (2008) as a part of the project titled "GRANT – support of doctoral students' research," financed by the European Social Fund. In her scientific work, she focuses on the use of stable isotope analyses in environmental research. She is the co-author of scientific publications in global periodicals, to include six articles published in periodicals included in the so-called "Philadelphia list" (*Applied Geochemistry, Environmental Chemistry Letters, Isotopes in Health and Environmental Studies, Atmospheric Environment, Environmental Pollution, Polish Journal of Environmental Studies*). She is also the co-author of reports from environmental analyses performed as a part of industry-science cooperation (2005, 2008, 2009) and Environmental Impact Reports for Investment Projects (2009, 2010).

Dr Marek Kasprzak – graduate of the Institute of Geography and Regional Development (IGRD) of the Wrocław University, specialized in physical geography. In September 2009, he completed his doctoral studies in the Geomorphology Department of the IGRD. His doctoral thesis pertained to the geomorphologic effects of floods. He actively participated in many scientific conferences, to include the most important European symposia in the field of Earth Sciences – the European Geosciences Union in Vienna (2006 and 2007).. He completed numerous training courses in the field of environmental hazards, to include the *FORM-OSE Post-Graduate Training School on Multi-Risks: Concepts to approach multiple hazards and risk*s organized by the Rheinische-Friedrich-Wilhelms-Universität in Bonn, Germany (2006); the *Environmental Analysis and Geomorphologic Mapping for a Sustainable Development* in Ethiopia (2008) and the 1st Mid-European Summer School on Geomorphology: "Complex Response of Earth Surface Processes to Environmental Change" in Heimbuchenthal, Germany (2010). He is the author of several articles which have been reviewed and published in academic periodicals and monographs as well as dozens of other publications focusing mostly on geomorphology, geology, hydrology, and environmental hazards. Besides academic and didactic work at the University of Wrocław, he gained experience working for one and a half years as an assistant in a company that prepares environmental impact reports and assessments for projects, ecophysiography, environmental reviews, and other documents pertaining to environmental protection and management. He continues this track in his career by cooperating, for over a year, with the Fundeko company.

Dr inż. Wojciech Ciurzycki – a graduate of the Warsaw University of Life Sciences (SGGW) in the field of forestry, holds a doctorate in forest sciences, and currently works as an adjunct at the Independent Forest Botany Institute at the Forestry Faculty of the SGGW. He specializes in the broadly defined geobotany and conducts scientific research mostly in the field of floristics,

phytosociology, plant ecology, and protection of the plant cover. He is the author and co-author of over ten scientific papers published in domestic and foreign periodicals, over ten conference reports and popular science articles, the co-author of 2 textbooks and 2 chapters in academic books. He cooperates with various institutions in the field of environmental protection practice. He is the author of over ten expert opinions, inventories and valuations, as well as flora and plant mappings prepared for environmental impact assessments for various projects, as well as inventories of protected areas prepared by the Bureau for Forest Management and Geodesy. He teaches several classes at the Forestry Faculty, the Inter-faculty Environmental Protection Studies, and Inter-faculty Tourism and Recreation Studies, as well as the Environmental Compensation for Natura 2000 postgraduate studies. He also conducts training courses in protection of plant cover at State Forests inspectorates.

Dr nauk prawnych Anna Haładyj (Ph.D. at law) – an adjunct at the Chair of Environmental Management Law, Faculty of Law, Canon Law, and Administration of the John Paul II Catholic University of Lublin. Her areas of scientific interest include the participation of the public in environmental protection and assessments of impact on the environment and on Natura 2000 areas.

Mgr inż. Władysław Kiełbasa – an expert in nuclear energy, graduate of the Faculty of Mechanics, Power Generation and Airspace Engineering of the Warsaw University of Technology, specialized in power generation systems and equipment (completed his studies under an individual programme focusing on nuclear power plants).

He has worked for 31 years in the power generation sector; for 15 years he has been involved in the Żarnowiec nuclear power plant project (from the project preparation, through construction and preparation for operation, until liquidation of the construction site).

His positions included manager of the Nuclear Safety Department and later the acting head engineer in charge of preparation for the operation and nuclear safety in the operation preparation department at the Żarnowiec nuclear power plant. In particular, he was responsible for the licensing process of the Żarnowiec nuclear power plant with respect to nuclear safety and for preparation for the operation with respect to supervision of the operation of the reactors and of the nuclear fuel management system. He also coordinated the cooperation with the five foreign expert missions evaluating various aspects of the Żarnowiec project (technical, safety, and economic aspects): 3 IAEA missions (Pre-OSART – Pre-Operational Safety Analysis and Review Team, Żarnowiec Site Safety Review Mission, Safety Review of the Containment of the Żarnowiec Nuclear Power Plant), as well as missions of Belgatom/Tractebel and Siemens.

He was also a member of the Nuclear Generation Study Committee (10.NUCLE) UNIPEDE and the International Research Team for VVER Reactor Physics.

He also won a scholarship from the International Atomic Energy Agency (IAEA). He has competed relevant training and obtained a license for the operating engineer on duty of a thermal power station (ZEC “Wybrzeże”) and then the operating engineer on duty of a nuclear power plant (Education and Training Centre for the Nuclear Power Sector and the Nuclear Power Plant in Novovoronezh, Russia). Moreover, he has completed a course in radiological protection and failure-prevention measures in Argonne National Laboratory (University of Chicago, USA), as well as a training course in nuclear safety and reactor physics at the Bohunice nuclear power plant in Slovakia). He has taught post-graduate courses in nuclear power generation co-organized by the IAEA and courses at the Gdańsk University of Technology, as well as courses during the 2nd and 3rd Nuclear Power Generation School (Warsaw, Gdańsk).

For the last 15 years he has been professionally involved in the hydroelectric sector and for 4 years he was the head consultant at Energoprojekt-Consulting S.A. Since 2005 he has been the author or co-author (in most cases with doc. dr inż. Andrzej Strupczewski) of a number of significant publications supporting the introduction of the Polish nuclear programme (including those ordered by the Ministry of Economy, the PSE S.A., and the PGE S.A.), to include those concerning siting studies and analyses.

Mgr inż. Łukasz Szkudlarek – completed university-level education in environmental engineering. Winner of the Maciej Nowicki award for the best graduates in the field of environmental protection. Winner of the award for the best graduates of the environmental engineering department. Many years of experience in performance of environmental protection projects. He completed post-graduate studies in contract management in accordance with international procedures (UE, World Bank, FIDIC). Entered into the central register of persons holding a construction license (no. 1377/10/UC/C). Five years of professional experience in preparation of environmental assessment and analyses documentation, to include that for projects financed by the European Union. Two years of professional experience in evaluation of programmes financed by the European Union.

Mgr Kacper Jancewicz – a doctoral student at the Earth Sciences and Environment Management of the University of Wrocław. He currently conducts research in the use of Geographic Information Systems in studying the presence of anemo-orographic phenomena in the Sudeten Mountains as well as cartographic editing using GIS tools. In 2009 he completed his master's degree in geography, specialty: cartography. His thesis titled "Atlas of the Kłodzko Region for bicycle tourism," prepared solely in the ArcInfo software environment, won the 3rd award in the 26th master's degree theses competition organized by the Polish Geographical Society.

Mgr inż. Dobrawa Wiktoria Ryng – graduate of the Wrocław University of Technology, Faculty of Geoengineering, Mining, and Geology, specialty: Management of Earth Resources and Environmental Protection. Her professional experiences connected with the mining industry has been confirmed by medium operating license in mining for open pit mines. She is the author or and co-author of environmental impact reports for projects and strategic documents. She focuses on issues related to the protection of atmospheric air and the acoustic climate.

Mgr inż. Andrzej Zając - graduate of the Wrocław University of Technology, Faculty of Mechanical and Power Engineering, Institute of Thermal Technology and Liquid Mechanics, specializing in boilers and turbines. He has many years of experience with operation and optimization of systems for co-firing of biomass at the EDF Polska group.

Mgr inż. Tomasz Chrapek – graduate of the Faculty of Geology, Geophysics, and Environmental Protection at the AGH University of Science and Technology in Krakow, specializing in environmental protection, hydrogeology, and engineering geology. Winner of the competition for best graduates of Polish universities in the field of environmental protection, organized by the Maciej Nowicki Foundation. Many years of professional experience in financing of environmental protection projects. He is an expert of the Infrastructure and Environment Operational Programme (1st and 2nd priority axis) and the Regional Operating Programme of the Silesia Province (protection of the Earth surface and waste management). Author of energy certificates and environmental impact reports for investment projects. He specializes in issues related to waste management.

Wojciech Błędowski – an ornithologist with over ten years of experience. He has worked for three years in the ornithologist team led by Professor Tomasz Wesołowski in the Białowieski National Park. He cooperates with the Ornithology Station of the Museum and Institute of Zoology of the Polish

Academy of Sciences. He holds a bird ringing license for all bird species. His professional focus is environmental monitoring and environmental impact assessments.

Mgr Danuta Mruk – graduate of the Department of Natural Sciences and Environmental Management, Geography major (specialty: physical geography) and the Philology Department, Journalism and Public Communication major (specialty: public relations) of the University of Wrocław. In 2010 she defended a master's thesis titled "Phytoremediation and washout of heavy metals from the soil under the influence of an electric field." She participated in the work on the *Strategic Environmental Assessment Report for the Polish Nuclear Programme* – she edited and verified Analysis no. 2 and verified the questions and answers given in the public consultations.

3 ANALYSIS AND EVALUATION OF THE IMPACT OF RADIOACTIVE EMISSIONS FROM NUCLEAR POWER PLANTS

AUTHORS: *Andrzej Strupczewski, Władysław Kiełbasa*

TABLE OF CONTENTS:

3.1 Analysis of radioactive emission levels

3.1.1 Emissions during regular operation

3.1.1.1 Emissions from PWR and Generation II BWR reactors - current experiences

According to the principles adopted by the US Atomic Energy Commission in the middle of the 20th century, which was the very beginning of the development of the nuclear energy sector, a person may not be exposed to significant additional hazards as a result of operation of a nuclear power plant, and the risk to the public resulting from the operation of a nuclear power plant must be comparable with the risk resulting from other methods of power generation and may not significantly increase of the total risk to the public. To achieve this objective, it was agreed that the doses around a nuclear power plant must be limited so that the average risk of cancer in the population living within the radius of 16 km caused by the doses does not exceed 0.1% of all the cancer cases due to all other causes.[6.] At that time, 2 out of one thousand persons in the USA died out of cancer, so the permissible hazard ratio caused by nuclear power plants for the critical population group[7] was on average equal to 2×10^{-6} per person per year. Since then, the emissions of fission products from nuclear reactors into nuclear power plant surroundings have continuously decreased. Fig. 7.1.1 shows the reduction in the emissions of inert gases and Fig. 7.1.2 shows the reduction in the emissions of iodine and radioactive dusts into the atmosphere from nuclear power plants with PWR reactors (data from the UNSCEAR report[8]).

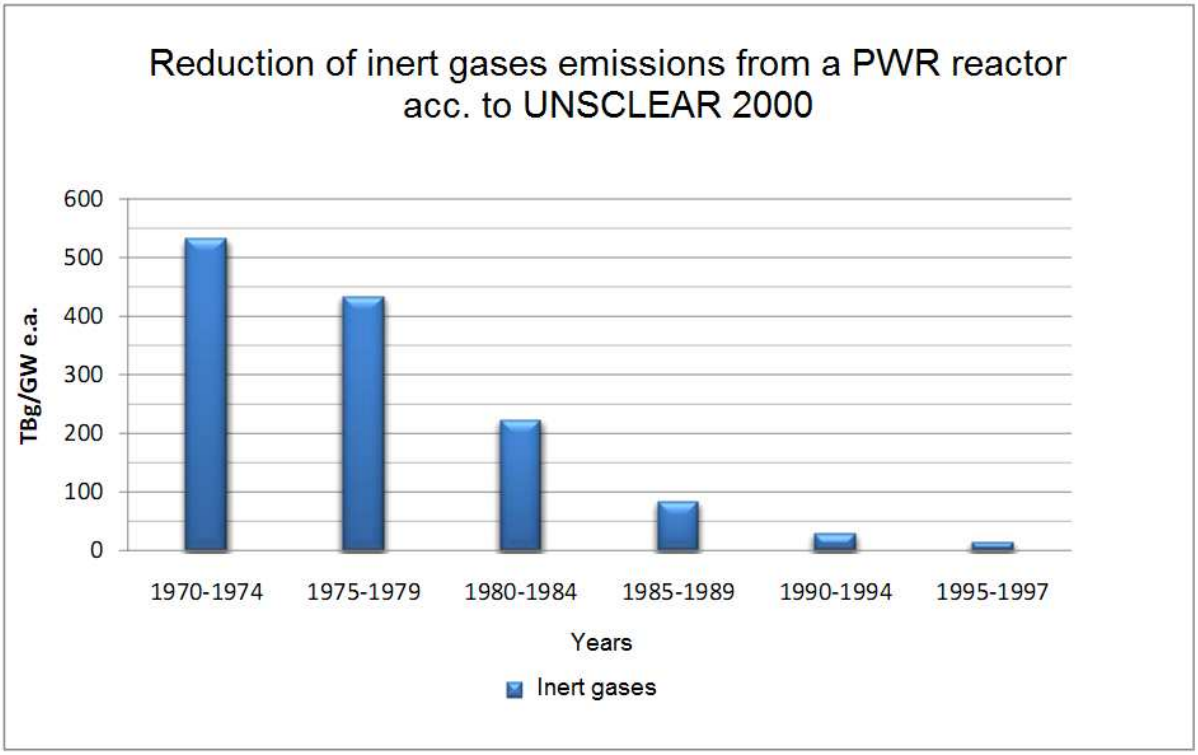

Fig. 3.11. **Reduction of inert gas emissions from PWR reactors; numerical data from the UNSCEAR report[8]**

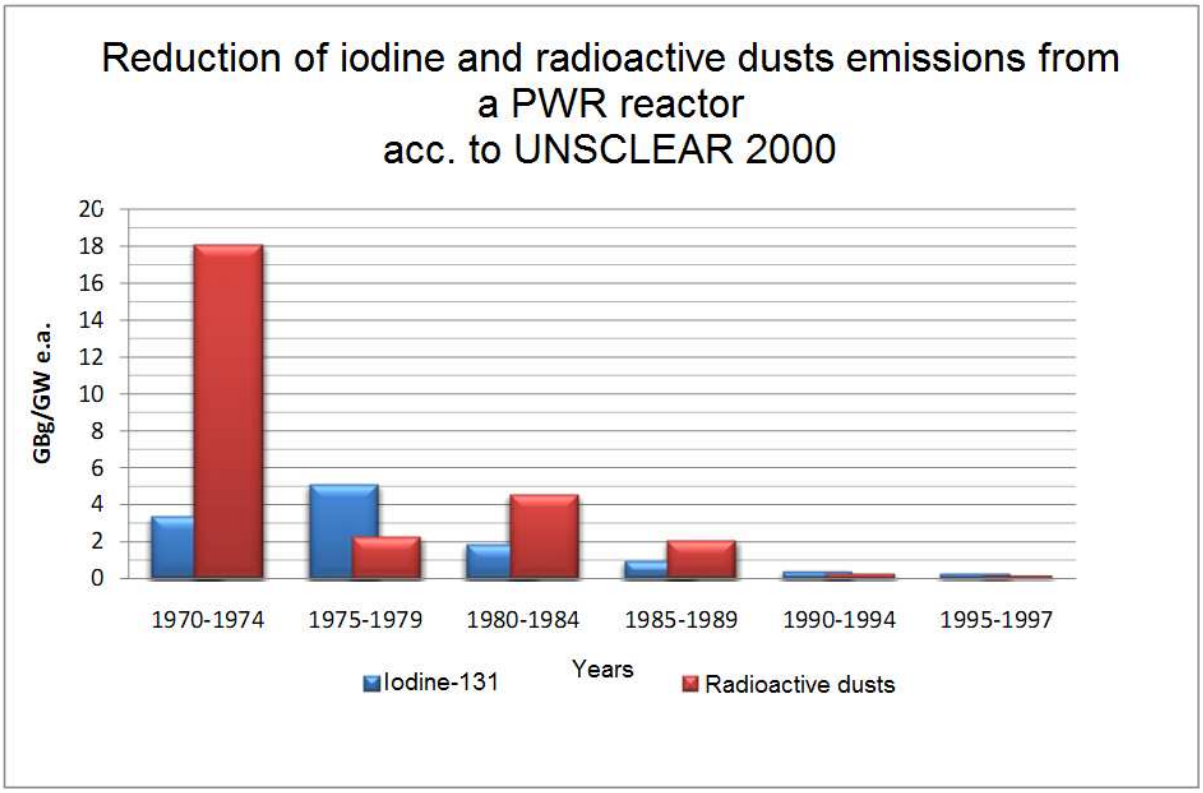

Fig. 3.1.2. Reduction of iodine and radioactive dust emissions from PWR reactors; numerical data from the UNSCEAR report[8]

It should be mentioned that not all fission products are equally hazardous. The most hazardous are radioactive dusts (which contain such elements as caesium or strontium), which get into the human body and (due to the long decay period) remain there for a long time. Iodine is less hazardous; even though it accumulates in the thyroid, it decays fairly quickly (the half-life of the J-131 isotope is 8 days and that of other isotopes – even shorter). Examinations of many persons who were irradiated with iodine for diagnostic or treatment purposes did not demonstrate any increase in cancer incidence[9]. Nevertheless, iodine, as a fairly air-borne element, is a typical hazard which is countered in an effort to lower the doses emitted at nuclear power plants.

The least hazardous are releases of inert gases, which do emit gamma and beta radiation but are released into the environment and do not accumulated in the human body. Figure 7.1.3 (data from the NEA report[10]) shows a comparison of the risk of cancer caused by releases of the same amount of radioactivity (measured as the number of radioactive decay instances per second, i.e. Bq) in inert gases (krypton Kr, xenon Xe), iodine (J), and caesium (Cs). The risk associated to caesium is the highest, because its half-life is 30 years and it remains in the environment long after iodine and inert gases are completely eliminated.

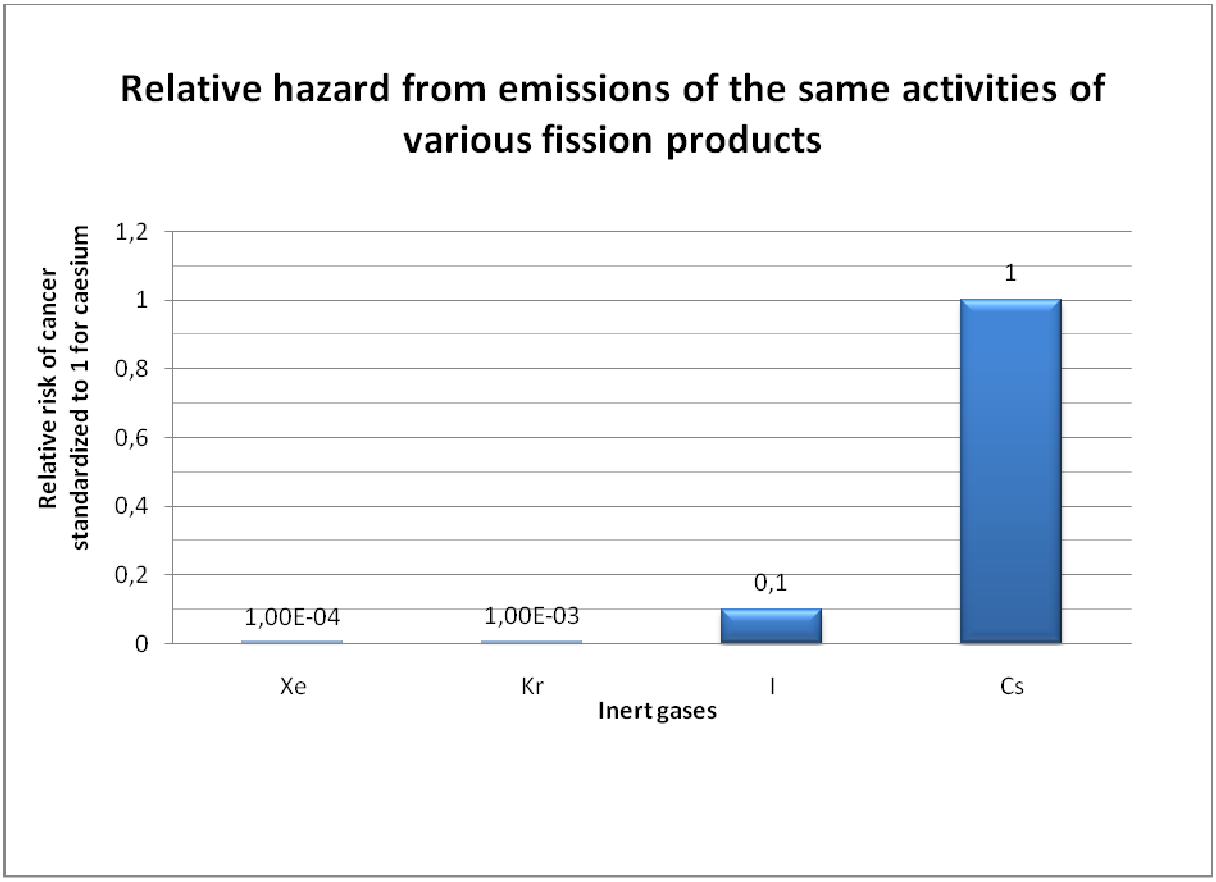

Fig. 7.1.3. Relative risk of cancer caused by release of a certain amount of radioactivity in fission products, normalized to one for caesium. As can be seen, iodine is less hazardous and the hazard on the part of inert gases is negligible. The data for the diagram was taken from the NEA report[10].

Considering the above, Figures 7.1.1 and 7.1.2 must be analyzed again. According to the report for the Convention on Nuclear Safety[11], the average release from PWR nuclear power plants in EU countries in 2003 per unit of generated electric energy, was equal to 4.9 GBq/GWh for inert gases, 0.000025 GBq/GWh for iodine, and 0.000042 GBq/GWh for aerosols. Evidently, the two latter values are over one hundred thousand times less than the former one. Release of the most hazardous isotopes is prevented the most effectively.

The nuclear power plants which are the most representative example of the development of the nuclear energy sector in Europe are the French nuclear power plants. Their total capacity is 62.8 GWe, which is approximately two times more than the capacity of all Poland's power plants. The average release of iodine and aerosols from the French nuclear power plants in 2000 was approximately 0.4% of the permissible yearly release value.[12] Release of liquid radioactive waste was equal to approx. 0.5% of the permissible value. The newer the reactors, the lower the release levels. For example, the nuclear power plants in Chooz and Civaus, which have the latest generation reactors, with the total capacity of 4 x 1450 MWe, release in total less than 4 TBq of inert gases and tritium and less than 0.4 GBq of iodine and radioactive dusts.

Fig**. 3.11.** Reduction of inert gas emissions from PWR reactors; numerical data from the UNSCEAR report[8]

Fig. 7.1.1 shows that as early as the end of the 20th century, the average global release of inert gases was approx. 13 TBq/GWe/year. The French nuclear power plants achieved a ratio of less than 1

TBq/GWe/year in total for inert gases and tritium, and less than 0.1 GBq/GWe/year in total for iodine and radioactive dusts! These values cannot be shown on Fig. 7.1.1 because they are on the horizontal axis.

Taking into account the improved technology of nuclear power plants, for the most recently built nuclear power plants with 1450 MWe reactors, the French nuclear regulatory body imposed limits that are 10 times lower than those for the previous 1300 MWe power units. While the previous limits for nuclear plants with 1300 MWe reactors were 110 GBq in total for iodine and aerosols and 3,300 TBq in total for inert gases, tritium and C-14, the limits for the new plants in Chooz and Civaux are 11 GBq and 330 TBq, respectively. Moreover, considering that the EJ released only small parts of the limiting values, France reduced the permissible emission values in general. Power units which renewed their licenses after 1995 must observe limits that are lower than those in force previously. Examples of the permissible values for 2 x 1300 MWe nuclear plants according to the old regulations and the present regulations are shown in Table 7.1.1.

Table 7.1.1. Permissible and actual emissions at French nuclear power plants operating under original permits (old limits) and current permits, renewed in accordance with new regulations (new limits)[12].

NUCLEAR POWER PLANT	Golfech, 2X1300 MWE (old limits)		Flamanville, 2X1300 MWE (new limits)	
	Limit	Actual emission	Limit	Actual emission
Inert gases, TBq/year	1,650	2.74	45	0.90
Tritium, TBq/year	1)	1)	5	2.03
Carbon C-14, TBq/year	1)	1)	1.4	0.416
Iodine, GBq/year	55	0.083	0.8	0.108
Aerosols, GBq/year	2)	2)	0.8	0.0049

1) The values were included in the "Inert gases" item
2) The values were included in the "Iodine and aerosols" item

As the table shows, the new limits are 30 times lower than the previous ones. The old limits were quite adequate from the point of view of human health, but the French government highlights the fact that thanks to the technological developments the old limits no longer make sense because the actual emissions were significantly lower. This is why new limits, 2 to 40 times lower (depending on the isotope and the plant) were introduced.

Other countries also systematically reduce the permissible emission levels.

3.1.1.2 Emissions during normal operation of an EPR nuclear power plant

3.1.1.2.1 Principles of optimization adopted as a basis for the design of an EPR reactor

The Franco-German design of an EPR reactor (Evolutionary Pressurized Reactor) was intended to meet stringent requirements regarding the emissions of radioactive substances imposed by the nuclear regulatory bodies of both countries. The requirements were formulated in the Technical Guidelines of October 2000 and included, among others:

- reduction of the exposure of the public to radioactive emissions into the atmosphere and water;
- reduction of activity and volume of materials removed as radioactive waste;
- required evaluation of measures aimed to reduce emissions with regards to the quantity of waste resulting from such measures.

The objectives were achieved through optimization and use of experiences, in particular those pertaining to selection of materials, primary loop chemistry, corrosion deposit reduction, processing of radioactive wastewater and gases, and neutralization of solid radioactive waste.

These requirements as well as other recommendations of the French nuclear regulatory body were adopted by the Electricité de France (EDF) group and became a basis for the solutions adopted by the EPR reactor designers in 1999 and 2000, namely:

- a 30% reduction of liquid radioactive waste compared to the average values for 13 MWe reactor power units, with the exception of C-14 carbon and H-3 tritium;
- reduction of cobalt content and improvement of the chemical composition in the primary loop in order to reduce leaks and the amount of waste.

3.1.1.2.2 Principles of evaluation of expected emissions from an EPR reactor

The expected level of emissions from an EPR reactor can be calculated in comparison with reference values based on experiences related to the operation of the existing French and German reactors and on an evaluation of the design improvements that were implemented.

The evaluation process consists of three stages:

- determination of a reference value based on the experiences;
- evaluation of improvements to the design of the EPR reactor;
- determination of expected emission values for the proposed operating parameters of the EPR reactor and the maximum possible emission values.

The experiences are used as a reference point, as it provides more realistic values than any theoretical considerations which, because of the complexity of the process of production, processing, and emission of radioactive substances, produce significantly higher values. The optimization of emission of radionuclides, which has been implemented for many years in nuclear power plants, demonstrates that it is possible to reduce emissions of radioactive isotopes by using appropriate operation strategies.

The reactors used as starting points for comparisons were the 1300 MWe EDF reactors which are the most fully tested and stable. Eight nuclear power plants, in operation over a period of 3 years, from 2001 to 2003, were considered.

In order to take into account the improvements to the design of the EPR reactor, it was compared with the existing 1300 MWe reactors in the following areas:

- primary emissions of fission products;
- main emission routes and their quantitative evaluation;
- design solutions that influence emissions in the EPR reactor and in the existing 1300 MWe power units.

3.1.1.2.3 Parameters of the EPR reactor used in the calculations

The capacity of the EPR reactor was assumed as follows:

- 4,500 MW thermal;

- 1,735 MWe gross;
- 1,630 MWe net.

The amount of energy generated was determined assuming a certain value of the Kd load factor. The value of this factor for 1,300 MWe reactors was **85%,** while for the EPR reactor it is expected to be **91%.**

The yearly energy production for the EPR reactor is 1,630 MW*8,760 h*0.91 ≈ 13,000 GWh.

3.1.1.2.4 Emissions of liquid radioactive waste[13]

3.1.1.2.4.1 Expected emissions of tritium

In the French reactors, 99% of tritium is in liquid state. In PWR reactors (the EPR reactor in question is one of them) the tritium production rate is a nearly linear function of the quantity of generated energy. Tritium is emitted as a result of two factors:

- electron capture in boron B-10 in the in the PWR reactors power adjustment system;
- neutron capture in lithium Li-6 (intended for maintaining proper pH in the coolant of the primary loop), whose quantity is proportional to the quantity of boron in the coolant.

Moreover, in the case of PWR reactors, there is no tritium filtration process implemented on an industrial scale, and the half-life of this isotope (over 12 years), as well as the significant volume of the liquid, makes it impossible to keep it in storage reservoirs until it decays.

Considering the high capacity of the EPR reactor and the deep burnup of nuclear fuel, the primary loop contains a large quantity of boron B-10 which is necessary for slow compensation of changes in reactivity resulting from the fuel burnup. Production control of liquid tritium is intended to maintain its speed on the same level as in 1,300 MWe units. It is achieved by implementing the following design solutions:

- strong poisoning of nuclear fuel with gadolinium (which allows for a reduction of boron B-10 concentration and, consequently, a reduction in tritium production);
- use boron enriched to 30-40% with boron B-10 (this leads to a reduction in the number of boron nuclei without reducing the number of B-10 atoms; because the quantity of lithium is proportional to the total quantity of boron and not to the quantity of the boron B-10 isotope, reduction of the total number of boron atoms leads to a reduction of the number of lithium atoms in the coolant and, eventually, to a reduction in tritium production).

Poisoning of the fuel with gadolinium reduces the concentration of boron B-10 which is the main source of tritium. On the other hand, it reduces the operation time between fuel changes, which has negative economic effects. An in improvement was implemented in the EPR reactor, which has resulted in the length of a fuel cycle equal to 18 months, with a reduction of concentration of the boric acid by 180 ppm, which corresponds to a reduction of activity of tritium by 6 TBq at a loss of 3 days of full-capacity operation.

The analysis of the expected operation tritium emissions considers two options:

- the average value, assuming uniform distribution from the beginning of the cycle during the year and a realistic load factor Kp = 91%;
- the maximum value, assuming that the reactor works continuously for 12 months at the load factor Kp = 100%.

Moreover, two possible lithium concentrations were considered, namely 3.5 ppm and 6 ppm.

The average yearly production of tritium for the operation strategies in question is within the range of 51 TBq/year – 54 TBq/year. The maximum yearly production of tritium is 75 TBq/year.

3.1.1.2.4.2 Releases of liquid carbon C-14

Production of carbon C-14 is a result of neutron irradiation of oxygen O-17 contained in water and, to a lesser extent, of nitrogen contained in the primary loop coolant. The quantity of produced carbon C-14 depends on the volume of irradiated water, the capacity of the reactor, and the content of nitrogen in the primary loop coolant. In PWR reactors, carbon C-14 filtration is not used on an industrial scale.

The experiences related to operation of power units with 1,300 MWe reactors in the years 2001-2003 in France demonstrate that the average yearly release of C-14 is in the range of 15.5-16.2 GBq, which - given the amount of energy generated in those units equal to 9,800 GWh a year - translates into 1.76 Bq/kWh.

More carbon C-14 is produced in an EPR reactor due to its larger capacity and size. Moreover, in EPR reactor reservoirs a gas cylinder is used, containing nitrogen as opposed to hydrogen, together with the pressurizer. This reduces the risk related to presence of hydrogen in the system. Production of C-14 is evaluated for a number of different possible nitrogen concentrations in the primary loop coolant, assuming in particular that it is equal to the concentration in the pressurizer, i.e. max. 27 ppm (or in the ECCS reservoirs, i.e. max. 12 ppm), at the load factor equal to 91%.

Table 7.12. Yearly production of carbon depending on the concentration of nitrogen in the primary loop of an EPR reactor

Concentration of nitrogen (ppm)	EPR (KP = 91%) Yearly production of carbon C-14 (GBq)
1	405
10	444 (34 Bq/kWh)
12	453
27	518

Reduction of nitrogen content in the coolant can be achieved by limiting the flow through the pressurizer and, if necessary, by using a vent installed on the relief line. Consequently, the evaluations of the EPR reactor assume that concentration of nitrogen in the primary loop will be equal to 10 ppm. It is assumed that all the carbon C-14 that will be generated will be released from the reactor. This translates to the average expected yearly release of C-14 to sewage equal to **23 GBq** (1.76 Bq/kWh x 13 TWh).

The maximum values of release from an EPR reactor is assumed, based on experiences with operation of 1,300 MWe units, to be equal to **95 GBq/year**.

3.1.1.2.4.3 Release of other radionuclides into wastewater[13]

Experiences related to operation of 1,300 MWe units demonstrate that releases of iodine isotopes were in the range of 4-18 MBq, with the average value of 0.7 mBq/kWh. The values are very low and cannot be detected with measurement devices. Therefore, instead of measurement data, the evaluation of EPR reactors uses threshold values. Releases of other fission products (FP) and activation products (AP) were in the range of 0.4-1.2 GBq, with the average value of 61 mBq/kWh.

An evaluation of the design improvements to EPR reactors demonstrated that they assure reduction of radioactivity released to wastewater (with the exception of carbon C-14 and tritium) by at least 10% compared with the best 1,300 MWe units.

In the case of liquid iodine isotopes it makes no sense to extrapolate the measurement results from the existing units since the values are so low that threshold values are used instead. Thus, the expected releases are 7 MBq/year which constitutes a 20% improvement in relation to the energy generated.

In the case of other fission products and activation products, it was assumed that the design improvements will lead to a 10% reduction of release values. The release values do not need to be correlated with the amount of generated energy. Consequently, two approaches are used to evaluate the release values:

First, release values per unit of generated energy: 61 mBq/kWh*13 TWh*0.9 = 0.71 GBq.

Second, gross yearly releases equal to 0.54 GBq.

Given this dual approach and the unreliability of measurements due to the low values of the measured parameters, in the design of EPR reactors it was assumed that the expected operation release of fission products and activation products will be equal to **0.6 GBq/year**.

The total value of releases of iodine isotopes and other fission products and activation products is much lower than 1 GBq which was defined by EDF as the design objective for the EPR reactor.

The maximum values of releases of liquid iodine isotopes are equal as those for 1,300 MWe units, i.e. **50 MBq/year**. **This constitutes a reduction of release by unit of generated energy by 24%.**

The maximum values of release of other liquid fission products and activation products reflect the design improvements to the EPR reactors which have resulted in a 10% reduction compared to 1,300 MWe units; based on the current experiences with the operation this leads to expected emissions equal to **10 GBq/year**.

3.1.1.2.5 Releases of radioactive gases[13]

Like in the case of release of liquids, in spite of the great progress achieved in reducing releases of radioactive gases from the existing French reactors, the EPR project involved efforts to reduce them even further, so that their impact on the environment is much smaller than the natural background radiation and the permissible limit values.

3.1.1.2.5.1 Releases of gaseous tritium

Experiences with the existing 1,300 MWe units demonstrate that releases of gaseous tritium are in the range of 0.77-1.86 TBq, with the average value of 91 Bq/kWh.

Unlike in the 1,300 MWe reactors, in EPR reactors there is no intermediate system for washing the tanks of the Emergency Core Cooling System (ECCS). Consequently, production of gaseous tritium in EPR reactors is due mostly to evaporation of water in the fuel storage pool, similarly to the 900 MWe series reactors and the N type reactors (1,450 MWe).

Assuming that the maximum concentration of tritium in the fuel pool is the same as that in the existing power units and taking into account the actual surface area of the pool, the value of release of tritium was calculated to be equal to 0.35 TBq/year. Moreover, taking into account the speed of evaporation and the quantity of steam which condensates on cooler coils in the reactor building ventilation system and the value of releases related to the in-containment refuelling water storage tank (IRWST), which is used as a boric acid tank in emergency situations, the value of tritium release was calculated to be in the range of 0-0.5 TBq/year.

The expected average releases of gaseous tritium was assumed to be 0.5 TBq/year, which constitutes a 60% improvement in the value per unit of generated energy compared with 1,300 MWe units.

The maximum yearly release of gaseous tritium was assumed based on the experiences with 900 MWe units and the N4 series units to be equal to **3 TBq/year** which constitutes a 45% in the value per unit of generated energy compared with 1,300 MWe units.

3.1.1.2.5.2 Release of gaseous carbon C-14

The experiences with the 1,300 MWe units operated in the years 2001-2003 have demonstrated that the average calculated values of gaseous carbon C-14 releases were in the range of 210-250 GBq or, on average, 24 Bq/kWh.

Extrapolation of these values for EPR reactors results in releases related to total energy generation equal to 24 Bq/kWh * 13 TWh = 312 GBq. The value must be added to the second value resulting from the use of a nitrogen blanket in the pressurizer, equal to 117 GBq for a scenario with average nitrogen content of 27 ppm , or 43 GBq for a more realistic scenario with average nitrogen content of 10 ppm.

The sum of these two values is in the range of 350-400 GBq. The EPR design assumes the value to be **350 GBq/year**. It is larger by about 10% than the values for 1,300 MWe units, which is due to the improvement safety of the power unit thanks to replacing hydrogen with nitrogen.

The maximum release of gaseous carbon C-14 for 1,300 MWe units is 700 GBq/unit/year. When the design requirements of the EPR reactor were taken into account, the assumed maximum release of gaseous carbon C-14 was equal to 900 GBq/year. The value in relation to the amount of generated energy is the same as for 1,300 MWe units.

3.1.1.2.5.3 Release of other gaseous radionuclides (iodine, inert gases, other fission products and activation products)

The experience with operation of 1,300 MWe units demonstrates that release of gaseous iodine isotopes is in the range of 16-110 MBq or, on average, 4.6 mBq/kWh.

The values for inert gases are between 0.26 and 7.75 MBq or, on average, 80 mBq/kWh.

Release of other gaseous fission products was between 2 and 11 MBq or, on average, 0.3 mBq/kWh. The release was related mostly to release of aerosols. These are very small values which are on the borderline for measuring abilities.

After analyzing the design improvements to EPR reactors compared with 1,300 MWe reactors, the average expected yearly release of iodine was assumed to be equal to 50 MBq/year, those of inert gases – 0.8 TBq/year, and those of other fission and activation products (mostly aerosols) – 4 MBq/year.

The maximum values for iodine isotopes are 400 MBq/year, for inert gases – 22 TBq/year, and for other fission products and activation products (mostly aerosols) – 340 MBq/year.

3.1.1.2.6 Maximum release values

As demonstrated above, the maximum release values have a certain margin compared to the expected operation values and comprise releases that may occur in the case of small leaks, drainage of the system for repair purposes, or change of chemism of water in order to meet the operation requirements. The efforts to set a realistic level for this margin were based on the experiences with operation of the currently active reactors.

The maximum values are – similar to the expected values of release during operation – based on the experiences with 1,300 MWe reactors, taking into account the design improvements.

Tables 7.1.3 and 7.1.4 show the expected average values of release during normal operation and the maximum values.

Table 3.1.1 Expected average value and maximum yearly value of release into water from EPR reactors

Expected yearly operation release into water	Average [GBq]	Maximum [GBq]
Tritium	52,000	75,000
Carbon C-14	23	95
Iodine isotopes	0.007	0.05
Other fission products and activation products	0.6	10

Table 3.1.2 Expected average value and maximum yearly value of release into the atmosphere from EPR reactors

Radionuclides	Expected yearly operation release into the atmosphere [GBq]	Maximum yearly release to the atmosphere [GBq]
Tritium	500	3,000
Carbon C-14	350	900
Iodine isotopes, total	0.05	0.400
Inert gases, total	800	22,500
FP/AP* – total	0.004	0.340

** FP/AP : other fission products or activation products which emit beta or gamma radiation*

The following spectrum is used in the case of radionuclides present in the group of major gaseous releases:

Table 7.1.5. 3.1.3. Spectrum of radionuclides released into the atmosphere

For maximum yearly release into the atmosphere	Spectrum of radionuclides
Tritium H-3	100%
Carbon C-14	100%
Iodine I-131	45.6%
I-133	54.4%
Inert gases	
Kr-85	13.9%
Xe-133	63.1%
Xe-135	19.8%
Ar-41	2.9%
Xe-131m	0.3%
FP/AP	
Co-58	25.5%
Co-60	30.1%
Cs-134	23.4%
Cs-137	21%

Table 3.1.4. Spectrum of maximum yearly release of radioactive substances from EPR reactors into water

X	Radionuclide	Spectrum
Tritium	H-3	100%
Carbon C-14	C-14	100%
Iodine	I-131	100%

FP/AP	Co-58	20.7%
	Co-60	30%
	Cs-134	5.6%
	Cs-137	9.45%
	Mn-54	2.7%
	Sb-124	4.9%
	Sb-125	8.15%
	Ni-63	9.6%
	Te-123m	2.6%
	Others	0.6%

The analysis cover the following chemical forms of radionuclides:

- In the case of gaseous emissions, it was assumed that tritium is present in the form of steam in the atmosphere.
- Carbon C-14 has the form of M-type particulate atmospheric carbon.
- Iodine isotopes have an inorganic form (I_2) because this is the most hazardous form for living organisms.
- In the case of releases of liquids, it was assumed that tritium has the form of a water solution.

3.1.1.3 Release during normal operation of a nuclear power plant with an AP reactor[14]

3.1.1.3.1 Radioactive waste management systems

Radioactive waste management systems of an AP1000 reactor assure proper handling of solid waste (solid radwaste system - WSS), liquid waste (liquid radwaste system – WLS), and gaseous waste (gaseous radwaste system – WGS).

The WLS system collects, processes, and controls liquid waste and consists of sedimentation tanks, circulation pumps, control equipment, etc. The main process in treating liquid waste is ion exchange.

The WGS collects, processes, and controls gaseous waste which may be radioactive or contain hydrogen, e.g. gases removed in the process of venting the reactor coolant and the reactor coolant drain tank (RCDT). The gaseous waste is kept in the system to assure decay of short-lived radionuclides and then it is forwarded to charcoal filters and discharged to the atmosphere through a ventilation stack.

Solid waste does not cause releases outside of the nuclear power plant.

3.1.1.3.2 Fission products

When defining the design release of fission products, it is assumed that a significant leak has occurred in the jacket of the fuel element, larger than that expected during normal operation. It is assumed that small jacket defects occur in fuel rods which generate 0.25% of the core (the term "0.25% fuel defect" is used). The parameters used to calculate the concentration of fission products in the reactor coolant are mentioned in **Błąd! Nie można odnaleźć źródła odwołania.**.

3.1.1.3.3 Corrosion products

Activity of corrosion products is defined on the basis of extrapolation data of reactors that are currently in operation and is independent of the level of defects of fuel elements. Concentrations of

corrosion products and concentrations of fission products in reactor coolant are mentioned in **Błąd! Nie można odnaleźć źródła odwołania.**.

3.1.1.3.4 Tritium

The concentration of tritium in reactor coolant depends on a number of processes, namely:

- infiltration from the fuel (tritium is produced together with fission products) to the coolant through the jacket or defects in the jacket;
- reaction of neutrons with the boron dissolved in the coolant;
- absorption of neutron in the burned poison;
- reaction of neutrons with the lithium dissolved in the coolant;
- reaction of neutrons with deuterium in the reactor coolant.

The first two processes are the main sources of tritium in reactor coolant.

Tritium is present in reactor coolant in connection with hydrogen, i.e. a tritium atom replaces a hydrogen atom in a water particle, which makes it hard to separate from the coolant. The maximum concentration of tritium in the coolant is less than 3.5 µCi/g, due to losses connected with leaks and control discharge of tritium into the environment.

3.1.1.3.5 Nitrogen N-16

Activation of oxygen in the coolant results in creation of nitrogen N-16 which emits strong gamma radiation. Due to its short half-life, equal to 7.11 s, N-16 does not constitute a hazard to the surroundings of a nuclear power plant. After reactor shutdown, N-16 quickly decays and does not constitute a source of radiation inside the safety containment.

3.1.1.3.6 Activity of the second loop adopted as a design basis

Defects of pipes in the steam generator result in leaks of the coolant from the primary loop to the secondary loop. The resulting radioactivity in the steam loop depends on the rate of the leaks from the primary loop to the secondary loop, the decay coefficient of a given nuclide, and the rate of flow through the steam generator.

Błąd! Nie można odnaleźć źródła odwołania. shows a realistic evaluation of activity in the primary loop of an AP1000 reactor.

Table 3.1.5. Parameters used in design calculations of fission products activity in an AP1000 reactor

Parameter	Value
Core thermal power (MWt)	3,400
Liquid volume of reactor coolant (m3)	271
Average coolant temperature at full power (°C)	300.5
Flow rate in the cleaning loop (m3/h)	
Maximum	22.7
Normal	20.7
Effective flow through the cation bed demineralizer, yearly average (m3/h)	2.7
Nuclide release coefficients (product of the fraction of defective fuel rods and the coefficient of escape of fission products)	
Equivalent fraction of the core power generated in fuel rods with small jacket defects (fraction of defective fuel)	0.0025

Parameter	Value
Coefficient of escape of fission products during work at full capacity (s-1):	
Kr and Xe	6.5 x 10-8
Br, Rb, I, and Cs	1.3 x 10-8
Mo, Tc, and Ag	2.0 x 10-9
Te	1.0 x 10-9
Sr and Ba	1.0 x 10-11
Y, Zr, Nb, Ru, Rh, La, Ce, and Pr	1.6 x 10-12
Mixed-bed demineralizer in the volume and chemical composition adjustment system	
Resin volume (m3)	1.4
Coefficients of isotope decontamination in the demineralizer:	
Kr and Xe	1
Br and I	10
Sr and Ba	10
Other isotopes	1
Cation-bed demineralizer in the volume and chemical composition adjustment system	
Resin volume (mt3)	1.4
Coefficients of isotope decontamination in the demineralizer:	
Kr and Xe	1
Sr and Ba	1
Rb-86, Cs-134, and Cs-137	10
Rb-88, Rb-89, Cs-136, and Cs-138	1
Other isotopes	1
Initial concentration of boron (ppm)	1,400
Operation time (effective hours of operation at full capacity)	12,492

Table 3.1.6. Activity in reactor coolant adopted as a basis for the design of the AP1000 reactor

Nuclide	Activity (µCi/g)	Nuclide	Activity (µCi/g)
Kr-83m	1.8×10^{-1}	Rb-88	1.5
Kr-85m	8.4×10^{-1}	Rb-89	6.9×10^{-2}
Kr-85	3.0	Sr-89	1.1×10^{-3}
Kr-87	4.7×10^{-1}	Sr-90	4.9×10^{-5}
Kr-88	1.5	Sr-91	1.7×10^{-3}
Kr-89	3.5×10^{-2}	Sr-92	4.1×10^{-4}
Xe-131m	1.3	Y-90	1.3×10^{-5}
Xe-133m	1.7	Y-91m	9.2×10^{-4}
Xe-133	1.2×10^{2}	Y-91	1.4×10^{-4}
Xe-135m	1.7×10^{-1}	Y-92	3.4×10^{-4}
Xe-135	3.5	Y-93	1.1×10^{-4}
Xe-137	6.7×10^{-2}	Zr-95	1.6×10^{-4}
Xe-138	2.5×10^{-1}	Nb-95	1.6×10^{-4}
Br-83	3.2×10^{-2}	Mo-99	2.1×10^{-1}
Br-84	1.7×10^{-2}	Tc-99m	2.0×10^{-1}
Br-85	2.0×10^{-3}	Ru-103	1.4×10^{-4}
I-129	1.5×10^{-8}	Rh-103m	1.4×10^{-4}

Nuclide	Activity (µCi/g)	Nuclide	Activity (µCi/g)
I-130	1.1×10^{-2}	Rh-106	4.5×10^{-5}
I-131	7.1×10^{-1}	Ag-110m	4.0×10^{-4}
I-132	9.4×10^{-1}	Te-127m	7.6×10^{-4}
I-133	1.3	Te-129m	2.6×10^{-3}
I-134	2.2×10^{-1}	Te-129	3.8×10^{-3}
I-135	7.8×10^{-1}	Te-131m	6.7×10^{-3}
Cs-134	6.9×10^{-1}	Te-131	4.3×10^{-3}
Cs-136	1.0	Te-132	7.9×10^{-2}
Cs-137	5.0×10^{-1}	Te-134	1.1×10^{-2}
Cs-138	3.7×10^{-1}	Ba-137m	4.7×10^{-1}
Cr-51	1.3×10^{-3}	Ba-140	1.0×10^{-3}
Mn-54	6.7×10^{-4}	La-140	3.1×10^{-2}
Mn-56	1.7×10^{-1}	Ce-141	1.6×10^{-4}
Fe-55	5.0×10^{-4}	Ce-143	1.4×10^{-4}
Fe-59	1.3×10^{-4}	Pr-143	1.5×10^{-4}
Co-58	1.9×10^{-3}	Ce-144	1.2×10^{-2}
Co-60	2.2×10^{-4}	Pr-144	1.2×10^{-2}

The above values of activity are used to design the shields and the radioactive waste management system. In the event that 1% of fuel rods are defective (which corresponds to the maximum capacity of the liquid and gaseous radioactive waste management systems), the above-mentioned values must be multiplied by 4, with the exception of activity of corrosion products (Cr-51, Mn-54, Mn-56, Fe-55, Fe-59, Co-58, and Co-60).

Błąd! Nie można odnaleźć źródła odwołania. – Błąd! Nie można odnaleźć źródła odwołania. show the expected release values of iodine, inert gases, and other fission products from the AP1000 reactor into the atmosphere and into water, and a comparison with the release values from other PWR reactors.

Table 3.1.7. Expected yearly release of air-borne iodine isotopes into the atmosphere

Released activity, GBq/a						
		Building/ventilated area				
Nuclide	Gaseous waste management system	Safety containment building	Ancillary building	Engine building	Condenser venting system	Total release
I-131	$7.4*10^{-3}$	$1.9*10^{-2}$	$1.8*10^{-1}$	$2.4*10^{-3}$	$9.6*10^{-4}$	$2.1*10^{-1}$
I-133	$1.1*10^{-2}$	$7.4*10^{-2}$	$2.6*10^{-1}$	$7.4*10^{-4}$	$3.0*10^{-3}$	$3.5*10^{-1}$

Total activity of volatile iodine isotopes: 6E-01 GBq/year.

Table 3.1.8. Release of gaseous fission products into the atmosphere from the AP1000 reactor (Ci/year)

Inert gases	Active gases system	Building/ventilated area			Condenser venting system	Total
		Safety containment	Ancillary building	Turbine building		
Kr-85m	0	3.0	4.0	0	2.0	3.6*10
Kr-85	$1.65*10^{3}$	$2.4*10^{3}$	2.9*10	0	1.4*10	$4.1*10^{3}$

Inert gases	Active gases system	Building/ventilated area			Condenser venting system	Total
		Safety containment	Ancillary building	Turbine building		
Kr-87	0	9.0	4.0	0	2.0	1.5*10
Kr-88	0	3.4*10	8.0	0	4.0	4.6*10
Xe-131m	$1.42*10^{2}$	$1.6*10^{3}$	2.3*10	0	1.1*10	$1.8*10^{3}$
Xe-133m	0	8.5*10	2.0	0	0	8.7*10
Xe-133	3.0*10	$4.5*10^{3}$	7.6*10	0	3.6*10	$4.6*10^{3}$
Xe-135m	0	2.0	3.0	0	2.0	7.0
Xe-135	0	$3.0*10^{2}$	2.3*10	0	1.1*10	$3.3*10^{2}$
Xe-138	0	1.0	3.0	0	2.0	6.0
					Total	$1.1*10^{4}$
Also:						
H-3 released with gases						350
H-14 released with gases						7.3
Ar-41 released as a result of ventilation of the safety containment						34

The release of tritium is 1.3 x 10^{13} Bq/year and the release of carbon C-14 – 27 x 10^{10} = 0.27 TBq/year.

Table 3.1.9. Comparison of release values of radioactive gases from an AP1000 reactor with release values from other nuclear power plants

	AP1000	South Texas 1	Braidwood 1	Cook 1	Vogtle 1	Sizewell B
Total release per 1,000 MWe per year, in GBq	10,311	7,692	561	12,571	2,184	70,115

Table 3.1.10. Release of aerosols into the atmosphere from the AP1000 reactor (Ci/year)

Inert gases	Active gases system	Building/ventilated area			Total
		Safety containment	Ancillary building	Turbine building	
Cr-51	$1.4*10^{-5}$	$9.2*10^{-5}$	$3.2*10^{-4}$	$1.8*10^{-4}$	$6.1*10^{-4}$
Mn-54	$2.1*10^{-6}$	$5.3*10^{-5}$	$7.8*10^{-5}$	$3.0*10^{-4}$	$4.3*10^{-4}$
Co-57	0	$8.2*10^{-6}$	0	0	$8.2*10^{-6}$
Co-58	$8.7*10^{-6}$	$2.5*10^{-4}$	$1.9*10^{-3}$	$2.1*10^{-2}$	$2.3*10^{-2}$
Co-60	$1.4*10^{-5}$	$2.6*10^{-5}$	$5.1*10^{-4}$	$8.2*10^{-3}$	$8.7*10^{-3}$
Fe-59	$1.8*10^{-6}$	$2.7*10^{-5}$	$5.0*10^{-5}$	0	$7.9*10^{-5}$
Sr-89	$4.4*10^{-5}$	$1.3*10^{-4}$	$7.5*10^{-4}$	$2.1*10^{-3}$	$3.0*10^{-3}$
Sr-90	$1.7*10^{-5}$	$5.2*10^{-5}$	$2.9*10^{-4}$	$8.0*10^{-4}$	$1.2*10^{-3}$
Zr-95	$4.8*10^{-6}$	0	$1.0*10^{-3}$	$3.6*10^{-6}$	$1.0*10^{-3}$
Nb-95	$3.7*10^{-6}$	$1.8*10^{-5}$	$3.0*10^{-5}$	$2.4*10^{-3}$	$2.5*10^{-3}$
Ru-103	$3.2*10^{-6}$	$1.6*10^{-5}$	$2.3*10^{-5}$	$3.8*10^{-5}$	$8.0*10^{-5}$
Ru-106	$2.7*10^{-6}$	0	$6.0*10^{-6}$	$6.9*10^{-5}$	$7.8*10^{-5}$
Sb-125	0	0	$3.9*10^{-6}$	$5.7*10^{-5}$	$6.1*10^{-5}$
Cs-134	$3.3*10^{-5}$	$2.5*10^{-5}$	$5.4*10^{-4}$	$1.7*10^{-3}$	$2.3*10^{-3}$
Cs-136	$5.3*10^{-6}$	$3.2*10^{-5}$	$4.8*10^{-5}$	0	$8.5*10^{-5}$
Cs-137	$7.7*10^{-5}$	$5.5*10^{-5}$	$7.2*10^{-4}$	$2.7*10^{-3}$	$3.6*10^{-3}$
Ba-140	$2.3*10^{-5}$	0	$4.0*10^{-4}$	0	$4.2*10^{-4}$
Ce-141	$2.2*10^{-6}$	$1.3*10^{-5}$	$2.6*10^{-5}$	$4.4*10^{-7}$	$4.2*10^{-5}$

The total activity of aerosols is 46.76 x 10-3 Ci/year, i.e. 0,173 x 10^{10} Bq = 1.7 GBq/year.

Table 3.1.11. Comparison of release values of liquid radioactive waste from the AP1000 reactor, with the exception of tritium, with release values from European nuclear power plants in the years 1995-1998

X	Unit	AP1000	Sizewell B	All PWR	Magnox and AGR Nuclear Power Plants	All BWR
No. of power units	0	1	73	30	10	
Minimum GBq/GWa	1.1	16	0	2	0	
Average GBq/GWa	2.4	21.8	4.9	12.2	65.5	
Maximum GBq/GWa	3.5	28	61	28	599	

Table 3.1.12. Comparison of release values of liquid radioactive waste from an AP1000 reactor with release values from other nuclear power plants

	AP1000	South Texas 1	Braidwood 1	Cook 1	Vogtle 1	Sizewell B
Total release per 1,000 MWe per year, in GBq	33,374	46,331	49,094	44,500 /	40,450	50,503

3.1.1.4 Emissions during normal operation of an ESBWR nuclear power plant

3.1.1.4.1 Sources of radioactivity

In an ESBWR (Economic Simplified Boiling Water Reactor), there is only one large radioactivity source inside of the primary containment, that is the reactor core. Another source of radioactivity is the control rod drives used for precise control of reactivity. An ESBWR has no circulation pumps outside or inside the reactor containment, a circular core sampling system, or heat exchangers, which may become contaminated in the course of normal operation.

The list of radioactivity sources in the primary containment of an ESBWR reactor presented in the report[15] submitted for consideration by the British nuclear regulatory authority do not include sources resulting from accidental contamination, such as corrosion deposits or fission products deposits on surfaces of valves and other elements of the primary loop.

3.1.1.4.1.1 Sources in the engine building (turbine building)

The main source of radiation in the turbine building is nitrogen N-16 generated after neutron capture and proton emission, which is contained in the steam rising above the reactor container. This isotope causes significant gamma radiation from elements through which the steam flows, with the dose of approx. 0.2-0.5 Sv/hour on the surface of pipelines and containers. The remaining sources of radiation in the turbine building are the venting system and the condenser and supply water system.

3.1.1.4.2 Releases from ESBWR reactors during normal operation

The radioactivity release values from boiling water reactors was determined based on many years of experience with operation of boiling water reactors according to the ANSI/ANS-18.1 standard. The observations made when switching from old design fuel to new design fuel were also taken into

account. The radionuclides included in the design bases were divided into fission products and activation products.

3.1.1.4.2.1 Inert gases produced as fission products

The typical concentrations of 13 inert gases produced as fission products which are present in steam discharged from the reactor container are defined in the ANSI/ANS-18.1 Source Term Standard.[16] The concentration of such gases in the reactor water are negligibly low in normal operating conditions, because all gases emitted into the coolant are quickly eliminated with steam and moved to the main turbine condenser. Considering the quick elimination of such gases, the expected composition of the gaseous fission products in steam is independent of the reactor's design.

Table 3.1.13. Parameters assumed for determining releases of radioactivity in ESBWR reactors

Parameter	Value
Total release of 13 inert gases after 30 minutes (t30)	3,700 MBq/s
Normal operation rate of release of inert gases (t30)	740 MBq/s
Rate of release of radioactive iodine I-131 from the core assumed as the design basis	26 MBq/s
Expected rate of release of radioactive iodine I-131 from the core	3.7 MBq/s
Scale factor for concentration of I-131 in the coolant	5
Concentration of N-16 at the core outlet (the design basis must be the same as the normal operation value)	1.85 MBq/gm w/o HWC, 9.25 MBq/gm w/HWC
Rate of release of argon Ar-41 assumed as design basis	2.0 MBq/s
Normal operation rate of release of argon Ar-41	0.4 MBq/s

Błąd! Nie można odnaleźć źródła odwołania. shows the parameters adopted as the design basis for concentration of inert gases in steam.

Table 3.1.14. Parameters adopted as the design basis for concentration of inert gases in steam

Isotope	Decay coefficient	Concentration in steam		Release after T=30 minutes	
	Per hour	MBq/g	microCi/g	MBq/g	microCi/s
Kr-83m	3.73E-1	5.4E-05	1.5E-03	1.1E+02	2.9E+03
Kr-85m	1.55E-1	9.1E-05	2.5E-03	2.0E+02	5.5E+03
Kr-85	7.37E-6	3.6E-07	9.8E-06	8.9E-01	2.4E+01
Kr-87	5.47E-1	3.0E-04	8.1E-03	5.6E+02	1.5E+04
Kr-88	2.48E-1	3.0E-04	8.1E-03	6.5E+02	1.7E+04
Kr-89	1.32E+1	1.9E-03	5.2E-02	6.4E+00	1.7E+02
Xe-131m	2.41E-3	3.0E-07	8.1E-06	7.3E-01	2.0E+01
Xe-133m	1.30E-2	4.5E-06	1.2E-04	1.1E+01	2.9E+02
Xe-133	5.46E-3	1.3E-04	3.4E-03	3.1E+02	8.4E+03
Xe-135m	2.72E+0	4.0E-04	1.1E-02	2.5E+02	6.8E+03
Xe-135	7.56E-2	3.5E-04	9.4E-03	8.1E+02	2.2E+04
Xe-137	1.08E+1	2.4E-03	6.4E-02	2.6E+01	6.9+02
Xe-138	2.93E+0	1.4E-03	3.7E-02	7.7E+02	2.1E+04
Total		7.3E-03	2.0E-01	3.7E+03	1.0E+05

Concentrations of inert gases in steam after 30 minutes, when the decay of the fission products occurs, are used as a standard measure of leaks from fuel elements.

The typical rate of release of inert gases, which is equal to 3,700 MBq/s, after 30 minutes of decay, has been successfully used to design systems for processing the released gases in BWR reactors[17]. The rate was determined based on operational experiences, taking into account the effect of current design solutions.

3.1.1.4.2.2 Iodine isotopes produced as fission products

For many years, the design basis adopted for BWR reactors was the rate of iodine I-131 release from fuel in the core equal to 26 MBq/s[17]. However, based on the experiences, such a high rate of iodine release takes place only in the event that the core is operated with significant defects in the fuel elements. The rate of iodine isotopes release for the ESBWR reactor was determined based on the ANS standard[18].

Table 3.1.15. Concentrations of iodine isotopes in water and steam of an ESBWR reactor adopted as the design basis

Isotope	Decay coefficient	Concentration in water		Concentration in steam	
	Per hour	MBq/g	microCi/g	MBq/g	microCi/g
I-131	3.59E-3	3.9E-04	1.1E-02	7.9E-06	2.1E-04
I-132	3.03E-1	3.7E-03	9.9E-02	7.4E-05	2.0E-03
I-133	3.33E-2	2.7E-03	7.2E-02	5.3E-05	1.4E-03
I-134	7.91E-1	6.8E-03	1.8E-01	1.4E-04	3.7E-03
I-135	1.05E-1	3.8E-03	1.0E-01	7.6E-05	2.1E-03

The assumed ratio of concentration of iodine in steam to concentration of iodine in water (*carryover ratio*) is assumed to be equal to approx. 0.02.

3.1.1.4.2.3 Other fission products

This category includes products other than inert gases and iodine, among others transuranic nuclides. Some of the fission products are products of decay of inert gases produced in steam and condensate. One transuranic element which may be detected in significant quantities is Np-239. After introducing appropriate coefficients for concentrations that are typical for BWR reactors in the ANS standards [18], the concentrations for the ESBWR were obtained – See **Błąd! Nie można odnaleźć źródła odwołania.**. The ratio of concentration of these nuclides in steam to their concentration in water is less than 0.001. Thus, concentration in steam can be calculated by multiplying the concentration in water by the 0.001 factor.

Table 3.1.16. Concentrations of fission products in water in ESBWR reducers adopted in the design

Isotope	Decay coefficient	Concentration in water	
	(per hour)	(MBq/g)	µCi/g
Rb-89	2.74E+0	6.9E-04	1.9E-02
Sr-89	5.55E-4	1.7E-05	4.5E-04
Sr-90	2.81E-6	1.2E-06	3.1E-05
Y-90	2.81E-6	1.2E-06	3.1E-05
Sr-91	7.31E-2	6.4E-04	1.7E-02
Sr-92	2.56E-1	1.5E-03	4.1E-02
Y-91	4.93E-4	6.6E-06	1.8E-04
Y-92	1.96E-1	9.3E-04	2.5E-02
Y-93	6.80E-2	6.4E-04	1.7E-02
Zr-95/Nb-95	4.41E-4	1.3E-06	3.6E-05
Mo-99/Tc-99m	1.05E-2	3.3E-04	8.9E-03
Ru-103/Rh-103m	7.29E-4	3.3E-06	8.9E-05
Ru-106/Rh-106	7.83E-5	5.0E-07	1.3E-05
Te-129m	8.65E-4	6.6E-06	1.8E-04
Te-131m	2.31E-2	1.6E-05	4.4E-04
Te-132	8.89E-3	1.6E-06	4.5E-05
Cs-134	3.84E-5	4.5E-06	1.2E-04
Cs-136	2.22E-3	3.0E-06	8.0E-05
Cs-137/Ba-137m	2.63E-6	1.2E-05	3.2E-04

Isotope	Decay coefficient (per hour)	Concentration in water (MBq/g)	µCi/g
Cs-138	1.29E+0	1.4E-03	3.8E-02
Ba-140/La-140	2.26E-3	6.6E-05	1.8E-03
Ce-141	8.88E-4	5.0E-06	1.3E-04
Ce-144/Pr-144	1.02E-4	5.0E-07	1.3E-05
Np-239	1.24E-2	1.3E-03	3.6E-02

Table 3.1.17. Parameters adopted as a basis for calculation of release of radioactivity into the atmosphere from an ESBWR reactor

Parameter	Value (keep in mind that 1 Ci = 3.7 x 10^{10}Bq)
Sources of inert gases after t=30 min	740 MBq/s (20,000 µCi/sec)
Rate of release I-131	3.7 MBq/s (100 µCi/sec)
Power unit load coefficient	0,92
Release from turbine sealing system:	25 g /h
I-131	0.81 Ci/a per µCi/g I-131 in the coolant
I-133	0.22 Ci/a per µCi/g I-131 in the coolant

Table 3.1.18. Value of activity in the demineralizer of the RWCU system

Class	Isotope	MBq	Class	Isotope	MBq
Class 2	I-131	9.85E+06	Class 6	Sr-89	2.74E+06
	I-132	1.09E+06		Sr-91	8.10E+05
	I-133	6.97E+06		Sr-92	5.39E+05
	I-134	7.77E+06		Y-91	1.08E+06
	I-135	3.31E+06		Y-92	4.28E+05
				Y-93	8.64E+05
Class 3	Rb-89	2.30E+04		Zr-95	2.33E+05
	Cs-134	8.09E+05		Nb-95	1.51E+04
	Cs-136	6.41E+04		Mo-99	2.85E+06
	Cs-137	2.3E+06		Tc-99m	2.59E+05
	Cs-138	4.90E+04		Ru-103	3.94E+05
	Ba-137m	6.56E+01		Rh-103m	4.12E+02
				Rh-106	5.49E-01
Class 4	N-16	6.03E+02		Ag-110m	5.14E+04
				Te-129m	6.83E+05
Class 5	Na-24	6.36E+05		Te-131m	6.44E+04
	Cr-51	4.29E+07		Te-132	1.68E+04
	Mn-54	1.88E+06		Ba-140	2.66E+06
	Mn-56	1.28E+06		La-140	3.49E+05
	Fe-59	6.54E+05		Ce-141	4.95E+05
	Co-58	3.08E+06		Ce-144	1.58E+05
	Co-60	1.26E+07		Pr-144	1.88E+01
	Cu-64	8.11E+05		W-187	1.54E+05
	Zn-65	5.10E+07		Np-239	9.74E+06
				Total	1.64E+08

Błąd! Nie można odnaleźć źródła odwołania. shows the activity of radionuclides in the turbine condenser. The activity values are much higher in the filters.

Table 3.1.19. Values of radionuclides in the turbine condenser of an ESBWR reactor

Isotope	Activity MBq	Isotope	Activity MBq
Kr-85m	1.49E+04	P-32	1.09E+00

Isotope	Activity MBq	Isotope	Activity MBq
Kr-85	5.98E+01	Cr-51	8.16E+01
Kr-87	4.93E+04	Mn-54	9.52E-01
Kr-88	4.93E+04	Mn-56	6.24E+02
Kr-89	3.14E+05	Fe-55	2.73E+01
Xe-131m	4.93E+01	Fe-59	8.16E-01
Xe-133m	7.32E+02	Co-58	2.73E+00
Xe-133	2.09E+04	Co-60	5.44E+00
Xe-135m	6.58E+04	Ni-63	2.73E-02
Xe-135	5.68E+-4	Cu-64	7.95+01
Xe-137	3.89E+05	Zn-65	2.73E+01
Xe-138	2.25E+05	Sr-89	2.73E+00
Total	1.21E+06	Sr-90	1.90E-01
Class 2		Y-90	1.90E-01
I-131	1.29E+03	Sr-91	1.05E+02
I-132	1.21E+04	Sr-92	2.50E+02
I-133	8.72E+03	Y-91	1.09E+00
I-134	2.23E+04	Y-92	1.52E+02
I-135	1.26E+04	Y-93	1.05E+02
Total	5.70E+04	Zr-95	2.18E-01
Class 3		Nb-95	2.18E-01
Rb-89	1.14E+02	Mo-99	5.40E+01
Cs-134	7.34E-01	Tc-99m	5.40E+01
Cs-136	4.89E-01	Ru-103	5.44E-01
Cs-137	1.95E-01	Rh-103m	5.44E-01
Cs-138	2.28E+02	Ru-106	8.16E-02
Ba-137m	1.95E+00	Rh-106	8.16E-02
Total	3.47E+02	Ag-110m	2.73E-02
Class 4		Te-129m	1.09E+00
N-16	1.26E+08	Te-131m	2.69E+00
Class 5		Te-132	2.71E-01
H-3	6.08E+04	Ba-140	1.09E+01
		La-140	1.09E+01
		Ce-141	8.16E-01
		Ce-144	8.16E-02
		Pr-144	8.16E-02
		W-187	8.03E+00
		Np-239	2.17E+02
		Total	1.27E+08

Table 3.1.20. Value of radionuclides collected on filters of the ion exchanger of an ESBWR reactor

Isotope	Activity MBq
Class 2	
I-131	1.94E+06
I-132	2.15E+05
I-133	1.41E+06

Isotope	Activity MBq
I-134	1.52E+05
I-135	6.57E+05
Class 3	
Rb-89	2.28E+02
Cs-134	9.10E+03
Cs-136	1.18E+03
Cs-137	2.52E+04
Cs-138	9.55E+02
Ba-137m	6.48E-01
Class 6	
Sr-89	3.86E+03
Sr-90	4.93E+02
Y-90	3.17E-01
Sr-91	1.58E+03
Sr-92	1.05E+03
Y-91	1.66E+03
Y-92	8.39E+02
Y-93	1.66E+03
Zr-95	3.50E+02
Nb-95	2.47E+02
Mo-99	5.61E+03
Tc-99m	5.06E+02
Ru-103	6.68E+02
Rh-103m	7.93E-01
Ru-106	1.93E+02
Rh-106	1.06E-03
Te-129m	1.21E+03
Te-131m	1.26E+02
Te-132	3.30E+01
Ba-140	5.18E+03
La-140	6.82E+02
Ce-141	8.76E+02
Ce-144	1.88E+02
Pr-144	3.67E-02
Np-239	1.91E+04
Total	4.47E+06

Table 3.1.21. Releases into the atmosphere from the systems of an ESBWR reactor

Nuclide	Reactor building	Turbine building	Radioactive waste building	Vacuum pumps	Turbine sealing	Gas extraction system	Secondary containment
Kr-83m						1.4E-04	3.7E+01
Kr-85m	6.9E+04	5.7E+05				6.8E+03	1.5E+02
Kr-85						4.3E+06	3.3E+01
Kr-87	4.6E+04	1.4E+06				8.6E-10	1.4E+02
Kr-88	9.2E+04	2.1E+06				1.5E+01	3.0E+02
Kr-89	4.6E+04	1.3E+07	6.7E+05				3.7E+01

Kr-90							1.3E+01
Xe-131m						1.1E+05	1.8E+01
Xe-133m						8.1E-01	8.5E+01
Xe-133	2.5E+06	3.4E+06	5.1E+06	1.9E+07		8.3E+05	5.0E+03
Xe-135m	1.4E+06	9.2E+06	1.2E+07			4.3E-37	3.7E+01
Xe-135	2.9E+06	7.6E+06	6.4E+06	7.4E+06			1.2E+03
Xe-137	4.1E+06	2.3E+07	1.9E+06				5.5E+01
Xe-138	1.8E+05	2.3E+07	4.6E+04				1.2E+02
Xe-139							1.6E+01
I-131	9.4E+02	5.2E+03	3.4E+02	1.8E+03	4.7E+01		6.8E+03
I-132	8.5E+03	4.6E+04	3.0E+03				9.9E+02
I-133	6.2E+03	3.4E+04	2.2E+03		8.4E+01		6.5E+03
I-134	1.5E+04	8.4E+04	5.5E+03				6.9E+02
I-135	8.6E+03	4.7E+04	3.1E+03				2.9E+03
H-3	1.3E+06	1.3E+06					2.6E+05
C-14							
Na-24							5.4E-01
P-32							1.3E-01
Ar-41							
Cr-51	2.7E+01	2.2E+01	1.7E+01				1.1E+01
Mn-54	3.4E+01	1.5E+01	9.8E+01				1.7E-01
Mn-56							1.1E+00
Fe-55							4.7E+00
Fe-59	9.5E+00	2.4E+00	7.3E+00				1.2E-01
Co-58	7.3E+00	2.4E+01	4.9E+00				4.4E-01
Co-60	1.2E+02	2.4E+01	1.7E+02				9.4E-01
Ni-63							4.7E-03
Cu-64							6.9E-01
Zn-65	1.2E+02	1.5E+02	7.3E+00				4.6E+00
Rb-89							2.0E-02
Sr-89	1.2E+00	1.5E+02					4.3E-01
Sr-90	2.4E-01	4.9E-01					3.3E-02
Y-90							3.3E-02
Sr-91							6.7E-01
Sr-92							4.6E-01
Y-91							1.7E-01
Y-92							3.7E-01
Y-93							7.2E-01
Zr-95	2.4E+01	9.8E-01	2.0E+01				3.5E-02
Nb-95	2.4E+02	1.5E-01	9.8E-02				3.3E-02
Mo-99	1.6E+03	4.9E+01	7.3E-02				2.4E+00
Tc-99m							2.2E-01
Ru-103	1.0E+02	1.2E+00	2.4E-02				8.2E-02
Rh-103m							8.2E-02
Ru-106							1.4E-02
Rh-106							1.4E-02
Ag-110m	5.9E-02						1.3E-07
Sb-124	1.2E+00	2.4E+00	1.7E+00				
Te-129m							1.6E-01

Te-131m					5.5E-02
Te-132					1.4E-02
Cs-134	1.1E+02	4.9E+00	5.9E+01		1.3E-01
Cs-136	1.2E+01	2.4E+00			5.8E-02
Cs-137	1.5E+02	2.4E+01	9.8E+01		3.4E-01
Cs-138					8.5E-02
Ba-140	5.4E+02	2.4E+02	9.8E-02		1.3E+00
La-140					1.3E+00
Ce-141	2.2E+01	2.4E+02	1.7E-01		1.2E-01
Ce-144					1.3E-02
Pr-144					1.3E-02
W-187					1.3E-01
Np-239					8.3E+00

Table 3.1.22. Comparison of releases from an ESBWR reactor with the limit values according to USA regulations, 10CFR 20

X	Releases to the atmosphere from an ESBWR reactor	Concentration	Limit value acc. to 10CFR20
Nuclide	**MBq/yr**	**Bq/m^3**	**Bq/m^3**
Kr-83m	3.73E+01	2.36E-06	2.E+06
Kr-85m	6.50E+05	4.12E-02	4.E+03
Kr-85	4.29E+06	2.72E-01	3.E+04
Kr-87	1.45E+06	9.17E-02	7.E+02
Kr-88	2.18E+06	1.38E-01	3.E+02
Kr-89	1.40E+07	8.90E-01	4.E+01
Kr-90	1.25E+01	7.94E-07	4.E+01
Xe-131m	1.10E+05	6.97E-03	7.E+04
Xe-133m	8.59E+01	5.44E-06	2.E+04
Xe-133	3.11E+07	1.97E+00	2.E+04
Xe-135m	2.27E+07	1.44E+00	1.E+03
Xe-135	2.43E+07	1.54E+00	3.E+03
Xe-137	2.90E+07	1.84E+00	4.E+01
Xe-138	2.32E+07	1.47E+00	7.E+02
Xe-139	1.57E+01	9.93E-07	4.E+01
I-131	1.51E+04	9.57E-04	7.E+00
I-132	5.89E+04	3.74E-03	7.E+02
I-133	4.88E+04	3.09E-03	4.E+01
I-134	1.06E+05	6.72E-03	2.E+03
I-135	6.14E+04	3.89E-03	2.E+02
H-3	2.80E+06	1.78E-01	4.E+03
C-14	3.54E+05	2.24E-02	1.E+02
Na-24	5.42E-01	3.44E-08	3.E+02
P-32	1.34E-01	8.50E-09	2.E+01
Ar-41	2.85E+02	1.81E-05	4.E+02
Cr-51	7.73E+01	4.90E-06	1.E+03
Mn-54	1.47E+02	9.29E-06	4.E+01
Mn-56	1.07E+00	6.80E-08	7.E+02
Fe-55	4.72E+00	2.PPE-07	1.E+02

X	Releases to the atmosphere from an ESBWR reactor	Concentration	Limit value acc. to 10CFR20
Nuclide	MBq/yr	Bq/m^3	Bq/m^3
Fe-59	1.94E+01	1.23E-06	2.E+01
Co-58	3.70E+01	2.35E-06	4.E+01
Co-60	3.18E+02	2.02E-05	2.E+00
Ni-63	4.74E-03	3.01E-10	4.E+01
Cu-64	6.93E-01	4.39E-08	1.E+03
Zn-65	2.80E+02	1.78E-05	1.E+01
Rb-89	2.01E-02	1.27E-09	7.E+03
Sr-89	1.48E+02	9.38E-06	7.E+00
Sr-90	7.65E-01	4.85E-08	2.E-01
Y-90	3.27E-02	2.07E-09	3.E+01
Sr-91	6.72E-01	4.26E-08	2.E+02
Sr-92	4.63E-01	2.93E-08	3.E+02
Y-91	1.74E-01	1.10E-08	7.E+00
Y-92	3.68E-01	2.33E-08	4.E+02
Y-93	7.23E-01	4.58E-08	1.E+02
Zr-95	4.49E+01	2.85E-06	1.E+01
Nb-95	2.44E+02	1.55E-05	7.E+01
Mo-99	1.66E+03	1.05E-04	7.E+01
Tc-99m	2.23E-01	1.41E-08	7.E+03
Ru-103	1.04E+02	6.58E-06	3.E+01
Rh-103m	8.24E-02	5.22E-09	7.E+04
Ru-106	1.35E-02	8.56E-10	7.E-01
Rh-106	1.35E-02	8.56E-10	4.E+01
Ag-110m	5.86E-02	3.71E-09	4.E+00
Sb-124	5.37E+00	3.40E-07	1.E+01
Te-129m	1.63E-01	1.03E-08	1.E+01
Te-131m	5.50E-02	3.49E-09	4.E+01
Te-132	1.41E-02	8.91E-10	3.E+01
Cs-134	1.78E+02	1.13E-05	7.E+00
Cs-136	1.47E+01	9.31E-07	3.E+01
Cs-137	2.69E+02	1.70E-05	7.E+00
Cs-138	8.50E-02	5.39E-09	3.E+03
Ba-140	7.82E+02	4.96E-05	7.E+01
La-140	1.29E+00	8.19E-08	7.E+01
Ce-141	2.66E+02	1.69E-05	3.E+01
Ce-144	1.35E-02	8.53E-10	7.E-01
Pr-144	1.35E-02	8.53E-10	7.E+00
W-187	1.29E-01	8.21E-09	4.E+02
Np-239	8.28E+00	5.25E-07	1.E+02

3.1.1.5 Releases in the course of normal operation of a nuclear power plant in Poland according to the provisions of the Atomic Energy Act

3.1.1.5.1 Requirements set forth in Polish regulations

According to the draft regulations which are to be adopted in Poland, the general objective of atomic nuclear safety is to protect individuals, the society, and the environment by establishing and maintaining effective measures protecting them from radiologic threats. For a nuclear facility, the above-mentioned general objective translates into the basic objective of radiological protection, namely assuring that in the course of normal operation the exposure to ionizing radiation inside the building and to radiation doses resulting from release of radioactive substances are maintained on the lowest achievable level and below the maximum permissible values.

The same requirement - to maintain the doses resulting from release of radioactive substances on the lowest reasonably achievable level and below the maximum permissible values – in force in the United Kingdom for which the above-mentioned designs of EPR, AP1000, and ESBWR reactors were elaborated. The measures taken in the design of each of these reactors measures are sufficient to meet the requirement.

According to further texts of draft regulations of the Council of Ministers, in the case of nuclear power plants with light water reactors or with reactors with pressure channels, the typical safety functions performed by appropriate designs, systems, and equipment consist in "limiting the discharge or release of radioactive waste and radioactive substances that can be found in the air to values that are below the defined limits, in all operation states."

According to §193, a nuclear facility must be provided with appropriate systems for processing radioactive liquids and gases so as to maintain the quantities and concentrations of release of radioactive materials within certain limits in the course of normal operation and of anticipated operational occurrences. The principle of maintaining releases of radioactive materials on the lowest reasonably achievable level must also be observed. In particular, it is necessary to provide appropriate possibility to store gaseous and liquid discharges containing radioactive substances, especially if it is expected that disadvantageous environmental conditions in the vicinity of the site may cause extraordinary restrictions on their release into the environment.

The release limits for nuclear power plants are set by nuclear regulatory authorities. In Poland, the limits have not been defined. Therefore, one can only compare the release values for nuclear power plants with the limits in force in other countries of the European Union. The limits which are in force in France are very stringent and are the product of many years of work on improving the reactors and the operation procedures.

What follows is a verification if the reactors which have been analyzed in the earlier parts of the present document meet these limits.

3.1.1.5.2 Comparisons of three studied types of reactors

Błąd! Nie można odnaleźć źródła odwołania. shows the results of the comparison of the EPR, AP1000, and ESBWR reactor.

Table 3.1.23. Comparison of the limit values defined for new nuclear power plants in France with release values from the nuclear power plant in Flamanville 1, 2 (Generation II reactors) and for Generation III reactors with EPR, AP1000, and ESBWR reactors

NUCLEAR POWER PLANT	Flamanville, 2x1,300 MWe (new limits)		EPR, 1650 MWe	AP1000, per 1000 MWe	ESBWR[19] acc. to the table
Isotopes	Limit	Actual emission	Anticipated/Maximum	Anticipated	Anticipated

Inert gases, TBq/year	45	0.90	0.8 / 22.5	10.3	153
Tritium, TBq/year	5	2.03	0.5/3	13	2.8
Carbon C-14, TBq/year	1.4	0.416	0.35/0.9	0.27	0.35
Iodine, GBq/year	0.8	0.108	0.05/ 0.4	0.6	29
Aerosols, GBq/year	0.8	0.0049	0.004/0.34	1.7	4.6

The EPR reactor has the lowest anticipated release values of inert gases, tritium, iodine isotopes, and aerosols. The AP1000 reactor has the highest anticipated release values of inert gases, tritium, iodine isotopes, and aerosols. The release values of inert gases and iodine isotopes from the ESBWR reactor are two orders of magnitude larger, and of aerosols - three orders of magnitude larger than release values from the EPR reactor; the release values of tritium and carbon C-14 from these two types of reactors are approximately on the same level. The results of the comparison conform to the expectations, because reactors with only one water-steam look cannot prevent releases into the atmosphere as effectively as reactors with two loops separated with a partition consisting of heat exchange pipes in the vapour generator. Unlike in pressurized water reactors, where only a small part of the reactor coolant is vaporized (and discharged through the chemical and volume control system, CVSS, in order to continuously clean and control the boron in the primary loop), in boiling water reactors the whole reactor coolant is continuously evaporated in the turbine condenser. Nevertheless, emissions from ESBWR reactors do result in exceeding the permissible concentrations of radionuclides which are set forth in the 10CFR20 American federal regulations.

For each of the reactors, the radiation doses received by the inhabitants are defined in Chapter 7.3.

3.1.2 Emissions in transient and accident conditions

3.1.2.1 Emissions in transient and accident conditions from Generation II nuclear power plants

3.1.2.1.1 Characteristics of possible accidents in nuclear power plants with water reactors

According to their frequency, possible accidents in Generation II nuclear plants can be divided into three groups:

- moderately frequent accidents;
- rare accidents;
- borderline accidents which should never happen but are assumed in the analysis as borderline cases in order to determine the potential possibility of release of radioactive substances.

In the case of moderately frequent accidents, such as coolant pressure drop as a result of accidental opening of a relief valve, loss of flow in the main supply water circuit, loss of mains power supply, or a single operator error, the situation should be controlled by way of actions that do not lead to consequences larger than reactor shutdown. After the disturbance is eliminated, the nuclear power plant should be capable of resuming operation. Such events may not cause a breach of any of the three barriers which limit the spread of fission products or lead to category 2 and 3 accident conditions, in the absence of other simultaneous accidents.

In the event of rare accidents, such as significant leak of coolant in the primary loop, loss of forced coolant flow, erroneous movements of control rods, or erroneous loading of fuel, there may be small damage to a part of the fuel in the core, and the quantity of released fission products may be larger than in the course of normal operation. The released products may not, however, cause any hazards related to the use of pastures, farmland, etc. by people in areas located outside of the prohibited

area. An accident of this type may not cause a category 3 accident condition or lead to the loss of tightness of the primary loop or the safety enclosure.

In the case of borderline accidents, such as rupture of the main primary loop pipeline or secondary loop pipeline, jamming of a pump rotor, or shooting of the control rod, the maximum quantity of fission products defined in regulations concerning the siting of nuclear plants may be released into the environment, but the safety systems that limit the consequences of accidents, shut the reactor down and cool it must remain in good working order.

3.1.2.1.2 Hazard after design condition accidents and hypothetical accidents in Generation II nuclear power plants

The safety principles adopted when designing, building, and operating nuclear power plants have proven to be so effective, that despite the experiences gathered in the course of over ten thousand of reactor-years of operations of nuclear power plants with water moderators and coolants, so far there have been no accidents where any member of the staff or the public would lose their life or health as a result of radiation exposure. The Chernobyl accident did involve loss of health and lives, but it occurred in a reactor which was fundamentally different than a light-water reactor, similar to military reactors which were designed to make plutonium. This accident cannot be included in the health balance of the purely civilian nuclear power sector.

As far as light-water reactors are concerned, the accident with the most serious consequences was the accident at the Three Mile Island (TMI) nuclear power plant, where the reactor core was completely destroyed to the extent that operation of the plant was no longer possible. Nevertheless, the health impact of this particular accident was negligible.

During another accident, that in the Browns Ferry nuclear power plant in 1975, which was caused by a technician who was checking the tightness of a cable duct, the fire destroyed most connections important from safety standpoint. As a result, the emergency core cooling system and all other core water injection systems were lost. The fire was eventually suppressed by injecting water to the cable duct, which involved the risk of short circuits in power cables and of further deterioration of the situation. Despite the very extensive damage in the reactor systems, the fire did not cause any loss of health or lives of any staff members or the public.

An analysis of the conclusions from the fire has resulted in a number of improvements in fire detection and suppression systems in all nuclear power plants, which in many cases required many months of outage. Once this process was completed, the fire safety level in existing nuclear power plants improved significantly. Also, new nuclear power plants were built taking into account the conclusions from the Browns Ferry accident.

Other accidents in nuclear power plants were of more limited scopes and did not lead to serious damage to fuel or release of fission product. Even then damage similar to that which initiated the accident at the Three Mile Island took place, the operators - aware of the errors made at the TMI – brought the nuclear power plant into a safe shutdown condition and prevented any damage to the core. .

According to the criteria adopted by the US Nuclear Regulatory Commission (NRC), the calculated frequency of accidents with core meltdown must be lower than 10^{-4}/reactor-year, and the calculated value of any release of fission products producing, at the distance of 0.8 km from the reactor, a dose absorbed by the whole body in excess of 0.25 Sv, must be less than 10^{-6}/reactor-year[20]. The requirements set forth by US power companies are even more stringent and set the target of reducing the frequency of accidents involving core meltdown to 10^{-5}/reactor-year. In European Union countries, power companies elaborated guidelines similar to the American guidelines, to be adopted as a basis for designing new nuclear power plants[21].

The requirements of the nuclear regulatory authorities are different in different countries, but new nuclear power plants meet the most stringent of them. For example, according to the 2008 decree of the government of Finland[22], the maximum dose for the critical group of people living in the surroundings of a nuclear power plant must not exceed 5 mSv after a class 2 design–basis accident which occurs less than one time per 1000 years of a reactor's operation and 20 mSv after a comprehensive combination of defects which a nuclear power plant must withstand without core meltdown. Releases of fission products in the case of a design-basis accident must not lead to restrictions in the use of the ground and food. The dose after a severe accident has not been defined, but the limits of release of fission products have been set forth. After an accident involving core meltdown, the limit value for release of radioactive substances is such a release that does not cause either severe damage to health among members of the public in the vicinity of the nuclear power plant or long-term restrictions of the use of large areas of soil and water[23].

After 50 years of experiences with the operation of nuclear power plants which are built and operated in accordance with the safety principles defined in Western countries and popularized by international agencies (IAEA), one can conclude that the nuclear power sector is one of the safest sectors of the industry. On the other hand, the example of the RBMK reactors in the former USSR and the Chernobyl accident demonstrate that any deviations from safety rules are not permissible.

Thus, even though the basic principles of safety and the nuclear safety are set above political motives, the defence-in-depth system guarantees that a nuclear power plant will remain safe even in the event of defective equipment and human error.

The organization and safety culture in European Union countries and in Poland appear to guarantee that in practice operation of a nuclear power plant does not lead to any risk to the natural environment or to the health or lives of the people living in its vicinity.

3.1.2.2 Types of accidents adopted in regulations recommended for Poland

The types of accidents which are adopted in the regulations currently recommended for Poland reflect the progress achieved in enhancing the safety of power reactors and sets appropriately high requirements for Generation III and Generation III+ reactors that may be built in Poland.

Art. 36 (f) (2) of the draft amendment of the Atomic Energy Act of 29 November 2000 (Journal of Laws of 2007, no. 42, item 276, as amended), has the following wording:

"The restricted-use area around a nuclear facility covers an area outside of which:

1) during operation of the nuclear facility, which includes regular operation and expected operation events, the yearly effective dose from all routes of exposure does not exceed 0.3 milisivert (mSv);

2) in the event of an accident without core meltdown, there will be no need to evacuate the public or to implement long-term restrictions in the use of soil and water around the power plant, or the "yearly effective dose caused by external radiation from the cloud and from deposits, and as a result of exposure through the respiratory tract, will not exceed 20 milisivert (mSv)"[24].

Art. 3 includes a definition of a severe accident:

"severe accident – accident conditions at a nuclear facility that are more severe than design-basis accidents and lead to significant degradation of the reactor core and, potentially, to significant releases of radioactive substances."

§2 (28) (b) of the draft Regulation of the Council of Ministers concerning requirements for safety analyses conducted prior to applying for a permit to build a nuclear facility and for the content of safety report for a nuclear facility divides accident conditions into the following groups:

- design-basis accidents, which can be further divided into:
  - category 1 design-basis accidents;
  - category 2 design-basis accidents;
- beyond design-basis accidents, which can be further divided into:
  - accidents without significant degradation of the core;
  - severe accidents.

Moreover, the draft Regulation of the Council of Ministers concerning the nuclear safety and radiological protection requirements that must be observed in nuclear facility designs defines design extension conditions as "a collection of sequences of beyond design-basis accidents, selected on the basis of deterministic and probabilistic analyses, for which the design principles and criteria are different than for design-basis accidents, which includes:

- complex sequences,
- selected severe accidents."

Item 6 of the aforementioned draft of the regulation provides that:

The design of a nuclear facility must assure limiting of release of radioactive substances outside of the safety containment of the reactor in accident conditions so that:

in the event of design-basis accidents, no interventions are required at a distance larger than 800 m from the reactor;

in the event of design extension conditions it is not necessary to:

a) take early intervention measures during release of radioactive substances from the safety containment at the distance of more than 800 m away from the reactor;

b) take mid-term intervention measures at any time at the distance of more than 3 km away from the reactor;

c) take long-term intervention measures at the distance of more than 800 m away from the reactor.

Thus, according to the regulations to be in force in Poland, design-basis accidents must be controlled by the reactor safety systems before the core is damaged. Accidents where the core becomes damaged, either partly (for example in the process of fuel melting occurring for a certain time, with the fuel remaining inside the reactor containment), or completely (for example where full core meltdown occurred with the fuel remaining inside the reactor containment (in an AP1000 reactor) or where the molten core is cooled down and kept in the core catcher inside the safety containment (in an EPR reactor)), are classified as severe accidents.

In the regulations in force in most countries of the European Union and adopted by the International Atomic Energy Agency, similarly to the Polish regulations draft, it is considered that nuclear power plant's safety systems are designed so as to limit the consequences of an accident to damage of the jackets of some fuel element and to stop the spread of the accident before core meltdown occurs. The values of emissions and doses during such accidents are demonstrated for the UK EPR reactor. They will be used as reference values for reactors that may be built in Poland.

In US regulations, on the other hand, it is assumed that, despite the presence of safety systems, accidents in a reactor may lead not only to damage to the fuel jacket, but also to partial fuel meltdown. The values of emissions and doses after such accidents are defined in reactor designs submitted to the US nuclear regulatory authorities for approval, e.g. the US EPR or the US AP1000. The values will be used as a measure of the hazard that may occur in Poland after accidents involving core meltdown, that is after events defined in the Polish regulations as severe accidents.

3.1.2.3 Values of releases in the event of design-basis accidents in nuclear power plants with EPR reactors

3.1.2.3.1 Main Design Objectives

Since 1992, Framatome and Siemens, in cooperation with the EDF and the major German power sector operators, have worked on developing the European Pressurized Reactor, also referred to as the Evolutionary Pressurized Reactor (EPR).

Two Design Objectives for the EPR reactor were defined.

Based on a thorough evaluation of the different solutions regarding passive safety systems, a decision was made to define the EPR reactor using an evolutionary approach, that is based on the experiences related to the operation of about 100 nuclear power plants built by Framatome and Siemens.

An objective which was equally important as the adoption of an evolutionary approach was to assure competitiveness of electric energy generation compared to other alternative energy sources. The EPR is to guarantee a significant reduction of the cost of generation of electricity compared to most modern nuclear power plants and large gas-steam power plants. In order to achieve this objective, a decision was made to design high-capacity units, in the order of 1,600 MWe.

The safety is assured by separable systems working in a direct mode. Four separate, redundant loops of all safety systems are installed in four separate buildings or safety building parts which are separated with resistant physical barriers and with assured strict separation (spatial and physical separation) so as to prevent their simultaneous failure caused, for example, by internal factors. Such a quadruple redundancy of the main loops of the safety systems assure flexibility with regards to making the design meet the requirements regarding maintenance and thus reducing the standard reactor downtime. In new generation nuclear power plants, new additional elements and functions are used to meet the safety criteria imposed by relevant nuclear safety authorities regarding improved protection during accidents and incidents, to include reactor core meltdown and its radiological effects, as well as resistance to external threats, in particular plane crashes and earthquakes.

Thus, the evolutionary approach selected by the designers of the EPR reactor constitutes an optimum mix of proven solutions from the most extensive available experiences and innovative solutions necessary to meet new requirements, especially as regards safety.

Class 1,600 MWe EPR reactors are characterized by high efficiency, reduced construction time, longer operation period, improved and more flexible use of fuel, and improved availability, which translates into their outstanding competitiveness with regards to costs per 1 kW of installed power and per 1 kWh of generated energy.

3.1.2.3.2 Recommendations of French and German nuclear safety authorities:

According to the principles set forth by French and German nuclear safety authorities for the next generation PWR reactors, an EPR reactor meets the following criteria:

"Evolutionary" design aimed at taking advantage of the accumulated experience in designing and operating PWR units which are currently in use in France and Germany and in countries where Framatome and Siemens have exported their technologies (Belgium, Brazil, China, Korea, South Africa, Spain, and Switzerland). The EPR is based mostly on the experiences related to the French technology used in N4 reactors and the German Konvoi technology.

Increased safety level. On the one hand, the likelihood of a reactor core meltdown was reduced by improving the availability of the safety systems; on the other hand, the design incorporates solutions aimed at reducing the radiological consequences in the event of a severe accident. In the event of an accident involving no core meltdown, the architecture of the peripheral buildings and the ventilation system eliminate the need to implement protective measures in relation to people living in the vicinity of a damaged nuclear power plant unit. In the event of the highly unlikely, but still considered, situation where the reactor core melts down in low-pressure conditions, the strengthened reactor building and the unique equipment mitigating the consequences of the accident will reduce the radioactive emissions. Only some very limited protective measures would be required. Moreover, the design of the reactor and the concept of the safety containment eliminate the possibility of situations which would cause large emissions at early stages of an accident.

In the case of an EPR reactor, the likelihood of an accident causing a reactor core meltdown – already very low in previous generation reactors – is reduced even further.

Taking potential operation problems into account at early stages of the design works. At the base stage of design works thorough analyses were performed to reduce as much as possible the collective radiological exposure of the nuclear power plant's staff. Maintenance of equipment has been made more effective by assuring easy access. Moreover, the design takes into account the human factor so as to minimize the chance for human errors in the operation of a power unit with an EPR reactor.

3.1.2.3.3 The requirements concerning protection of the environment from radiation adopted in the EPR design in accordance with the European Utility Requirements (EUR)

3.1.2.3.3.1 EUR requirements

The design of an EPR reactor has Design Conditions[25] that cover normal operation, incidents, and accidents, as well as design extension conditions. A separate category is a severe accident which will be described in Chapter 4.3.

In normal operation it is expected that the Design Objectives will assure observance of the applicable limit values of doses for the employees and the public defined by national or international regulatory authorities or licensing authorities. The criteria for incident conditions and accident conditions have values which are considered as appropriate to assure licensing in countries participating in the elaboration of the EUR requirements.

The Design Objectives for design extension conditions are determined so as to avoid the need to take significant protective measures outside of the nuclear power plant. The criteria for limiting the impact of such conditions which exceed the design assumptions would result in limited effects on the public.

In incident conditions, like during normal operation, the dose resulting from direct radiation from the reactor must not exceed 0.1 mSv/a. This value is the same irrespective of the reactor's capacity.

3.1.2.3.3.2 Objectives concerning emissions in accident conditions

Emissions in accident conditions, evaluated using the best evaluation method, must not exceed the targets set for each of the accident categories defined in Appendix B to the EUR document[26]. The

relations between the accident categories and the frequency of their initiating events are shown in table 2 of the EUR document[26].

The targets are defined as a linear combination of releases in each of the isotope reference groups. The criteria and the methods of evaluating acceptability of such releases are shown in Annex B for all the accident conditions*.

The emission thresholds are also defined for design extension conditions.

Appendix B lists the acceptance criteria for all accidents within the category 3 and 4 Design Conditions. The criteria are selected so that the releases which do not exceed them constitute little hazard to people in the vicinity of the nuclear power plant and do not require intervention measures affecting the public at the distance of over 800 m away from the reactor. It is also expected that the economic consequences in such cases will be very limited (to the area of several square kilometers and one crop harvest).

Limited impact criteria for design extension conditions accidents.

The meeting of the criteria is verified in the following manner:

- The releases from the nuclear power plant into the atmosphere are divided into 9 isotope reference groups;
- The releases are added and compared with the criterion value in accordance with the following formula:

$$\sum_{i=1}^{9} R_{ig} \cdot C_{ig} + \sum_{i=1}^{9} R_{ie} \cdot C_{ie} < criterion$$

In the formula for the linear combination of releases, Rig and Rie are the total releases during the period of release from the safety containment (on the ground level and from the ventilation stack) for the 9 isotope reference groups.

For sequences where the safety containment remains intact, the period of release must be determined based on the pressure in the containment and the value of the releases. Rig and Rie have three different values, depending on the objective to be verified.

In the case of objective no. 1, R(1)ig and R(1)ie are releases during the first 24 hours; **in the case of objective no. 2**, R(2)ig and R(2)ie are releases during the first 4 days, and in the case of objective no. 3, R(3)Ig and R(3)ie are releases caused by a severe accident. Cig and Cie are coefficients listed in **Błąd! Nie można odnaleźć źródła odwołania.** and **Błąd! Nie można odnaleźć źródła odwołania.**, related to the effects of unit releases into the environment.

To verify if the economic consequences of possible accidents are within the permissible limits, only three isotopes are considered. The releases for each of them are compared to an independent criterion.

Table 3.1.24. Limited impact criteria assuring that no early intervention measures are required more than 800 m away from the reactor.

Isotope group	Release coefficient at the ground level, C_{ig}	Release coefficient at the ventilation stack, C_{ie}
Xe-133	$6.5*10^{-8}$	$1.1*10^{-8}$
I-131	$5.0*10^{-5}$	$3.1*10^{-6}$
Cs-137	$1.2*10^{-4}$	$5.4*10^{-6}$
Te-131m	$1.6*10^{-4}$	$7.6*10^{-6}$

Sr-90	$2.7*10^{-4}$	$1.2*10^{-5}$
Ru-103	$1.8*10^{-4}$	$8.1*10^{-6}$
La-140	$8.1*10^{-4}$	$3.7*10^{-5}$
Ce-141	$1.2*10^{-3}$	$5.6*10^{-5}$
Ba-140	$6.2*10^{-6}$	$3.1*10^{-7}$

$$\sum_{i=1}^{9} R_{ig} \cdot C_{ig} + \sum_{i=1}^{9} R_{ie} \cdot C_{ie} < 5 \cdot 10^{-2}$$

Rig and Rie (expressed in TBq): accumulated releases during the first 24 hours after a design extension conditions accident.

Table 3.1.25. Limited impact criteria assuring that no delayed intervention measures are required more than 3 m away from the reactor.

Isotope group	Release coefficient at the ground level, C_{ig}	Release coefficient at the ventilation stack, C_{ie}
Xe-133	0	0
I-131	$1.2*10^{-6}$	$3.5*10^{-7}$
Cs-137	$5.6*10^{-6}$	$8.9*10^{-7}$
Te-131m	$3.8*10^{-6}$	$7.0*10^{-7}$
Sr-90	$9.9*10^{-7}$	$3.2*10^{-7}$
Ru-103	$1.3*10^{-6}$	$2.2*10^{-7}$
La-140	$2.9*10^{-6}$	$4.8*10^{-7}$
Ce-141	$4.5*10^{-6}$	$8.1*10^{-7}$
Ba-140	$1.5*10^{-6}$	$2.5*10^{-7}$

$$\sum_{i=1}^{9} R_{ig} \cdot C_{ig} + \sum_{i=1}^{9} R_{ie} \cdot C_{ie} < 3 \cdot 10^{-2}$$

R_{ig} and R_{ie} (expressed in TBq): cumulative releases during the first 4 days after the accident.

Table 3.1.26. Limited impact criteria assuring that no long-term intervention measures are required more than 800 m away from the reactor.

Isotope group	Release coefficient at the ground level, C_{ig}	Release coefficient at the ventilation stack, C_{ie}
Xe-133	0	0
I-131	$1.2*10^{-5}$	$7.8*10^{-7}$
Cs-137	$6.5*10^{-5}$	$3.4*10^{-5}$
Te-131m	$2.6*10^{-5}$	$1.3*10^{-6}$
Sr-90	$1.4*10^{-5}$	$7.2*10^{-7}$
Ru-103	$2.3*10^{-5}$	$1.2*10^{-7}$
La-140	$7.9*10^{-5}$	$4.1*10^{-6}$
Ce-141	$7.6*10^{-5}$	$4.0*10^{-6}$
Ba-140	$1.1*10^{-5}$	$5.9*10^{-7}$

$$\sum_{i=1}^{9} R_{ig} \cdot C_{ig} + \sum_{i=1}^{9} R_{ie} \cdot C_{ie} < 1 \cdot 10^{-1}$$

For each reference isotope, the sum of releases on the level of the ground and of the ventilation stack over the whole time of accident should be comparable with reference values listed in the table below.

Table 3.1.27. Criteria of limited impact of an accident on economic effects

Isotope	Reference value (TBq)

I-131	4,000
Cs-137	30
Sr-90	400

3.1.2.3.4 The objectives limiting releases in the event of category 3 and 4 design basis accident.

In the case of accidents classified as category 3 and 4, the same general approach is used as in the case of Design Extension Conditions in order to demonstrate that the project will achieve the following objectives:

- no actions are required more than 800 m from the reactor;
- the economic impact of the accident is limited.

The methodology is similar to that used with regards to the previous three objectives. The difference is the possibility to consider fewer isotopes. It is recommended that releases from a nuclear plant be divided into three isotope groups, added, and compared with one criterion, in accordance with the following formula:

$$\sum_{i=1}^{3} R_{ig} \cdot C_{ig} + \sum_{i=1}^{3} R_{ie} \cdot C_{ie} < kryterium$$

where:

R_{ig} and R_{ie} are total release values at the ground level and at the level of the ventilation stack for the three reference isotopes during the whole period of release from the safety containment.

C_{ig} and C_{ie} are indicators defined in the tables below, related to the impact of unit releases on the environment. The indicators can be used for all design extension conditions accidents.

Table 3.1.28. Objectives related to releases during design-basis accidents aimed to exclude the need to take intervention measures at a distance of more than 800 m.

Isotope group	Release coefficient at the ground level, C_{ig}	Release coefficient at the ventilation stack, C_{ie}
Xe-133	$1.5*10^{-8}$	$3.0*10^{-9}$
I-131	$8.1*10^{-5}$	$5.5*10^{-6}$
Cs-137	$1.5*10^{-4}$	$8.1*10^{-5-}$

$$\sum_{i=1}^{3} R_{ig} \cdot C_{ig} + \sum_{i=1}^{3} R_{ie} \cdot C_{ie} < 1 \cdot 10^{-3} \; forclass\ 3\ DBC$$

The value of the criterion for class 4 accidents is five times larger because such accidents occur much less frequently.

$$\sum_{i=1}^{3} R_{ig} \cdot C_{ig} + \sum_{i=1}^{3} R_{ie} \cdot C_{ie} < 5 \cdot 10^{-3} \; forclass\ 4\ DBC$$

The release values are in TBq units.

The limitations listed in the table below affect the activity of crops and milk after an accident and are more stringent than the limitations of the impact of design extension conditions on the health of people. They pertain to only two isotopes - I-131 and Cs-137 – and have the following values:

Table 3.1.29. Objectives limiting releases considering the economic consequences.

Isotope	Objective for release on the ground level, TBq	Objective for releases at the level of the ventilation stack, TBq.
I-131	10	150
Cs-137	1.5	20

The same objectives apply to class 3 and 4 accidents.

3.1.2.3.5 Assumptions made when evaluating the values of releases from a reactor

Activity in the primary loop coolant can be determined based on the maximum values used in the technical specification for French nuclear power plants. In the case of an EPR reactor, the following values are used:

Activity of iodine in the primary loop in steady state equal to 20 GBq/Mg of iodine-131. Activity of iodine in the primary loop in transient state after change of capacity (iodine peak): 150 GBq/Mg equivalent activity of iodine-131 according to the formula:

I-131 eq = I-131 + I-132 / 30 + I-133 / 4 + I-134 / 50 + I-135 / 10.

Activity in the secondary loop can be calculated based on the following assumptions:

- Coolant activity in the primary loop is equal to the maximum values stipulated in the technical specification;
- Water leaks from the primary loop to the secondary loop at the rate of 20 l/h or 0.48 m^3/day.
- Blowdown of steam generators in accordance with the parameters for full-capacity operation
- The lift coefficients taken into account at the transfer of activity from water to the vapour phase in steam generators are as follows:
  - All inert gases contain in the water move to the vapour phase;
  - As far as other radionuclides are concerned, there are the following options:
    - In a steam generator with ruptured heat exchange pipe, the lift coefficient is 1%. In a steam generator with no ruptured pipes, the lift capacity coefficient is assumed to be equal to 0.25%.
    - The values of releases in the case of ruptured fuel element jacket assumed in the calculations are listed in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.1.30. Rate of release of activity in the case of a defective fuel jacket in an EPR reactor.

	Rate of release from UO2 fuel assumed in the calculations		Rate of release from MOX fuel assumed in the calculations	
Isotope	**Burnout ≤ 47 GWd/t**	**Burnout > 47 GWd/t**	**Burnout ≤ 33 GWd/t**	**Burnout > 33 GWd/t**
Kr-85	8 %	25 %	8 %	50 %
Other inert gases	2 %	8 %	2 %	15 %
Bromine, rubidium, iodine, caesium	2 %	8 %	2 %	15 %

Deposition of fission products: In the formulas describing the deposition of aerosols and molecular iodine in the safety containment, it is assumed that the deposition constants are equal to, respectively, 0.035/h and 0.014/h.

The rates of leaks through the safety containment is assumed, for the interior containment of an EPR reactor (with steel lining) to be equal to 0.3% of its volume per day at the design pressure (5.5 bar).

The efficiency of the filters is assumed to be as follows:

High-efficiency filters: inert gases – 0%; aerosols (to include iodine aerosols) – 99.9%; all other substances – 0%.

High-efficiency filters plus iodine trap: inert gases – 0%; organic iodine – 99%; molecular 99.9%; aerosols (to include iodine aerosols) – 99.9%.

3.1.2.3.6 Accidents involving steam loop rupture outside of the safety containment.

The analysis of the consequences of the steam pipeline rupture outside of the safety containment of an EPR reactor is based on the assumption that immediate rupture with two-side leak takes place in the valve room in one of the safety buildings, upstream of the main cut-off valve on the steam conduit. The pressure in the ruptured pipeline immediately drops to a value at which the defective pipeline is cut off.

Release into the atmosphere occurs through the overflow valves on the steam generators which were not damaged and continues for 8 hours, while release from the damaged steam generator continues for 9 hours (after that time the temperature on the secondary side of the steam generator drops to 100 ^{o}C).

In the event that a peak iodine release occurred before the accident or as a result of the accident, the following pessimistic assumptions are made:

- The control valve on the steam pipeline in loop no. 3 is damaged in the open position, which leads to the pipes in the steam generator being uncovered for 30 minutes and to a direct release of fission products into the atmosphere as a result of flashing in steam generator no. 3. The release ends when the main cut-off valve is closed on the steam conduit, when the pressure in the steam collector drops below 41 bar.

- Loss of one of the two mutually redundant iodine filtration systems in the control room.

- Loss of external power supply is not assumed because the option with available power supply is more hazardous from the standpoint of release of radioactivity into the environment. Moreover, it is assumed that the pipeline was ruptured upstream of the main cut-off valve so that the seizure of this valve in the open position cannot be considered as a single defect.

Of the various situations being considered, the most dangerous is the one with the iodine peak occurring earlier so that the concentration of iodine in the primary loop coolant has reached the maximum value equal[27] to iodine-131 dose equivalent (DE) of 60 μCi/g.

Another scenario under consideration is one where iodine peak occurs simultaneously with the accident involving pipeline rupture, when the rate of release of iodine into the primary loop increases 500-fold.

Moreover, a scenario considered in the case of a severe accident involves simultaneous burn-through of some fuel jackets and meltdown of a part of the fuel; however, such scenarios go beyond design-basis accidents and will not be discussed in this section.

To be on the safe side, it is also assumed that leaks from the primary loop into the ruptured steam generators are released directly into the atmosphere through the ceiling of the building, where the safety valves of the steam generators are located, without any reduction or restriction on the release. Such releases continue for 9 hours, until the loop cools down to 100 ^{o}C.

The potential radiological consequences of such an accident are limited to 2.7 mSv at the boundary of the restricted-use area.

3.1.2.4 Nuclear power plants with AP 1000 reactors

According to the safety documentation of the AP1000 reactor submitted in the United Kingdom[28], the greatest radiological impact results from an accident with rupture of the main pipeline of the primary cooling loop. Such an accident does not involve core meltdown but only to rupture of the fuel jackets. The radiological impact of other design-basis accidents is even less severe.

The fractions of fission products released from the gap between the jacket and the fuel pellets are defined in the US nuclear regulatory authorities 10CFR and in the Regulatory Guide 1.183. **Błąd! Nie można odnaleźć źródła odwołania.** lists such fractions, together with the fractions of releases that occur if the accident is not limited to rupture of fuel jackets but involves fuel meltdown.

In the case of design-basis accidents which do not involve fuel meltdown, the relative emission values (per unit of reactor's capacity) in AP1000 reactors do not exceed the relative emission values for EPR reactors. At the same time, one must keep in mind that the electric capacity of EPR reactors, which is equal to 1,650 MWe, is greater than that of AP1000 reactors, which is equal to approx. 1,100 MWe. Thus, one can assume, with accuracy that is sufficient for the purpose of this evaluation, that emissions calculated for an EPR reactor and for accidents not involving core meltdown are representative of other Generation III reactors.

The calculations included in the safety reports for the AP100 reactor are based on the assumption, which conforms to the requirements set forth by the US nuclear regulatory authority, that a radiological assessment assumes that first the activity contained in the primary loop coolant is released into the containment in 10 minutes, then the activity contained in the gap under the fuel jacket is released in 30 minutes, and then the core melts down in 1.3 hours while releasing the activity contained in the fuel. These assumptions are not appropriate for AP1000 reactors which are designed specifically to prevent core meltdown. Nevertheless, since such assumptions correspond to the traditional model of a severe accident which is defined in the NRC guidelines RG 1.183[29] and RG 1.145, they will be used to describe the consequences of severe accidents in Generation III reactors.

Table 3.1.31. Fractions of fission products released during a maximum design-basis accident assumed in the safety analyses of AP1000 reactors according to the RG 1.183.

Group of activity in a PWR reactor core	Fraction releases inside the safety containment		
	Early release from the gap under the jacket	Release during the core meltdown phase	Total release
Inert gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkali metals	0.05	0.25	0.3
Tellurium	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Precious metals	0.00	0.0025	0.0025
Cerium	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

The proportion of the different forms of iodine is assumed in accordance with the model defined in the NUREG-1465[30] document, namely:

- Aerosol 95%
- Molecular 4.85%
- Organic 0.15%

The quantity of fission products in the core is assumed for burnout corresponding to the end of a fuel campaign, with reactor power 2% higher than the nominal power[31]; the values are defined in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.1.32. Quantity of fission products in the core of an AP1000 reactor for thermal power of the core equal to 3,468 MWt (2% above the design power of 3.400 MWt)

Nuclides activity (Ci)			Nuclides activity (Ci)		
Iodine	I-130	$3.66*10^{6}$	Inert gases	Kr-85m	$2.63*10^{7}$
	I-131	$9.63*10^{7}$		Kr-85	$1.06*10^{6}$
	I-132	$1.40*10^{8}$		Kr-87	$5.07*10^{7}$
	I-133	$1.99*10^{8}$		Kr-88	$7.14*10^{7}$
	I-134	$2.18*10^{8}$		Xe-131m	$1.06*10^{6}$
	I-135	$1.86*10^{8}$		Xe-133m	$5.84*10^{6}$
Caesium	Cs-134	$1.94*10^{7}$		Xe-133	$1.90*10^{8}$
	Cs-136	$5.53*10^{6}$		Xe-135m	$3.87*10^{7}$
	Cs-137	$1.13*10^{7}$		Xe-135	$4.84*10^{7}$
	Cs-138	$1.82*10^{8}$		Xe-138	$1.65*10^{8}$
	Rb-86	$2.29*10^{5}$	Sr & Ba	Sr-89	$9.66*10^{7}$
Tellurium	Te-127m	$1.32*10^{6}$		Sr-90	$8.31*10^{6}$
	Te-127	$1.02*10^{7}$		Sr-91	$1.20*10^{8}$
	Te-129m	$4.50*10^{6}$		Sr-92	$1.29*10^{8}$
	Te-129	$3.04*10^{7}$		Ba-139	$1.78*10^{8}$
	Te-131m	$1.40*10^{7}$		Ba-140	$1.71*10^{8}$
	Te-132	$1.38*10^{8}$	Cerium	Ce-141	$1.63*10^{8}$
	Sb-127	$1.03*10^{7}$		Ce-143	$1.52*10^{8}$
	Sb-129	$3.10*10^{7}$		Ce-144	$1.23*10^{8}$
Ruthenium	Ru-103	$1.45*10^{8}$		Pu-238	$3.83*10^{5}$
	Ru-105	$9.83*10^{7}$		Pu-239	$3.7*10^{4}$
	Ru-106	$4.77*10^{7}$		Pu-240	$4.94*10^{4}$
	Rh-105	$9.00*10^{7}$		Pu-241	$1.11*10^{7}$
	Mo-99	$1.84*10^{8}$		Np-239	$1.93*10^{9}$
	Tc-99m	$1.61*10^{8}$			

Regardless of the formal NRC regulations which require including the release of fission products after partial fuel meltdown in the activity emitted from AP1000 reactors after design-basis accidents, one can expect releases from AP1000 reactors to be actually similar to those from EPR reactors. This will be the assumption in evaluation of releases in the case of severe accidents in the following parts of the present document.

In the case of accidents involving rupture of the steam loop outside of the safety enclosure, the radiological consequences can be assessed assuming that before the accident the reactor worked with the fraction of damaged fuel elements equal to 0.25% of the total fuel in the core and that due to the leaks in the heat exchange pipes in the steam generators the iodine activity gradually collected in the secondary loop. It is assumed that once the steam collector was ruptured, the flow of supply water to the damaged steam generator will be cut off and the generator will become dewatered. The iodine isotopes transported with the primary loop coolant into the secondary loop are emitted

directly into the atmosphere. The reactor is cooled by evaporating the water from the steam generators that were not damaged.

If the iodine peak occurred before the accident, it is assumed that in the secondary loop coolant there is iodine I-131, in concentration equal to dose equivalent of 0.1 μCi/g, which constitutes 10% of the maximum concentration of iodine in the primary loop coolant in a balanced state. The concentration of alkali metals in the secondary loop water is also assumed to be equal to 10% of the concentration in the primary loop coolant in a balanced state. The rate of the leaks through each steam generator prior to the accident is assumed to be on the level of 150 gallons of coolant a day (or 0.567 m^3/day), which is an significantly exaggerated value.

The leaks take place through three paths:

- The secondary loop water is released into the environment as steam through the ruptures in the pipeline and it lifts the iodine and alkali metals isotopes it contains.
- Any leaks of the coolant from the primary loop to the secondary loop through the steam generator are released into the environment and (in principle) lift the iodine and alkali metals isotopes they contain (not taking into account the separation of iodine between the liquid phase and the gaseous phase or the iodine deposits on pipeline walls.
- The coolant which leaks from the primary loop into the steam generator that was not damaged is mixed with the secondary loop coolant and increases the concentration of activity in the secondary loop water. Even though when steam flows out of the steam generator that was not damaged iodine is separated into the liquid phase and the gaseous phase, the present analysis makes the cautious assumption that any activity transmitted into the secondary loop becomes released.

It is assumed that the time of increased releases of iodine into the primary loop (iodine peak) prior to the accident is 6 hours. This causes an increase of activity in the primary loop to 280 μCi/g dose equivalent Xe-133. The calculated dose values (see section 7.4.2.3) are larger than those from EPR reactors.

3.1.2.5 Nuclear power plants with ESBWR reactors

3.1.2.5.1 Types of accidents in ESBWR reactors

According to the American ANSI standard, design-basis accidents include events with frequency higher than 1 time per one million years.[32] Consequently, the design of the ESBWR reactor assumes that events that are less frequent are not included among design-basis accidents. This assumption conforms to the provisions of the draft regulation of the Polish Council of Ministers.

Anticipated operational occurrences (AOO) are events that, according to the 10CFR, take place one time during the useful life of a reactor. Because the useful life of an ESBWR reactor is 60 years, AOO cover all events whose frequency is 1 per 60 years; nevertheless, the design of an ESBWR reactor assumes a broader range, 1 time per 100 years (a more conservative assumption).

The provisions of the 10 CFR define accidents as any and all events which involve damage to one of the barriers preventing the release of fission products, which leads to radiological consequences outside of the nuclear power plant.

Because the regulations which set forth the actions taken by nuclear regulatory authorities when analyzing the safety of a nuclear power plant (Standard Review Plan) require the design basis to include all initiating occurrences involving a single defect or operator error, the ESBWR reactor

design takes into consideration normal operation, to include anticipated operating occurrences, namely:

- infrequent incidents;
- accidents;
- external events;
- natural phenomena.

The most dangerous are the following accidents:

- loss of coolant (pipeline rupture) inside the safety containment;
- rupture of the main steam collector outside of the safety containment;
- accidents occurring during handling of spent fuel.

3.1.2.5.2 Accident involving damage to 1000 fuel rods in an ESBWR reactor

For a number of accidents, the number of fuel rods which fail due to transition boiling was assumed to be limited to 1,000. Such cases include the following accidents:

- defective pressure controller – closing of the control valves and valves on the turbine bypasses;
- generator load discharge connected with failure of a turbine bypass;
- shutdown of a turbine with complete loss of a turbine bypass.

The assumptions made for the purpose of evaluating the emissions are shown in the table below.

Table 3.1.33. The parameters assumed to calculate emissions in ESBWR reactors in the event of failure of 1,000 fuel rods.

Parameter	Value
A. Power, MWt	4590
B. Number of fuel assemblies in the core	1132
C. Activity of fission products released into the coolant	Acc. to RG 1.183
E. Number of failed fuel rods	1000
F. Radial coefficient of unequal distribution for defective fuel rods.	1,5
II. Data and assumptions to evaluate the released activity	
A. Fraction of iodine released from damaged fuel.	10%
Fraction of gaseous fission products released from damaged fuel rods	10%
Fraction of alkali metals released from damaged fuel rods	12%
B. Fraction of iodine released from reactor coolant	10%
Fraction of inert gases released from coolant	100%
Fraction of alkali metals released from coolant	1%
C. Fraction of iodine released from condenser	10%
Fraction of inert gases released from condenser	100%
Fraction of alkali metals released from condenser	1%
Quantity of fission products released into the environment	See the table below

Table 3.1.34. Activity of isotopes released into the atmosphere of the safety containment from the primary loop of an ESBWR reactor (MBq) after failure of 1,000 fuel rods.

Radioisotope	Activity
Kr-85	$8.47*10^{7}$
Kr-85m	$1.88*10^{9}$
Kr-87	$3.63*10^{9}$
Kr-88	$5.11*10^{9}$
Rb-86	$1.94*10^{7}$
I-131	$6.82*10^{9}$
I-132	$9.92*10^{9}$
I-133	$1.40*10^{10}$
I-134	$1.55*10^{10}$
I-135	$1.32*10^{10}$

Table 3.1.35. Cumulative releases of fission products into the environment after failure of 1,000 fuel rods in an ESBWR reactor (MBq)

Isotope	2 hours	8 hours	24 hours
Kr-85	$7.07*10^{4}$	$2.82*10^{5}$	$8.44*10^{5}$
Kr-85m	$1.34*10^{6}$	$3.56*10^{6}$	$4.77*10^{6}$
Kr-87	$1.79*10^{6}$	$2.65*10^{6}$	$2.69*10^{6}$
Rb-86	1.61	6.42	1.89*10
I-131	$5.67*10^{4}$	$2.24*10^{5}$	$6.49*10^{5}$
I-132	$6.12*10^{4}$	$1.23*10^{5}$	$1.34*10^{5}$
I-133	$1.13*10^{5}$	$4.09*10^{5}$	$9.42*10^{5}$
I-134	$6.24*10^{4}$	$7.84*10^{4}$	$7.86*10^{4}$
I-135	$9.85*10^{4}$	$2.95*10^{5}$	$4.63*10^{5}$
Xe-133	$1.16*10^{7}$	$4.54*10^{7}$	$1.30*10^{8}$
Xe-135	$3.57*10^{6}$	$1.15*10^{7}$	$2.05*10^{7}$
Cs-134	$1.36*10^{2}$	$5.44*10^{2}$	$1.63*10^{3}$
Cs-136	4.73*10	$1.88*10^{2}$	$5.50*10^{2}$
Cs-137	8.84*10	$3.53*10^{2}$	$1.06*10^{3}$

3.1.2.5.3 Accidents with failure in the radioactive waste systems of ESBWR reactors

Tab. 3.1.36 Parameters assumed for the purpose of evaluation of radiological effects in the radioactive waste system in ESBWR reactors

Parameter	Value
A. Quantity of fission products	Tables 12.2-13a to 12.2-13g
B. Fraction of released iodine	100%
C. Duration of release	Immediate
II. Control room parameters	
A. Control room volume, m^3	$2.2*10^{3}$
B. Unfiltered air intake, litres	200
C. Length of stay indicators	Acc. to RG 1.183
C. Released activity	See the table below

Table 3.1.37. Releases into the atmosphere after a failure in the radioactive waste system (MBq)

Isotope	Activity (MBq)
I-131	$9.7*10^{4}$
I-132	$9.4*10^{3}$

Isotope	Activity (MBq)
I-133	$7.8*10^{4}$
I-134	$6.2*10^{3}$
I-135	$3.1*10^{4}$
Total for iodine isotopes	$2.2*10^{5}$

3.1.2.5.4 Accidents occurring during handling of spent fuel in ESBWR reactors

Table 3.1.38. Releases into the environment in the case of an accident occurring during handling of spent fuel in an ESBWR reactor

Isotope	Activity (MBq)
I-131	$4.4*10^{6}$
I-132	$2.9*10^{3}$
I-133	$2.8*10^{6}$
I-134	$3.2*10^{-2}$
I-135	$4.6*10^{5}$
Kr-85m	$7.8*10^{6}$
Kr-85	$1.5*10^{7}$
Kr-87	$5.8*10^{2}$
Kr-88	$1.2*10^{6}$
Xe-133	$1.1*10^{9}$
Xe-135	$6.6*10^{7}$

3.1.2.5.5 Accidents involving rupture of the steam collector of an ESBWR reactor outside of the safety containment

Such accidents do not cause damage to fuel. The only activity is the radioactivity in the primary coolant and steam loop present before the accident.

Releases into the environment are not filtered and occur directly from the turbine building into the atmosphere. The calculations were made assuming that the complete activity contained in the steam is released into the atmosphere.

Table 3.1.39. Releases into the atmosphere after rupture of the steam collector outside of the safety containment of an ESBWR reactor[33]

Isotope	Activity in balanced state, MBq	Peak iodine activity, MBq	Isotope	Activity in balanced state, MBq	Peak iodine activity, MBq
Co-58	$1.4*10^{3}$	$1.4*10^{3}$	Te-131m	$1.3*10^{3}$	$1.3*10^{3}$
Co-60	$2.7*10^{3}$	$2.7*10^{3}$	Te-132	$1.4*10^{2}$	$1.4*10^{2}$
Kr-85	1.7	1.7	I-131	$2.4*10^{5}$	$4.9*10^{6}$
Kr-85m	$4.4*10^{2}$	$4.4*10^{2}$	I-132	$2.3*10^{6}$	$4.6*10^{7}$
Kr-87	$1.4*10^{3}$	$1.4*10^{3}$	I-133	$1.7*10^{6}$	$3.4*10^{7}$
Kr-88	$1.4*10^{3}$	$1.4*10^{3}$	I-134	$4.2*10^{6}$	$8.5*10^{7}$
Rb-86	0.0	0.0	I-135	$2.4*10^{6}$	$4.7*10^{7}$
Sr-89	$1.4*10^{3}$	$1.4*10^{3}$	Xe-133	$5.9*10^{2}$	$5.9*10^{2}$
Sr-90	9.4*10	9.4*10	Xe-135	$1.6*10^{3}$	$1.6*10^{3}$
Sr-91	$5.2*10^{4}$	$5.2*10^{4}$	Cs-134	$3.7*10^{2}$	$3.7*10^{2}$
Sr-92	$1.2*10^{5}$	$1.2*10^{5}$	Cs-136	$2.4*10^{2}$	$2.4*10^{2}$
Y-90	9.4*10	9.4*10	Cs-137	$9.7*10^{2}$	$9.7*10^{2}$
Y-91	$5.5*10^{2}$	$5.5*10^{2}$	Ba-139	0.0	0.0

Isotope	Activity in balanced state, MBq	Peak iodine activity, MBq	Isotope	Activity in balanced state, MBq	Peak iodine activity, MBq
Y-92	$7.6*10^4$	$7.6*10^4$	Ba-140	$5.5*10^3$	$5.5*10^3$
Y-93	$5.2*10^4$	$5.2*10^4$	La-140	$5.5*10^3$	$5.5*10^3$
Zr-95	$1.1*10^2$	$1.1*10^2$	La-141	0.0	0.0
Zr-97	0.0	0.0	La-142	0.0	0.0
Nb-95	$1.1*10^2$	$1.1*10^2$	Ce-141		$4.0*10^2$
Mo-99	$2.7*10^4$	$2.7*10^4$	Ce-143		0.0
Tc-99m	$2.7*10^4$	$2.7*10^4$	Ce-144	4.0*10	4.0*10
Ru-103	$2.7*10^2$	$2.7*10^2$	Pr-143	0.0	0.0
Ru-105	0.0	0.0	Nd-147	0.0	0.0
Ru-106	4.0*10	4.0*10	Np-239	$1.1*10^5$	$1.1*10^5$
Rh-105	0.0	0.0	Pu-238	0.0	0.0
Sb-127	0.0	0.0	Pu-239	0.0	0.0
Sb-129	0.0	0.0	Pu-240		0.0
Te-127	0.0	0.0	Pu-241	0.0	0.0
Te-127m	0.0	0.0	Am-241	0.0	0.0
Te-129	0.0	0.0	Cm-242	0.0	0.0
Te-129m	$5.5*10^2$	$5.5*10^2$	Cm-244	0.0	0.0

3.1.2.6 Reference nuclear power plant for conditions present in Poland according to the requirements set forth in the Atomic Energy Act

The values listed above indicate that emissions during design-basis accidents for the three analyzed types of reactors are significantly different. The largest difference can be seen when comparing the effects of rupture of the steam pipeline outside of the safety enclosure. While the emissions associated with such accidents in EPR and AP1000 reactors are not large, the emissions in ESBWR reactors are large (especially the release of iodine dissolved in the medium circulating in the primary loop). Even though, after an accident involving rupture of the steam pipe in an ESBWR reactor, the main cut off valves become closed (which prevents the loss of coolant in the core), the inert gases and the iodine isotopes which were in the loop outside of the containment at the moment when the accident started are released into the environment. If the content of iodine in the loop before the accident is in a balanced state, the release into the atmosphere is 2,4*105; if an iodine release peak occurred before the accident (for example due to the transient state during reactor start-up), the release into the atmosphere is estimated to be $4.9*10^6$ MBq. As will be described in the chapter on the radiological consequences of design-basis conditions, such high releases in ESBWR reactors result in high doses, which do fit within the limits permissible according to the US regulations, but much higher than the limit doses permissible in most countries of the European Union. In the case of an AP1000 reactor, the content of iodine in the steam loop is much lower, because the only sources of iodine are the leaks through the heat exchange pipes in the steam generator, which should remain tight.

In the case of an EPR reactor, the additional reductions of emissions result from the design characteristics of this reactor, which were described in section 7.4.1.4.

The regulations in force in Poland and their proposed modifications do not define permissible emissions during design-basis accidents and, instead, use the limit doses at the boundary of the restricted-use area. Consequently, the present document also focuses on the definition of dose limit values.

3.1.3 Emissions in the event of severe accidents

3.1.3.1 Nuclear power plants with EPR reactors

3.1.3.1.1 Frequency of severe accidents in EPR reactors

An analysis of the characteristics of EPR reactors demonstrates a very high conformance of the reactor with the EUR requirements.[34] In particular, the EPR reactor meets the requirements concerning low frequency of accidents involving core failure, as defined in section 2.1-2.6 of the EUR.

The probability objectives defined in the EUR are as follows:

- total frequency of core failure due to all causes must be less than 10^{-5} per reactor-year (RY);
- total frequency of exceeding the criteria for limited impact of radioactivity releases on the environment (Criteria for Limited Impact - CLI), defined in Annex B to the EUR, must be less than 10^{-6} per reactor-year;
- total frequency of sequences which may potentially lead to early rupture of the safety containment or to very large releases of radioactivity must be much less than the aforementioned value of the criteria for limited impact (10-6 per reactor-year).

The results of the evaluation of the UK EPR reactor are as follows: [34]

- the total core damage frequency CDF = $6.8*10^{-6}$/R-Y, to include due to internal failures in the plant $6.1*10^{-7}$, and due to external threats^{-7}.$2*10^{-7}$;
- the total frequency of exceeding the criteria for limited impact is less than 10^{-6}/R-Y;
- the frequency of delayed safety containment failures is $5.3*10^{-8}$/R-Y; the frequency of early safety containment failures is $3.9*10^{-8}$/R-Y. This is, respectively, 9% and 6% of the frequency of core damage frequency caused by internal accidents.

3.1.3.1.2 Value of emissions after a severe accident in an EPR reactor

According to US regulations, analyses of US EPR reactors assume the maximum design-basis accident to be one involving rupture of the primary loop where failure of fuel jackets is followed by fuel meltdown. The course of such an accident is defined in the RG 1.183 guidelines, which assume that the coolant leak out period (10 minutes) is followed by failure of the fuel jacket and release of the fission products from the gap between the jacket and the fuel (30 minutes), and then by fuel meltdown (90 minutes). Thus, an accident of this type is a severe accident with fuel meltdown.

Table 3.1.40. Phases of the accident after primary loop rupture according to US regulations (RG 1.183)

	PWR		BWR	
Phase	**Start**	**Duration**	**Start**	**Duration**
Release from under the jacket	30 s	30 minutes	2 minutes	30 minutes
Early fuel meltdown	30 minutes	90 minutes	30 minutes	90 minutes

The fractions of fission products released during the particular phases of the accident stipulated for LOCA type accidents in a PWR reactor in the RG 1.183 are shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.1.41. Fractions of fission products released into the safety containment during the particular phases of a LOCA type accident in a PWR reactor

Accident phase	Release from under the	Early release during core	Total release

	jacket	meltdown in the reactor containment	
Inert gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkali metals	0.05	0.25	0.30
Tellurium	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Precious metals	0.00	0.0025	0.0025
Cerium	0.00	0.0005	0.0005
Lanthanum	0.00	0.0002	0.0002

The design of an EPR reactor includes a number of safety measures which are intended not only to reduce the frequency of severe accident but also to limit their impact. As a result, the values of releases from EPR reactors are limited, as stipulated in the EUR.

3.1.3.2 Nuclear power plants with AP 1000 reactors

Release values during accidents involving core meltdown in AP1000 reactors are (similarly to the EPR reactors), defined in the US nuclear regulatory authorities guidelines, RG 1.183. The quantity of activity in the core of an AP1000 reactor and the fractions of activity that, according to RG 1.183, are emitted from the core of an AP1000 reactor during an accident involving core meltdown are stipulated in the tables above.

The fractions of activity that are released from the safety containment into the environment of the nuclear power plant depend on the condition of the containment. The probability analysis for an AP1000[35] reactor considers the following containment condition categories:

3.1.3.2.1 IC (intact containment) release category.

It is assumed that leaks from the containment occur as provided for in the design, taking into account the external cooling of the molten core by the reactor containment, with control of the concentration of hydrogen in the containment and passive cooling of the containment. The most likely path for the leaks runs from the containment to the auxiliary building. To model the deposition of aerosols in the ancillary building, the release reduction coefficient equal to 1/3 was introduced. In the additional evaluation which does not include this coefficient (marked as Direct), the release values were calculated assuming that the releases occur directly into the reactor's surroundings.

3.1.3.2.2 BP (bypass) release category – safety containment bypass.

It is assumed that fission products flow from the core through the ruptured pipes in the steam generator into the secondary loop and then into the atmosphere through the safety valve which is stuck in the open position. This is a category of very large early releases outside of the safety containment (Large Early Release Frequency - LERF) for the AP1000 reactor.

3.1.3.2.3 CI release category

This category covers accidents where the safety containment is not insulated from the surroundings. As a result, the fission products released from the core to the inside of the safety containment also escape into the nuclear power plant's surroundings. CI releases contribute to the frequency of LERF releases.

3.1.3.2.4 CFE release category

This category covers releases of fission products caused by failure of the safety containment during a severe accident which may occur during core meltdown and relocation. Fission products are released into the containment and, before they settle on its internal surface, the containment is ruptured (e.g.

due to burning of hydrogen or a steam explosion) and the fission products escape into the environment. This category is also taken into consideration when calculating the frequency of LERF releases.

3.1.3.2.5 CFI release category

This category covers those releases of fission products that are due to rupture of the containment after core meltdown and relocation, within 24 hours after the core meltdown process starts, e.g. as a result of burning of hydrogen or long-time heating of the containment with the after-heat. The fission products are released into the interior of the safety containment. The atmosphere in the containment is well mixed and before containment failure the process of aerosols deposition begins.

This category contributes to large releases but it is not a process included in the LERF release category.

3.1.3.2.6 CFL release category

This category covers release of fission products into the environment after the safety containment is damaged after 24 hours as a result of increased pressure inside the containment due to the after-heat. The containment cracks due to the lack of cooling of the containment. Before the containment is damaged, the process of deposition of aerosols on surfaces inside the containment takes place. This is an example of a large, although not early, release of fission products.

3.1.3.2.7 CFV release category

This category covers releases release of fission products into the atmosphere due to ventilation of the containment after 24 hours. The operator removes some air from the containment in order to lower the pressure. This leads to large, although not early, release of radioactivity. The probability analyses assume that the frequency of such events in AP1000 reactors is equal to zero.

3.1.3.2.8 Direct release category

This category covers release of fission products directly into the environment, without deposition of aerosols in the auxiliary building.

Table 3.1.42. Fractions of fission products released from the inside of the safety containment into the environment within 24 hours after a severe accident of an AP1000 reactor for different categories of safety containment failure.

Cat.	Xe, Kr	CsI	TeO_2	SrO	MoO_2	CsOH	BaO	La_2O_3	CeO_2	Sb
IC	1.0E-3	1.2E-5	9.5E-6	1.1E-5	1.3E-5	1.1E-5	1.2E-5	1.3E-6	1.5E-6	1.3E-5
BP	1.0E-0	3.2E-1	2.5E-1	3.6E-3	4.5E-2	2.1E-1	8.9E-3	1.3E-4	8.0E-4	2.2E-1
CI	6.4E-1	4.6E-2	2.1E-2	2.0E-2	4.0E-2	1.8E-2	3.2E-2	2.4E-4	7.4E-4	2.7E-2
CFE	8.1E-1	5.7E-2	3.2E-2	3.5E-3	1.4E-2	5.5E-2	5.3E-3	6.5E-5	2.5E-4	2.3E-2
CFI	8.0E-1	3.3E-3	5.0E-3	2.2E-2	9.3E-3	3.3E-3	1.7E-2	8.3E-3	1.1E-2	7.2E-3
CFL	1.3E-3	1.2E-5	8.5E-6	1.7E-5	1.7E-5	1.1E-5	1.7E-5	8.5E-6	9.0E-6	1.7E-5
DIRECT	3.0E-3	3.6E-5	2.9E-5	3.3E-5	3.9E-5	3.3E-5	2.8E-5	3.9E-6	4.5E-6	3.9E-5

3.1.3.3 Nuclear power plants with ESBWR reactors

In the case of boiling water reactors - similarly to pressurized water reactors – a representative severe accident is considered to involve rupture of the primary loop with meltdown of a part of fuel (in accordance with the assumptions defined in the methods of evaluation of releases after reactor accidents set forth in the US RG 1.183 guidelines).

3.1.3.3.1 Rupture of primary loop inside the safety containment (LB LOCA)[36]

Table 3.1.43. Data and assumptions for analyses of accidents involving rupture of the primary loop of an ESBWR reactor inside the safety containment

I. Data and assumptions for evaluation of fission products releases	
A. Power, MWt	4,590
B. Fraction of activity in the core released during the accident	RG 1.183, Table 1
C. Chemical composition of iodine isotopes	
Iodine in molecular form, I2, %	4.85
Molecular (mostly CsI), %	95
Organic, %	0.15
D. Time until fuel meltdown starts, min.	20
E. Releases from the core	Table 15B-1
II. Data and assumptions for evaluation of radioactivity releases	
A. Primary safety containment	
Leak value, %/day	0,5
Fraction of the leak flowing into the reactor building (leak, %/day)	0.98 (0,49)
Fraction of the leak flowing into the ring around the containment (leak, %/day)	0.02 (0,01)
Volume, m^3	7,206
Rate of elimination of molecular iodine, h-1 (0-12 h)	0.92
Rate of elimination of aerosols, h-1	
0 – 0.333 h	0.0
0.333 – 0.833 h	5.0
0.833 – 2.333 h	3.0
2.333 – 3.0 h	1.0
3.0 – 4.0 h	0.8
4.0 – 5.0 h	1.0
5.0 – 6.0 h	0.6
6.0 – 7.0 h	0.4
7.0 – 9.5 h	0.2
9.5 – 12.0 h	0.1
>12.0 h	0.0
B. Reactor building	
Leak rate, %/day	50
Mixing indicator, %	40
Volume, m^3	60,500
C. Condenser data	
Free volume, m^3	6,230
Fraction of volume significant to the course of the accident, %	20
Coefficients of iodine elimination	
Molecular, %	99.5
Molecular, %	99.5
Organic, %	0
D. Data of the main cut-off valve on the steam collector, MSIV	
Leaks of the MSIV valve (total on all lines), m3/min.	0.0623
Deposition coefficients	0 (Not Credited)
III. Control room parameters	
A. Control room volume, m^3	2,460
B. Flow rate, m^3/min.	2.83
C. Unfiltered leak into the control room, m^3/min.	0.0113
D. Indicators of stay time in the control room	
0 – 1 day	1.0
1 – 4 days	0.6
4 – 30 days	0.4

3.1.3.4 Emissions during severe accidents from the types of reactors under consideration compared to the requirements in different countries

Different emission limits in the case of severe accidents have been defined in each country; in many cases instead of emission limits, limits of permissible doses to individuals or collective doses to the public were used. In order to make the requirements uniform, EU utilities companies introduced the EUR guidelines which define the limits of emissions after design basis conditions and after severe accidents (described in section 7.1.2.3.3.1).

For the purpose of comparison, data for several selected countries is provided below:

Canada

The Large Release Frequency (LRF) outside of the nuclear power plant caused by internal events (accidents) in a heavy water CANDU-9 reactor which require evacuation of inhabitants of the adjacent areas: 10^{-6}/R-Y[37].

Finland

The natural safety features of a nuclear power plant must guarantee that even in the case of core meltdown the fraction of solid fission products released into the atmosphere does not exceed 0.1% of their content in the core[38]. When applying for a permit to build a new nuclear power plant, the investor must prove that during a severe accident the release will not exceed 10 TBq of Cs-137 with the maximum permissible frequency of 10^{-6}/R-Y.

France

The likelihood that a nuclear power plant becomes a source of unacceptable radiological consequences must not exceed 10^{-6}/R-Y.

Germany

The total frequency of beyond-design condition failures must be within 10-4 to 10^{-5}/R-Y, and the frequency of accidents with large early release of radioactivity must be at least 10 times lower.

Japan

The area where, in the case of a hypothetic accident, the effective dose for an adult exceeds 250 mSv and the equivalent dose in the thyroid exceeds 1,500 mSv, is a restricted-use area.[39]. Usually, the radius of such an area, which corresponds to the restricted-use area in the Polish technology (basically with no permanent residents) is 400 m, and the radius of a low population density area is 1,000 m.

Russia

In new nuclear power plants, the likelihood of accidents with consequences exceeding the permissible limits (evacuation of a population centre of more than 100,000 inhabitants) must be less than 10-7/R-Y.[40] The release limit in severe accidents is considered to be 100 TBq of Cs-137 or 1000 TBq of I-131. In the case of smaller releases at the distance of over 25 km from a nuclear power plant, additional intervention measures are not required, with the exception of restrictions of consumption of local food products.[41]

The likelihood of death due to the presence of a nuclear plant which is less than 10-6/year is considered to be negligible.[42]

Sweden

The limit core damage frequency must be less than 10-5/R-Y and the safety containment must remain intact. Releases of radioactive substances during a severe accident, which cause contamination of the ground, such as caesium and iodine, must be less than 0.1% of their content in the core. This limit applies to nuclear power plants with thermal power equal to MWt; for larger capacity plants it must be appropriately adjusted.[43] Accident sequences of extremely low probability do not have to be considered.[44]

United Kingdom

According to the HSE definition, a severe accident is one which leads to a dose of 100 mSv at the distance of 3 km.[45] According to the paper by M. J. Lewis[46], release of 0.1% of caesium and iodine from the Sizewell B nuclear power plant (which was equal to 3,000 TBq of I-131 and 200 TBq of Cs-137) causes similar consequences. In moderate weather conditions, such a release does not lead to exceeding the 1st Emergency Reference Level (ERL) more than 3,500 m away from the nuclear plant; in bad weather conditions, the dose in the thyroid gland will exceed 1 ERL at the distance of up to 15 km.

USA

The frequency of large releases, which are defined as releases causing an effective dose in excess of 250 mSV at the distance of 800 m within 24 hours, must be less than 10-6/R-Y.[47] [48] This release corresponds approximately to the release of 0.1% of iodine and caesium content in the reactor core.

EUR

As early as during the initial formulation of assumptions to the EUR, it was assumed that the frequency of large early releases must be less than 10-6/R-Y and that sequences with very large releases, exceeding 000 TBq of I-131 and 100 TBq of Cs-137, must be even much more infrequent.[49]

The current limits set forth by the EUR assume that during a severe accident not only iodine and caesium but also other radionuclides will be released from the core. Therefore, the limits set forth in the EUR refer to a collection of radioisotopes and not only to iodine or caesium.

The probability criteria adopted in the EUR are as follows:

• cumulative frequency of core damage: less than 10^{-5}/R-Y;

• cumulative frequency of large release of radionuclides in excess of the limited impact limits: less than 10^{-6}/R-Y;

• much lower frequency of early or larger releases.

The EPR reactor meets the EUR requirements as well as the Finnish requirements which provide that release of caesium must not exceed 100 TBq. It can be concluded that reactors which will be built in Poland will need to meet similar requirements. However, the final decisions will be made taking into account both the type of reactor and the selected location, with its weather characteristics and the distance of the restricted-use boundary.

3.1.4 Emissions and radiation doses after decommissioning, during dismantling and after dismantling of a nuclear power plant

3.1.4.1 Principles of dismantling of nuclear plant

Until October 2010, 100 uranium ore mines, 80 nuclear power reactors, over 250 research reactors, and many fuel cycle installations have been decommissioned. Some of them, e.g. Maine Yankee,

were completely dismantled to a "greenfield" condition[50]. As the experiences demonstrate, most components of a nuclear power plant are not radioactive or very contaminated. Most metal recovered during the dismantling may be recycled. There are tried and tested methods and equipment for safe disassembly. Their effectiveness has been verified on projects in various parts of the world. Dismantling works proceed effectively, according to the budget and the schedule.

The area occupied by a nuclear plant is much smaller than the areas needed for other power generation sources, especially those using renewably energy sources (RES). Consequently, there is no reason for hasty disassembly and dismantling of the reactor. Despite the current tendency to dismantle nuclear facilities just a few years of their final shutdown, delaying dismantlement by 20-50 years does not lead to any negative consequences, while significantly reducing the exposure of the staff to radiation and the cost of dismantling.

When a nuclear plant ends its useful life, it must be dismantled in observance of safety and radiological protection principles.[51] According to the US practice and regulations, the owner of a nuclear power plant can choose from among the following three options:

DECON (immediate dismantling) - soon after the nuclear power plant is shut down, its equipment, structure, and elements containing radioactive contaminants are removed or decontaminated so as to be able to hand over the area for further use without continued control by the nuclear regulatory authorities (NRC).

SAFSTOR (safe storage) – the nuclear facility is kept under control to allow time for decay of radioactive substances and then is dismantled.

ENTOMB (entombment) - protection of the nuclear power plant that allows for depositing the radioactive material present at the plant for ever (without supervision). This usually involves reducing the size of the area where radioactive materials are placed and closing the facility in the so-called sarcophagus, i.e. a durable structure (made, e.g. from concrete) which will last for a sufficiently long period to eliminate any problems with the residual radioactivity. The elements contaminated with radioactivity are permanently enclosed in the area of the former nuclear power plant, in durable structural materials, and monitored until their radioactivity diminishes to a level that allows for transferring the facility to be used without supervision of the nuclear regulatory authorities.

The license holder may decide to choose a combination of the first two methods and to dismantle and decontaminate some parts of the nuclear power plant, while leaving other parts under supervision in accordance with the SAFSTOR concept. The decision may depend on factors that are not directly related to radioactive decay (e.g. availability of facilities for disposal of low-activity radioactive waste).

According to US regulations, a nuclear plant must be dismantled no later than 60 years after its decommissioning.

3.1.4.2 Dismantlement of the Maine Yankee nuclear power plant

An example of successful dismantlement of a nuclear plant is the works conducted at the Maine Yankee plant with a 900 MWe PWR reactor which had safely generated 119 TWh of electricity between 1972 and 1996. The plant, built in Wiscasset, was the largest power plant in Maine. It was finally shut down in August 1997 when continued operation became no longer economically viable. Maine Yankee was one of the US's 69 PWR reactors (the remaining reactors are BWR reactors). Its dismantlement involved removal of 105 million kilograms of waste, to include 68 million kilograms of concrete. Over 50% of the waste (approx. 60 million kilograms) was radioactive. The safety containment was a typical large nuclear power plant containment, with the volume of approx. 70,000 m^3 (enough to fit a large school gymnasium inside). The thickness of the containment at the base was

about 120 cm and decreased to approx. 60 cm at the top, with concentric layers of steel reinforcement bars. The weight of the containment was approx. 34 million kilograms.

To remove the major equipment from the containment, the workers had to use diamond blade saws. One of the easiest tasks was to remove the major nuclear equipment, such as the reactor pressure vessel and the three steam generators located in the core of the plant. The steam generators were removed as one piece. On the other hand, the reactor pressure vessel, which is a huge boiler made from carbon steel with a stainless steel lining and a metal frame holding the core and directing the water flowing around the fuel, was cut into pieces with water jets and cutting tools. The works were performed under water with remotely controlled tools. The technology to cut large metal equipment was provided by the French Framatome ANP company.

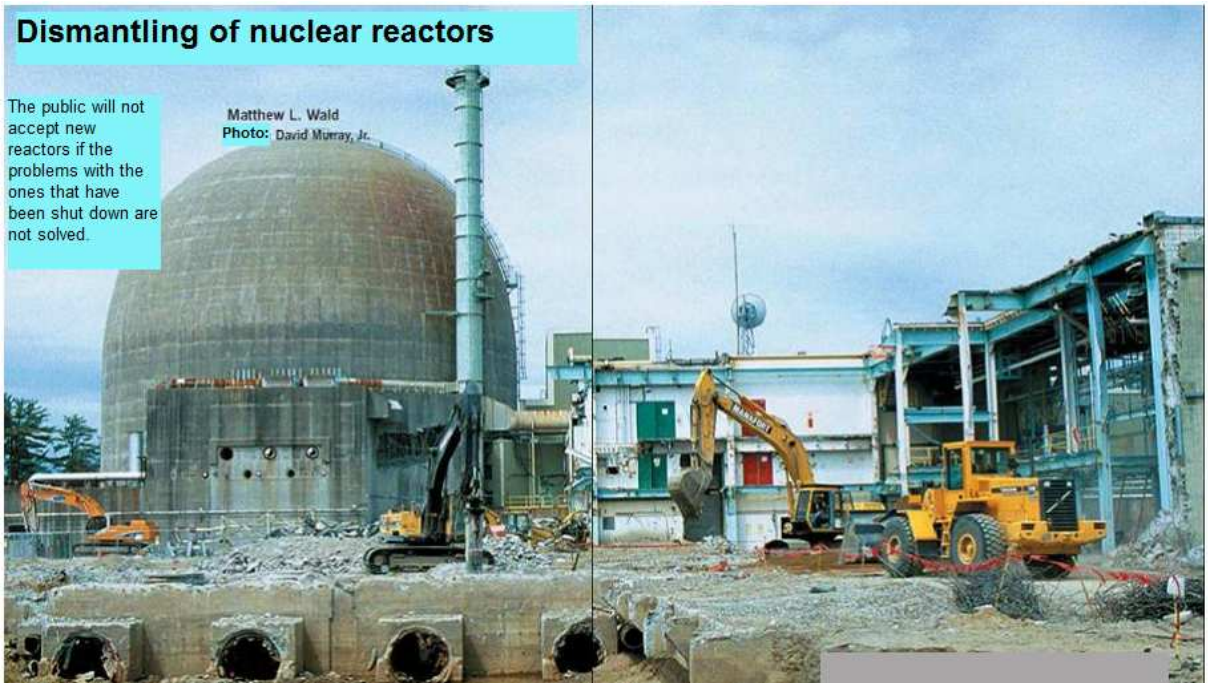

Figure 3.1.1. Demolition of the Maine Yankee nuclear power plant; figure from the work by M. Wald[52]

Figure 3.1.2. After dismantlement of the Maine Yankee nuclear power plant – a green field, additional radiation level below 0.1 mSV/year (the difference between Krakow and Wrocław is 0.37 mSv/year).

Afterwards, the reactor core was filled with cement, to reduce the possibility that the internal parts will lose stability in the coming centuries. The vessel was lifter and moved onto a barge which carried it to a low-activity radioactive waste storage site. Only containers filled with spent fuel remained at the former nuclear power plant. They were placed in a 6 acre lot. The containers form a new Independent Spent Fuel Storage Installation (ISFSI). This is one of numerous such installations in the USE. Their establishment at nuclear power plant sites is due to the delay in commissioning of a high-activity radioactive waste deposit site in Yucca Mountain.

Figure 3.1.3. Containers with spent fuel left at the former Maine Yankee site.

The residual radiation allowed by federal and state regulations is so low that the management of the nuclear power plant decided that it was necessary to measure the natural background radiation level. This will allow avoiding the need to eliminate radionuclides which were present at the site regardless of the presence of the plant.

According to the Maine Yankee dismantlement design, the maximum cumulative irradiation of workers is 11.15 person-Sv for all the works performed at the plant site. The value can be compared to the dose equal to 4.40 person-Sv a year when the last reactor fuel exchange was performed prior to the shutdown.[53]

The Maine Yankee reactor was the first large nuclear energy reactor which was fully dismantled to the so-called "green field" condition, i.e. to a condition where the site can be used for any purpose, to include food production. The works were performed according to schedule and within the budget. The site was completely decontaminated, to an activity level that is lower than that required by the NRC. A particularly important achievement during the dismantlement of the Maine Yankee plant was the outstanding cooperation of all the parties interested in reclaiming the site. The main achievements in their cooperation are:

- there were no accidents resulting in lost work time over the whole project performance period, i.e. more than 3 years;
- the dismantlement was completed with the cumulative dose received by the workers equal to 50% of the limit dose set forth by the NRC;
- the site was cleaned to the extent that the dose value is lower than the target value of 0.1 mSv/year;
- for the first time in history, explosives were used to demolish the reactor safety containment;
- the waste generated during the dismantlement of the plant was safely transported by train, truck, and on river barges;
- the project involved the largest ever campaign of transport of spent nuclear fuel from a wet store to a dry store;
- over 200 acres were handed over to be used for conservation and environmental education purposes;
- 400 acres owned by the plant were transferred to be used for business purposes.

Figure 3.1.4. Collection of soil samples from former Maine Yankee nuclear power plant site[54]

3.1.4.3 Status of dismantling of nuclear plants in the USA

Other nuclear power plants followed the example of Maine Yankee. In total, ten nuclear power plants have been dismantled in the USA, namely: Big Rock Point, Fort St. Vrain Nuclear Generating

Station, Haddam Neck - Connecticut Yankee, Shoreham Nuclear Power Station, CTVR (Pressurized Tube, Heavy Water), Pathfinder (Superheat BWR), Maine Yankee Atomic Power Station, Saxton, Trojan, and Yankee Rowe Nuclear Station. Additional nuclear power plants are in the immediate dismantling phase (DECON): Enrico Fermi, Humboldt Bay, San Onofre, and Three Mile Island 2. Moreover, 9 nuclear plants have been decommissioned and are being dismantled in accordance with the SAFESTOR concept. These are: Dresden Nuclear Power Station, Unit 1, GE VBWR (Vallecitos), Indian Point Unit 1, LaCrosse Boiling Water Reactor, Millstone Nuclear Power Station, Unit 1, N.S. Savannah, Peach Bottom Unit 1, Rancho Seco Nuclear Generating Station, Zion Nuclear Power Station, Units 1 and 2.

The overall characteristics of the status of dismantling of nuclear power plants in the USA is shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.1.44. The status of dismantlement of nuclear power plants in the USA as of 2008.

Type	Name	Location	Capacity [MWt]	Shutdown date	Status	Total dismantlement
BWR	Shoreham	Wanding River, NY	2,436	28 June 1989	immediate dismantlement	1994
	Dresden 1	Morris, IL	700	31 October 1978	safe storage	2036
	Millston 1	Waterford, CT	2,011	21 July 1998	safe storage	2056
HTG	Fort St. Vrain	Platteville, CO	842	18 August 1989	immediate dismantlement	1992
PWR	Yankee Rowe	Franklin Co, MA	600	1 October 1991	immediate dismantlement	2007
	Maine Yankee	Wiscasset, ME	2,700	6 December 1996	immediate dismantlement	2005
	Haddam Neck	Meriden, CT	1,825	5 December 1996	immediate dismantlement	2007
	Trojan	Ranier, OR	3,411	9 September 1992	immediate dismantlement	2005
	San Onfore 1	San Clemente, CA	1,347	30 November 1992	immediate dismantlement (ongoing)	2045
	Rancho Seco	Herald, CA	2,772	7 June 1989	immediate dismantlement (ongoing)	2009
	Indian Point 1	Buchanan, NY	615	31 October 1974	safe storage	2026
	Zion 1	Zion, IL	3,250	21 February 1997	safe storage	2026
	Zion 2	Zion, IL	3,250	19 September 1996	safe storage	2026
	Three Mile Island 2	Middletown, PA	2,770	28 March 1979	License in place only	2036

3.1.4.4 Experiences from dismantlement of nuclear power plants

In the last 40 years, significant experience has been gathered in dismantling of various types of nuclear installations.[55]

European reactors

When dismantling graphite-gas reactors in Chinon, Bugey, and St. Laurent, Electricite de France selected the option with partial disassembly and the final disassembly delayed by 50 years. Because

other nuclear power plants are in operation on the same lots, monitoring the decommissioned reactors does not involve significant additional costs.

In the United Kingdom, the disassembly of 25 reactors has been started. The first reactors are two Magnox type reactors at the Berkeley nuclear power plant (2 x 138 MWe) which were shut down for economic reasons in 1989, after 27 years of operation. The removal of fuel from these reactors was completed in 1992. The reactor buildings are ready for the SAFESTORE phase. In the future, they will be dismantled and the land will be levelled and developed. Similar works are ongoing in other British nuclear power plants.

In Spain, the graphite-gas reactor at the 480 MWe Vandellos-1 plant was decommissioned in 1990 after 18 years of operation due to a fire in the engine house which made repair of the plant not cost-effective. In 2003 ENDRESA completed the 2nd phase of dismantling of the reactor, which makes it possible to hand over large parts of the site for other uses. After 30 years of safe storage (SAFESTOR), when the level of activity decreases by 95%, the remaining part of the plant will also be dismantled. The cost of the works, planned to be performed over a period of 63 months, is 93 million Euros.

In Japan, the Tokai-1 reactor, based on the design of the British Magnox type reactor, is undergoing dismantlement (after 30 years of operation which ended in 1998).. After storage for 5-10 years, the unit will be disassembled and the land will be handed over for other uses. The total cost is estimated to be 25 billion yen, or approx. 250 million USD.

Germany has opted for immediate dismantling of the shut down Greifswald nuclear power plant, where 5 reactors of the WWER-440 type were in operation. Similarly, the 100 MWe nuclear power plant in Niederaichbach in Bavaria was dismantled and the site was handed over for unrestricted farm use (“green field”) in mid-1995.

The 250 MWe Gundremmingen-A boiling water reactor was the first German commercial power reactor and was in operation in the years 1966-1977. Works on dismantling of this reactor started in 1983 and the more contaminated elements were dismantled in 1990, using the underwater cutting method. The works proved that dismantling can be safe, inexpensive, and fast, and that most materials can be reused.

Reactors in the USA

Of all reactors dismantled in the USA, 14 power reactors were dismantled using the SAFESTOR method and 10 - using the DECON method. The relevant procedures were defined by the Nuclear Regulatory Commission (NRC). The USA has significant experience with dismantling of nuclear power plants. In total, 31 reactors have been shut down and dismantled in the USA. The reclaimed sites were transferred for use, while maintaining pools for temporary storage of spent fuel there. Such pools can be eliminated only when the US Department of Energy collects the spent fuel (which is officially its property) and transports it to the Yucca Mountain repository.

The Rancho Seco nuclear power plant (a 913 MWe PWR reactor) was shut down in 1989, and in 1995 the NRC approved its dismantlement in accordance with the SAFESTOR method. However, at a later date the owner of the plant opted for gradual dismantlement and currently the works on dismantling the plant are advanced.

In the case of plants with several power units, the preferred strategy is to monitor the first shut down unit until the others end their useful lives. This allows for dismantling them at the same time and for optimum use of the staff and specialized equipment needed for cutting and remote disassembly (which leads to significant reduction of the cost).

Fourteen years after its complete cleaning, the second power unit of the Three Mile Island (TMI-2), which was destroyed because of the 1972 accident, it was dismantled according to the SAFESTOR method. The TIM-2 unit will be monitored until the operation license of power unit 1 expires in 2014 so that both power units can be dismantled at the same time. The SAFESTOR method was also implemented in power unit no. 1 in San Onofre which was shut down in 1992. It was to be dismantled when the licenses expire for units 2 and 3 in 2013, but after the changes implemented by the NRC, its dismantling started earlier and is not ongoing. The DECON method was used to dismantle the 60 MWe reactor in Shippingport which was in commercial operation between 1957 and 1982. This was an showcase of safe and inexpensive dismantling of a nuclear power reactor with fast transfer of the reclaimed land to other uses. Removal of the fuel was completed after 2 years and after 5 years the land was transferred for use without any restrictions. Due to the fairly small size, the reactor vessel was removed in its entirety. In larger plants, the vessel must be cut into parts prior to shipment.

The immediate dismantling using the DECON method was also used in the case of the 330 MWe high–temperature gas reactor at Fort St. Vrain which was shut down in 1989. The dismantling was performed under a 195 million USD contract, which means that the cost was below 0.01 USD per kWh, even though the plant was in operation for only 16 years. The project was completed according to schedule and the land was transferred for use in 1997. This was the first high-capacity power reactor dismantled in the USA.

At the 1,180 MWe PWR reactor Trojan nuclear power plant in Oregon, the dismantling was performed by the power company which owned the reactor. The plant was shut down in 1993. The removed steam generators (1995) and reactor vessels, including the internal elements (1999) were transported to Hanford. The lot was transferred for unrestricted use in 1005, with the exception of the spent fuel storage pool. The cooling towers were demolished in 2006.

At the Yankee Rowe nuclear power plant, the 167 MWe PWR reactor was shut down in 1991, after 30 years of operation. The dismantling was performed in accordance with the DECON method and completed in 2006. The lot was transferred for unrestricted public use in August 2007, with the exception of 2 hectares which were kept as a storage yard for spent fuel.

Another DECON type project was the dismantling of the Maine Yankee nuclear power plant, with an 860 MWe PWR reactor which was shut down in 1996 after 24 years of operation. The safety containment was demolished in 2004 and, with the exception of 5 hectares intended for a spent fuel dry store, the site was transferred for no-restriction public use. The works were performed in accordance with the budget and the schedule. The 590 MWe PWR reactor at the Connecticut Yankee nuclear power plant was shut down in 1996, after 28 years of operation. The demolition started in 1998 and was completed in 2006.

Disassembly of the remaining power units is ongoing.

3.1.4.5 Doses during disassembly of a nuclear plant and after its dismantling

The doses during disassembly of a nuclear power plant are small. For example, the cumulative dose expected for the dismantling of the Maine Yankee plant was 5.7 person-Sv, which is two times less than the limit dose set forth in the Environmental Impact Statement. In practice, the total cumulative dose during dismantling of the Maine Yankee plant was lower and equal to only 2.7 person-Sv.[56]

The work safety is also high. For instance, during the dismantling of the Maine Yankee plant, during the more than 2 million work hours there was not a single case of work lost as a result of an accident.

Moreover, the public with no professional connections with a nuclear power plant is not exposed to significant radiation doses. For the Maine Yankee plant it was assumed that the residual radioactivity

after dismantling of the plant may not lead to a dose larger than 0.1 mSv/year through all exposure paths, to include 0.04 mSv/year through groundwater.[57] The difference in the natural radiation background value in Finland (7 mSv/year) and in Poland (2.5 mSv/year) is 4.5 mSv/year. What this means that living for one year at a former nuclear power plant site will result in the same additional dose as an 8-day trip from Poland to Finland!

In the case of nuclear reactors, approximately 99% of activity is related to the fuel, which is removed from the reactor after its final shutdown. Besides surface contamination, the remaining activity comes from activation products, such as the steel which is exposed to neutron radiation. Activation products contain such radioactive isotopes as Fe-55, Co-60, Ni-63, and C-14. The first two isotopes are highly radioactive and emit gamma radiation. However, their half life is short and after 50 years their activity is much lower. As a result their hazard to employees is reduced practically to zero.

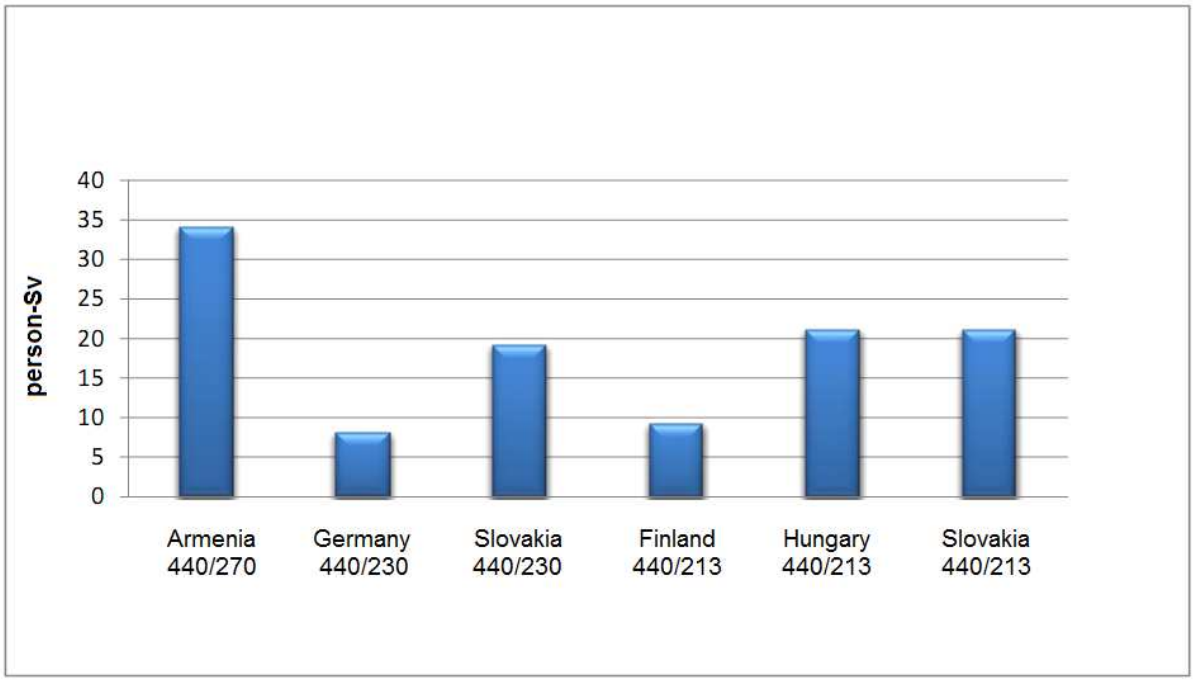

Figure 3.1.5. Expected equivalent collective doses of radioactivity in the case of immediate disassembly and dismantling of a nuclear plant with WWER reactors[58]

In the case of delayed disassembly, the doses are significantly smaller, which can be seen by comparing **Błąd! Nie można odnaleźć źródła odwołania.** and **Błąd! Nie można odnaleźć źródła odwołania.**. The lowest value has been defined by Finland; it is slightly higher than 2 person-Sv for a 20 years long period of cooling and removal of the reactor vessel, the internal elements, and the steam generators without cutting. The highest value, i.e. 21 person-Sv, was defined by Slovakia for a 50 years long period of cooling and cutting the primary loop on the disassembly site[58].

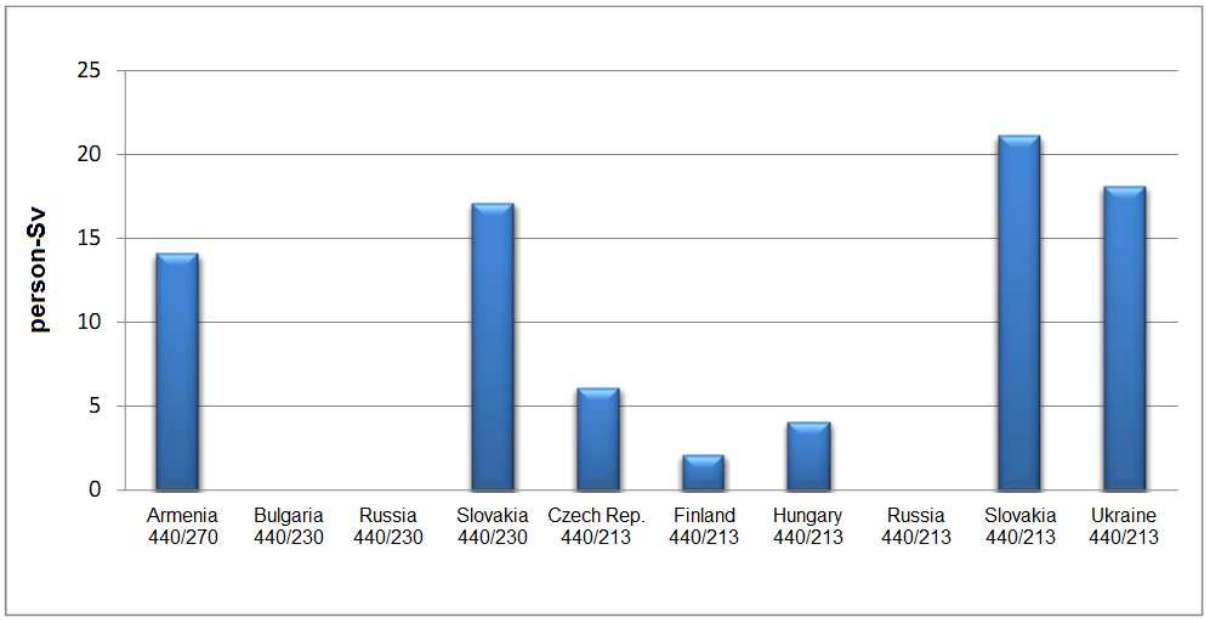

Figure 3.1.6. Expected equivalent collective doses of radioactivity in the case of delayed disassembly and dismantling of a nuclear plant with WWER reactors[5858].

In none of the dismantled nuclear power plants did emissions into the environment constitute a problem.

3.1.4.6 Emissions and contamination anticipated in the case of Generation III and III+ reactors

The radiological effects of dismantling of Generation III reactors will be evaluated using the example of an EPR reactor.[59]

3.1.4.6.1 Selection of materials

Reduction of activation of materials starts at the stage of design by taking actions to reduce the abrasion of materials and replacing materials with high content of cobalt (stellites) with materials without any of this element. Radioactive cobalt is the main cause of irradiation of workers during dismantling of a nuclear power plant. Thus, for example:

- in steam generator piper one can use alloy 690 (INCONEL) (with cobalt content below 0.018%) instead of INCONEL alloy 6000 (with cobalt content below 0.05%);
- in steel elements exposed to radiation, the content of cobalt is to be limited to 6 ppm;
- the content of silver in steel and alloys is also reduced and seals covered with silver are replaced with graphite seals (silver is a significant source of radiation for several years after plant shutdown);
- the use of antimony in seals is also reduced.

Increasing the tightness of fuel jackets significantly affects the classification of radioactive waste as it limits release of beta and alpha radiation emitters.

Hazardous materials

Reducing the use of hazardous materials is particularly important with regards to materials which may be activated, because removal of mixed waste is very difficult. This applies, in particular, to corrosive, toxic, and inflammable substances, heavy barium concrete, and combustible metals (e.g. Zircaloy) which require particular care in the process of cutting and packing, as well as to fibrous

materials. The use of porous materials must be avoided due to the difficulties with their decontamination.

3.1.4.6.2 Design requirements

The purpose of design modifications implemented in Generation III reactors is to reduce the radiation doses received by workers during dismantling of a nuclear power plant, by reducing the time of stay in the vicinity of highly active elements and by increasing the speed of their removal from the installation.

Numerous pieces of equipment, such as measurement instruments in the core, steam generations, reactor circulation pumps, stabilizers, heat exchangers, etc., are designed for easy disassembly. In the case of most such equipment located in areas that are not available due to the level of radiation, the possibility of their removal in their entirety is studied, which requires proper designing of their connections and access doors. The design of the reactor well and the core catcher makes it possible to fill the reactor well with water, which allows for the reactor to be disassembled under water. Placing the in-containment refuelling water storage tank (IRWST) below the reactor vessel allows for collecting any water leaks occurring during disassembly of the internal elements of the reactor. The thermal insulation of the primary loop is modular and easy to disassemble.

A number of operations have been designed so as to facilitate the dismantling of the plant, e.g. drainage of the fuel pool and the steam generators, movement of fuel from the reactor building to the fuel building, solid, liquid, and gaseous waste systems, ventilation, fire protection measures, drainage of chambers and floors, power supply, compressed air system and water supply system, and the related systems and circuits. Division of the systems into four parallel circuits facilitates disassembling one circuit after the other, while maintaining the supply of the ancillary systems required in the fuel building and the auxiliary systems building. A number of improvements facilitate disassembly of large elements. An analysis of their disassembly options was performed as early as during the design phase. Improvements were also implemented to facilitate staff access, as well as shade and dismountable screens which limit the exposure of people during dismantling works. Other improvements limit the spread of contamination in the plant's systems.

Overall, one can expect that dismantling of Generation III reactors will be easier and cheaper that the ongoing dismantling of Generation III reactors. As far as environmental impact is concerned, event the present methods of dismantling of nuclear plants produce very good results; the continued progress in limiting radioactive contamination and the quantity of activated material indicates that the impact will be reduced even further in the future. In conclusion, neither emissions nor large dosed during and after dismantling of nuclear power plants constitute a hazard to the environment and the public.

3.2 Evaluation of direct and indirect radiological hazard paths in emergency situations

3.2.1 Direct radiation

The main source of radiation during operation of a nuclear reactor is the reactor core, where the fission reactions take place. During every fission reaction, parts of the fission are emitted: two or three neutrons, as well as gamma and beta radiation. After radioactive decay of fission parts, further emission of gamma, beta, and alpha occurs. This is the so-called direct radiation. In order to protect the nuclear power plant's staff, the reactor core is surrounded with huge screens which fully protect people inside the plant from such radiation.

Another source of radiation hazard is radioactive substances that leak in small quantities into the primary loop through microscopic openings in the fuel element jackets and are produced in the coolant as a result of activation of oxygen and pollutants (such as corrosion and erosion products)

flowing with the coolant through the core. Therefore, the whole primary loop in a pressurized water reactor (PWR) is enclosed in a safety containment which constitutes another protection screen.

The safety containment also provides protection against direct radiation in the case of an accident (event the least likely accidents involving primary loop rupture and core failure). In the case of an accident in a PWR reactor, radioactive substances can escape from the core and the reactor's primary loop into the safety containment. This is why the thickness of the screening wall in the safety containment is selected so as to stop direct radiation from all fission products that can be released from the core. The design of the safety containment, to include the structural elements and the equipment located inside the containment, is calculated – with appropriate margins – for accident parameters (pressure and temperature) and for impacts and loads that may occur during an accident, as well as for loads from external occurrences and events (to include appropriate combinations of loads). Also, the containment must be appropriately leak-tight at maximum accident pressure (leaks from Generation III reactor containments may not exceed 0.25-0.30% of containment volume per day). In order to verify that these requirements are met, pressure tests and leak tests are performed (during the commissioning and then periodically during the operation). The hazard related to direct radiation outside of the containment is small and drops fast, in proportion to the distance from the containment.

The effectiveness of safety containments was confirmed by the analysis of the consequences of the Three Mile Island accident. This was the only accident during over 10,000 years of total operation of PWR and BWR reactors where core meltdown occurred and where fission products escaped into the containment. In spite of this, the proportion of direct radiation in the total exposure of the employees and the public was negligibly low.

The main source of hazard during an accident in a reactor is the radioactive substances that escape from the nuclear power plant and are carried in the air and in the water, and then are inhaled or swallowed with food by people.

3.2.2 **Hazard from leaks of radioactive substances from a nuclear power plant**

3.2.2.1 Release of radioactive isotopes into the atmosphere

A radioactive cloud affects people directly with beta and gamma radiation and exposes them to intake of isotopes, mostly through the lungs, but also through the skin. A radioactive cloud also causes formation of radioactive deposits on the soil and on plants. Radioactive isotopes are absorbed by plants and, once the plants are harvested, they are consumed by people with food. Releases during a nuclear power plant accident include:

- direct radiation from the cloud, inhalation, and contamination of skin caused by isotopes contained in the radioactive cloud;
- fallout of radioactive substances from the cloud onto open water reservoirs;
- fallout of radionuclides onto the ground and their seeping into the ground water, animal fodder, and edible plants.

Radioactive contamination of water is hazardous to people who are in water or on the bank and, most importantly, radioactive isotopes are absorbed by organisms living in water and are consumed by people in fish and other food from water plants and animals. Moreover, water used for irrigation of fields and for drinking contaminates crops and milk and, if drunk by people, leads to radionuclides depositing in their bodies.

3.2.2.2 Release of radioactive isotopes in liquid waste

In the case of a leak of radioactive liquid waste, the ground water becomes contaminated in a similar fashion as in the case of fallout from a radioactive cloud. The likelihood of a leak during normal operation is low, but it is considered in the case of a severe accident involving core meltdown which leads to the risk of melting through the base of the safety containment and of radioactive substances leaking into the ground layers beneath the containment. In order to prevent such an occurrence, Generation III reactors are provided with appropriate systems which protect the whole containment, to include its foundation slab.

Even though during an accident radioactive substances may be released into the water and the soil, the most likely risk is connected with releases into the atmosphere. After such a release, the public can be exposed to direct radiation from the radioactive cloud and by inhaling the radioactive dust from the cloud. As the cloud disperses, radioactive particles settle on the ground or are quickly washed by rain or snow. The public may then be exposed to radiation from radioactive deposits on the surface of the ground, to inhaling dusts which is lifted off the ground and suspended in the air, and to consuming contaminated food or water. The degree if such a hazard largely depends on the process of dispersion of radionuclides in the atmosphere.

3.2.3 Phenomena taken into account when calculating plume dispersion

The mechanism of dispersion of the radioactive plume in the air – which is considered by the NCR as one of the main safety features of a nuclear power plant location – was described very early, in the first regulatory guidelines, namely RG 1.3 and 1.4.[60] The process of atmospheric dispersion is dealt in these guidelines with a wide safety margin. However, in 1979, after obtaining new experimental data, the NCR issued the RG 1.145 guidelines which introduced significant changes which reduce the overly pessimistic assumptions made in the previous guidelines, i.e. RG 1.3 and RG 1.4. The RG 1.145 guidelines take into accounts meanders of the plume, relation between the dispersion and the direction of wind and the distribution of frequency of wind directions in a given location.[61]

The meteorology data needed to determine the atmospheric dilution factor χ/Q includes the speed and the direction of wind and the class of atmospheric stability. The data should be based on hourly averages collected over the period of a whole year. Wind direction is divided into 16 sections of the wind rose, 22.5 degree each. Atmospheric stability is determined based on the difference in temperatures in the vertical direction between the plume release height and the 10 m level. At small wind speed, the speed of mixing of atmosphere layers depends mostly on the ration of the vertical temperature gradient to the gradient corresponding to the adiabatic cooling of the air with its decompression as the height increases.

For dry air, the adiabatic temperature gradient is approx. -0.65^{o}C/100 m. When the actual gradient is higher (as shown on **Błąd! Nie można odnaleźć źródła odwołania.** a), the difference between the density of gases released into the atmosphere and the density of the air increases with increasing height, which in turns increases the uplift pressure of the gases. The mechanism is analogous in the other direction: when gases fall down, the adiabatic compression does not lead to the gasses heating up fast enough to reach the temperature of the surrounding air, which leads to an increase in the speed of their downward movement.

Thus, if the temperature gradient has a large negative value, all vertical movements of gases are accelerated, which leads to large turbulences in the plume (**Błąd! Nie można odnaleźć źródła odwołania.** a). When the value of the temperature gradient is reduced, the plume becomes more stable (**Błąd! Nie można odnaleźć źródła odwołania.** b). If the gradient has a positive value, the plume can travel over ten kilometers without any significant dilution (**Błąd! Nie można odnaleźć źródła odwołania.** c). A changing temperature gradient leads to the creation of either a rising plume (**Błąd! Nie można odnaleźć źródła odwołania.** d) - which is the most advantageous from the point of view of elimination of radioactive waste – or a fumigating plume (**Błąd! Nie można odnaleźć**

źródła odwołania. e) – which is the most dangerous because it causes contamination of the ground surface at large distances from the source (even though radioactive gases are emitted at large heights).

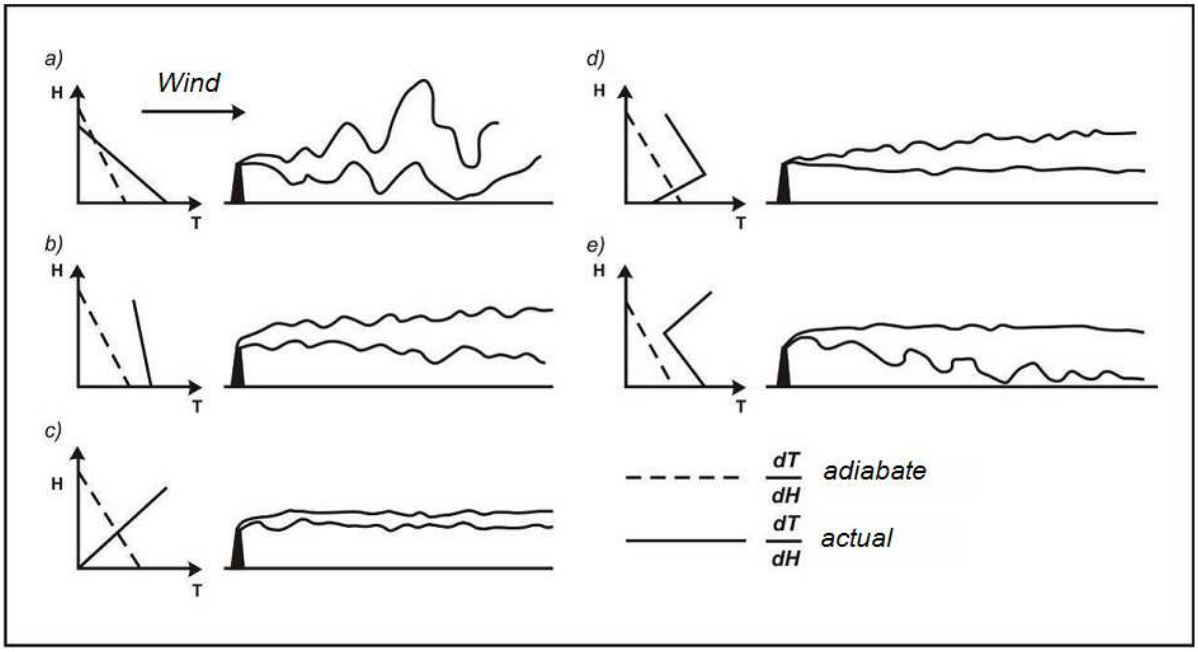

Figure 3.2.1. Impact of the vertical temperature gradient on the behaviour of a gas plume

The temperatures of atmosphere layers change in a daily cycle as the Earth absorbs the solar radiation heat during the day and warms up the lower layers of the atmosphere during the night. During the day, the plume is often unstable (looping), and during the night the surface of the Earth cools down faster than the atmosphere, which leads to plume inversion.

For the purpose of calculation, the atmosphere stability conditions are divided into categories, which are shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.2.1. Relation between stability of the atmosphere and the weather[62]

Wind speed, m/s	**Insolation during the day**			**Night conditions, cloud cover**	
	high	moderate	sparse	dense	sparse
2	A	A-B	B		
3	A-B	B	C	E	F
4	B	B-C	C	D	E
5	C	C-D	D	D	D
6	C	D	D	D	D
A - high instability			D - neutral conditions		
B - moderate instability			E - low stability		
C - low instability			F - moderate stability		

For each of the categories, the parameters which describe the concentration of radioactive isotopes were defined, namely σy and σz, i.e. horizontal and vertical standard deviation of the distribution of the radioactive cloud density depending on the distance from the source of emission.

Assuming that gaseous fission products are released from a nuclear power plant in a uniform fashion over a longer period of time at the rate of Q (Bq/s), one can determine the concentration of radioactive substances in the cloud χ (Bq/m^3) using the Sagendorf formula[63].

$$\frac{\chi}{Q} = 2{,}032\sum_{ij}\{\frac{n_{ij}}{Nxu_i\sum_{zj}(x)}\exp(\frac{-h_e^2}{2\sigma^2_{zj(x)}})\}$$

where:

h_e - effective height of the emission point [m],

n_{ij} - time [h], during which a given wind direction, with speed i and atmosphere stability category j, was observed,

N – total observation time [h],

u - average wind speed at the height h_e [m/s],

x - distance from the emission point [m],

$\Sigma_{zj}(x)$ - elevation of the cloud with correction reflecting the impact of buildings, used only for ground level emissions [m]. In the case of emissions from a stack, $\Sigma_{zj}(x) = \sigma z,j$.

In the case of emission points situated higher than twice the height of the neighbouring buildings, the effective emission height he can be calculated using the following formula:

$h_e = h_{stack} + h_{elev} - h_{gr} - c$

where:

h_{elev} - elevation of the cloud above the emission point, depending on the outlet speed w_o, the temperature, and the stack diameter d [m],

h_{gr} - maximum elevation of the ground in relation to the base of the stack [m],

c - correction to take into account low outlet speed [m],

w_o -s vertical outlet speed of the plume [m/s][64].

In the case of accidents involving emission points situated less than 2.5 times the height of the neighboring buildings Dz, it is assumed that the source is situated on the ground level. Consequently, the effect of turbulences caused by buildings must be reflected.

3.2.4 **Calculation of the atmospheric dilution factor**

The RG 1.145 guideline requires calculating the atmospheric dilution coefficient **χ/Q** at the boundary between the restricted-use area around the nuclear power plant for 2 hours after the accident.

Releases through discontinuities in the containment include all releases taking place at heights lower than 2.5 times the height of the neighboring buildings. In the case of atmospheric conditions in the neutral D or stable E, F, and G class, when the wind speed is less than 6 m/s, one can consider horizontal meandering of the plume. The **χ/Q** value can be determined by selectively using a set of

formulas describing the atmospheric dilution factor at the ground level on the centreline of the radioactive plume:

3.2.4.1 The case of fumigation

In the case of sites located 3,200 m or more away from large water reservoirs (a sea, an ocean), it must be assumed that at the time of the accident the fumigation conditions are present for a period of 0.5 h. If the χ/Q value with fumigation is higher than that without fumigation, then in the time between 0 h and 0.5 h the value with fumigation must be used. Then, in the period between 0.5 h and 2 h, the χ/Q value without fumigation must be used.

3.2.4.2 Determination of the χ/Q value for a given location

The values of χ/Q which are exceeded for no more than 5% of the total time around the excluded zone can be determined in the following manner:

One must determine the total cumulative distribution of probability of certain values of χ/Q. Then, one must make a graph of the function of the probability that a certain value of χ/Q will be exceeded. On the curve, one must select the value of χ/Q which is exceeded for 5% of the time.

The value of χ/Q used in the calculation is assumed to be the higher of the following values: the maximum value of χ/Q for the sector or 5% of the value of χ/Q for the location.

3.2.5 Methods of calculating atmospheric dispersion of radioactive substances released during continuous operation of a nuclear power plant.

Calculation of the hazard related to emissions into the atmosphere during normal operation of a nuclear power plant can be done in a similar fashion to calculation for accident conditions. However, it must take into account appropriate atmospheric dispersion factors, calculated for long periods and not for several or several dozen hours (as in the case of calculation of hazards after an accident). The value of atmospheric dilution factors for exposures over the course of a year is much lower due to changes in the wind direction. Moreover, releases of radioactivity occur at the stack level, which means that the exposure at the ground level in the vicinity of the plant is lower.

Various dispersion models are used in the calculations. In models with stable average wind direction, it is assumed that the stable wind lifts and disperses the radioactive plume in the whole area in question in accordance with the wind direction at the emission point. The most popular is the model with a straight-line trajectory, with Gauss distribution, where one assumes that the wind speed and the atmospheric stability category at the emission point are characteristic of the atmospheric conditions along the whole path of the radioactive plume.

3.2.5.1 Emission types

At short distances from the nuclear power plant, the yearly concentration of radioactive substances at the ground level are strongly dependent on the type of emission. The longer the distance from the emitter, the less the values are dependent on this factor.

In typical nuclear power plants, release of gaseous radioactive substances from high stacks cause the maximum concentration at the ground level at the distance of 1-3 km, whereas releases at the ground level usually cause concentrations of radioactivity which decrease monotonically as the distance from the plant increases. In some conditions, the radioactive plume may be pulled by a turbulence in the aerodynamic shade of a building and may fall to the ground at a short distance from the plant. In other conditions, the plume may be elevated above the emission point.

Methods have been developed to evaluate the effective emission height to calculate the concentration of radioactive substances for all distances along the path of the wind. Important parameters include the initial emission height, the location of the emission point in relation to the

obstacles in the air flow, the dimensions and the shape of the emission site, the initial speed of the plume in the vertical direction, the heat content of the leaking gases, the ambient temperature, the wind speed, and the atmospheric stability class.

If the leak is pulled into the aerodynamic wake of a building, it is usually assumed that the leak is mixed in the air turbulence. The mixing zone can be a plume with the initial cross-section equal to a half or more of the cross-section of the building.

3.2.5.2 Process of removal of radioactive substances from the plume

As the leak moves from the emission point along the wind direction, a number of processes take place, which cause a reduction of the concentration of radioactive substances below the concentration corresponding to the very process of atmospheric diffusion. The processes include radioactive decay, dry deposition, and wet deposition.

Radioactive decay depends on the half-life and the time of movement of the radioactive substances. In the case of short-lived radionuclides, the time of the plume's flow from the emission point to the receptor has a large effect on reduction of the radiation hazard. In the case of long-lived substances, the time of movement is practically insignificant.

All substances settle through sorption on the surface of the ground, but the speed of dry transfer for inert gases, tritium, carbon-14, and organic iodine compounds is so low that their removal from the plume is negligible within the radius of 80 km from the emission point. Molecular iodine and substances which have the form of aerosols settle on the ground much faster. The process of transfer of radioactive substances to the dry surface of the ground can be described with the following formula:

concentration of radionuclides in the plume [Bq/m^3] x speed of transfer to the ground [m/s] = speed of settling [Bq/m^2s].

The speed of transfer is in direct proportion to the wind speed. Consequently, the settling speed is independent of the wind speed, since the concentration of radionuclides in the radioactive plume is inversely proportional to the wind speed.

Dry deposition is a continuous process. On the other hand, wet deposition occurs only during precipitation. Nevertheless, dry deposition is not as effective as wet deposition. In most locations, precipitation occurs only for few hours a year. Consequently, despite the higher efficiency of wet deposition, calculation of doses for long periods which take into account only dry deposition are not very different from calculations taking into account wet deposition. Still, wet deposition may be an important factor in calculation of doses caused by emissions from a stack in locations where the rainy season corresponds to the local cattle grazing season.[65]

The model with a constant wind direction does not describe the effects of time and space changes in the flow of air in the area in question. Unlike the model with variable wind trajectory, the model with constant wind direction can be based only on the meteorological data obtained from one meteorological station.

Usually, the analyzed area is located within 80 km from the plant. Therefore, when using the constant wind direction model, one must study the characteristics of the air flow and check if the measurements at the given location are representative for the conditions present between the plant and the nearest receptors (usually within 8 km) and for the conditions within 80 km from the plant.[66]

3.2.5.3 Dry deposition

Radioactive materials are divided into four groups according to their deposition speeds. These are:

- Inert gases (Kr, Xe) – do not undergo either dry or wet deposition;
- Molecular iodine (I_2 vapour in the air);
- Organic iodine, e.g. methyl iodide CH_3I;
- Aerosols (radionuclides in the form of aerosols or deposited on aerosols, e.g. aerosol iodine or metal oxides).[67]

Contamination of the surface beneath a radioactive cloud can be characterised using the deposition speed vd, defined as the ratio of the speed of deposition of contaminants on the surface dCd/dt to the concentration of radionuclides in the surrounding air χ. The speed of deposition on a standard surface of pastures, at the wind speed at the height of 10 m equal to U_{10}= 4 m/s is stipulated in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.2.2. Speed of deposition of radionuclides (m/s) on the surface of pastures at the wind speed of 4 m/s' data from the RODOC code.

Type of surface	Molecular iodine	Organic iodine	Aerosols
Pastures	$8*10^{-3}$	$0.1*10^{-3}$	$1.0*10^{-3}$

3.2.5.4 Wet deposition

Assuming that elimination of radioactive substances by rain or snow proceeds uniformly in the whole height of the radioactive cloud, the rate of elimination at any distance from the source of emission is proportional to the quantity of material reaching that distance.[68] Concentration in the air can be determined by using the modified size of the source in the following formula:

$$Q = Qo \exp(-\Lambda t)$$

where:

t - precipitation time [s];

Λ - sweeping coefficient [s^{-1}] in proportion to rain intensity I [mm/h]; Λ = α*I, whereas α depends on the characteristics of the material carried in the cloud, e.g. on the aerodynamic diameter of particles and solubility of the gas in water.

According to the methods provided for in the universally recognized RODOS standard,

$$\Lambda\ (1/s) = \alpha\ (\ I/\ 1\ mm/h)^b$$

whereas the values α and b for various radionuclides are listed in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.2.3. Parameters α and b used to determine wet deposition speed

Group of radionuclides	α	b
Inert gases	0	0
Aerosols	8 E-5	0.8
Molecular iodine	8 E-5	0.6
Organic iodine	8 E-7	0.6

In the event of rainfall, the quantity of material remaining in the cloud is

$Q = Qo \exp(-t\Lambda) = Qo \exp(-\Lambda x/u)$

where:

- x – distance from the emission point [m];
- u - wind speed [m/s].

It is assumed that sweeping occurs uniformly from the whole height of the cloud.

3.2.5.5 Impact of surface roughness

Table 3.2.4. Typical roughness values assumed in calculation codes

Surface type	Surface roughness, (M)
City buildings	1.0 - 3.0
Coniferous forest	1.3
Arable land (summer)	0.2
Arable land (winter)	0.1
Meadows (summer)	0.1
Meadows (winter)	0.001
Water	0.0001

3.2.6 Parameters of atmospheric dispersion for typical locations in Poland

Table 3.2.5. Frequency of atmosphere stability classes in the lowland parts of Poland

Stability class	Frequency	Most disadvantageous speed [m/s]
A convection	0.006	1
B	0.06	2
C	0.17	5
D neutral	0.6	5
E stable	0.07	3
F	0.08	2
G very stable	0.014	1

According to the aforementioned recommendations set forth in the NRC guidelines, for the purpose of calculating the dispersion in a given location, it is assumed that the weather conditions within the 95% envelope are present in the whole lot. Therefore, for the location as a whole, we will use the F class and wind speed equal to 2 m/s, which results in a 98.6% weather conditions envelope.

The following assumptions can be made in calculations for a typical location.

Source geometry:	point
Class of atmospheric stability	F
Wind speed at the height of 10 m	2 m/s
Type of terrain	flat
Terrain roughness	pastures

σy coefficients	acc. to the RG 1.145[69]
Emission elevation	ground level
Reactor building	50 m (wide) x 60 m (high)
Precipitation	none
Emission time	2 hours
Critical group	adults

The atmospheric dispersion coefficient for the distance of 1,000 m from the reactor, based on the calculation method recommended in the RG 1.145 guidelines for the above-mentioned parameters is:

$\chi/Q = 9{,}57*10^{-5}$ s/m3;

the coefficient for the distance of 500 m is:

$\chi/Q = 2{,}49*10^{-4}$ s/m^3.

If the value of the wind speed is assumed to be 1 m/s, the atmospheric dispersion coefficient values will be two times larger.

To demonstrate the effect of the distance, **Błąd! Nie można odnaleźć źródła odwołania.** shows the atmospheric dispersion coefficients for the distances of 500, 914, and 2,500 m for short releases lasting for one hour, for longer releases lasting 24 hours, and for long-term releases, measured for the Darlington nuclear power plant in the years 1997-2000. Apparently, the nature of the changes in the coefficient at different distances is similar to that shown above based on the formulas recommended by the NRC.

Table 3.2.6. Example atmospheric dispersion data for the Darlington nuclear power plant[70]

Time of release	χ/Q for 500 m (s/m^3)	χ/Q for 914 m (s/m^3)	χ/Q for 2500 m (s/m^3)
Short, 1 h	164 x 10^{-6}	83 x 10^{-6}	23 x 10^{-6}
Medium, 24 h	12.7 x 10^{-6}	6.09 x 10^{-6}	1.60 x 10^{-6}
Long	6.41 x 10^{-6}	2.48 x 10^{-6}	0.493 x 10^{-6}

3.3 Impact of radioactive emissions from a nuclear power plant during normal operation

3.3.1 Nuclear power plants with EPR reactors

3.3.1.1 Assumptions

The first stage of radiological assessments involves determination of the dose for the critical group, i.e. the group of people who are likely to receive the highest radiation exposure. For the various radionuclides, the dose per limit release (DPUR) coefficient [μSv*y^{-1}/Bq*y-1] is used, taking into account the location and the exposure chain through the air or water path. Four group of people are considered, i.e. embryo, baby, child, and adult. In order to calculate the coefficient – regardless of the location – designers of the EPR reactor use data from the Federal Guidance Report 12. To calculate exposure through inhaling and consuming food, the effective dose coefficient for absorption of radionuclides is taken from the EC BSS Council Directive 96/29/EURATOM. The dose

coefficients used to evaluate the effective dose for an embryo, after absorption of radionuclides by the mother, were taken from the ICRP publ. 88.

For all release scenarios defined in this methodology, the DPUR is calculated for each radionuclide, each exposure path, and each age group, in accordance with the following formula:

DPUR p,r,a = CPURr x Hp,a x DFr,a

where:

- *DPUR p,r,a* is the dose per unit of release for a given exposure path, radionuclide, and age group [µSv*y^{-1}/Bq*y^{-1}];
- *CPURr* is the concentration of activity per unit of release in the given material (Bq*kg^{-1} (l^{-1} or m^{-3}) / Bq*y^{-1});
- *Hp,a* is the parameter describing the coefficients pertaining to the exposure route for a given age group, i.e. the rate of breathing [m^3/y] or the rate of food consumption [kg/y], or the time of stay in a given location [h/y];
- *DFr,a* is the dose per unit of consumption or absorption by inhalation [µSv/Bq], or the coefficient of exposure to external radiation [µSv*h^{-1} / Bq*kg^{-1}] for each radionuclide and each age group.

The total value of the DPUR for all exposure paths present in the given case is calculated for each age group. For each radionuclide, the DPUR value is compared for different age groups and the worst value is selected. Therefore, in the first approximation, the selected doses are doses for the most exposed group of embryos, babies, children, or adults.

3.3.1.2 The yearly dose of direct radiation of the most exposed person in the critical group

In this case, no measurement data is available and the direct radiation dose was calculated.

Because the outdoor limit of the dose received by the public is 1 mSv/year, one can calculate the dose value at the boundary of the restricted-use area by using the 1/r ratio.

Direct radiation $DR = D \cdot (SFi \cdot FTi \cdot SFo \cdot FTo)$

where:

- D - dose [mSv/y] at the distance of 1 m from the surface of the building;
- SFi,o - shielding factor indoors and outdoors;
- FTi,o - fraction of time spent indoors and outdoors;
- r - distance of the critical group from the reactor.

For a person who lives 100 m from the reactor, the DR dose is:

$$DirectRadiation = \frac{1}{100} \cdot 1 \cdot (0.1 \cdot 0.5 + 1 \cdot 0.5) = 5.5 \mu Sv / y$$

In the opinion presented in Chapter 11 of the environmental impact documentation for the UK EPR reactor[71], it is assumed that all releases of radioactive materials occur in a continuous manner, with the same intensity throughout the year, and that they continue for 50 years.[72] The maximum gaseous and liquid releases from an EPR power unit are shown in **Błąd! Nie można odnaleźć źródła**

odwołania. and **Błąd! Nie można odnaleźć źródła odwołania.**, while the spectrum used to evaluate the isotope content of radionuclides is shown in **Błąd! Nie można odnaleźć źródła odwołania.** i **Błąd! Nie można odnaleźć źródła odwołania.**.

Making a number of simplifications resulting in pessimistic results, the designers of the EPR reactor calculated the doses for the most exposed family equal, as shown in the table below, to approx. 63 mSv a year. The designers referred to this stage of evaluation as stage 2. However, once more precise data was used, it turned out that the calculated values were much too high. More accurate evaluations, referred to as stage 3, are shown below in sections 7.3.1.3 – 7.3.1.11.

Table 3.3.1. Total dose for the critical group caused by an EPR rector and received through different exposure paths (µSv.y$_{-1}$), calculated based on simplified data which lead to excessively high results

Exposure path	Dose
Air, family living close to a nuclear power plant	11.4
Family of a fisherman living at the shore	46.1
Family living close to a nuclear power plant, exposed to direct radiation	5.5
Total dose for the critical group [µSv/y]	63.0

3.3.1.3 Location characteristics

A set of location characteristics needed to conduct the assessment of the impact of a nuclear plant on the environment is shown in **Błąd! Nie można odnaleźć źródła odwołania.**. The characteristics are selected so as to assure the possibility of locating an EPR reactor in many areas, in particular taking into account the typical geographical conditions and the impact of the plant in both coastal and inland regions. The parameters determining the dispersion of gaseous releases and liquid releases are presented below.

3.3.1.3.1 Parameters of gaseous releases

The receiving points of radiation impact on people and on food are located at the distance of 500 m from the emission point from the reactor. The effective height of the ventilation stack is assumed to be equal to 20 m. This is due to the conservative approach to the analysis of atmospheric dispersion parameters. The height of the EPR reactor building is approx. 60 m, and the ventilation stack extends several meters above the building. Due to air swirling in the wake of the building, the effective height of emission from the stack is lower. In the safety analysis for an EPR reactor[71], safe stack height is assumed to be 1/3 of the height of the building (hence the height of 20 meters).

The predominant weather category is assumed to be the D class of atmospheric stability which is characteristic of coastal locations. The standard sweeping and deposition coefficients and roughness of the surface typical of farming regions were selected in accordance with the guidelines set forth in the RP 72 document.

3.3.1.3.2 Liquid releases dispersion parameters

Local waters, referred to as local area, are defined based on the most restrictive values of each parameters for each of the potential locations. The smallest volume occurs together with the largest depth, the longest shoreline, the smallest mass exchange coefficient, the smallest deposit load, and the highest deposition speed.

Table 3.3.2. Location parameters of a nuclear power plant with an EPR reactor

Location characteristics	Parameter value
Receiving point for releases into the atmosphere [m]	500
Receiving point for food products [m]	500
Location boundary [m]	100
Category of wind stability according to the Pasquille scheme	70 % D
Deposition speed [m/s]	$1*10^{-3}$, $1*10^{-2}$ (I), 0 (inert gases)

Sweeping coefficient [s^{-1}]	$1*10^{-4}$
Surface roughness [m]	0.3
Local area depth [m^3]	$3*10^8$
Local area depth [m]	20
Length of local area shoreline [m]	$3*10^4$
Rate of volume exchange for local area [m^3/y]	$1.1*10^{10}$
Local area load with suspended deposits [Mg/m^3]	$5*10^{-6}$
Rate of deposition in the local area [Mg/m^2/y]	$1*10^{-2}$
Density of deposits in the local area [Mg/m^3]	2.6
Rate of bioturbation in the local area [m^2/y]	$3.6*10^{-5}$
Rate of diffusion in the local area [m^2/y]	$3.15*10^{-2}$

3.3.1.4 Yearly doses from gaseous releases from an EPR reactor for the most exposed member of the public

3.3.1.4.1 Critical group

It was assumed that the critical group, i.e. the group most likely to receive the highest exposure, is a farmer's family living 500 m from the point of release from the nuclear power plant. It was assumed that adults and children spend a significant part of their time outdoors.

Table 3.3.3. Data for evaluation of the consequences of gaseous releases from an EPR reactor

Parameter	Adult	Child	Baby
Shielding factor[73] for gamma radiation from the cloud	0.2	0.2	0.2
Shielding factor for gamma radiation from the deposits	0.1	0.1	0.1
Time during the year [h/y]	8,760	8,760	8,760
Fraction of time spent indoors	0.5	0.8	0.9
Speed of breathing [m^3/h]	1.12	0.64	0.22

The exposure paths considered are:

- Internal irradiation caused by inhalation of radionuclides located in the radioactive cloud and inhalation of nuclides which settled on the ground and were lifted up again;
- External irradiation from radionuclides located in the radioactive cloud;
- External irradiation from radionuclides settled on the surface of the ground;
- Internal irradiation caused by consumption of food containing radionuclides deposited on the ground.

3.3.1.4.2 Consumption of food

A farmer's family consumes two types of food with the highest contamination level in the largest possible quantities and all other land-based foods in accordance with the NRPB-W41. It is assumed that the two food products are made 100% locally and that all other food products are 50% made locally and 50% imported from other regions which are not contaminated.

It turns out that the largest share in the dose comes from consumption of cow milk and vegetables. Therefore, these two types of food are considered as the "top two" most contaminated and most often consumed types of food products.

Table 3.3.4. Rate of consumption of food products exposed to air-path radioactive contamination from an EPR reactor

Parameter	Adult	Child	Baby

Fraction of food produced locally – "Top 2"	1	1	1
Fraction of food produced locally – all other food	0.5	0.5	0.5
Consumption of green vegetables [kg/person/year] – average values	35	15	5
Consumption of root vegetables [kg/person/year] – average values	60	50	15
Consumption of fruit [kg/person/year] – average values	20	15	9
Consumption of mutton [kg/person/year] – average values	8	4	0.8
Consumption of offal [kg/person/year] – average values	5.5	3	1
Consumption of beef [kg/person/year] – average values	15	15	3
Consumption of milk [kg/person/year] – average values	95	110	130
Consumption of dairy products [kg/person/year] – average values	20	15	15
Consumption of green vegetables [kg/person/year] – critical values	80	35	15
Consumption of root vegetables [kg/person/year] – critical values	130	95	45
Consumption of fruit [kg/person/year] – critical values	75	50	35
Consumption of mutton [kg/person/year] – critical values	25	10	3
Consumption of offal [kg/person/year] – critical values	20	10	5.5
Consumption of beef [kg/person/year] – critical values	45	30	10
Consumption of milk [kg/person/year] – critical values	240	240	320

3.3.1.4.3 Yearly dose from air-path exposure for the most exposed person in the critical group

The consumption of grains is not analyzed because there is no information that the grains are grown, milled, and consumed locally. Dairy products are not considered to be the "top two" products because they include cheese, which usually is not produced locally, and milk beverages which are included in the cow and sheep milk.

Table 3.3.5. Yearly dose for the most exposed persons from a farmer's family due to releases into the atmosphere from an EPR reactor

	Dose from breathing [µSv/y]	Dose from land-based food consumption [µSv/y]	Dose from external exposure from the cloud [µSv/y]	Dose from external exposure from the cloud [µSv/y]	Total dose [µSv/y]
Adult	2.4E-01	3.6E+00	4.7E-02	3.9E-02	4.0E+00
Child	1.9E-01	4.2E+00	3.0E-02	2.0E-02	4.4E+00
Baby	1.3E-01	7.7E+00	2.4E-02	1.3E-02	7.8E+00

In the case of a farmer's family, the total dose for the different age groups is, respectively, 3.9, 4.4, and 7.9 µSv/year. The largest share of the dose comes from food, which contains carbon-14 (milk – 62% and vegetables – 17%).

3.3.1.5 Yearly doses for the most exposed members of the public due to water leaks

3.3.1.5.1 Critical group and environmental parameters

The most exposed group is members of a fisherman's family where the adults spend time fishing near the shore and the children play on the shore. The characteristics of the family according to the NRBP-W41 and ICRP66 are as follows:

Table 3.3.6. Characteristics of the habits of a fisherman's family taken into consideration in the evaluation of doses from liquid releases from an EPR reactor.

Parameter	Adult	Child	Baby
Fraction of time spent in the local area	1	1	1
Fraction of time spent in the regional area	0	0	0

Fraction of sea food caught in the local area	1	1	1
Fraction of sea food caught in the regional area	0	0	0
Time spent on the beach [h/y]	2,000	300	30
Speed of breathing on the sea [m^3/h]	1.69	1.12	0.35

3.3.1.5.2 Exposure paths

- Consumption of sea fish, crustaceans, and bivalves caught in the local waters;
- Inhaling the suspension of sea water drops in the air during stay on the beach;
- External exposure to deposits on the beach (three age groups) and to deposits on fishing equipment (adults only).

3.3.1.5.3 Consumption of food

The fisherman's family eats food from the sea in the maximum quantities, as shown in the table below:

Table 3.3.7. Habits of a fisherman's family pertaining to food from the sea assumed in the calculation of doses from liquid releases from an EPR reactor

Parameter	Adult	Child	Baby
Consumption of sea fish [kg/person/year]	100	20	5
Consumption of shellfish [kg/person/year]	20	5	0
Consumption of bivalves [kg/person/year]	20	5	0

3.3.1.5.4 Results of calculations

Table 3.3.8. Yearly doses for the most exposed fishermen's families [µSv/y] for various exposure paths from EPR reactors

	Dose from consumption of sea food (µSv y^{-1})	Dose from external exposure (µSv y^{-1})	Dose from inhaling sea splatter (µSv y^{-1})	Total dose (µSv y^{-1})
Adult	1.4E+01	3.2E+00	1.7E-09	1.7E+01
Child	4.2E+00	4.8E-01	2.2E-10	4.7E+00
Baby	1.4E+00	4.8E-02	1.5E-11	1.4E+00

3.3.1.6 Yearly doses for the most exposed members of the public received from all releases

The following three possible scenarios were taken into account to determine the maximum doses:

- A critical group of people living inland who also consume sea food with moderate intensity;
- A critical group of fishermen who also consume locally produced land-based food with moderate intensity;
- The most exposed person: a local inhabitant exposed to releases into the atmosphere and into the water.

3.3.1.7 Persons living inland who also consumes sea food

The dose from consuming sea food caught locally is defined with the assumption that:

- the fraction of sea food caught in the local area is equal to 1;

- the rate of consumption of sea food is moderate, as stated in the table below.

Table 3.3.9. Rate of seafood consumption

Parameter	Adult	Child	Baby
Consumption of sea fish [kg/person/year]	15	6	3.5
Consumption of shellfish [kg/person/year]	1.75	1.25	0
Consumption of bivalves [kg/person/year]	1.75	1.25	0

3.3.1.7.1 Results of calculations

Table 3.3.10. Dose received as a result of consumption of sea and land-based food in the vicinity of an EPR reactor

	Dose from consumption of sea food (average) (mSv y^{-1})	Dose from the land path of exposure (mSv y^{-1})	Total dose (mSv y^{-1})
Adult	1.9E+00	4.0E+00	5.9E+00
Child	1.2E+00	4.4E+00	5.6E+00
Baby	1.0E+00	7.8E+00	8.8E+00

3.3.1.8 Fishermen's families also consuming land-based food

It is assumed that 50% of food grown inland comes from local sources and all food is consumed in moderate quantities. The persons are not exposed to a radioactive plume by way of inhalation or exposure to external radiation from the plume and radionuclide deposits released to the atmosphere.

Calculation results:

Table 3.3.11. Dose received as a result of consumption of sea and land-based food in the vicinity of an EPR reactor

	Dose from consumption of land-based food (average) (µSv y^{-1})	Dose from the sea-path exposure (µSv y^{-1})	Total dose (µSv y^{-1})
Adult	1.4E+00	1.7E+01	1.8E+01
Child	1.6E+00	4.7E+00	6.3E+00
Baby	2.3E+00	1.5E+00	3.8E+00

3.3.1.9 The most exposed inhabitant

It is assumed that the family (adults, children, and babies) who lives the closest (500 m) from a nuclear power plant is exposed to releases into the atmosphere and into the sea. Therefore, the dose for the critical group is the sum of the radiation dose from the atmosphere and the dose received due to eating sea food.

Table 3.3.12. Comparison of the maximum exposure of critical group in different categories of inhabitants of the vicinity of an EPR reactor

	Fisherman's family (µSv y^{-1})	Farmer's family (µSv y^{-1})	Local inhabitant (µSv y^{-1})
Adult	1.7E+01	4.0E+00	2.1E+01
Child	4.7E+00	4.4E+00	9.1E+00
Baby	1.5E+00	7.8E+00	9.3E+00

3.3.1.10 The dose from direct radiation

Outside of buildings, the dose to the public is limited to 1 mSv per year. This value is based on the assumption that a work year is equal to 2,000 man-hours: hence the 0.5 µSv/h value. For simplicity's sake, it was assumed that this dose is present at the distance of 1 m from the external surface of buildings. The dose for a receiver located at the r distance can be calculated using the $1/r^{74}$ function, i.e. by assuming that the relation between the dose and the distance from the reactor is inversely proportional. It turns out that the value of the dose is 0.001 µSv/h.

In order to calculate the yearly dose from direct radiation per person in the critical group, one must evaluate the shielding effect of his or her house. In the United Kingdom, the proper location (shielding)[73] factor is equal to 0.1 (according to the IRA methodology, table D). The total yearly dose for the critical group is equal to the value of the dose in the receiving person, multiplied by the stay time, taking into consideration the reduced value of the dose during the receiving person's stay indoors.

$$Dose = D \times (LF_i \times O_i + LF_o \times O_o)$$

where:

- D - value of the dose in the location with the highest radiation level on the land [µSv/h];
- LFi,o – Location (shielding) factor taking into account the ration of radiation values indoors and outdoors;
- Oi – time of stay indoors [h/y];
- Oo – time of stay outdoors [h/y].

It is assumed that adults spend 50% of their time outdoors, working in the field adjacent to the lot on which the nuclear power plant is situated. The anticipated dose is 4.8 µSv/y, as shown below:

$$Dose = D\times (LF \times O + LF_o \times O) = 0.001\times (0.1\times 4380 +1\times 4380) = 4.8\ \mu Sv/y$$

Children spend 20% of time outdoors. Thus, the expected yearly dose is 2.5 µSv/y, as shown below:

$$Dose = D\times (LF \times O + LF \times O) = 0.001\times (0.1\times 7008 +1\times 1752) = 2.5\ \mu Sv/y$$

Babies spend 10% of time outdoors. The anticipated dose for babies is 1.7 µSv/y, as shown below:

$$Dose = D\times (LF \times O + LF \times O) = 0.001\times (0.1\times 7{,}884 +1\times 876) = 2.5\ \mu Sv/y$$

Table 3.3.13. Yearly dose for people exposed do direct radiation from an EPR reactor

Parameter	Adult	Child	Baby
Yearly dose for people who are the most exposed do direct radiation [µSv/year]	4.8	2.5	1.7

The maximum anticipated exposure due to direct exposure in the critical group is 5 µSv/year for people who live 500 m from a nuclear power plant. The value can be compared to the exposure to external radiation from Earth and space sources, which is equal to 700 µSv/year.

Table 3.3.14. Yearly dose for the critical group in the vicinity of an EPR reactor

	Local inhabitant (µSv y^{-1})	Direct radiation (µSv y^{-1})	Critical group (µSv y^{-1})
Adult	2.1E+01	4.8E+00	2.58E+01
Child	9.1E+00	2.5E+00	1.16E+01
Baby	9.3E+00	1.7E+00	1.10E+01

The assumption that the critical group of adults will consume the maximum quantity of sea food, milk, and vegetables results in a large safety margin and is very unlikely. Moreover, the parameters used in the calculation are very exaggerated which results in a dose which is slightly larger than 20 μSv/year.

3.3.1.11 Conclusions

The largest possible yearly dose is 25 μSv/year for an adult in a critical group. This value demonstrates that the radiation hazard from an EPR reactor is within limits for the public defined by the ICRP (1,000 μSv/year) and within the limits defined by the nuclear regulatory authorities (300 μSv/year). It must be remembered that we all receive an average yearly radiation dose equal to 3,400 μSv/year.

3.3.2 Nuclear power plants with AP 1000 reactors

The yearly doses resulting from exposure through the air leak during operation of an AP1000 reactor at the boundary of the restricted-use area (800 m) are equal to 0.021 mSv/year for gamma radiation and 0.1 mSv/year for beta radiation. The doses were determined based on the average yearly atmospheric dispersion coefficient $\chi/Q = 2{,}0 \times 10^{-5}$ s/m^3. The doses are lower than those allowed according to the US regulations 10CFR50, Appendix I, which are equal to 0.1 mSv/year for gamma radiation and 0.2 mSv/year for beta radiation.

3.3.3 Nuclear power plants with ESBWR reactors

During normal operation of an ESBWR reactor, the effective yearly dose for the critical group at the nuclear plant site is equal to 0.53 mSv and at the boundary of the restricted-use area – 0.002 mSv from the water path exposure and 0.01 mSv from the air path exposure.[75]

3.3.4 Reference nuclear power plant for conditions present in Poland according to the requirements set forth in the Atomic Energy Act

According to the requirements set forth in the draft of the modified Atomic Energy Act, the maximum dose that a critical group of people living at the boundary of the restricted-use area can receive is equal to 0.3 mSv/year.

The review of the yearly doses caused by normal operation of EPR, AP1000, and ESBWR reactors presented above demonstrates that at the distance of 800 m from the reactor the doses are lower than the permissible value (0.3 mSv/year).

The calculated anticipated total dose for an adult in the critical group is equal to

0.025 mSv/year **for an EPR reactor** (taking into account all exposure paths, for persons consuming sea and land-based food products contaminated with radiation and living 500 m away from the nuclear plant);

for an AP1000 reactor: air path exposure: 0.021 mSv/year for gamma radiation and 0.1 mSv/year for beta exposure (in total: 0.121 mSv/year);

for an ESBWR reactor: 0.002 mSv from water path exposure and 0.01 mSv from air path exposure (in total: 0.012 mSv/year).

The above three reactors have been recommended for Poland and are being considered by the investor. Thus, it can be assumed that the future Polish nuclear power plant will emit a dose larger than 0.3 mSv/year at the boundary of the restricted-use area 800 m away from the reactor. For other types of reactors that may be recommended in the future, it will be necessary to conduct similar analyses and demonstrate that the value of their radiation doses is not larger than those mentioned above.

As the above discussion indicates, the doses emitted during regular operation of an EPR reactor are very low. If we consider the input from atmospheric emissions, deposits on fields and the contaminated food produced on land, as well as fish exposed to radiation coming from nuclear power plants, inhabitants of highest-risk areas within 500 m around the reactor will be exposed to radioactive doses amounting to about 26 microsieverts, i.e. much less than the difference in annual doses between average towns and cities in Poland. For instance, the average dose of external radiation in Kraków is higher by 390 microsieverts than in Wrocław. This means that by moving to Krakow, an inhabitant of Wrocław would receive an additional 10 times larger radiation dose, compared to the dose he would have received if a nuclear power plant was built in Wrocław so close to her home that the plant's fence would be next to her window.

A comparison of the maximum dose received from an EPR reactor with the difference of doses in different cities in Poland is shown on **Błąd! Nie można odnaleźć źródła odwołania.**.

Still, nobody in their right mind would shy away from going to Kraków for fear of higher radiation. By the same token, we are not afraid of going to Zakopane, where radiation levels are even higher. Thus, it can be concluded that the small additional radiation present in the close vicinity of a nuclear power plant during its normal operation is not a problem for the ecosystem or for the health of people.

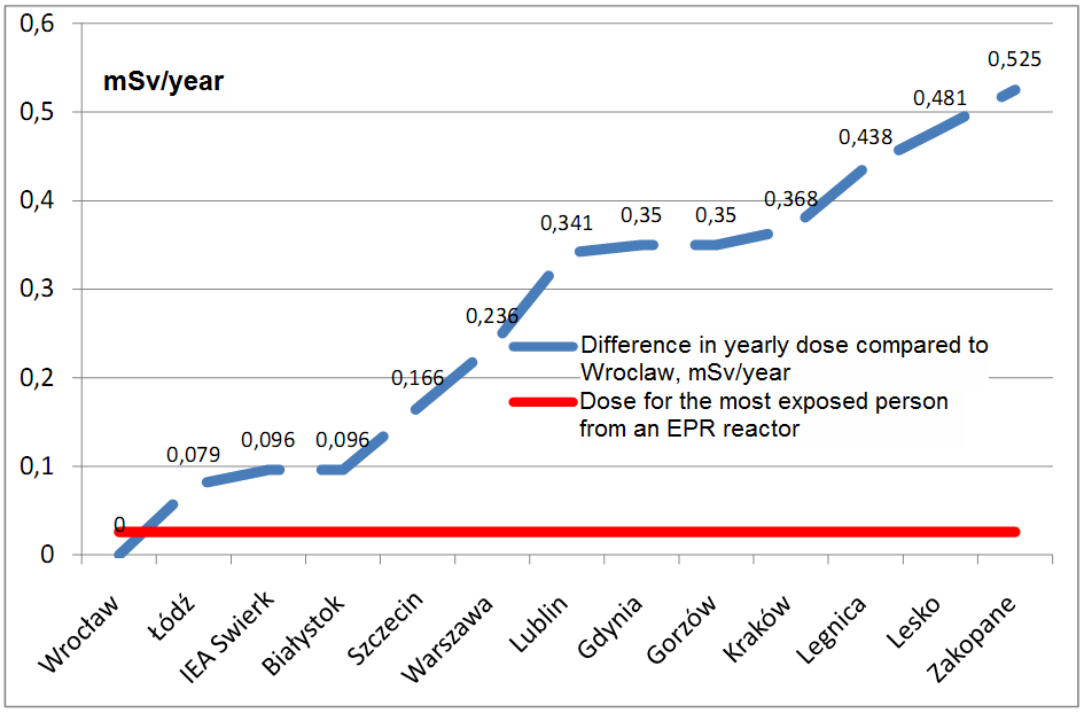

Figure 3.3.1. Comparison of the different external radiation doses in different cities in Poland with the additional radiation dose that may be received by the most exposed person from all exposure paths due to the operation of an EPR reactor.

3.4 Impact in transient and accident conditions

3.4.1 Nuclear power plants with EPR reactors

3.4.1.1 Assumptions for dose calculations

- The releases were evaluated using conservative methods and based on conservative assumptions regarding the primary activity, the rate of failure of fuel jackets, etc.

- When calculating the radiological effects (the doses), one makes realistic assumptions, so as to obtain fairly reasonably conservative assessment of the radiological effects of the transient conditions in question.

- The calculations of the effective doses cover all potential exposure paths, external exposure to radiation from clouds and deposits, internal exposure from inhalation and consumption of contaminated products. The value of the effective dose is calculated for a period of 50 years.

- After 7 days. The doses in this phase correspond to the exposure of the most exposed person located in the direct vicinity of the reactor during release of radioactive substances. The value of the effective dose received by inhalation and external exposure to radiation from clouds and deposits on the ground has been calculated for an EPR reactor at the distance of 500 m from the reactor. This must be underscored because the radius of the restricted-use area planned for Poland is 800 m, which means that the doses will be relatively smaller. Moreover, the equivalent dose to the thyroid by inhalation is calculated for an adult and for a 1 year old baby.

- After 50 years. The dose represents the integrated effects over the lifetime of the exposed person. In addition to the dose received during the passage of the cloud, it also considers the doses received due to the long-lasting contamination of the ground. Persons who live near a power plant are exposed to external radiation from deposits on the ground and to internal radiation resulting from consumption of contaminated food for 50 years. The dose is calculated for locations 2 km away from the plant.

3.4.1.2 The principles of radiation protection adopted in the EPR design in reference to design-basis accidents

In accordance with the general principle, when considering design-basis incidents and accidents, the more often an occurrence may take place, the least extensive its radiological consequences must be. In observance of this principle, the EPR reactor has been designed so as to significantly reduce the releases in transient conditions during incidents and accidents.

Each of the four types of occurrences has the following associated radiation protection objectives:

In PCC1 operation conditions (normal operation conditions) and in PCC2 transient conditions, the normal operating limits, namely the yearly dose of 0.3 mSv, must not be exceeded.

The radiological objectives are the same for PCC3 and PCC4 accidents; they are based on the principle that intervention measures should be unnecessary, i.e. the requirements set forth in the EUR document must be met. However, restrictions in the use of food coming from the surrounding area may be required.

Intervention measures at the early stage of an accident include staying in a shelter, evacuation, and administration of iodine pills. The ICRP recommends staying in a shelter when the avoided dose is in the range of 5 to 50 mSv; it recommends administering iodine pills when the equivalent avoided dose to the thyroid is in the range of 50 to 500 mSv. During accidents of this type, long term intervention measures, such as resettling of the population, are not allowed.

The 2008 safety analysis of the UK EPR reactor states that the French government adopted threshold values of 10 and 50 mSv (effective dose), respectively, for staying in a shelter and for evacuation, and the dose of 10 mSv (equivalent dose to the thyroid) for administering iodine pills. In the decree issued in December 2009[76], the Minister of Health of the French Republic defined the following doses at which intervention measures are undertaken.

Staying in a shelter – effective dose 10 mSv;

Evacuation (temporary) - effective dose 50 mSv;

Administration of stable iodine - equivalent dose to the thyroid 50 mSv.

The restriction on consumption of food are the same as in other European Union countries.

Thus, for class 3 and class 4 of an accident, the EPR design states an effective dose and an equivalent dose to the thyroid.

3.4.1.3 Atmospheric dispersion factors adopted in the calculations for the EPR reactor

The values of the doses are strongly dependent on the atmospheric dispersion factor (ADF). According to the data included in the US EPR reactor safety report, the values of the ADF for the American option are preliminarily assumed to be the same as those mentioned in the safety report of the Flamanville 3 power unit. The values are stipulated below.

Table 3.4.1. Atmospheric dispersion factors - χ/Q

Place and time where dispersion is to be determined	Value
Maximum average value during the year (800 m – boundary of the restricted-use area)	≤4.973E-06 s/m^3
Accident conditions	
0-2 h (800 m)	≤1E-03 s/m^3
0-2 h (2,400 m)	≤1.75E-04 s/m^3
2-8 h (2400 m)	≤1.35E-04 s/m^3
8-24 h (2400 m)	≤1.00E-04 s/m^3
1-4 day (2400 m)	≤5.40E-05 s/m^3
4-30 day (2400 m)	≤2.20E-05 s/m^3

Please note the high value of the atmospheric dispersion factor χ/Q= 1E-03 s/m3 assumed for the calculation of hazard after accidents of an EPR reactor at the boundary of the restricted-use area (radius 800 m) in the period of 0-8 hours after the accident. The anticipated values of the factor for typical conditions in Poland are, as shown above, for 1000 m - approx. χ/Q= 1E-04 s/m3; and for 500 m – approx. 2,5 E-4 s/m^3. What this means is that in Poland, at the distance of 500 m from the reactor, the anticipated dose within 8 hours after an accident in an EPR reactor is approx. 4 times less than that calculated in the analyses of the UK EPR reactor for the ADF χ/Q= 1E-03 s/m^3.

Similarly, a high value of the ADF was assumed in the calculations for the UK EPR reactor for the distance of 2,400 m.

Why do the safety analyses for the EPR reactor assume such high χ/Q values? This does not mean that EPR reactors cause high χ/Q values. The values characterize the location of a nuclear power plant and not the reactor type. Such high χ/Q values mean that all locations where the χ/Q values are smaller than those listed in **Błąd! Nie można odnaleźć źródła odwołania.** are suitable for the construction of an EPR reactor from the point of view of radiological restrictions on the doses received from the air after reactor accidents.

For the period of 24 hours after an accident and the distance of 800 m the χ/Q value is not stipulated because, according to US regulations, calculations are performed for 2 hours periods only. Nevertheless, for the distance of 2,400 m, it is stated that extension of the time from 2 hours to 24 hours results a 1.75 times increase of the ADF value. In the case of the Darlington nuclear power plant, the reduction of the ADF when the time is extended is even larger. Thus, the ADF for 800 m defined in Poland for 24 hours will be at least 1.75 times smaller than that for 2 hours.

3.4.1.4 Results of calculations

The doses received by the critical population group in the case of class 2, 3, and 4 design-basis accidents in a UK ERP reactor are shown in **Błąd! Nie można odnaleźć źródła odwołania.** [77] .

Table 3.4.2 . Doses received by the critical population group in the case of design-basis accidents in a UK ERP reactor

Occurrence		Distance 500 m, short-term dose, 7 days		Distance 2 km, long-term dose, 50 years
Incidents - Category 2:				
	Dose	Adult (Sv)	Child (Sv)	Adult (Sv)
Loss of vacuum in the condenser	Effective To the thyroid	$1.9\ 10^{-5}$ $2.2\ 10^{-4}$	$2.5\ 10^{-5}$ $4.0\ 10^{-4}$	$6.9\ 10^{-5}$ $1.7\ 10^{-5}$
Accidents – Category 3”				
Primary loop pipeline rupture outside of safety containment	Effective To the thyroid	$5.6\ 10^{-6}$ $1.7\ 10^{-5}$	$6.0\ 10^{-6}$ $2.9\ 10^{-5}$	$6.8\ 10^{-6}$ $1.3\ 10^{-6}$
Pipe rupture in steam generator – 1 pipe	Effective To the thyroid	$1.9\ 10^{-4}$ $3.3\ 10^{-4}$	$2.0\ 10^{-4}$ $5.9\ 10^{-4}$	$1.2\ 10^{-4}$ $2.5\ 10^{-5}$
Accidents – Category 4:				
Severe accident involving loss of coolant during operation	Effective To the thyroid	$2.9\ 10^{-4}$ $2.4\ 10^{-4}$	$2.3\ 10^{-4}$ $3.9\ 10^{-4}$	$1.4\ 10^{-4}$ $1.9\ 10^{-5}$
Loss of coolant when reactor shut down	Effective To the thyroid	$2.3\ 10^{-5}$ $9.3\ 10^{-5}$	$2.2\ 10^{-5}$ $1.5\ 10^{-4}$	$1.4\ 10^{-4}$ $7.0\ 10^{-6}$
Multiple failures in auxiliary building during earthquake	Effective To the thyroid	$3.8\ 10^{-4}$ $2.1\ 10^{-4}$	$3.8\ 10^{-4}$ $3.1\ 10^{-4}$	$7.3\ 10^{-5}$ $1.7\ 10^{-5}$
Pipe rupture in steam generator – 2 pipes	Effective To the thyroid	$4.6\ 10^{-4}$ $1.1\ 10^{-3}$	$4.8\ 10^{-4}$ $1.9\ 10^{-3}$	$5.0\ 10^{-4}$ $8.6\ 10^{-5}$
Accident during refuelling	Effective To the thyroid	$5.5\ 10^{-3}$ $1.8\ 10^{-4}$	$5.5\ 10^{-3}$ $2.7\ 10^{-4}$	$6.1\ 10^{-4}$ $2.0\ 10^{-5}$

As the data above shows, the largest effective doses at the boundary of the restricted-use area adopted in the design of the EPR reactor occur after class 4 accidents. In the case of the most severe accident involving primary loop rupture, namely an accident where 2 heat exchange pipes in the steam generator are ruptured, the effects received within 7 days after the accident are equal to 4,8 * 10^{-4} Sv for a child and 4,6 * 10^{-4} Sv for an adult. These values are higher than those for an accident involving rupture of the main primary loop pipeline because when the primary loop pipeline is ruptured the fission products escape into the safety containment, whereas rupture of pipes in a steam generator result in the fission products from the primary loop escaping to the plant’s surroundings through the safety valves of the primary loop and bypassing the safety containment. In safety analyses of Generation II PWR reactors, accidents involving rupture of pipes in a steam generator are considered to be potentially the worst and design-basis accidents assumed rupture of only one pipe. Rupture of two pipes was classified as hypothetical accident.

The analysis of such an accident assumes that the maximum concentration of iodine in the primary loop prior to the pipe rupture corresponds to the previous peak of iodine emission which causes the concentration of iodine in the primary loop coolant to reach the maximum value right before the accident; the maximum value is equal to 150 GBq/t in units of equivalent I-31 iodine activity (I131eq), which is defined as:

$$(I-131eq) = I131 + I132/30 + I133/4 + I134/50 + I135/10$$

The radiological consequences of this accident are effectively mitigated in the EPR reactor thanks to the following design improvements:

The pressure in the medium head safety injection system (MHSI) is below the tripping value for the safety valves in the steam loop, which lowers the quantity of water that may flow from the primary loop to the secondary loop in a defective steam generator.

The defective steam generator is identified based on a simple symptom of its condition, i.e. on the level of water in the generator. The resulting signal automatically initiates lifting of the tripping level of the main relief valves on the steam side, which eliminates the leak from the primary loop to the secondary loop. Prior actions of the operator, consisting in manual switching based on the level of activity in the defective steam generator, are possible but not included in the safety analysis.

The capacity of the steam generator on the secondary loop side is increased, which extends the time for taking actions to prevent filling of the secondary loop side of the steam generator with water.

Thanks to these safety measures introduced to the EPR reactor, an accident involving simultaneous rupture of 2 pipes is qualified as a design-basis accident (i.e. an accident where the design of the reactor guarantees that the safety of the surroundings of the reactor will be maintained); in fact, the effective dose within 7 days after such an accident is limited to 0.48 mSv. The maximum equivalent dose to the thyroid during such an accident is equal to 1.2 mSv for an adult and 1.9 mSv for a child. These values are much lower than the values where, according to the decree of the French Minister of Health**Błąd! Nie zdefiniowano zakładki.**, intervention measures must be undertaken.

The largest potential threat takes place in the case of an accident occurring during handling of fuel, since in such a case it is assumed that the whole fuel assembly fails as it is in danger of becoming damaged after the refuelling container is dropped. Because refuelling operations are conducted with the safety containment open, the released fission products are released into the atmosphere around the plant and are not trapped in the containment. At the distance of 500 m from the reactor, the maximum effective dose for a child within 7 days is equal to 5.5 mSv and the equivalent dose to the thyroid is equal to 0.27 mSv. Both these values are lower than the values that trigger intervention measures. Evidently, from the point of view of doses received after design-basis accidents of an EPR reactor that do not involve core meltdown, the radius of the restricted-use area could be less than 800 m and be just under 500 m.

The doses at the distance of 2 km from the reactor calculated for a 50-year period for an adult are equal to 0.61 mSv for a fuel handling accident and 0.5 mSv for an accident involving simultaneous rupture of two pipes in a steam generator. These values are much lower than the values triggering any intervention measures.

Moreover, according to the note made in the previous section, when the atmospheric dispersion factor values χ/Q for the actual location are lower than those assumed in the UK EPR design, the dose values will be lower than those stipulated in **Błąd! Nie można odnaleźć źródła odwołania.**.

Błąd! Nie można odnaleźć źródła odwołania. shows the doses received in the case of an accident involving rupture of the steam pipeline outside of the safety containment, analyzed in accordance with US requirements.

Table 3.4.3. Doses at the exclusion area boundary (EAB), low population zone (LPZ), and in the master control room (MCR) after an accident involving rupture of the steam pipeline outside of the safety containment of an EPR reactor, calculated according to the US NRC method (note: the values in parentheses are permissible dose limit values according to US regulations)

	Total effective dose (TEDE) (REM) and dose limit			
Receiving	Iodine peak prior	Simultaneous	Failure of 3.3% of fuel	Meltdown of

point	to the accident	iodine peak	jackets	0.58% of fuel
EAB	0.24 (25)1	0.27 (2.5)	5.3 (25)	5.8 (25)
LPZ	0.06 (25)	0.20 (2.5)	2.6 (25)	2.8 (25)
MCR	0.52 (5)	0.72 (5)	4.5 (5)	4.5 (5)

3.4.2 Nuclear power plants with AP 1000 reactors

3.4.2.1 Atmospheric dispersion factors assumed in the design of the UK AP1000 reactor

The atmospheric dispersion factors assumed in the design of the UK AP1000 reactor are shown in **Błąd! Nie można odnaleźć źródła odwołania.**[78].

Table 3.4.4. Atmospheric dispersion factors assumed in the analysis of the AP1000 reactor

Atmospheric dispersion factor - $X/Q_{(E)}$	
Exclusion area boundary - 800 m from the reactor (0-2 h)	5.1×10^{-4} s/m^3
Exclusion area boundary (average value during the year)	2.0×10^{-5} s/m^3
Low population zone boundary	
0 - 8 h ,	2.2×10^{-4} s/m^3
8 - 24 h,	1.6×10^{-4} s/m^3
24 - 96 h,	1.0×10^{-4} s/m^3
96 - 720 h,	8.0×10^{-5} s/m^3

The value of the χ/Q factor assumed for the exclusion area boundary is two times lower than the value assumed in the UK ERP design. On the other hand, the χ/Q values assumed for the UK AP100 reactor for the low population zone are slightly lower than those for the EPR reactor. This must be remembered when considering the results of the calculation of doses after the AP1000 reactor accidents.

3.4.2.2 Consequences of steam generator tube rupture in an AP1000 reactor[79]

The safety analyses distinguish two cases:

– when the iodine peak is caused by the accident;

– when the accident occurs at a time when the iodine peak has continued for 8 hours.

In the first case the effective dose within 2 hours after the accident at the exclusion area boundary does not exceed 11 mSv and at the boundary of the low population zone – does not exceed 8 mSv within 30 days after the accident. These are small fractions of the permissible dose for such cases according to US regulations[80] (250 mSv).

In the latter case, the doses are 22 mSv and 13 mSv respectively and are within the permissible dose limits according to US regulations.

If the anticipated atmospheric dispersion factor at the exclusion area boundary for a typical location in Poland, which is equal to 10^{-4} s/m3, were used, it would turn out that the dose from an AP1000 reactor is five times lower and is within the permissible dose limits according to the draft regulations to be adopted in Poland. However, in worse weather conditions, it may be necessary to enlarge the exclusion area beyond the 800 m radius.

3.4.2.3 Consequences of complete rupture of the steam pipeline outside of the safety containment of an reactor[81]

In the event that the iodine peak is caused by the pipeline rupture, the calculated total effective dose equivalent (TEDE) is 11 mSv at the exclusion area boundary within 2 hours after the accident and 20 mSv at the boundary of the low population zone within 30 days after the accident. The doses are small fractions of the permissible dose according to US regulations, which is equal to 250 mSv (10 CFR Part 50.34). A small fraction, according to the NCR guidelines (Standard Review Plan) is 10% or less. In the event that the iodine peak occurs prior to the accident, the TEDE values are 10 mSv at the exclusion area boundary and 8 mSv at the low population zone boundary.

3.4.2.4 Consequences of sudden rupture of the main pipeline of the primary loop in an AP1000 reactor

In calculations made on the basis of assumptions made according to the best knowledge, an accident involving rupture of the large diameter pipeline rupture in an AP1000 reactor does not lead to failure of the fuel jackets. This is in line with the results of many experimental studies.

The calculated maximum jacket temperature, which with 95% confidence will not be exceeded, is 1,002.78 oC; thus, the value is lower than the permissible value of 1,204.44 oC.

The maximum local depth of oxidation of the jacket is equal to 2.25% (whereas the permissible depth is 17%). The maximum oxidation of the jackets in the whole core is 0.2%, which is also less than the permissible value which is ≤ 1%. The geometry of the core remains intact to the extent that allows for continued cooling of the core.

The radiological consequences remain small, similar to accidents in an EPR reactor. On the other hand, given the assumptions imposed by US regulations, where, regardless of the operation of the emergency core cooling system, fuel meltdown will occur, such an accident is considered as a severe accident involving fuel meltdown in section 9.4.2.

3.4.3 Nuclear power plants with ESBWR reactors

3.4.3.1 Principles of classification of design-basis accidents in ESBWR reactors

The radiation hazard after accidents in ESBWR reactors has been evaluated in accordance with the requirements set forth in US regulations. Due to the fact that no safety reports have been elaborated for ESBWR reactors in accordance with EUR requirements, the present document uses data from a report published for the NRC, with the dose values recalculated to correspond to the anticipated requirements to be set forth in Polish regulations.

In the event of an accident according to the 10 CFR 50.34(a)(1), the total effective dose equivalent[82] according to the SRP 15.0.1 and RG 1.183 must be limited to 0.025 Sv, 0.063 Sv, and 0.25 Sv, depending on the category of the accident, as shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.4.5. Categories of accidents in ESBWR reactors and the corresponding permissible total effective dose equivalent values according to US regulations

Accident	0.025 Sv	0.063 Sv	0.25 Sv
Fall of spent fuel container		x	
Rupture of a small diameter tube containing primary loop coolant outside of the safety containment	x		
Rupture of the feedwater pipeline outside of the safety containment	x		
Rupture of the coolant cleaning loop and the after-shutdown cooling loop outside of the safety containment	x		

Fall of the control rod		x	
Rupture of the steam collector outside of the safety containment	x		
Rupture of the primary loop inside the safety containment			x
Fuel handling accident		x	
Leak from the radioactive gases system	x		

The atmospheric dispersion factor values assumed in the ESBWR reactor safety analysis are shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.4.6. Data for calculating the dispersion and the radiation doses in the case of an ESBWR reactor

A. Meteorology – the value of the atmospheric dispersion factor:	
Exclusion area boundary, 800 m	2.00E-03 s/m^3
External boundary of low population zone, 0–8 hours	1.90E-04 s/m^3
8 – 24 hours	1.40E-04 s/m^3
1 – 4 days	7.50E-05 s/m^3
4 – 30 days	3.00E-05 s/m^3

It must be noted that the value of the atmospheric dispersion factor at the exclusion area boundary used in the analyses of the ESBWR reactor is equal to 2*10^{-3} s/m^3, which it two times more than the value for the EPR reactor and four times more than the value for the AP1000 reactor.

3.4.3.2 Doses calculated for design-basis accident[83]s in ESBWR reactors

In the case of failure of 1,000 fuel rods, the dose at the exclusion area boundary (800 m away from the reactor) is less than 0.025 Sv.

Table 3.4.7. Results of calculation of doses after failure of 1,000 fuel rods in an ESBWR reactor

Exposure location and time	Maximum effective dose calculated REM / (mSv)	Criterion of acceptance of effective dose REM / (mSv)
Exclusion area boundary throughout the time of flow of radioactive plume	1.56E-01 /(1.56)	2.5 (25)
External boundary of the low population zone throughout the time of flow of the radioactive plume	5.94E-02/ (0.594)	2.5/ (25)

As **Błąd! Nie można odnaleźć źródła odwołania.** shows, defects which lead to failure of the jackets of 1,000 fuel rods in an ESBWR reactor cause doses at the exclusion area boundary (1.56 mSv) and at the low population zone boundary (0.594 mSv) which are lower than the permissible values defined in US regulations and lower than the permissible values designed in the draft Polish regulations. The analysis of the accidents assumes that fission products will be released from the jackets and that partial fuel meltdown will not occur.

Accidents in the radioactive gas system will also not result in doses that exceed the permissible values, as shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.4.8. Coefficients for failures leading to releases from the radioactive gas system

Data for calculation of dispersion of the dose	
A. Atmospheric dispersion factor	
At the exclusion area boundary, 800 m (s/m^3)	2.0E-03
In the reactor control room (s/m^3)	Table 2.0-1
B. Assumptions regarding conversion of the dose	RG 1.183

Table 3.4.9. Calculated radiation doses after a failure in the radioactive gas system

Exposure location and time	Maximum effective dose calculated REM / (mSv)	Criterion of acceptance of effective dose REM/(mSv)
Exclusion area boundary, 2 hours	7.2 E-02 / (0.72)	2.5 / (25)
External boundary of the low population zone, 30 days	7.2E-02 / (0.72)	2.5 / (25)

On the other hand, an accident with fuel container failure during manipulation with the safety containment open leads to doses which, given the large atmospheric dispersion factor values assumed in the ESBWR reactor safety analyses, exceed the values allowed in the draft Polish regulations. The calculations results are shown in **Błąd! Nie można odnaleźć źródła odwołania.**. Even higher are doses after accidents involving rupture of the steam loop outside of the safety containment. This is due to the relatively high concentration of fission products in the steam loop, which is the natural outcome of the operation of the ESBWR reactor which, unlike the EPR reactor or the AP1000 reactor, has one loop, instead of two. Consequently, the releases of iodine and other fission products dissolved in the steam from ESBWR reactors are much larger than from PWR reactors.

Table 3.4.10. Calculated doses after a failure during fuel handling in an ESBWR reactor

Exposure location and time	Maximum effective dose calculated REM / (mSv)	Criterion of acceptance of effective dose REM/(mSv)
Exclusion area boundary, 2 hours	3.6 / (36)	6.3 /(63)
External boundary of the low population zone throughout the time of flow of the radioactive plume	3.6 / (36)	6.3 /(63)
Dose received by the operator in the control room through the time of the accident	2.3 / (23)	5.0 /(50)

Table 3.4.11. Results of radiological analysis of consequences of steam collector rupture in an ESBWR reactor

Exposure location and time	Maximum effective dose calculated REM / (mSv)	Criterion of acceptance of effective dose REM/(mSv)
At the exclusion area boundary throughout the time of flow of radioactive plume		
With iodine activity peak prior to the accident	12,6 (126)	25 / (250)
With iodine activity in a balanced state	0,7 (7)	2,5 / (25)
At the external boundary of the low population zone throughout the time of flow of the radioactive plume		
With iodine activity peak prior to the accident	12,6 (126)	25 / (250)
With iodine activity in a balanced state	0,7 (7)	2,5 / (25)
Dose received by the operator in the control room through the time of the accident	4,5 (45)	5 / (50)

3.4.4 Reference nuclear power plant for conditions present in Poland according to the requirements set forth in the Atomic Energy Act

According to the recommended requirements included in draft Polish regulations, similar to the EUR requirements, nuclear power plants should be designed so that outside of the exclusion area, during a design-basis condition accident (without core meltdown) it is not necessary to implement any intervention measures. Regulation of the Council of Ministers of 27 April 2004[84] defines the doses at which appropriate intervention measures with specific effectiveness must be undertaken. The respective values are shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.4.12. Intervention levels according to the Polish regulations

Dose that may be received with no intervention measures			Type of intervention measures
Value	Type of dose	Time	
100 mSv*	Effective	7 consecutive days	Evacuation
10 mSv*	Effective	2 consecutive days	Instructions to stay in sealed indoor premises
100 mSv*	To the thyroid	-	Administration of stable iodine
30 mSv*	Effective	30 days	Temporary resettlement
10 mSv*	Effective	30 days, 2 years after the accident	Permanent resettlement
1000 mSv*	Effective	Lifetime**	Permanent resettlement
When the radioactive substance content in food exceeds the values stipulated in Annex 1 to the Regulation			Ban on consumption of contaminated food
When the content of caesium in animal feed or water exceeds the values stipulated in Annex 2 to the Regulation			Ban on giving contaminated feed and water to animals and on pasturing cattle on the contaminated land

** With the exception of the dose received with food*
*** Adults - 50 years; children – 70 years*

The basic evaluation criteria for the three types of reactors in question are the dose values at the exclusion area boundary in reference to the two types of intervention measures and levels defined in the aforementioned Regulation of the Council of Ministers, namely:

- staying in sealed indoor premises – the doses within 2 days after the accident may not exceed 10 mSv;
- evacuation – the doses within 7 days after the accident may not exceed 100 mSv.

As shown in sections 7.4.1 – 7.4.3, the doses at the exclusion area boundary after accidents which do not involve fuel meltdown are equal to:

EPR reactor: χ/Q (0-2 h) $=1*10^{-3}$ s/m3, the effective dose within 7 days equal is equal to 5.5 mSv in the case of refuelling accident; the dose to the thyroid is equal to 1.9 mSv in the case of simultaneous rupture of 2 tubes in the steam generator. In the case of rupture of tubes in the steam generator – which is the accident with the most extensive radiological consequences for the other two generator types - in an EPR reactor all factors that increase the hazard have been considered, to include the preceding peak of iodine emission into the primary loop.

AP1000 - χ/Q (0-2 h) = 1*10-3 s/m3, the effective dose within 2 hours after the accident is equal to 11 mSv for an accident involving rupture of the steam pipeline without prior peak of iodine emission into the coolant; or 10 mSv in the case of iodine emission prior to the steam pipeline rupture. At the distance of 2,400 m from the reactor, the doses within 30 days are 20 mSv and 8 mSv, respectively.[85]

ESBWR: χ/Q (0-2 h) = 2*10-3 s/m3, the effective dose within 2 hours after an accident involving failure of 1,000 fuel rod jackets at the exclusion area boundary is 1.56 mSv; after a fuel handling accident – 36 mSv; and after rupture of the steam loop with earlier iodine peak – 126 mSv (which is approx. 6 times more than in the case of an AP1000 reactor). Even though the atmospheric dispersion factor assumed for the ESBWR is two times larger than that assumed for the EPR reactor, it is evident that the consequences of steam loop rupture in the ESBWR reactor are severe. Considering the fact that in specific locations in Poland the value of the atmospheric dispersion factor is several times lower than the value assumed for the ESBWR reactor, such accidents would require people living at the distance of 800 m from the reactor to at least stay in sealed indoor premises and perhaps to be temporarily evacuated. The right decision can be made only after determining the characteristics of the atmospheric dispersion in a specific location. Of course, an ESBWR can be

located in Poland without the need for evacuation, but it will require a significantly larger exclusion area.

Thus, the values shown in table 7.4.11 will be used as the reference values.

3.5 Impact in the event of severe accidents

3.5.1 Nuclear power plants with EPR reactors

3.5.1.1 Assumptions for evaluation of radiological hazards after accidents according to US regulations

With regards to accidents other than those involving rupture of the primary loop, the assumptions are more conservatives than those in regulations in force in most European Union countries and in regulations to be adopted in Poland. At the stage of licensing of a reactor design, the designer performs analyses which demonstrate the extent of defects caused by a given accident at which the doses in reference points will not be exceeded. Such reference points are usually:

- the exclusion area boundary located 800 m from the reactor;
- the low population zone boundary[86] located 2,400 m from the reactor;
- the reactor master control room located in the building adjacent to the safety containment for which conservative assumptions are made concerning the movement of air and the atmospheric dispersion factors.

Depending on the type of accident, one of these reference points is the location where the dose is close to the limit value defined in US regulations. In the other points, the anticipated dose is lower than the permissible dose. The permissible doses after severe accidents are not defined either in Polish regulations or in regulations of other EU countries; however, they are defined in US regulations. The permissible dose depends on the anticipated frequency of the accident; for the maximum accident, it is equal to 250 mSv and for more frequent accidents – 63 mSv or 25 mSv. These values are much higher than the doses for design-conditions accidents according to the EUR requirements, which is due to the formal requirements regarding the characteristics of the accidents in question. The main types of accidents in EPR reactors according to the US approach are shown below.

3.5.1.2 Characteristics of the main types of accidents considered in the design of the US EPR reactor

1. The loss of coolant accident (LOCA) in the US EPR reactor is considered in accordance with the assumptions defined in the RG 1.183 document.

It is assumed that a loss of off-site power (LOOP) occurs simultaneous with the loss of coolant, and that the time sequence of the releases conforms to the requirements of the RG 1.183, Table 4. US EPR reactors are licensed under the condition that leaks are detected before rupture of the primary loop. Despite this, the radiological consequences of an accident are determined as if this condition was not met. This is important, because it is assumed that in the first 305 seconds air will be removed from the space between the primary and the secondary safety containment. This process does not include filtration, which means that releases occurring immediately after an accident are removed into the surroundings on the outside of the plant.

The releases consists of the activity in the coolant at the moment of rupture of the pipeline (with the limit content of iodine isotopes corresponding, from the point of view of radiation hazard, to the

dose equivalent of iodine-131 equal to 1 µCi/gm, the dose equivalent of inert gases equivalent, from the point of view of radiation hazard, to dose equivalent of xenon-133 equal to 210 µCi/gm) and of releases from the reactor fuel, in accordance with the course of releases defined in the RG 1.183 guidelines. In addition to releases from underneath the jacket for a period of 30 minutes, the releases from the fuel include releases from melting fuel for a period of 90 minutes. Thus, an accident of this type is a severe accident with fuel meltdown.

All releases occur to the primary atmosphere in the safety containment. Their elimination from the atmosphere in the containment occurs as a result of natural decay, settling on the wall and on the floor, escape in the first 10 seconds after the accident through the ventilation system, and escape through any openings in the remaining time of the accident. It should be noted that the designers of the EPR reactor did not take advantage of the possibility to increase the rate of elimination of iodine from the atmosphere inside the containment by using an encasement sprinkling system. The rate of the leaks from the containment is 0.25% per day for the first 24 hours and then it drops by 50% in the following days. The fact that fission products are stopped in buildings around the containment is not taken into account. The atmospheric dispersion factor values for the LOCA conditions are shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

Table 3.5.1. Atmospheric Dispersion Factors (χ/Q) used in the safety report for the US EPR[87] reactor for LOCA accidents

Location and time	Value of the χ/Q factor
Maximum average value during the year (800 m – boundary of the restricted-use area)	≤4.973E-06 s/m^3
Accident conditions	
0-2 h (800 m)	≤1E-03 s/m^3
0-2 h (2400 m)	≤1.75E-04 s/m^3
2-8 h (2400 m)	≤1.35E-04 s/m^3
8-24 h (2400 m)	≤1.00E-04 s/m^3
1-4 day (2400 m)	≤5.40E-05 s/m^3
4-30 day (2400 m)	≤2.20E-05 s/m^3

2. Rupture of a small diameter pipeline (sampling line, dia. 6 mm) outside of the safety containment does not lead to fuel meltdown or jacket failure because the leak is too small and the coolant replenishment system keeps the core under water and properly cooled. However, the rupture causes water with fission products to leak out (it is assumed that an iodine peak occurred before the rupture). The leak continues for 30 minutes before the operator intervenes. The accident results in the dose at 800 m (within 2 hours after the accident) and 2,400 m (within 30 days after the accident) to be 0.18 REM = 1.8 mSv and 0.3 REM = 3 mSv, respectively.

3. Rupture of tubes in the steam generator does not lead to fuel burnout because the core continues to be properly cooled. Two courses accidents of this type are assumed:

- Due to the prior iodine peak, concentration of iodine in the primary loop coolant increases to reach the maximum operating value of 60 µCi/gm DE I-131 (acc. to the RG 1.183, Appendix F);
- With iodine peak caused by the accident in question, it is assumed that iodine concentration increases for 8 hours, which corresponds to a 355-fold increase of the rate of iodine emission into the coolant (acc. to the RG 1.183, Appendix F).

4. Rupture of the steam pipeline outside of the safety containment causes steam to escape from the defective steam generator for 9 hours and from the other steam generators for 8 hours after the accident, i.e. until the water temperature in the secondary loop drops to 99 ^{o}C. The analysis takes into account a prior iodine peak or an iodine peak caused by the accident, a jacket failure due to heat

exchange crisis and melting of the fuel centre in rods operating at the uneven thermal load coefficient equal to 1.7.

Various cases are also considered for this type of accidents:

- iodine concentration in the primary loop coolant at the time of accident is equal to 60 µCi/gm DE I-131;
- rate of iodine emission into the coolant increases 500 times;
- the jacket fails in 3.3% of fuel rods;
- 0.58% of fuel melts down.

The potential radiation consequences of such an accident that does not involve jacket failure and fuel meltdown are equal the limit values in the reactor control room. In the last two cases are considered, the doses in the control room reach 90% of the permissible dose. The radiological consequences of the accident are evaluated assuming that the whole uranium mass in the fuel rod melts and releases 100% of inert gases and 50% of halogens and alkali metals into the coolant.

The consequences of steam collector rupture are more severe than the consequences of feedwater pipeline rupture and, therefore, the latter case does not need to be considered.

5. Seizure of the rotor of the circulation pump leads to possible fuel rod failure due to heat transfer crisis. The purpose of the analysis is to determine the maximum fraction of jacket failures that does not lead to exceeding the level of 90% of the permissible dose in any of the receiving points. It turns out that the permissible fraction of jacket failure is equal to 9.5% and is limited by the dose at the distance of 800 m from the reactor.

6. Ejection of the control rod results in a sudden local power increase with simultaneous loss of tightness of the reactor vessel. The following two alternative scenarios are considered in accordance with the guidelines set forth in the SRP 15.0.3 and the RG 1.183:

- leaks through the safety containment into which the whole activity of radionuclides released from the failed fuel rod jackets and the overheated fuel escapes;
- leaks through the secondary loop into which the activity from the primary loop escapes due to the leaks in the steam generator tubes which are damaged in the plant's cooling phase (which occurs as steam is releases through the relief valves on the secondary side which are open due to the loss of the on-site AC power supply after the accident).

It is expected that during actual accidents leaks will occur simultaneously through both paths.

The analysis indicated that, due to the permissible doses at the distance of 800 m from the reactor, the fraction of failed fuel jackets may not be larger than 4%.

7. A fuel handling accident is considered with the assumption that it occurs at the beginning of refuelling, 34 hours after the reactor was shut down. It is assumed that a fuel assembly which has been used at the power peaking factor of 1.7 falls onto other assemblies, which leads to failure of all 265 fuel rod jackets in the dropped fuel assembly and to release of the whole quantity of gases and iodine contained in the gaps beneath the fuel rod jackets. It is assumed that the accident occurs either in the safety containment or in the fuel building. Other accidents occurring during handling of fuel (e.g. fall of the transport container into the fuel pool) are excluded thanks to proper design of the spent fuel transport equipment.

The fraction of released fission products from the gap beneath the fuel rod jacket correspond to the fuel with maximum burnup of 62 GW·d/tU (RG 1.183, Table 3, Footnote 11). The released activity flows through the 7 m deep layer of water which retains the alkali metals and reduces the release of halogens to less than 0.5% of the original value (RG 1.183, Appendix B, Section 2). The whole quantity of inert gases escapes into the atmosphere in the safety containment or in the fuel building. It is assumed that all releases into the plant's surroundings occur within 2 hours (RG 1.183, Appendix B, Section 4.1). This corresponds to the assumption that the air change rate is 2.5 times the building's volume per hour, which results in removal of 46.5% of the activity present in the air in the reactor building into the atmosphere within 15 minutes. This is equal to a release of 99.3% within 2 hours. It is assumed that the releases occur without filtration at the level of the ventilation stack base.

In US EPR reactor fuel with the maximum thermal load (radial peaking factor equal to 1.7), the maximum pressure of gases beneath the fuel rod jacket is less than the limit value of 8.4 MPa defined in the RG 1.25 document.

The period of cooling before the accident equal to 34 hours was selected so that the dose after the accident is less than 90% of the permissible value in the most exposed receiving point, which in this case is a person present at the distance of 800 m from a nuclear power plant. The dose received by the staff in the control room is fairly low, approx. 0.022 mSv, with continuous exposure throughout the time after the accident.

3.5.1.3 Protection of the foundation slab of the EPR reactor containment from melt-through in the event of a severe accident involving core meltdown

All Generation III reactors have a sturdy safety containment, whereas an EPR reactor has a containment which will withstand a collision with even the largest passenger aircraft. Moreover, nuclear power plants with EPR reactors have four safety systems located in four separate buildings, which provides an additional protection against attacks on the plant from the outside. However, the basic characteristic of Generation III reactors is the fact that, regardless of the extremely low likelihood of core meltdown as a result of an accident, it is assumed as a starting point that a core meltdown may occur and that appropriate reactor characteristics and technical measures are provided to protect the population against such a hypothetical accident.

All designs use safeguards against early sudden rupture of the safety containment, for example as a result of explosion of hydrogen emitted at high temperatures in chemical reactions between water and zirconium. A number of other systems ensure the reliable absorption of heat from the reactor and its safety containment. In AP1000 reactors, the shaft where the reactor is located can be filled with water in the event of a severe accident, in order to assure external cooling of the reactor vessel and, consequently, to prevent the glowing core from burning through the vessel. In EPR and ESBWR reactors, due to their larger power, such cooling would not be sufficient and, consequently, a concept of molten core catcher was introduced. The concept involves using a pool installed beneath the reactor vessel into which the molten core would leak and where it would spread over a large surface, thus facilitating the cooling of the molten layer of core materials.

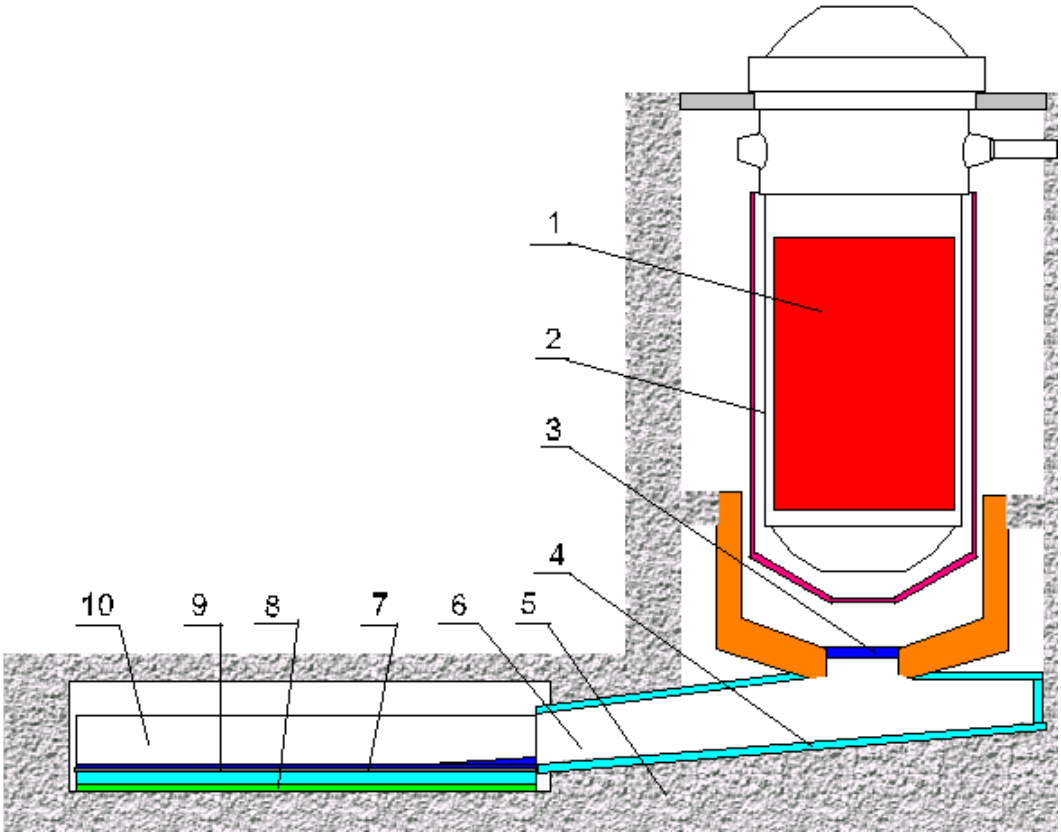

Figure 3.5.1. Core catcher in an EPR reactor

In the event of a severe accident involving core (1) meltdown and its escape from the reactor vessel (2), the molten core material melts through the lid (3) and flows through the channel (6) into the pool (10). The heat resistant material layer (4) protects the foundation slab (5) of the safety containment from being molten-through, and the pool (10) contains layers of concrete (7 and 9) which are cooled with water flowing through pipes (8).

3.5.1.4 Radiological effects of accidents with core meltdown (severe accidents) and without core meltdown in EPR reactors according to US regulations

Błąd! Nie można odnaleźć źródła odwołania. shows data on doses after severe accidents in EPR reactors.

Table 3.5.2. Radiological consequences of severe accidents with partial core meltdown in US EPR reactor (mSv TEDE[88])

Accidents without core meltdown and with partial core meltdown		Radiation dose, mSv (permissible value acc. to US regulations provided in the parenthesis)		
		800 m	2400 m	Control room
1. Loss of coolant accident (LOCA)		122 (250)	111 (250)	40 (50)
2. Rupture of small diameter pipeline outside of the safety containment		18 (25)	3 (25)	1 (50)
Steam generator tube rupture	3. with prior iodine peak	11 (250)	3 (250)	3 (50)
	4. with simultaneous iodine peak	7 (25)	5 (25)	6 (50)
Main steam line break	5. with prior iodine peak	2 (250)	1 (250)	5 (50)
	6. with simultaneous iodine peak	3 (25)	2 (25)	7 (50)
	7. with fuel jacket failure	53 (250)	26 (250)	45 (50)
	8. with partial fuel meltdown	58 (250)	28 (250)	45 (50)
9. Seizure of the rotor or breaking of the circulation pump shaft 2		23 (25)	9 (25)	13 (50)
10. Ejection of the control rod		57 (63)	35 (63)	43 (50)
11. Fuel handling accident		56 (63)	10 (63)	5 (50)

As one can see, the assumptions corresponding to severe accidents lead to a several-fold increase of the consequences of the same initiating occurrences which previously, when using the EUR methods, were possible to control with the safety systems to prevent core meltdown. In the event of a loss of coolant accident, when the emergency core cooling system is triggered, fuel meltdown is prevented and the effective dose for adults at the distance of 500 m within 7 days after the accident is 0.29 mSv and at the distance of 2,000 m within 30 days after the accident – 0.14 mSv. On the other hand, if the accident becomes a severe accident, the dose for adults at the distance of 800 m within 2 hours after the accident is 122 mSv.

The values of doses after other types of accidents change in a similar fashion. For example, steam generator tube rupture analyzed taking into consideration the operation of safety systems in accordance with the EUR methods results in an effective dose for adults at the distance of 500 m within 7 days after the accident equal to 0.19 mSv; if the accident becomes a severe one considered in US regulations, the dose at the distance of 800 m within 2 hours is equal to 11 mSv. The differences may be due to the values of releases and the assumed values of the atmospheric dispersion factor.

According to the EUR document, the frequency of severe accidents involving core meltdown must be less than one time in 100 000 years and the frequency of the ensuing releases of fission products must be less than one time in one million years. The maximum doses are equal to 250 mSv. In the opinion of designers of the EPR reactor, the frequency of accidents are lower than those set forth in the EUR; moreover, the radiation doses after such accidents will not exceed the permissible values according to the US requirements.

3.5.2 Nuclear power plants with AP 1000 reactors

3.5.2.1 Protection against burn-through of the reactor vessel in the event of a severe accident in an AP1000 reactor

The AP1000 reactor is designed so as to prevent the reactor vessel burn-through even if the core is not cooled inside the vessel and is melting. In such events, the vessel is cooled by the water that is poured into the reactor shaft, as shown on **Błąd! Nie można odnaleźć źródła odwołania.**.

The heat transferred from the vessel causes the water to evaporate. The steam condenses on the internal surface of the safety containment. Cooling of the external surface of the containment, described in Chapter 6, assures long-time heat transfer and safety of the population in the plant's vicinity.

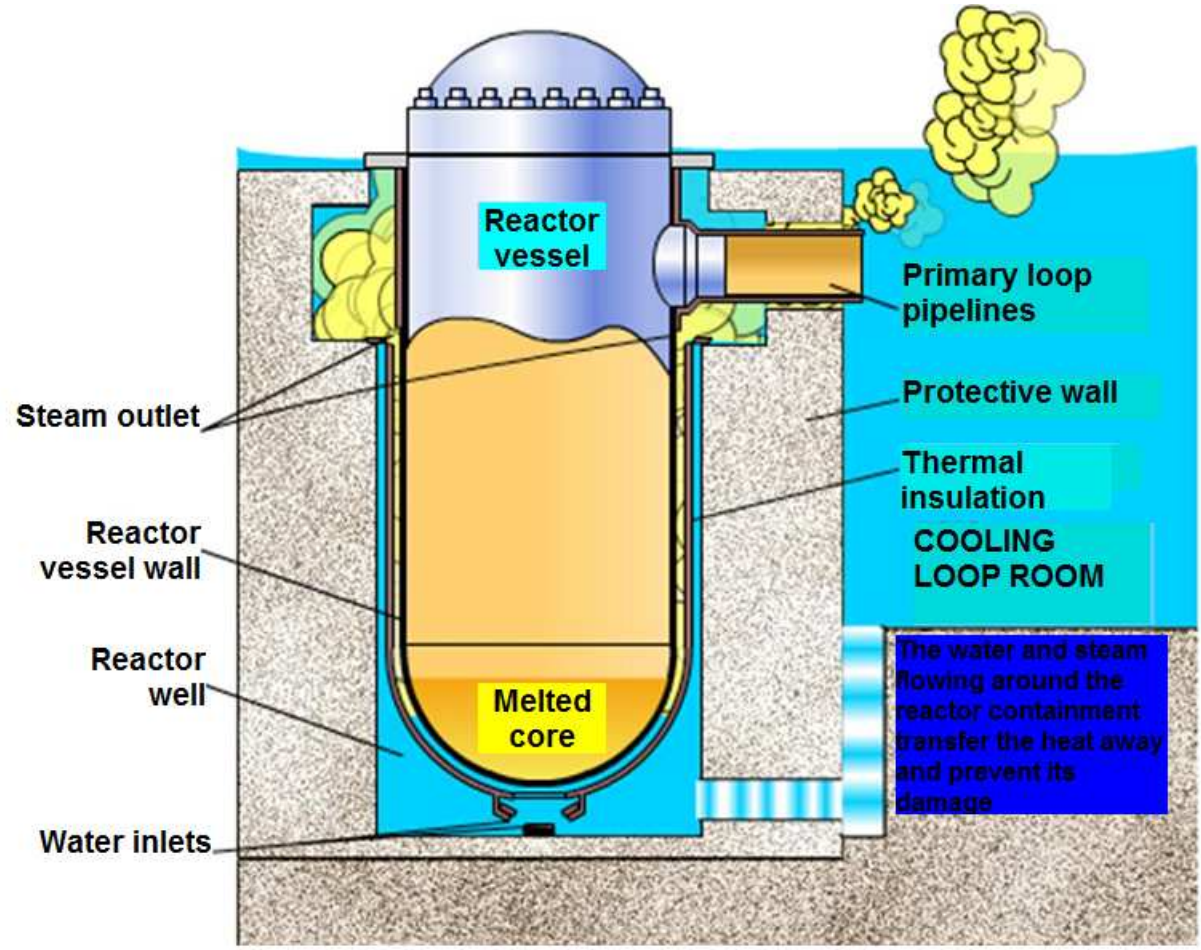

Figure 3.5.2. Diagram of heat transfer in the event of an AP1000 reactor accident, from the melted core through the reactor vessel to the water filling the reactor shaft

Nevertheless, according to the US regulations, for the purpose of evaluating the radiological consequences of an AP1000 reactor accident it is assumed that the core has melted and a part of the fission products are releases into the safety containment. This is an artificial assumption which was originally introduced to compare the effectiveness of safety containments and location conditions and not to determine the hazard to the population, but it may be used to evaluate the consequences of a severe accident, it one does occur in an AP1000 reactor.

3.5.2.2 Consequences of a severe accident in an AP1000 reactor

To distinguish the loss of coolant accident defined in accordance with US regulations with the accident defined in accordance with regulations in force in the UK and in other EU countries (to include Poland), the authors of the report for the UK AP100 reactor titled the accident analysis elaborated in accordance with US NRC regulations as “Radiological consequences of a loss of coolant accident with core meltdown” (table 15.6.5-3 in the UK AP1000 report).

Given the above-mentioned assumptions regarding releases during a severe accident and the assumed values of atmospheric dispersion factors, the effective dose received in 2 hours of stay in an exclusion area is equal to 246 mSv and the effective dose received in 30 days of stay after the accident on the low population zone boundary is equal to 234 mSv.

These doses are lower than the maximum permissible dose according to US regulations (250 mSv) but are much larger than the limit doses received during accidents that do not involve core meltdown according to the Polish regulations (10 mSv). Because the accident in question is a severe accident, one cannot assume that there is no need for “any intervention measures.” According to the

EUR and the proposed provision of the Regulation of the Council of Ministers, the limit values should refer to measures considered in the case of a severe accident, namely to evacuation (according to the Polish regulations the proposed value is 100 mSv) and to permanent resettling. However, it is evident that while according to US regulations an AP1000 reactor can be built on a site with the atmospheric dispersion factor at the boundary of the exclusion area equal to $5*10^{-4}$ s/m^3, according to the Polish regulations, such a site would be unacceptable (the atmospheric dispersion factor would have to be much lower).

The risk of reactor core failure and large radioactive releases in the case of the AP1000 reactor is very low. The total results for all accidents related to power supply shutdown, occurred due to external factors, fires, and floods are shown below:

Likelihood of core damage: 5 * 10-7/reactor-year

Likelihood of large releases: 6 * 10^{-8} /reactor-year[89]

3.5.3 Nuclear power plants with ESBWR reactors

The frequency of accidents involving core meltdown in an ESBWR is very low and equal to approximately:

CDF = 3 * 10^{-8}/R-Y.

Thanks to the passive cooling system of the safety containment and the molten core catcher (*BiMAC*), the relative frequency of safety containment damage in the event of a core meltdown (*Contingent Containment Failure Probability - CCFP*) is only 2.5%, which means that the likelihood of a large release of fission products is extremely low, in the order of 7.5 * 10^{-10}/R-Y.[90]

The low frequency of core failures and the low likelihood of safety containment failures guarantee that the radiation doses outside of the plant will remain on a very low level, even after severe accidents. **Błąd! Nie można odnaleźć źródła odwołania.** shows the radiation dose at the distance of 800 m from the reactor as a function of the likelihood for a typical location in the USA. As the figure shows, large releases of fission products do not occur even as rarely as one time per billion years.

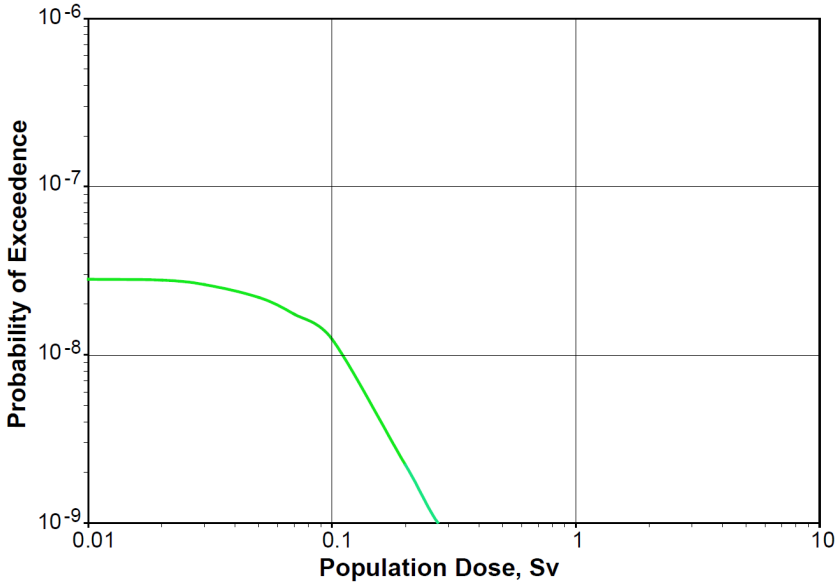

Figure 3.5.3. Radiation dose at the distance of 800 m from an ESBWR reactor as a function of likelihood for a typical location in the USA; source: description of the ESBWR reactor[90]

Błąd! Nie można odnaleźć źródła odwołania. shows the doses after an accident involving rupture of the main steam pipeline inside the safety containment.

Table 3.5.3. Calculated radiological consequences of a failure involving rupture of the cooling loop pipeline of an ESBWR reactor inside the safety containment

Exposure location	Meteorological conditions (s/m^3)	Maximum calculated effective dose (TEDE) (mSv)	Acceptance criterion acc. to 10 CFR 50.34(a)(1) TEDE (mSv)
Exclusion area boundary, 800 m	$2.00*10^{-3}$	130	250
Outer low population zone boundary	$1.9*10^{-4}$ (0-8 h*)	32	250
	$1.4*10^{-4}$ (8-24 h*)	59	
	$7.5*10^{-5}$ (1-4 d*)	111	
	$3.0*10^{-5}$ (4-30 d*)	177	

** The values shown in the table above do not take into account the additional 20 minutes of decay of fission products; therefore, the time shown in the table corresponds to the time from the beginning of the fuel failure and not from the beginning of the accident.*

3.5.4 Reference nuclear power plant for conditions present in Poland according to the requirements set forth in the Atomic Energy Act

In the event of a severe accident, emissions from Generation III reactors are limited thanks to the technical solutions and the built-in safety features, so that a long-term or severe exposure of the local population is not possible even for the core meltdown scenario assumed in the analysis. The special safety requirements adopted by the European power utilities (known as EUR – European Utility Requirements) assume that reactors must be safe not only during normal operation and design-basis accidents, but also during severe accidents involving core meltdown. he same requirements have been introduced in the proposed provisions of the Polish Atomic Energy Act and the related regulations of the Polish Council of Ministers. Reactors offered to Poland must conform to these requirements.

Thorough verification of all safety features is only possible after completion of an analysis of the reactor's documentation by the nuclear regulatory authorities. However, for the purposes of this study it was assumed that the results of analyses of three reactor designs by the EUR Committee and the nuclear regulatory authorities in the USA, Finland, France, China and the UK will be sufficient. AP1000 and ESBWR reactors offer considerable reduction of the frequency of serious accidents and have special solutions to prevent early and significant releases of radioactive materials after the reactor core melts. They meet the requirements concerning reduction of the likelihood of accidents and reduction of the hazard in the event that a serious accident does occur despite all of the preventive measures.

In general, one can expect that the reactors to be built in Poland will meet the requirements of the Polish standards providing that in the event of a severe accident involving core meltdown, there is no need to take early and long–term intervention measures (such as evacuation or permanent resettlement) outside of the exclusion area whose radius has been initially defined as approx. 800 m (depending on the actual local weather conditions and the type of reactor). Intervention measures with limited and medium-term scope, including administration of stable iodine pills, may be required after a severe accident within the low population zone – about 3 km around the reactor according to the EUR requirements (also depending on the local weather conditions and the reactor type).

Błąd! Nie można odnaleźć źródła odwołania. presents a summary of the parameters of radiological impact on humans and the environment for the nuclear power plant proposed for Poland, with the envelope including the results for Generation III reactors, taking into account the standards proposed for Poland.[91]

Table 3.5.4. Parameters of radiological impact on humans and the environment for the nuclear power plant proposed for Poland defined for the boundary of the exclusion area

Parameter		Results in the analyses performed for EPR	AP1000	ESBWR	Assumed for the nuclear plant in Poland
Atmospheric dispersion factor χ/Q for the distance of 800 m from the nuclear power plant and for the time of 2 h, s/m3		$1*10^{-3}$	$5.1*10^{-4}$	$2*10^{-3}$	$2.5*10^{-4}$
Radius of the restricted-use area, m		800	800	800	800
Annual dose during normal operation, mSv		0.025 mSv, 500 m from the plant	0.121 mSv, 800 m from the plant	0.012 mSv, 800 m from the plant	0.30 mSv, 800 m from the plant
Dose after an accident without core meltdown, 800 m from the plant, mSv	at χ/Q assumed in reports presented by suppliers of reactors	5	22	126	10
	at χ/Q assumed for the nuclear plant in Poland	1.4	10.8	15.8	
Dose after a serious accident with core meltdown, for 2 hours, for the assumed χ/Q, mSv	at χ/Q assumed in reports presented by suppliers of reactors	122	246	130	100
	at χ/Q assumed for the nuclear plant in Poland	30.5	120.6	16.3	
χ/Q for the boundary of a low population zone LPZ (2400 m) s/m^3					
0-2 h		$1.75*10^{-4}$	$2.2*10^{-4}$	$1.9*10^{-4}$	Data must be defined for the specific location, based on the annual cycle of meteorological measurements
2-8 h		$1.35*10^{-4}$	$2.2*10^{-4}$	$1.9*10^{-4}$	
8-24 h		$1.00*10^{-4}$	$1.6*10^{-4}$	$1.4*10^{-4}$	
24-96 h		$0.54*10^{-4}$	$1.0*10^{-4}$	$0.75*10^{-4}$	
96-720 h		$0.22*10^{-4}$	$0.8*10^{-4}$	$0.3*10^{-4}$	
χ/Q for the boundary of a low population zone LPZ s/m3, arithmetic average for 30 days		$2.63*10^{-5}$	$8.53*10^{-5}$	$3.87*10^{-5}$	
Dose after a serious accident with core meltdown, for 30 days, for χ/Q for the boundary of a low population zone LPZ, mSv		111	234	353	
Frequency of serious accidents involving high releases outside of the safety containment		Less than 10^{-6} /reactor-year	$6 * 10^{-8}$ /reactor-year	Less than 10^{-8} /reactor-year	Less than 10^{-6} /reactor-year

3.6 Case study – radiological impact of the Flamanville nuclear power plant on the ecosystem[92]

3.6.1 Radiological impacts in a maritime ecosystem

3.6.1.1 Surroundings of the Flamanville nuclear power plants with PWR reactors that have been in operation for 25 years

The description of the nuclear power plant with an EPR reactor is based on measurements conducted on reactors at the Flamanville plant, which have been in operation for over 25 years. Flamanville is located in France, at the west coast, in the vicinity of Cotentin, at the edge of rocky land. Its distinguishing features are large sand dunes cut by sharp protruding rocks in Cateret and Flamanville. The tides in this area are very strong. The maritime environment is of the oceanic type. The turbine condensers in the plant are cooled with sea water which is pumped through the inlet canal built on the shore and protected with two breakwaters. Wastewater from the plant's two power units are drained into the sea through underground tunnels over the distance of 500 and 600 m from the shoreline. The sea bottom in Flamanville drops sharply and reaches the bathymetric depth of -10 m CM[93] at the distance of 700 m and -20 m CM at the distance of approx. 4 km.

The radioecological study covers the shoreline from the town of Goury in the north to Catteret in the south and a fishing zone located approx. 5 km away from Flamanville.

Radioactivity in the environment was measured before and after the plant was commissioned. The purpose of the measurements was to determine the level of radioactivity and to detect radioactive deposits from the wastewater drained from the plant.

The reference conditions were determined between April 1981 and April 1982. The two power units of the Flamanville plant started operation in December 1985 and July 1986 respectively.

3.6.1.2 The measurement and assessment methods

Since 1981, a sampling and measurement campaign was conducted in the direct vicinity of the plant on a yearly basis. The studies were conducted by the Radiological Protection and Nuclear Safety Institute (Institut de Radioprotection et de Sûreté Nucléaire, IRSN). The information collected during the radioecology studies in 1996 and in the years 1997-2003 are presented below.

Before the operation of the nuclear power plant began, the condition of the maritime environment was determined in areas that were going to be affected by the radioactive wastewater drained from the plant. Most radioactivity was of natural origin. It comprised mostly K-40 isotopes and, to a lesser extent, Th-232 and U-238 isotopes.

Artificial isotopes present in the maritime environment included 14 radionuclides, of which Ru-106 and Cs-137 were the most common. The presence of these artificial radioisotopes was mostly due to the industrial wastewater from the nuclear installations operating at that time and from the atmospheric nuclear weapons tests conducted by the Chinese.

3.6.1.3 Inspections and measurements conduced at sea

Samples were collected in the maritime environment at the distance of 50 ma away from the wastewater discharge points, in open sea at the distance of 750 m from the wastewater discharge points, and in the groundwater in the area adjacent to the installations. Measurements of tritium did not demonstrate the presence of radionuclides above the detection level (37 Bq/l of water in 2003). The total beta activity was stable (approx. 11 Bq/l of seawater, maximum of 7 Bq/l of subsurface water in contact with seawater). This was mostly due to the activity of the K-40.

In the vicinity of Flamanville, radioactivity was also measured in the deposits as well as in caught fish, bivalves, and crustaceans.

In the years 1991-1995 the level of radioactivity did not change. The main component was K-40. Derivatives of the Th-232 and U-238 series were also detected. Be-7 of atmospheric origin was often detected in various organisms, with the exception of fish.

Artificial radioactivity was observed in the form of gamma emitters originating from the wastewater from the Flamanville nuclear power plant (Cs-137, Co-60, Ag-110m, Sb-125, and Mn-54), as well as from the spent nuclear fuel processing plant in La Hague and from nuclear weapons tests conducted in the previous years.

In 1991 a drop in the activity of artificial isotopes was reported, which indicated a reduction in the leaks from the spent nuclear fuel processing plant in La Hague and a reduced impact of the Chernobyl disaster. In the maritime environment, the presence of Co-60, Ag-110 m, Cs-137, and (more rarely) Ru-106 and Sb-125 was reported. The highest concentration of Ru-106 was detected on sea weeds.

In 1996, the presence of alpha radiation emitters was studied; Pu-238, Pu-239, and Am-241 was found in each of the 12 samples collected. The ratio of activity of the PU-238/PU-239 isotopes was between 0.25 and 0.70. It corresponded to a mid-range value between the nuclear weapons testing fallout in the past (0.05) and the ration characteristic of the processing of the spent nuclear fuel in La Hague (0.1 to 2.3).

3.6.1.4 Activity of radionuclides emitting alpha radiation

The activity of radionuclides emitting alpha radiation in different samples collected in the maritime environment was equal to:

- 77 to 1,251 mBq/kg of dry deposits,
- 58.5 do 250 mBq/kg of dry sea weeds,
- up to 25.9 mBq/kg of dry bivalves,
- 0.43 to 11.5 mBq/kg of wet crustaceans,
- 0.08 to 0.33 mBq/kg of wet fish.

3.6.1.5 Radionuclides emitting beta radiation

In some samples, beta activity caused by Sr-90 and Tc-99 was specifically searched for.

The concentrations of Sr-90 detected in bivalves and crustaceans were low and usually lower than the values measured before the nuclear power plant was commissioned. The activity of these radionuclides in the fish was lower than the sensitivity of the sensors. The Flamanville nuclear power plant does not emit Sr-90.

Of the artificial radionuclides, the highest activity level was measured in the case of Tc-99. It is a known fact that Tc-99 has high concentration levels in brown sea weeds and lobsters. The activity values measured in samples collected in the vicinity of Flamanville became gradually lower and were similar to the values measured in samples collected in the middle of the English Channel. Such activity is beyond doubt the result of the activity of the La Hague spent fuel processing plant.

3.6.1.6 Radionuclides emitting gamma radiation

Natural gamma radiation in different parts of the maritime ecosystem is caused mostly by K-40 as well as radionuclides from the U-328 and Th-232 series. The highest concentration of K-40 was found

on brown sea weeds (1,280 to 2,230 Bq/kg of dry material). Be-7 is present in deposits, sea weeds, and bivalves. It was not found in fish and crustaceans. Natural radioactivity is on the same level as before the plant was commissioned.

In 1996, artificial radioactivity was clearly connected with the presence of the following radionuclides:

- Am-243 and Ru-106, originating solely from the La Hague plant;
- Cs-137, Co-60, Mn-54, and Sb-125, which are present in the wastewater discharge from the Flamanville plant, but also in other sources (rainfall after nuclear weapons tests and the Chernobyl disaster in the case of Cs-137; the La Hague plant in the case of the remaining radionuclides);
- Ag-110m, mostly released by the Flamanville plant. Very low activity values were detected for this radionuclide in crustaceans and bivalves (0.08 to 0.654 Bq/kg of wet material) and in sea weeds (0.54 Bq/kg of dry material).

Given such low values, it was not possible to differentiate the radionuclides that were discharged from the Flamanville plant from the background of other nuclear installations located near the English Channel.

3.6.1.7 Analyses conducted in the years 1997-2003

Spectrometric testing of all the collected samples was conducted each year. From 2000 on, the radioecological monitoring also covered the activity of free tritium in brown sea weeds.

Radioactivity of the samples collected in the maritime environment in the vicinity of Flamanville (deposits, bivalves, and fish) was caused mostly by the radionuclides generated in the Earth crust (K-40, series of derivatives of Th-232 and U-238) and in the atmosphere (Be-7). In dry sea weeds, the activity of K-40 may reach the level of 1,700 Bq/kg of dry material.

In the period in question, the radioactivity of deposits and fish was caused only by Cs-137 and Co-60. In sea weeds, bivalves, and crustaceans, the two radionuclides were associated with Ru-106, Am-241, and Ag-110m.

The trace quantities of Agb-110m detected in sea weeds (0.07 to 0.6 Bq/kg of dry material), bivalves, and crustaceans (0.05 to 0.11 Bq/kg of dry material) were due to the operation of the Flamanville plant. Between 1991 and 2003, the concentration of these radionuclides was reduced by a factor of 10. In 2002 and 2003 they were detected in only 1 out of 12 samples. With the exception of Am-241, Ru-106, and Ag-108m, which originated solely from the La Hague plant, the origin of other radionuclides was impossible to determine due to the large number of possible sources (the Chernobyl disaster, the spent fuel processing plant, industrial waste, etc.).

From 2000 on, the content of tritium in brown sea weeds was thoroughly tested. Its value was from 1.4 to 7.7 Bq/dm^3. The activity measured in the vicinity of Flamanville was slightly higher than the background level in this area.

The measurements of I-131 demonstrated that its activity is below the sensitivity level of the measurement equipment (less than 3 Bq/kg of dry material).

3.6.1.8 Assessment of the measured values

Since the Flamanville nuclear power plant was commissioned, the whole radioactivity in the maritime ecosystem comes from the radionuclides generated naturally in the geosphere and the atmosphere.

The maritime environment in the vicinity of Flamanville is characterized by the presence of artificial radionuclides from the La Hague spent nuclear fuel processing plant, from the fallout after nuclear weapons tests, from deposits discharged from the Flamanville plant, etc.

Any leaks from the Flamanville plant contribute to the presence of artificial radionuclides in sea water. The proportion of such radionuclides is difficult to estimate.

Activity of artificial radionuclides is gradually reduced, thanks to the efforts of nuclear installation operators to reduce the level of radioactivity in the wastewater, as well as to the decrease of radioactivity in remains of nuclear weapons tests.

3.6.1.9 The anticipated effects of the operation of a power unit with an EPR reactor in a maritime ecosystem

The new power unit with an EPR reactor will not be a source of releases of radionuclides into the environment. The results of the operation of the two power units in the Flamanville plant, as well as the experiences gained from the measurements conducted in the vicinity of the 4 power units at the Atlantic coast (Paluel – 2 power units with PWR reactors and Gravelines – 2 power units with PWR reactors) confirm this statement. Evidently, operation of nuclear plants leads to detection of artificial radionuclides in maritime environments. However, such radionuclides are present only in a few locations (mostly in deposits and in sea plants) and their activity level is very often below 10 Bq/kg of dry mass. In sea fauna (bivalves, crustaceans, and fish) the activity of artificial radioisotopes has not exceeded 1 Bq/kg of wet mass since the year 2000. Such levels of activity are much below the natural radionuclide content in sea organisms.

The strategy adopted in the measurements and tests conducted in the vicinity of the Flamanville plant assured a good description of the condition of the local maritime flora and fauna. A similar strategy, with yearly measurement campaigns and radioecological reviews repeated at 10 year intervals will be continued in the future.

The results of the tests performed in the vicinity of Flamanville have not demonstrated any links of the radioactivity of maritime flora and fauna to any leaks from the nuclear power plant. Leaks of chemicals from the Flamanville plant have not disrupted the maritime ecosystem in the plant's vicinity, either. Irrespective of the chemical products discharged from the plant, the maximum concentration values of the leaks do not affect the maritime system because they constitute a small fraction of the natural contents of such products in the environment or are much lower than the reference values.

3.6.2 **Impact of the proposed EPR reactor on the land ecosystem**

3.6.2.1 Impact of the construction on the land ecosystem

No radioactive substances will be released into the environment at the construction stage.

3.6.2.2 Impact of the nuclear power plant's operation on the land radioecology

The measurements conducted in the vicinity of the two existing power units with PWR reactors at Flamanville have demonstrated that nearly the whole radioactivity in the land ecosystem around the Flamanville plant originates from the radionuclides that are naturally present in the geosphere and the atmosphere.

The artificial radionuclides detected on the land around Flamanville originate mostly from continuously falling aerosols generated during nuclear weapons tests in the atmosphere and during the Chernobyl disaster. This is why trace quantities of Cs-137, Sr-90, transuranic elements, H-3, and C-14 are detected. Moreover, the last two radionuclides are generated naturally.

In addition to the radionuclides falling from the atmosphere, artificial radionuclides are periodically emitted in liquid form into the sea (Cs-137, C-60, Pu-238) and are deposited on the beach due to the sea foam effect and the use of algae as fertilizer in the fields. Since 1996 the actual activity of Cs-137 has not exceeded 30 Bq/kg of dry mass, C-60 – 5 Bq/kg of dry mass, and transuranic elements – 0.2 Bq/kg of dry mass. Those radionuclides could have been released from the Flamanville plant or from the La Hague plant. The share in these values of the PWR reactors operated in power units 1 and 2 was so small that it was not possible to determine it; in the future it will most likely be impossible to determine the impact of the EPR reactor.

The radioecological measurements in the vicinity of the Flamanville plant have not lead to a detection of any increase in radioactivity that could be ascribed to the gaseous releases from power units 1 and 2.

Operation of an EPR reactor will evidently result in an increased of discharged radioactivity, but the emission levels will be as small as those from power units 1 and 2 and, consequently, it will not be possible to measure them.

The releases of radionuclides from power units 1 and 2 have not made any impact on land organisms.

3.6.3 **Conclusion**

- the very small emissions of non-radioactive gases will have no detectable impact on the quality of air in the area of Flamanville,
- operation of the EPR will have no significant impact on the current radioecological conditions around the nuclear power plant,
- nuclear waste will be reprocessed and stored in the power plant building to make sure that containers will not leave the controlled area without prior control and approval,
- radioactive waste will be transported by rail or by road only in final containers that will meet the requirements defined by the nuclear regulatory authorities,
- methods of transport of containers with radioactive waste will meet all the applicable requirements for transport.

3.7 Impact of small doses of radiation on living organisms

3.7.1 **Natural radiation on the Earth**

Radiation is a normal element of our daily lives. On the planetary scale, radon emitted from the earth in the gaseous form causes approx. 50% of the average individual daily dose and the remaining 40% id due to cosmic radiation and radioactive materials contained in the soil and entering the human body. This is not at all due to any nuclear accidents: radiation has been there since days immemorial and when life started on the Earth the intensity of radiation was much higher than it is now. Perhaps this is why radiation is a necessary element of life and many experiments confirmed that a complete lack of radiation causes plants and animals to stop growing and breeding.

To understand the situation regarding the current radiological protection regulations, we need to know how radiation is measured. Radioactivity describes the intensity of the source of radiation. The unit of measurement of radioactivity is curie (Ci), which was called so to commemorate Maria Skłodowska-Curie who discovered radium. One curie is the radioactivity of 1 gram of pure radium. Usually radioactivity values are much lower and they are measured in picocurie (one millionth part of a one millionth part of a curie). In 1 picocurie only approx. 2 atoms per minute undergo decay and emit radiation. The US Environmental Protection Agency (EPA) has recommended a limit for potable

water equal to 5 picocurie per litre. According to the relevant regulations, the radioactivity of water discharged from nuclear power plants may not be greater than 10 picocurie per litre. This appears to be reasonable.

However, one litre of regular sea water in which people swim when they go to the beach contains, on average, 350 picocurie. In other works, regular sea water is 35 times more radioactive than water discharged from a nuclear power plant. The average radioactivity of milk is 1,000 picocurie per litre. The radioactivity of salad olive is 5,000 picocurie per litre, which means that it is 1,000 times more radioactive than tap water, according to the EPA limit. Still, no one claims that sea water, milk, and salad olive pose a radiation hazard to the public.

Also, as far as the doses are concerned, the regulations limit the doses generated by human activities to values that are many times smaller than the natural radiation values present in nature. This is due to the caution of radiation protection experts who, in observance of the *primum non nocere* (first of all, do no harm) principle, are striving to assure that humans do not disturb the current natural balance. One must be aware, however, that radiation has been, is, and will be a natural phenomenon and it is not sure if its role is negative or that the opposite is true: radiation is beneficial and necessary for life.

The average natural background radiation in the world is 2.4 mSv/year[94] and the dose caused by humans (mostly by medical applications) is equal to 0.86 mSv/year. The operation of all the nuclear power plants in the world increases the average dose by a very minute value, approx. 0.006%.

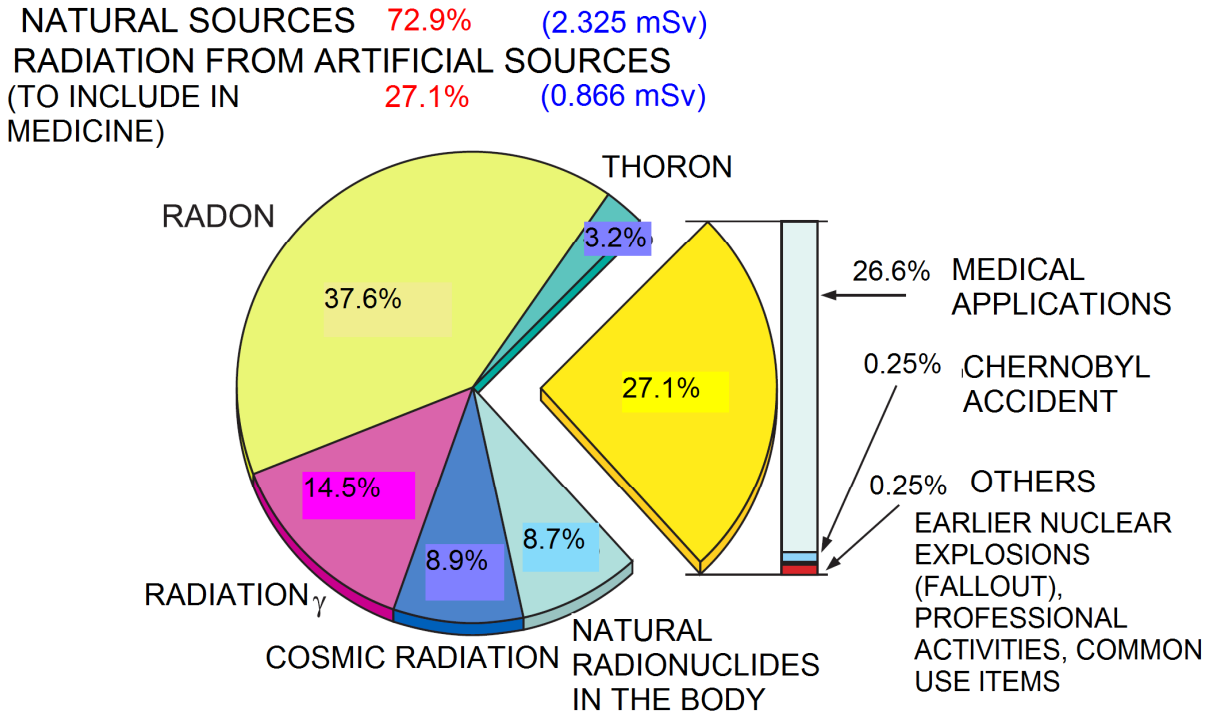

Fig. 7.7.1. Contribution of various sources of ionizing radiation to the average yearly effective dose (3.19 mSv) absorbed by the statistical inhabitant of Poland in 2009.[95]

The main components of natural radiation are:

cosmic radiation – its value is the larger the thinner is the layer of the atmosphere protecting people from radiation emitted by the stars – i.e. the higher the altitude at which a person is located. For example, in Zakopane, the yearly dose of cosmic radiation is 50% larger than that in Gdańsk. The dose at the sea level is 0.28 mSv/year;

soil radiation – soil contains radioactive elements which decay for millions of years, since the Earth formed; the average value in Poland is 0.46 mSv/year;

radiation from radon and the products of its decay - the values are very variable, depending on the composition of the soil; the global average value is 1.27 mSv/year;

radiation of radioactive elements – the elements are absorbed from food and beverages (such as potassium-40 or rubidium). This radiation causes human body to emit radioactivity. The dose from such external sources in human bodies is equal to 0.28 mSv/year.

The average value of natural radiation in Poland is equal to the average global values. The total effective dose in Poland is 3.19 mSv/year, of which 72.9% is natural radiation and 26.6 is radiation from sources used in medical applications. The effects of nuclear explosions in the atmosphere and the radiation caused by the Chernobyl disaster are equal to a total of approx. 0.5%. The contribution of the various sources of radiation to the total dose is shown in Fig. 7.7.1.

The fluctuations of background radiation values, caused mostly by differences in the content of radon in the soil, are very large, usually from 2 to 10 mSv; however, there are localities where the doses are much higher, up to several dozen mSv a year. For example, the background radiation in Sweden is twice as large as that in Poland and the background radiation in Finland - over 2.5 times larger.

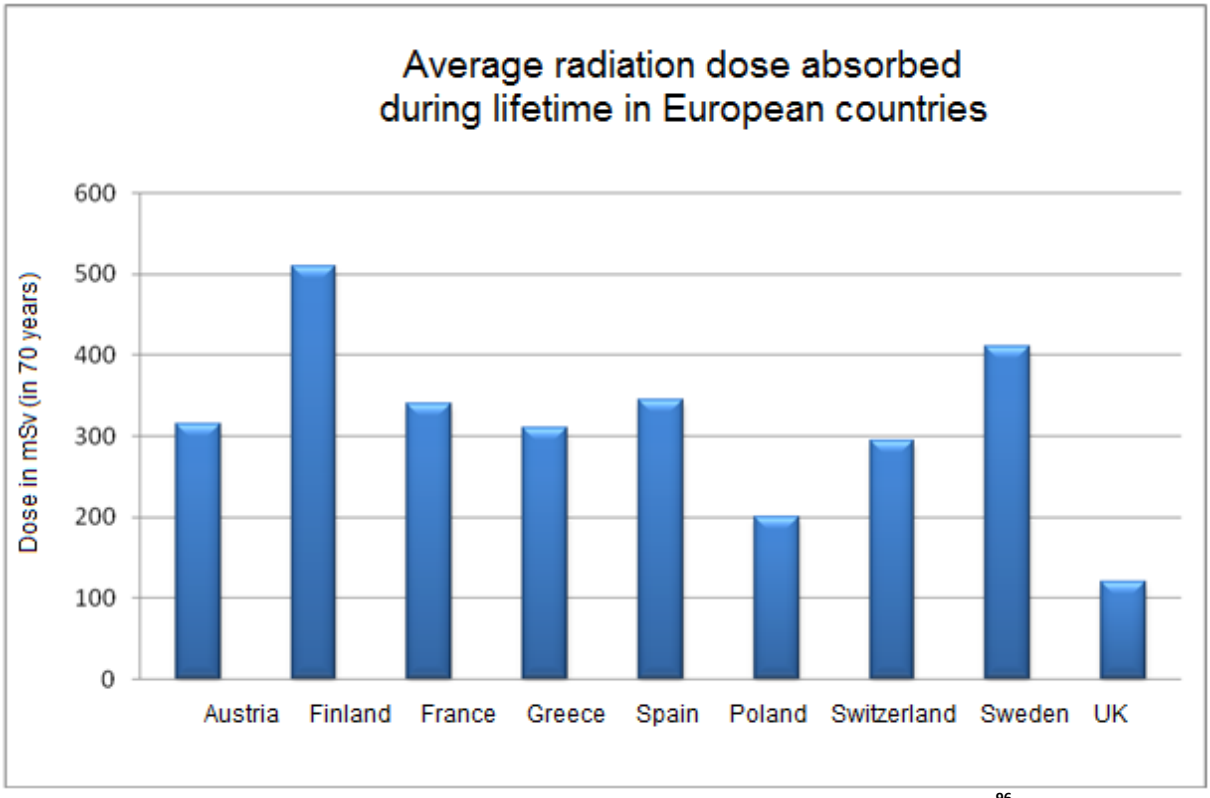

Fig. 7.7.2. Average doses absorbed by a person over a 70-year period in various countries of Europe [96].

In some areas in Brazil, India, and Iran, the yearly doses are much higher, up to 35 mSv (Kerala, India, and Guarapari, Brazil) or even 260 mSv (Ramsar, Iran).

Given such large differences, scientists have for many years conducted research in order to identify the negative consequences of higher natural background radiation to human health. So far, they have not been successful. Even in areas with the highest doses, the occurrence of cancer is not above average. To the opposite: surprisingly, at first site, it is often a little lower than average. This translates into difficulties in determining the scale of the effects of radiation – at small dozes the effects are simply unnoticeable! In order to have a basis for evaluations and comparisons, it is

assumed (as a hypothesis) that the relation between the effects of radiation and the dose, as described below.

3.7.2 **The hypothetical linear relation between the danger and the radiation dose**

Given the lack of detectable effects of small radiation doses and striving to assure utmost care in handling radioactive substances as well as to stop nuclear weapons tests, the International Commission on Radiological Protection (ICRP) introduced the hypothesis, referred to as the linear no-threshold (LNT) model, according to which hazards from small doses are the same as from large doses multiplied by the ratio of the doses and the appropriate proportionality factors. This model assumes that both somatic effects (cancer) and genetic effects of small doses are the result of mutations caused directly by ionizing radiation. At small doses, there is no data confirming directly the presence of a hazard. Therefore, the data must be extrapolated from data pertaining to large radiation doses, specifically to the effects of sudden irradiation with large radiation doses of the populations of Hiroshima and Nagasaki.

The LNT model has become the basic model used in radiological protection. It was used as a basis for formulating the principle of limiting radiation doses *as low as reasonably achievable (ALARA)* and the establishment of a very effective, albeit costly, system of barriers to prevent the spread of radiation from nuclear power plants.

As many more recent observations may suggest, the extrapolation assumed in the LNT model is excessively pessimistic. Research conducted on cancer development processes clearly indicates that cancer is a multi-stage disease and, as such, its nature is not linear but rather it resembles a curve with a threshold.

The LNT hypothesis is in contradiction with natural phenomena; in particular, it does not take into account hormesis, i.e. the fact that many occurrences and phenomena are beneficial to living organisms in small doses, even despite their harmful effects in large doses[97]. There are numerous examples: aspirin, which is beneficial when taken at a rate of one pill a day, even though several hundred pills taken at one time would be harmful; vitamins and microelements, which are necessary in small doses and harmful in large ones; sunlight, or even temperatures, which are advantageous to human health when in the range of 20-25^{o}C and deadly when they exceed 100 ^{o}C. The situation with radiation may be similar, as described in the next section.

3.7.3 **Hormesis – small radiation doses advantageous in cancer treatments**

Radiation is a necessary element of life and organisms and experiments where living organisms were screened from radiation have demonstrated that they suffered from diseases and died, whereas small doses of radiation were beneficial to their development. The positive impact of radiation on growth and breeding is common knowledge and has been discussed in dozens of publications on small radiation doses.

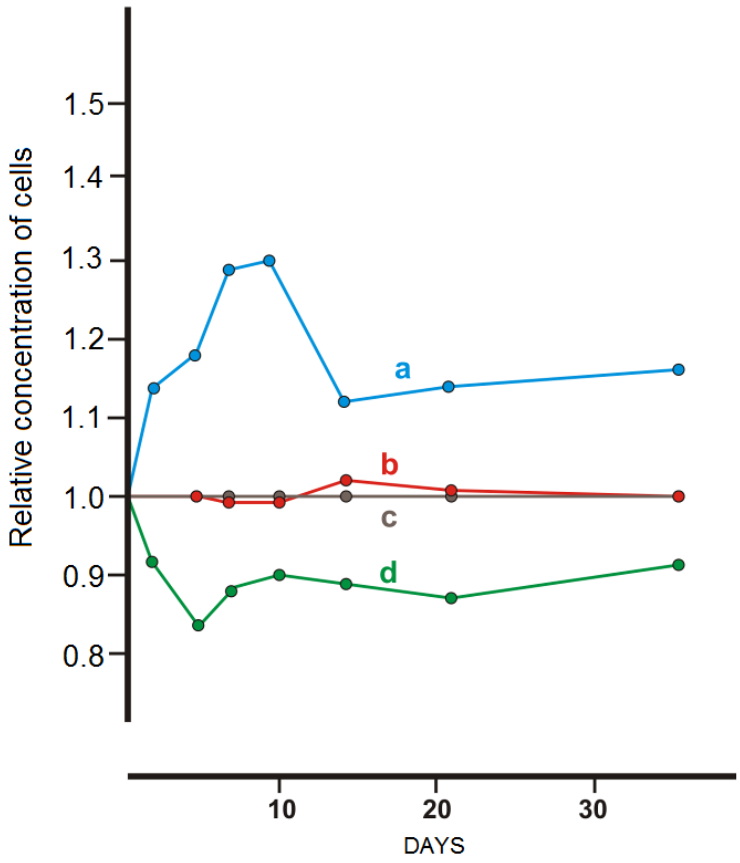

Fig. 7.7.3. Impact of screens blocking radiation, irradiation, and screens permeating some radiation on the relative concentration of Synechococcus lividus cells in experiments by professor Planel

a. Irradiated cells; b. Cells behind a screen but irradiated; c. Cells behind a screen and not irradiated.

The need for some radiation was demonstrated as early as 1960's when bacteria cultures were placed in environments without radiation – e.g. under 200 m thick rocks or behind 5-10 cm thick lead screens – and the K-40 isotope and other radioactive substances were eliminated from their nutrient medium. The yearly radiation doses were 1.65 mGy for the control group and 0.1 mGy for the cultures placed behind lead screens. It turned out that in the cultures that received no radiation the growth and breeding processes were stopped. After bacteria cultures were irradiated, with their location unchanged, the cultures began to breed normally[98].

Professor H. Planel and his team from the Medical Biology Laboratory in France conducted a series of such experiments; they used 5 or 10 cm thick lead screens[99]. The thicker were the screens, the slower was the growth of the tested bacteria and other organisms. An example of the results of their experiments is shown in **Błąd! Nie można odnaleźć źródła odwołania.**.

In another series of experiments, bacteria cultures were kept behind screens which blocked radiation. An clearly slower growth of the cultures was observed. Then, without changing the position of the cultures, approx. 1.5 mGy/year of radiation was introduced. As a result, the growth of the cultures returned to normal. In other experiments, the growth of plants and animals was demonstrated to be faster in the presence of radiation.

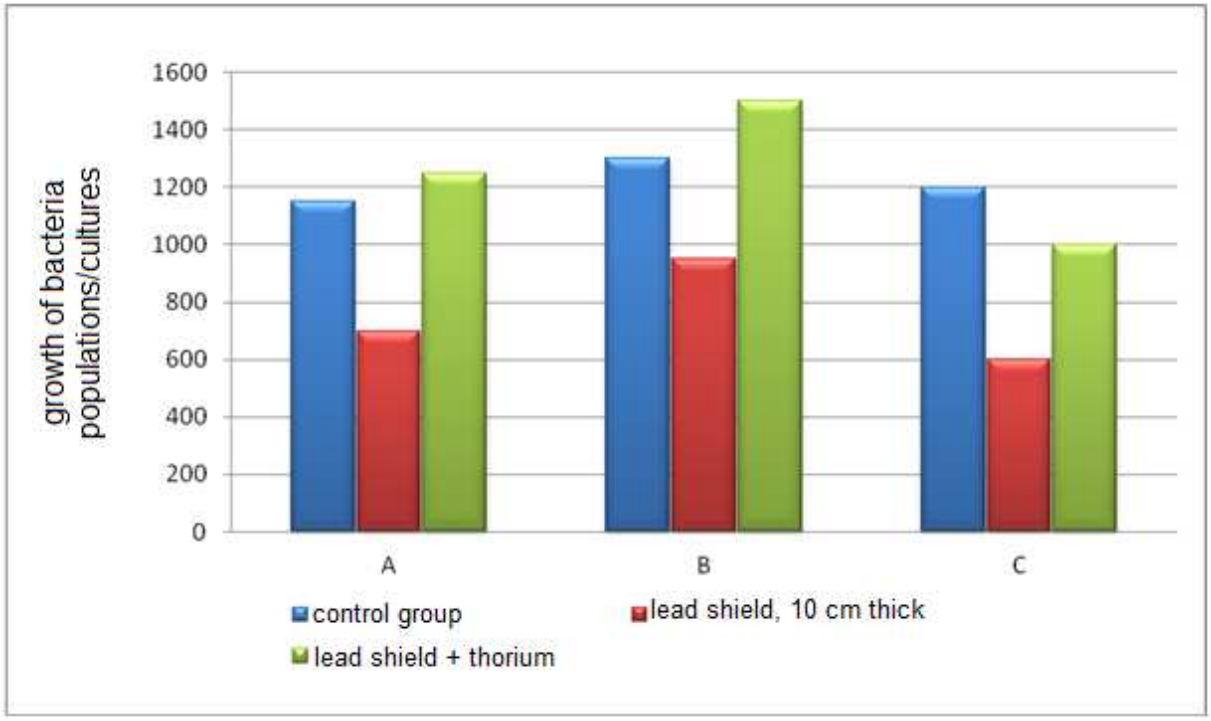

Fig. 7.7.4. Return of growth after radiation was reintroduced

Growth returned when bacteria cultures enclosed within lead screens received supplementary radiation from thorium equal to 1.59 mGy/year, which is comparable with a dose they would receive in natural conditions.

The scientists studying the role of hormesis emphasize that the LNT theory does not take into account the role of biological defence mechanisms that are stimulated by radiation. Life developed on Earth when the intensity of radiation from geological sources (uranium, thorium, potassium) and from internal sources in living organisms (K-40 potassium) was approximately 3.5 times higher than currently[100]. Thus, it is possible that our defence mechanisms work the most effectively in conditions where radiation values are much higher than they are now. In many experiences it was demonstrated that irradiation of organisms with small radiation doses enhances their immunity to cancer and is conducive to faster growth. The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has recognized the importance of hormesis and has issued a special report which recommended further studies of the positive role of radiation[101].

Given that the theoretical relations should correspond to the actual phenomena present in nature, below we present a review of the research results obtained for various groups of people irradiated with small doses.

3.7.4 Impact of small radiation doses on large groups of people

3.7.4.1 Research conducted in the USA

In the USA, the correlation between the background radiation and cancer mortality rates has been the subject of many research programs. The research has demonstrated that in all the states with an increased background radiation levels, cancer mortality rates are lower than average. This was confirmed by the results of research conducted by scientists who were not connected with the nuclear energy sector and who enjoyed impeccable reputation for honesty, such as Frigerio and Stowe[102] (Quakers), Hickey[103] from Argonne National Laboratory, and, in late 1990's, professor Bernard Cohen[104].

Frigerio and Stowe studied the rates of malignant cancer mortality in the 50 US states as a function of the background radiation. Before the research, it was expected that cancer mortality would

increase by approx. 350 deaths per 100,000 people for each additional each 1 mSv/year[105]. The results of their research proved the opposite.

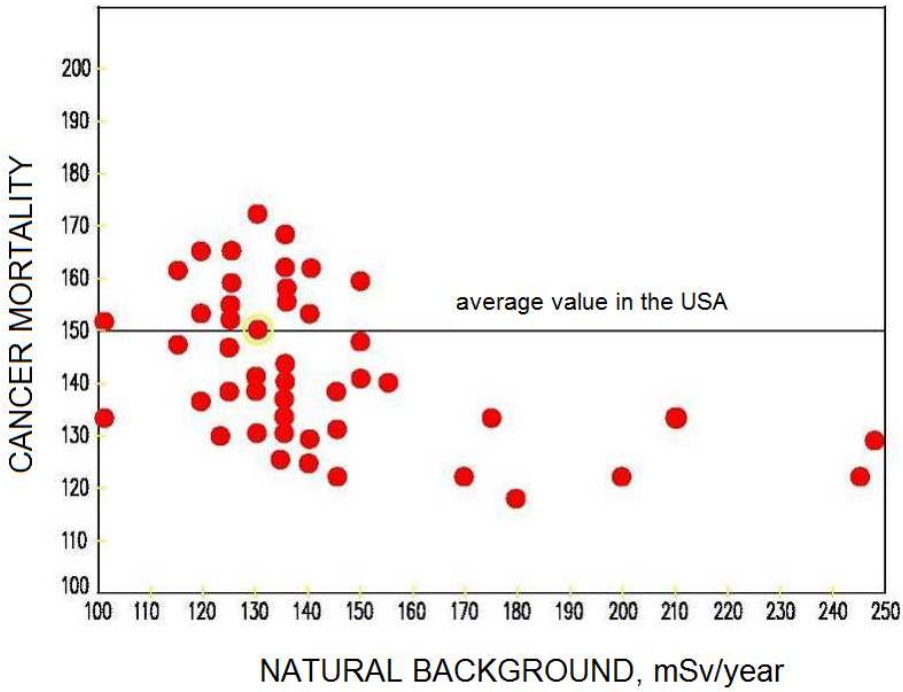

Fig. 7.7.5. Rates of cancer mortality as a function of the natural radiation background in various US states

The horizontal line and the circle indicate the average mortality and radiation background for the whole USA. Data from the work by Frigerio and Stowe, 1973.

As Fig. 7.7.5 shows, of the 14 states where the background radiation is higher than 1.4 mSv/year, **in 12 states the mortality rate was much LOWER than the average value** for the USE, in one – a little lower, and in one – a little higher.

Epidemiological research conducted in 1981 in 39 metropolitan areas and in 4 standard economic areas of the United States demonstrated that the rate of **lung and respiratory tract cancer mortality is lower in areas with the highest radiation levels**[103]**.**

Research of the impact of radon concentration in homes on lung cancer mortality [Cohen, 1995] covered 1,730 administrative districts in the USA inhabited by over 90% of the country's population. The results of Cohen's research demonstrated that **an increase in the concentration of radon does not lead to higher rates of lung cancer mortality** but rather the opposite: that cancer mortality is lower in areas with higher radon radiation levels.

In order to eliminate the impact of confounding factors, B. Cohen took into account those variables that may have an impact on the rates of lung cancer mortality, such as smoking, uncertainty of radon concentration data, impact of outliers, and 50 other socio-economic factors. Nevertheless, the slope of the curve remained negative for the scenarios which covered:

- only urban districts;
- only rural districts;
- the most wealthy districts or the poorest districts;
- the districts with the best or the worst health care services;

- and so on, for a total of 54 factors.

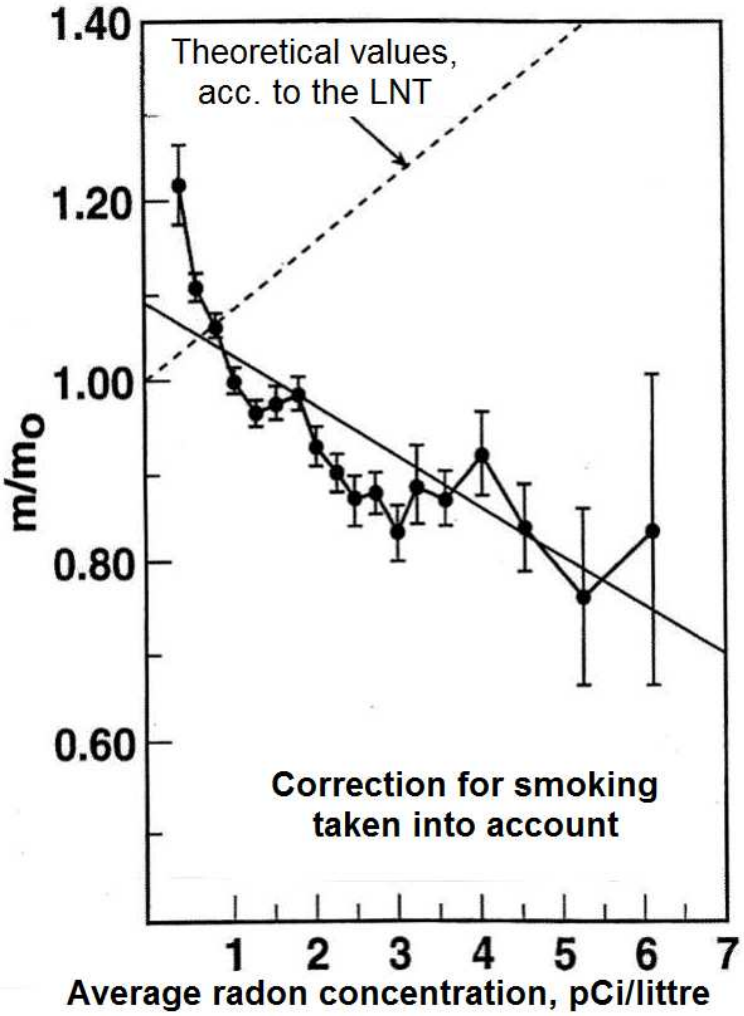

Fig. 7.7.6. Lung cancer mortality as a function of the average concentration of radon in USA administrative districts, compared to the mortality calculated in accordance with the LNT model presented in the BEIR IV report.

*- **m/mo** – the ratio of the mortality calculated in accordance with the LNT hypothesis and the mortality at 0 radon concentration, or the ratio of the mortality recorded during the research at the measured radon concentrations in homes to the mortality at the average radon concentration in homes in the USA equal to 1.7 pCi/litre.*

Cohen has also taken into account the impact of geography, altitude above the sea level, and the weather; nevertheless, the slope of the curve remained negative. Cohen's research raised a lot of interest among other scientists and efforts were made to call its result into question [Greenland, Robins 94][106], [Stidley Samet 93][107], [Lubin 02][108]. Nevertheless, Cohen was successful in refuting all the claims [Cohen] [109]. In particular, in response to the claim that introduction of averaging of the results for a large population is an example of an *"ecologic fallacy"* [110]. Cohen replied that his research was intended to answer the question of whether the LNT hypothesis is correct and that the very LNT hypothesis is based on integration of small or large doses for whole exposed populations, regardless of the doses' distribution. In Cohen's opinion, the results of his research clearly demonstrated that the LNG hypothesis does not correctly describe the reality and that calculating the number of hypothetical deaths caused in large populations by radiation is unreasonable.

An analysis of the radiation background on cancer mortality in the USA was presented by Jagger[111]. For the purpose of comparison, he selected three states with low background radiation levels

(Louisiana, Mississippi, and Alabama) and three states with high background radiation levels (Idaho, Colorado, New Mexico).

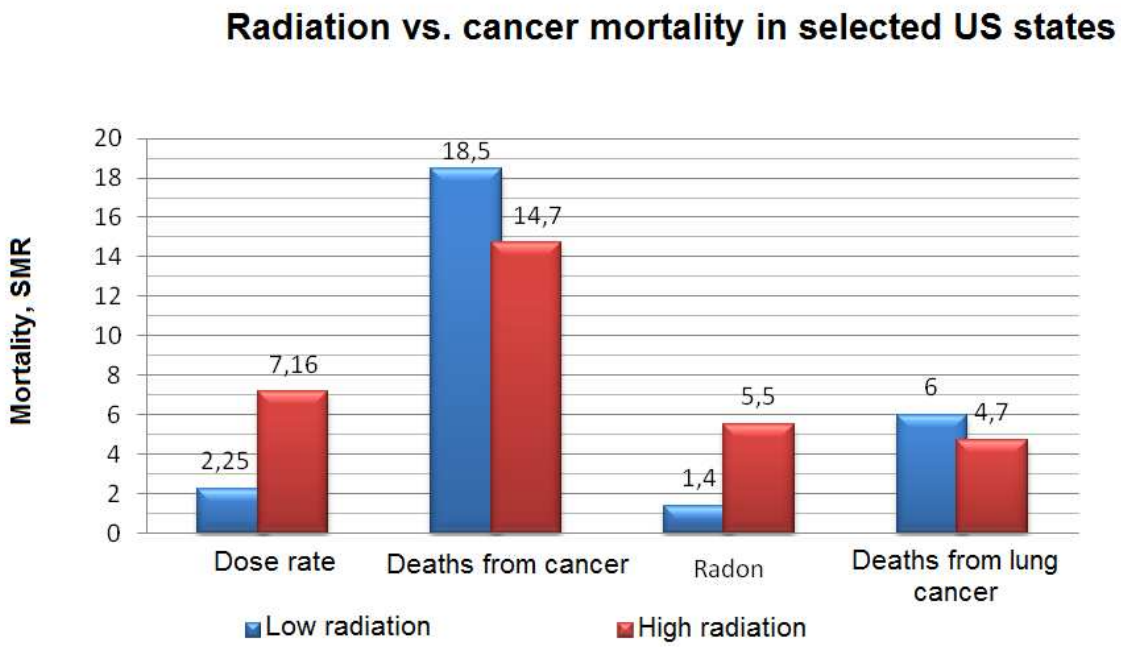

Fig. 7.7.7. Radiation vs. cancer mortality in selected US states

The average radiation doses in those states are, respectively 2.25 and 7.16 mSv/year, and the ration of radon concentration is 3.9 in open spaces and 5.2 in homes. According to the LNT hypothesis, one could expect that cancer and lung cancer mortality values would be higher in areas with high background radiation levels and high radon concentration values. In fact, the opposite is true, as shown on Fig. 7.7.7.

Another research[112] demonstrated that the actual lung cancer incidence in US states with the highest background radiation levels (CO, MT, ND, SD, UT, and WY) are, on average, 44 per year per 100,000 inhabitants, which is equal to only 14% of incidence calculated based on the LNT model. On the other hand, the incidence of the lowest background radiation levels (ID, OR, and WA), the average lung cancer incidence is 73 per year per 100,000 inhabitants, or 390% of the incidence anticipated according to the LNT model. Thus, the discrepancy between the actual numbers and the LNT model is huge: approximately 28 times larger or smaller. We would like to emphasize that "not only the reality is very different from the values anticipated according to the LNT model, but also the correlation with the background radiation is the opposite to that provided for in the LNT model."

Thus, the results of the research conducted in the USA confirm that in populations exposed to small radiation doses due to higher background radiation values there are no noticeable negative health impacts. On the contrary: in areas with high background radiation, the cancer mortality values are small.

3.7.4.2 Health effects in populations exposed to low radiation levels in China[113]

Research has been conducted in the high background radiation area (HBRA) near Yangjiang in China since 1972. The research covers two neighboring areas with the total surface area of 500 km^2, inhabited by 95,000 people who are exposed to radiation from monazite sands with high content of thorium. In the vicinity of those areas there is an area with low background radiation levels, which was selected as the control area (CA). In the control area, the average yearly dose of gamma radiation from external sources is 2 mSv and the doses in the HBRA – from 4.8 to 6.2 mSv. The

cumulative doses increase with age of the persons; consequently, 50 years old persons in the HBRA have absorbed the average cumulative effective dose of natural gamma radiation equal to approx. 274 mSv. Both areas are inhabited predominantly by farmers (93% and 94%). Their population structure is similar. All the environmental parameters are similar, too (e.g. the percentage of smokers in the HBRA is 37.9% and in the CA – 37.6%). Having taken into account the doses absorbed from food, the average yearly doses in the HBRA was determined to be equal to 6.4 mSv and in the control area – 2.4 mSv.

The cancer mortality values were 53.5/100,000 inhabitants in the CA and 6.3/100,00 inhabitants in the HBRA.

In order to better account for the effects of lengthy stay in areas with elevated background radiation levels, the cancer mortality (except for leukaemia) rates were compared for persons aged 40 to 70. The following values were determined:

- 168/100,000 inhabitants in the CA;
- 143.8/100,000 inhabitants in the HBRA.

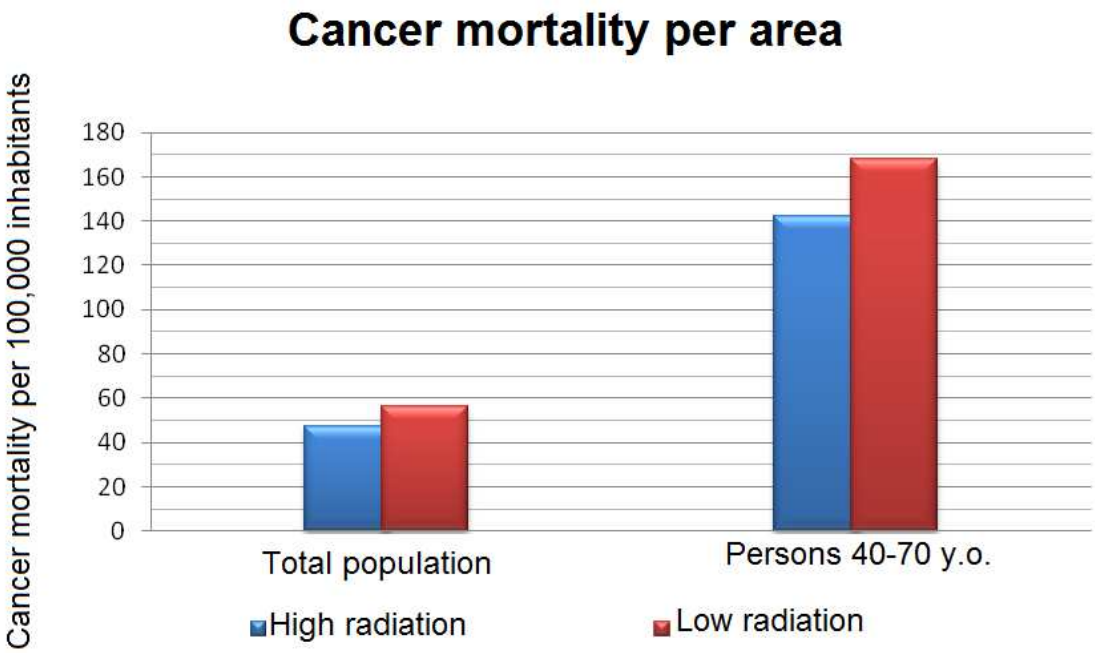

Fig. 7.7.8. Cancer mortality in the high (HBRA) and low (CA) background radiation levels in the vicinity of Yangjiang

Thus, the elevated radiation levels in the HBRA are associated with a REDUCED risk of death from cancer. Even though the research, conducted over the course of over 30 years, covered 100,000 persons, the differences turned out to be so small that they are statistically insignificant. It is beyond doubt that there is no increase in the risk of cancer.

A research conducted later[114] demonstrated that, even though no statistically valid conclusions can be drawn, further research conducted the previous research and strengthened the postulate that cancer mortality is lower in the HBRA than in the control area. An extensive research conducted by Chinese and Japanese scientists confirmed that inhabitants of the area with high background radiation levels absorb, over their lifetimes, an additional dose of external radiation equal to **320 mSv (6.4 mSv x 50 years).** No increase of cancer incidence was observed in this group; on the contrary,

the value of the optimum factor to evaluate the additional risk (excess relative risk, ERR) caused by radiation, was negative **(ERR = -0.11).**

In the description of their results, the researchers add that the confidence interval[115] (95% CI: -0.67 to 0.69) overlapped the confidence interval for the evaluation obtained for the cohort of inhabitants of Hiroshima and Nagasaki, **ERR = 0.40** (95% CI, 0.31 to 0.51); consequently, further research is necessary to obtain a larger statistical material and to reduce the uncertainty interval. Nevertheless, the research conducted in China for many years systematically leads to results that suggest beneficial impact of ionizing radiation of human health[116].

3.7.4.3 Research conducted in other countries

Similar conclusions were reached in the research conducted in the Kerala region in India[117,118] where an area with the presence of monazite sands with high thorium content (125 km^2) is inhabited by approx. 400,000 persons exposed to background radiation levels as high as 13 mSv/year. In the Kerala state, 98 various types of development anomalies were analyzed in a group of 37,000 new born babies. No significant differences were identified between the 26,000 babies in the high radiation area and the 11,000 babies in the normal radiation area.

Further research covered 50,000 new born babies from an area with high background radiation. It was found, that the frequency of anomalies is equal to 1.46%, which is slightly better than the frequency of anomalies among the 72,000 new born babies in Chennai (1.6%0 and among the 95,000 newborn babies in New Delhi, Baroda, and Mumbai (1.6% to 1.86%)[119].

The cytogenetic tests of newborn babies have been conducted since 1986. A comparison of the 9,493 newborn babies from the high radiation level area with the 1,737 newborn babies from areas with normal background radiation has not demonstrated any significant differences in the occurrence of chromosome aberrations. No connection between the frequency of chromosome aberrations and the radiation level of 1 to 35 mGy/year was found, either. Tests performed on rodents inhabiting the area of high background radiation levels in Kerala have not demonstrated any genetic effects that could be attributed to radiation[120].

Research was also conducted in other areas with radon radiation sources.

E.g. in Misasa in Japan[121], where the radiation level is elevated due to the presence of sources with significant contents of radon and radium, the normalized values of mortality from stomach cancer and other cancers are lower than in the control area with low background radiation, as shown in Fig. 7.7.9.

Such resorts as Bad Gatstein (Austria) and Boulder (USA) are used by radiation experts who come there yearly for health recovery purposes.

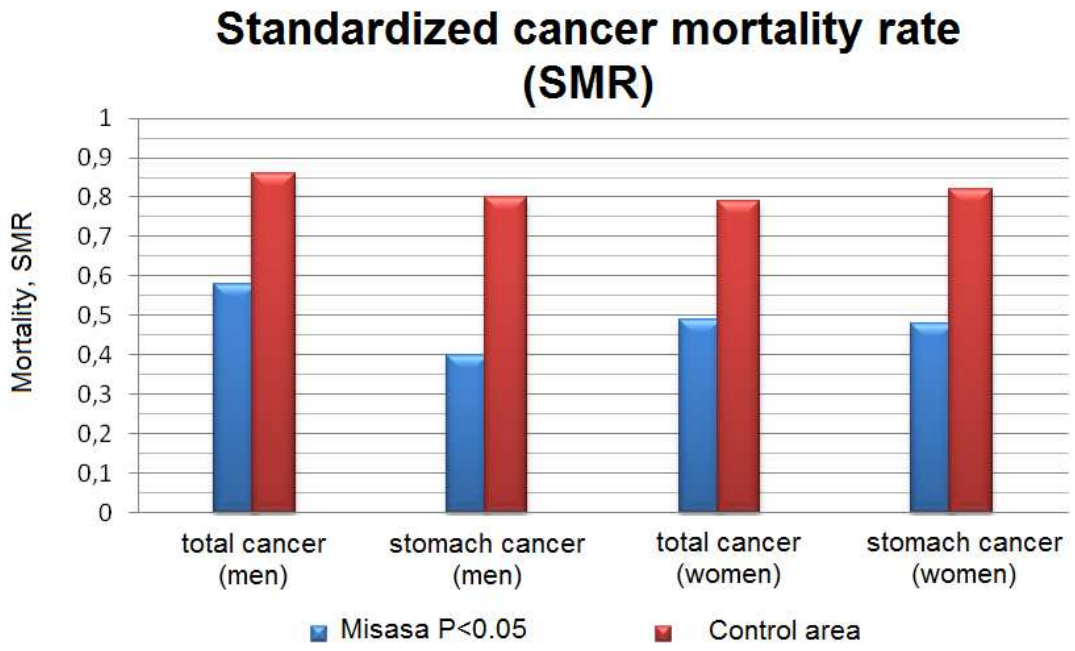

Fig. 7.7.9. Standardized cancer mortality rate for the Misasa radon resort and for the control area. Data acc. to Kondo[122]

Results that are not statistically sufficient to refute the hypothesis that all doses are harmful, but that demonstrate that mortality is lower in areas with elevated radiation levels are regularly obtained in research conducted in other countries. In no area with elevated background radiation level were higher cancer mortality rates detected. The incidence of chromosome mutations in the blood of inhabitants of such areas is sometimes elevated (but this is not a rule), but the incidence of cancer is lower than or the same as that in the control groups.

The relation between the chromosome aberrations and the value of the radiation dose has been tested in the high background radiation area near Ramsar, Iran. Approximately 2,000 persons received doses from 10 to 260 mGy/year, with the average value of 20 mGy/year. No positive relation between the chromosome aberrations and the value of the dose was identified. Also, no differences compared to inhabitants of nearby areas with low background radiation levels were found. Similar results demonstrating small significance of radiation compared with other factors were obtained in a high radiation level area in China[123].

According to the statement of the French National Academy of Medicine[124], certain data exists that confirms that elevated background radiation leads to a greater proportion of chromosome with aberration in the lymphocyte circulatory system, which constitutes a biological indicator of radiation exposure. One may not conclude, however, that this indicator pertains to health loss, as no increase of the risk of cancer, birth defects, or defects of newborn babies caused by cytogenetic effects was detected either in the thoroughly studied populations in the Kerala region in India or in the high background radiation level region in China. As the US NCRP[125] stated [NCRP 01], *"It must be concluded that cancer incidence in most populations exposed to low radiation doses is not significantly higher and in most cases the data indicates that it is lower."*

The hypothesis about the harmful effect of small radiation doses is based on the LNT model which has been questioned by many scientists and is contradictory to the experimental and epidemiological data. However, ecological research (conducted on large populations) does not allow for precise elimination of all the confounding factors and, so far, no research has been statistically valid, thus allowing it to eliminate the LNT hypothesis.

An excellent example demonstrating that elevated background radiation levels do not negatively influence people's health is shown in Fig. 7.7.10, which demonstrates the life expectancy for women in various countries as a function of electricity use. In Finland, where electricity use is high, people live much longer than in Poland, even though the background radiation there is among the highest in the world. As we can see, human health is not related to radiation levels.

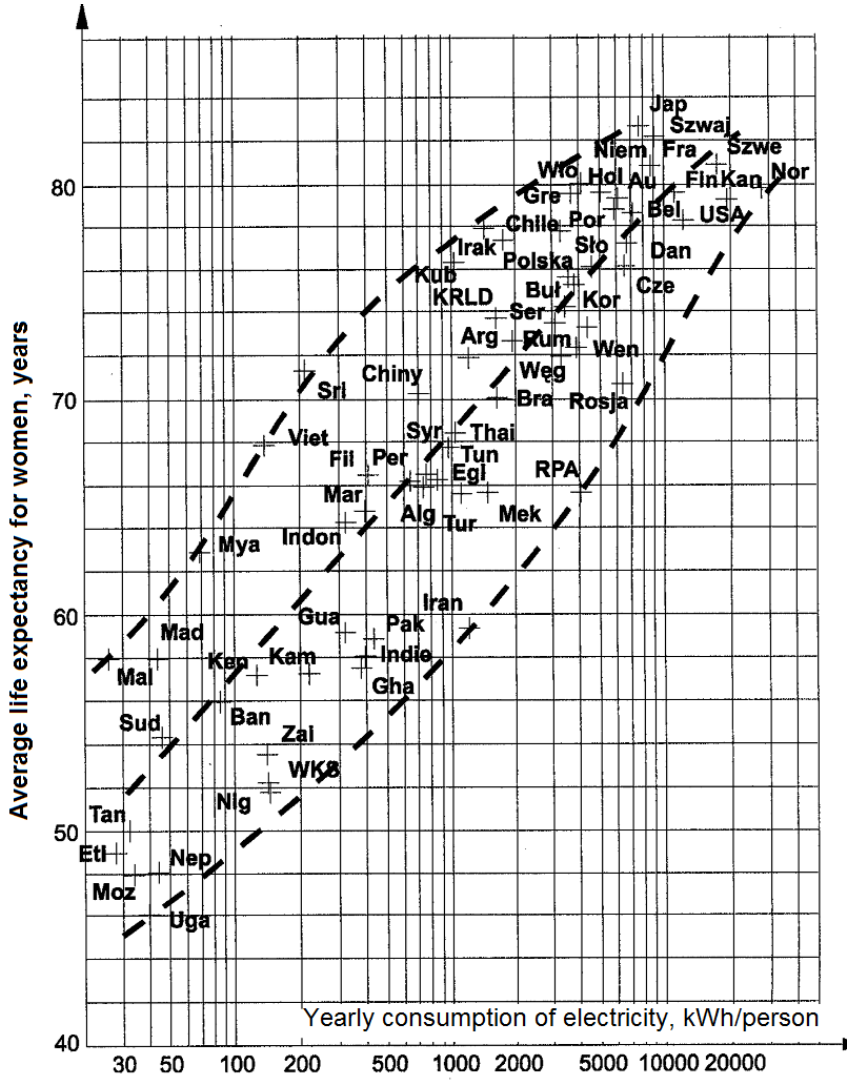

Fig. 7.7.10. Women's life expectancy as a function of electricity use

3.7.5 Research conducted on persons exposed to radiation at work

3.7.5.1 Lower cancer mortality rates among employees of the nuclear energy sector

Research involving 95,000 employees of the nuclear energy sector in the USA, Canada, and the United Kingdom, performed by the International Agency for Research on Cancer (IARC) demonstrated that within the range of small doses incidence of cancer does not increase, but rather decreases with the increase of the dose value by a factor of -7%/Sv. The relative mortality from cancer and leukaemia as a function of the cumulative dose absorbed in their lifetimes by employees exposed to ionizing radiation is shown in Fig. 7.7.11 prepared by the author based on the numerical data from the IARC work[126].

As one can see, an increase in mortality among employees exposed to radiation on their jobs occurred only in the case of very large doses, in the order of 400 mSv, and only with regards to leukaemia. This is a good example of the qualitative difference in the effects of small doses and large doses. The increase of incidence of for large doses is evident. However, doses emitted by nuclear power plants (1 mSv in a lifetime) do not cause any danger and the curves suggest that in this range of doses the mortality from cancer is reduced.

There are numerous statistically significant results of epidemiological research which demonstrate the hormetic effect of various factors, to include ionizing radiation. Based on the knowledge in the field as of 2005, it could be stated that "the hormetic model of the impact of a dose on the reaction of an organism is more common in toxicology than the threshold model"[127]. Despite the very large cohort and the many years of observation, the results of the IARC study have not achieved the statistical significance needed to prove that the LNT model should be abandoned. New data is being gathered systematically and will be published in the future. However, by now it is evident that small doses, which is what we are interested in with regards to radiation in the vicinity of nuclear power plants, there are no detectable negative effects on health.

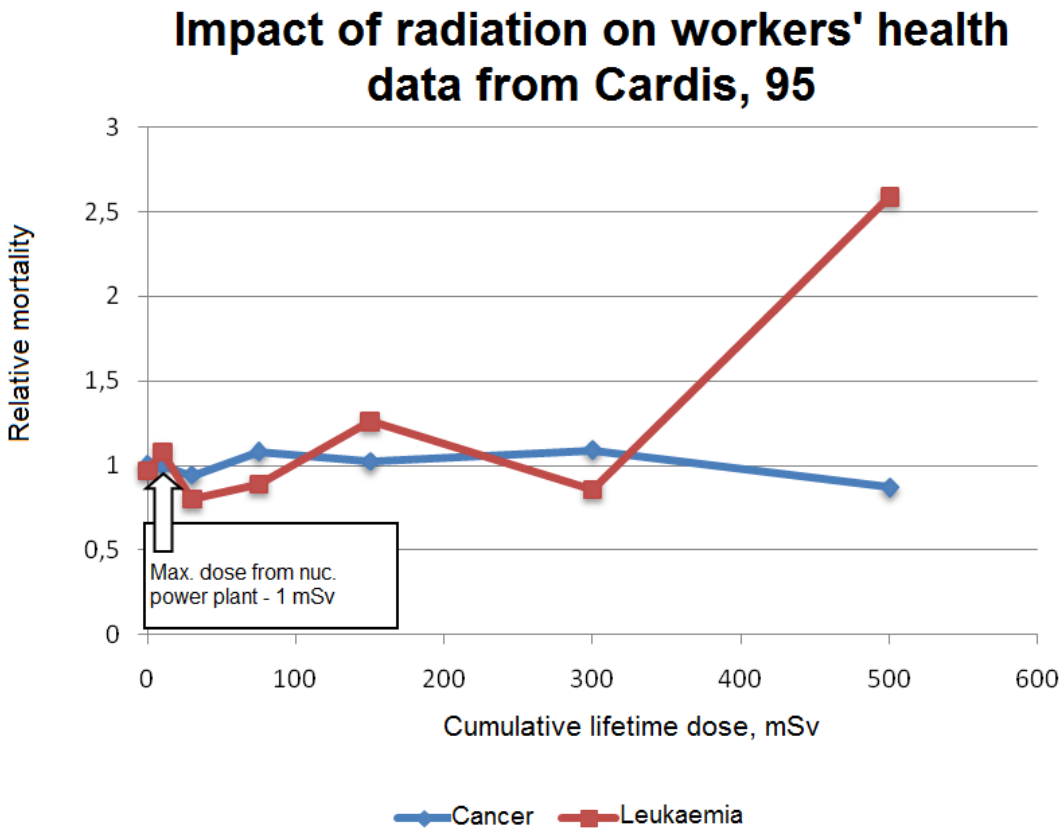

Fig. 7.7.11. Relative mortality (level „1" corresponds to the average mortality of employees not exposed to radiation) as a function of the additional dose absorbed in the course of work with radiation sources, accumulated during lifetime.

A research performed in Japan, covering 115,000 employees exposed to small radiation doses, demonstrated that both cancer incidence and general mortality in this population are lower than the average value for the corresponding group of men in Japan[128]. At the average cumulative dose of 13.9 mSv/person, the standardized mortality ratio (SMR) for the whole exposed population was SMR=0.83 for all the causes and SMR=0.89 for cancer. Thus, in Japan too cancer mortality among exposed employees is lower than the average.

Research of the impact of radiation on the health of employees of the nuclear sector are still performed as a part of international cooperation and the results from all countries are submitted to the International Agency for Research on Cancer (IARC). In 2007, the results of an analysis of deaths among 400,000 employees of the nuclear energy sector in 15 different countries were published. It turned out that the SMR ratio, as the average value of all the causes of deaths, was 62% for all deaths and for cancer among employees of the nuclear energy sector was only 19%, compared to 23% of all deaths[129] for the total population (see Fig. 7.7.12).

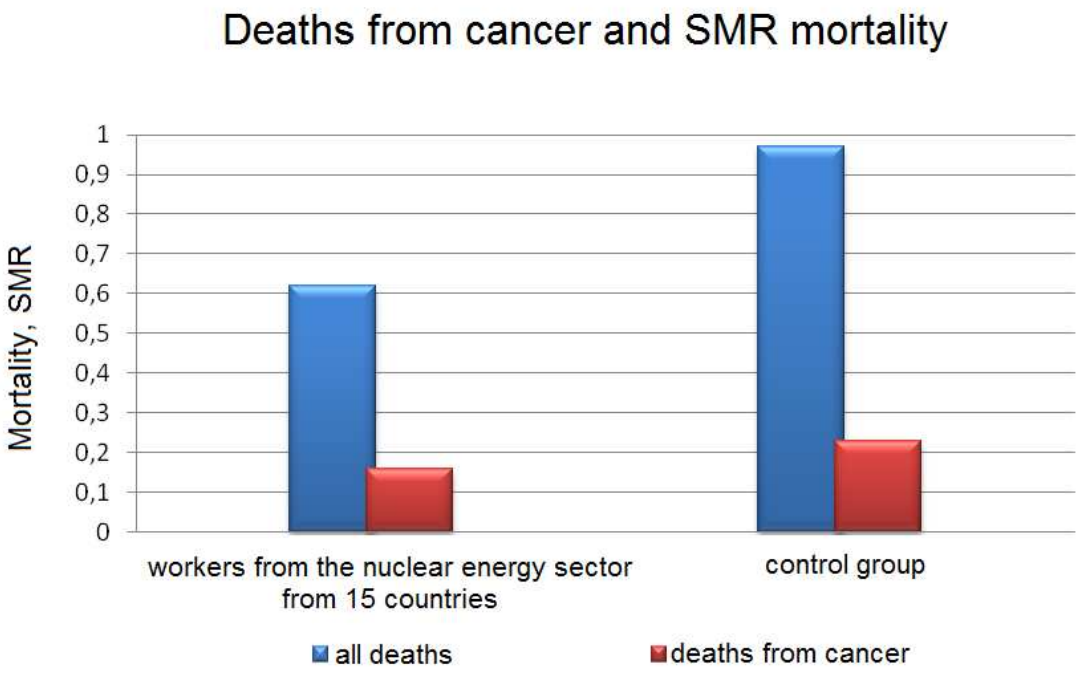

Fig. 7.7.12. Death from cancer and mortality (SMR) for employees of the nuclear energy sector from 15 countries and for the total population. Data from the IARC, figure based on the work by Fornalski and Dobrzyński[130] (with the authors' permission).

Thus, both the results of the research involving the 100,000 of employees presented in 1995 and the results of the research involving 400,000 employees presented in 2007 confirm that employees of the nuclear energy sector are more healthy and live longer than the average groups in the society of the same age characteristics. This is certainly a very encouraging facts for candidates for jobs involving work with radiation sources.

However, there are some doubts as to whether such good results are correlated with radiation or with other factors.

3.7.5.2 Can low incidence of diseases in persons exposed to radiation be explained using the healthy worker effect?

The supporters of the thesis that every dose of radiation is harmful claim that the low incidence of cancer among persons exposed to radiation is due to the healthy worker effect. The effect is the result of the fact that working persons are more healthy than the average person because persons who are ill do not work and are taken into account in comparisons between workers and the society as a whole.

The healthy worker effect can play a role in comparisons between groups of employees exposed to any factor with any external control groups, e.g. the total population of a country. It manifests itself

in the reduced incidence of diseases among the persons covered by the study as a result of selection of employees to eliminate persons who are not healthy. As a result, persons with poor health do not become workers and the average mortality of workers is lower than the mortality of the whole population. Of great significance are also socioeconomic factors and, for example, education. Persons who are more wealthy and better educated generally take a better care of their health and are less exposed to harmful factors, such as alcoholism.

This thesis is reasonable, but there is one exception to it: there is no selection of employees on the basis of susceptibility to cancer or other genetic disorders. Routine tests administered to employees do not include tomography or genetic tests that may detect cancer or susceptibility to them. This is very important for the analysis of the impact of ionizing radiation, as cancer is the most important element of any analyses. The hypothesis that the healthy worker effect has an impact on the IARC results is criticized by many scientists; in Poland it was refuted by experts of the Institute for Nuclear Problems, professor L. Dobrzyński (member of the UNSCEAR) and mgr K. Fornalski**Błąd! Nie zdefiniowano zakładki.**.

In their work, they also discussed the healthy worker survivor effect (HWSE) connected with the fact that the SMR is the lower the longer the employment time is. This can be explained by the fact that workers with poorer health gradually quit their jobs, which leads to a reduced average mortality rate among the remaining workers. However, the weighted average number of deaths in 15 countries does not confirm a significant drop of the SMR with time of employment (see Fig. **7.7.13**).

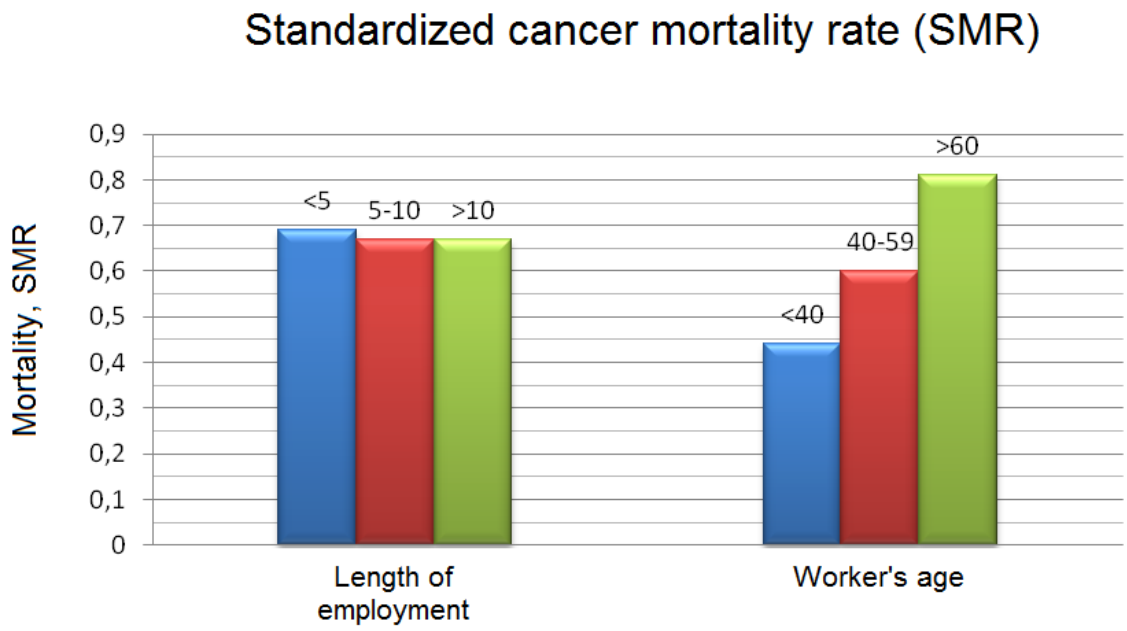

Fig. 7.7.13. Relation between the SMR and the time of employment and age of workers (figure based on the work by Fornalski and Dobrzyński[130], with the authors' permission)

This clearly proves that the HWSE does not exist in this cohort and, consequently, the HWE concept (which is related to the HWSE) is very dubious. Also, it can be seen that the SMR increases with the employees' age. This is not unusual, as it is related to the natural biological trend of every living creature.

Using the HWSE as an explanation for the reduction of the SMR appears to be unreasonable because it contradicts the claim that the HWE causes a greater reduction of the SMR at the early stages of employment than at late stages.

This leads to additional doubts as to the presence of the two effects.

The final argument against the HWSE is the fact described in the work by Berrington et al[131] who have analyzed mortality rates among British radiologists in the period of the last 100 years. The period is long enough to observe all the factors pertaining to deaths, in particular resulting from cancers, depending on the time of testing, time of employment, or the person's age. The work pointed at a drop in incidence of diseases for radiologists exposed to low doses. The control groups in all cases were selected to make the HWE factor neutral.

Another study, where the research groups and the control group were selected specifically to eliminate the healthy worker effect, was conducted to determine the impact of radiation on a large group of workers (28,000 persons) of the Shippingport shipyard. The whole research was designed, from the early stages, so as to avoid the impact of the HWE on the results. The US Department of Energy (DOE) has established a special committee for this purpose[132], and the list of its members – Professor Matanoski, Professor Cameron, etc. – is a guarantee that the works of the committee will be thorough and cohesive. The research has demonstrated that cancer mortality among persons irradiated with small doses (above 5 mSv) was 24% lower than in the control groups consisting of workers at the same shipyard who absorbed no radiation[133] (see Fig. 7.7.14).

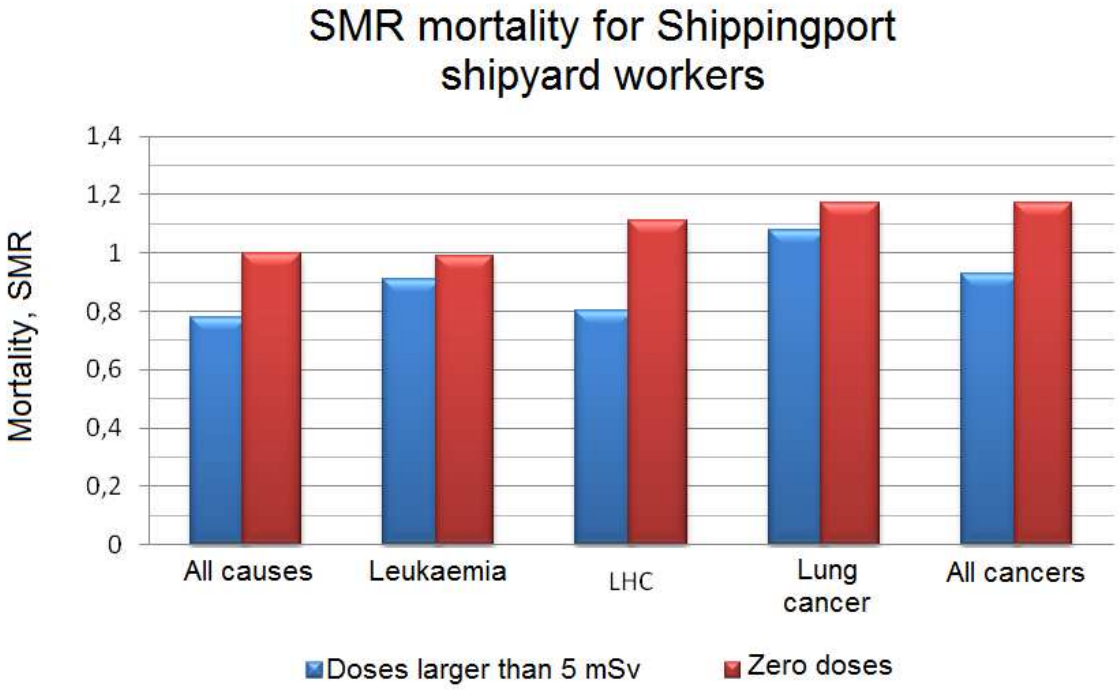

Fig. Mortality among shipyard workers at Shippingport

The selection of the control group from among the workers of the same shipyard has made it possible to eliminate the healthy worker effect. In the case of Shipingport, this effect could not have taken place as there are no reasons why workers of the same shipyard performing the same work (welding, erection, riveting, etc.) on nuclear ships could be “healthy workers” while the remaining workers are "unhealthy.”

3.7.5.3 Lower mortality among British radiologists

In the United Kingdom, extensive research among cancer mortality among radiologist has been performed. The research covered a period of 100 years (1897-1997) where the physicians absorbed various doses of radiation. As a result of the study, the standardized mortality ratio (SMR) was determined for deaths from all causes, deaths caused by cancer, and all deaths not caused by cancer

for the radiologists involved. The SMR values were compared with the SMR values for the following three groups:

(i) all men in England and Wales;

(ii) all men in social class I (to which physicians belong);

(iii) all male physicians.

As a group, the radiologists registered after 1920 (1921-1979) do not demonstrate any significant differences in the SMR values for deaths caused by cancer compared to other physicians. However, radiologists have a significantly lower SMR value for deaths caused by cancer than other men (SMR-0,63; p<0.001) and men from social class I (SMR=0.82; p<0.01). Moreover, radiologists registered after 1920 have a lower SMR value for deaths from all causes than other male physicians (SMR=0.91; p<0.01), men from social class I (SMR=0.91; p<0.01), and all men (SMR=0.72; p<0.001) [134].

The SMR for deaths caused by cancer for radiologists registered after 1955 was lower by 29% (which is statistically not significant) than the SMR for other physicians, while the SMR for deaths from all causes was much lower than the SMR for other physicians. Why would radiologists be more healthy than other physicians? Dr. Cameron, a recognized authority in the field of research on impact of radiation on people, suggests that the hypothesis that the good health of radiologists may have resulted from stimulation of their immune systems by radiation is justified:

“The study on British radiologists will not resolve the controversies concerning the soundness of the hypothesis based on the LNT model, but it confirms the doubts concerning the assumption that small radiation doses have no beneficial effects on the human body. This contradicts the current dogma that “all radiation doses are harmful” [135].

3.7.6 **Consequences of small radiation doses absorbed for medical purposes**

Medical diagnostic procedures often involve irradiation with small radiation doses. Extensive studies conducted on adult patients exposed to radiation for diagnostic purposes have not demonstrated an increase in incidence of diseases. For example, the analysis of data of 34,000 persons in Sweden who were administered J-131, which covered 653,000 person-years, demonstrated that at the average total dose of 1,100 mSv, the incidence of thyroid cancer did not change[136].

The study of the effects of thyroid hyperfunction using radioactive iodine conducted recently by the University of Birmingham and published in "The Lancet" reveals numerous facts that contradict the LNT model. The study, which involved a group of 7,414 adult patients treated in Birmingham in the United Kingdom in the years 1950-1991, with average cumulative dose of 308 MBq of I-131, identified 683 cases of cancer and 448 deaths caused by cancer in the years 1971-1991 among the patients. These values can be compared with the British statistics concerning cancer incidences and deaths caused by cancer for corresponding age groups, genders, and periods, which are equal to 761 and 499, respectively. The standardized incidence ration was 0.83 (95% confidence interval 0.77-0.90) and the standardized mortality rate was 0.90 (0.82 – 0.98). “Higher incidence and mortality were observed for the small intestine and thyroid cancers, but the absolute risk related to these types of cancer was small.” The scientists summed up their study with the following statement: “A decrease of the overall incidence and mortality from cancer among persons treated for thyroid hyperfunction with radioactive iodine is an encouraging occurrence."

A cohort study, involving 64,172 Canadian patients treated with multiple irradiations with small doses, which amounted in total from over ten mSv to several Sv, but where the instantaneous dose values were moderate (0.6 mSv/s), demonstrated that, as the author of the study stated, “there is no relation between the risk of death from cancer and the dose” [137]. A comparison with the cancer mortality among the Japanese who survived the dropping of nuclear bombs on Hiroshima and

Nagasaki and received very large single doses demonstrated that the nature of the risk related to small doses is very different. Fig. 7.7.15 shows the mortality for groups of persons who have absorbed total radiation doses in the following ranges: Group 1: 0.01-0.49 Sv; group 2: 0.50-0.99 Sv; group 3: 1.0 – 1.99 Sv; group 4: 2.00-2.99 Sv; and group 5: above 3 Sv.

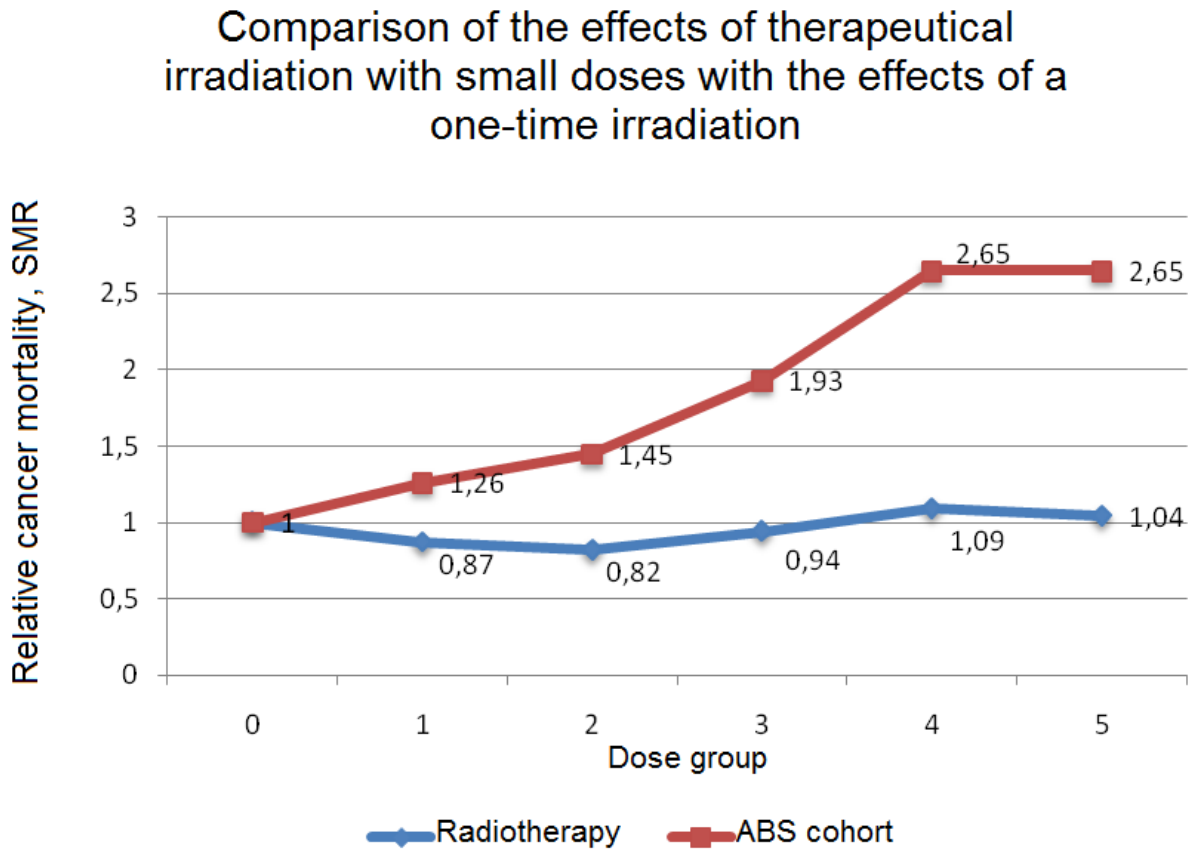

Fig. 7.7.15. Comparison of the effects of therapeutic irradiation with small doses with the effects of a single irradiation in Hiroshima and Nagasaki

In the case of the ABS cohort, the risk clearly increases with the dose. On the other hand, in the case of the cohort subject to radiotherapy with small dose rates, even though the total dose absorbed by the patients was the same as that in the ABS cohort, at small doses the cancer mortality values decrease. Only at high total doses the risk increases to levels above the average for periods who were not irradiated; however, it is still close to one, which is much lower than that for the ABS cohort. Similar results were obtained in a number of other studies.

3.7.7 The impact of irradiation with small doses on children

3.7.7.1 The impact of irradiation of parents on the health of children

In the group of children from Hiroshima and Nagasaki (1,263) who survived the explosions of the atomic bombs as foetuses and absorbed doses above 0.01 Gy (the average value was 0.309 Gy), there was no increase in the incidence of cancer and none of them died of leukaemia. In London and Edinburgh there were 9 cases of leukaemia among 39,166 irradiated children, while the number anticipated based on the average incidence of leukaemia in the United Kingdom (RR=0.86) was 10.5.

A large international epidemiologic study covering children that were begot by parents who, as children, underwent cancer treatment involving radiation, demonstrated that the incidence of genetic diseases among them is lower than that in the control group[138]. In the group of 5,559 children born in the USA and Denmark whose parents had been treated for cancer there were 239 of

genetic defects (0.43%), while the number in the control group (6564 children) was 306 (0.446%). The authors of the study emphasized that even though large doses of ionizing radiation are the cause of inherited effects in Drosophila flies and in mice, there is no evidence that irradiation of a human foetus leads to genetic defects in children.

In the ABS cohort where, after irradiation in Hiroshima and Nagasaki, 70,000 children were born, no genetic effects were identified. The research presented in the work by Boice confirms that no such effects should be expected in human populations. The authors cautiously conclude that "radiotherapeutic treatment of cancer diseases is not connected with a significant risk, if any, from the point of view of genetic defects in children."

3.7.7.2 The effects of foetal irradiation

Supporters of the LNT thesis use the consequences of foetal irradiation, in particular the results of the "Childhood Cancer Oxford Study" as the proof of validity of the LNT model. However, it is unknown whether children that have been exposed to X-rays *in utero* have been selected for the procedure due to suspected congenital diseases, i.e. whether the Oxford cohort is representative for normal foetuses. At the same time, the research involving the cohort of the Japanese survivors of the Hiroshima and Nagasaki bombs, referred to as the ABS cohort ("A-bomb survivors") has not demonstrated any carcinogenic effects of foetal irradiation, in the case of neither leukaemia nor solid tumours[139]. Similar experiments conducted on animals have not demonstrated carcinogenic effects of *in utero* irradiation with small doses. In numerous studies, the impact of X-ray radiation absorbed by foetuses *in utero* or by small children was analyzed. Doll and Wakeford concluded that *in utero* irradiation with doses equal to 5-10 mSv is connected with an increase of the incidence of leukaemia and solid tumours in children[140]. However, their work was subject to sharp criticism by Professor Mossman as early as mid-1990's. Since then, 19 case-control studies and 6 cohort studies have been completed, which have not demonstrated any significant increase in the incidence of cancer caused by small X-ray doses absorbed by a child before or soon after the birth[141].

3.7.7.3 Impact of irradiation of children according to tests conducted in the United Kingdom

In November 1999, the United Kingdom's National Radiological Protection Board stated that "The results of the new huge epidemiological study are not in line with the thesis that exposure of parents to radiation before conception of a child is the cause of leukaemia and non-Hodgkin lymphoma (LNHL) in children."

This hypothesis was proposed by the Gardner group in 1990. In response, the Committee on Medical Aspects of Radiation in the Environment (COMARE), established by the British government, recommended conducting a detailed study[142].

The NRPB's report titled "Cancer in the Offspring of Workers Exposed to Radiation" based on a study involving 36,000 children over a period of 30 years and an analysis of data concerning 120,000 workers exposed to radiation leads to the conclusion that the results of this study do not support Gardner's thesis[143].

In particular:

"No confirmation was found for higher risk for parents who absorbed a dose of 100 mSv or larger prior to conception or a dose of 10 mSv or larger within 6 months prior to conception.

Also, no relation between foetal irradiation and other categories of cancer in children was found**Błąd! Nie zdefiniowano zakładki.**. The both the 1994[144] report and the latest report[145] of the COMARE which uses the most sensitive statistical and mathematical methods confirm that "nothing indicates an increase in the incidence of any cancers in children within 25 km from nuclear power plants."

3.7.7.4 Thyroid cancers in children treated with X-rays

The research on thyroid cancer in children treated with X-rays is described in the article by E. Ron et al.; however, the research did not pertain to small doses, as the linearity range extended from doses above 10 Gy down to 0.1 Gy[146]. The authors mentioned "even up to 0.1 Gy," which means that they do not indicate that their work covers the range of doses discussed in the book, i.e. several mSv and less. Moreover, the authors of the aforementioned article recognize the existence and effectiveness of repair processes in human bodies by saying that "spreading dose over time (from a few days to >1 year) may lower risk, possibly due to the opportunity for cellular repair mechanisms to operate." Evidently, respected experts, such as E. Ron, J.H. Lubin, R.E. Shore, K. Mabuchi, B. Modan, L.M. Pottern, A.B. Schneider, M.A. Tucker, and J.D. Boice Jr, recognize the importance of repair processes and can differentiate between the effects of low dose rates and the effects of one-time exposure to large doses.

3.7.8 Achievements in research on biological processes occurring after irradiation of people

3.7.8.1 Comparison of permanent mutations caused by irradiation and those caused by metabolic processes

An analysis of the processes taking place in the human body indicates that radiation is not the only process that causes cell damage. Much more important are the normal metabolic processes which cause the production of over 100 million free radicals a day in each cell, which may cause damage to the DNA. The free radicals cause approximately a million DNA nucleotides a day in each cell. There is also damage caused by normal division of cells and DNA multiplication, as well as the loss of nearly 5,000 of purine cations a day per cell of a human body due to the destruction of their bonds caused by the normal warmth of the human body. Metabolism causes ten million times more cell mutations (repaired and not repaired) than natural radiation.[147]

Table 7.7.1. Comparison of the number of cell damages caused daily by metabolic processes and by radiation with dose rate equal to 1 mSv/year.

	Metabolism	Radiation (1 mSv/year)
Number of free radicals created in the vicinity of the DNA	100,000,000	
Number of DNA defects per cell	1,000,000	0.005
Ratio of the number of unrepaired DNA changes or DNA changes repaired with errors to the number of DNA defects	1 per 10,000	1 per 500
Number of DNA unrepaired DNA changes or DNA changes repaired with errors per cell	100	0.000,01
Ratio of the number of permanent mutations that are unrepaired or repaired with errors to the number of DNA changes that are repaired or repaired with errors	1 per 100	1 per 100
Number of permanent mutations that are unrepaired or repaired with errors per cell	1	0.000,000,1
Ration of mutations caused by radiation to mutations caused by metabolism	1 to 10,000,000	

Given that the due to radiation two DNA spirals at a time may become damaged, the ration of the number of DNA changes that are not repaired or repaired with errors to the number of DNA defects is 20 times larger for radiation than for metabolic processes. This fact is always emphasized by supporters of the LNT hypothesis in discussions concerning the effectiveness of repair processes. However, even though the number of DNA defects caused by metabolism is so huge that after the repair process the number of unrepaired permanent mutations caused by radiation is only one ten millionth of the number of unrepaired permanent mutations caused by metabolism.

In order for the organism to survive, it must have very effective methods of removing free radicals and to repair and eliminate DNA defects. The methods are also an effective response to ionizing radiation.

3.7.8.2 Repair processes in living organisms

Over the last decade we witnessed huge progress in our understanding of biological processes that serve the purpose of protecting the cells and the human body from radiation hazards. As it turns out, the nature of the defence mechanisms is variable and depends on the value of the dose. The supporters of the LNT hypothesis used to claim that both small and large doses cause similar DNA defects and that repair processes may sometimes lead to errors and, consequently, initiate carcinogenic processes. Currently, the French Academy of Sciences emphasizes that even though DNA defects in cells occur identically regardless of the dose rate, the defence processes on the cell, tissue, and organism level are different according to the rate and value of the dose[148].

In particular, at very small doses (less than several mSv), activation of defence processes by radiation causes increased immunity of the body to other dangers present in normal metabolic processes. For example, the effectiveness of removal of toxins, such as active oxidants, increases, which protects DNA from becoming damaged. While the number of DNA defects caused by metabolic processes reaches a million a day per cell, the number of radiation defects per cells at low radiation rates, e.g. 1 mSv/year, is approx. 0.005 a day. Even though radiation damage includes a larger proportion of double damage of the DNA strands than damages caused by metabolic processes, increasing the effectiveness of the biological defence mechanisms in our bodies has effects that are many times greater than the minimum relative increase in the danger to the body caused by small radiation doses.

Moreover, at small doses, no negative effects of irradiation of tissues are noticeable because damaged cells are not being repaired but eliminated by way of apoptosis, i.e. programmed death of cells containing unrepaired DNA damages. From the point of view of the organism (when the proportion of damaged cells is very small) this is the safest solution. According to the French Academy of Sciences, “elimination of damaged cells protects the body from potential malignant tumours.”

When the doses reach values exceeding several mSv, but less than approx. 100 mSv, defence mechanisms are activated and defective cells are eliminated or repaired in very effective processes. Such processes were first developed at the time life emerged on Earth; if it was not for them, no organism would endure the millions of cell defects per seconds. The effectiveness of the defence processes increases with the dose, so that in the range of over ten and several dozen mSv the hormesis effect may take place; the reduction cell defects caused by metabolic processes is much more important than possible imperfections of the repair processes. However, defects caused by radiation are of different nature than defects caused by metabolism: 1) the fraction of the double damage to DNA strands is larger; 2) there are clusters of defects caused by hydroxyl radicals; 3) the distribution of cell defects is more heterogenic. Such defects may be repaired with errors, although a number of studies indicate that the increase of effectiveness of defence processes is of the greatest importance[148].

At larger doses, above 100-200 mSv, the concentration of defective cells increases and the DNA repair processes may include errors, whose likelihood increases with the dose rate. If apoptosis is not initiated, errors in repair of the DNA may allow defective cells to survive and initiate the development of a neoplasm.

At doses above 500 mSv, the cell multiplication rate increases in order to compensate the loss of cells damaged by radiation. Fast division of cells interferes with the repair processes and the likelihood of erroneous repairs and development of a neoplasm increases.

Such differences in the repair processes explain why at small doses the impact of radiation on health may be positive, even though it is negative at large doses. Although the ICRP still supports the LNT hypothesis and it still constitutes the grounds for radiation protection regulations and comparative analyses, in the unanimous opinion of the French Academy of Sciences and the French National Academy of Medicine the current knowledge indicates that very small doses are not dangerous.

The French National Academy of Medicine emphasizes the fact that the most recent biological data indicates that the molecular and cellular processes that determine where a cell survives or undergoes mutagenesis depending on the dose value and rate are very complex and variable. Both the National Academy of Medicine and the Academy of Sciences, and many scientists alike, emphasize the fact that the hormetic model is the most suitable to describe the processes occurring after irradiation of people with small doses.

3.7.9 **Conclusion**

Long-term studies conducted in many parts of the world and among different populations have proven beyond any doubt that small doses of radiation – comparable to natural background radiation – have no negative impact on human health, including adults, children, and the offspring of persons exposed to radiation.

Still, until recently, comparative analyses would assume that every dose of radiation carries a risk that is proportional to the dose. All analyses performed until 2005, the results of which are quoted in this study, were based on this assumption.

The leading specialists in health protection call for additional studies and development of models that would explain the impact of small doses of radiation on human health. [UNSCEAR 1994, Sugahara, 1994]. Studies are underway, but in the meantime everyone agrees that small doses of radiation either have no negative impact at all or these impacts are undetectable even when studying the largest populations. On the other hand, many renowned scientists and most respectable institutions claim that the majority of results even suggest a beneficial effect of small doses of radiation. Thus, it is evident now that there are no reasons to be concerned about small radiation doses.